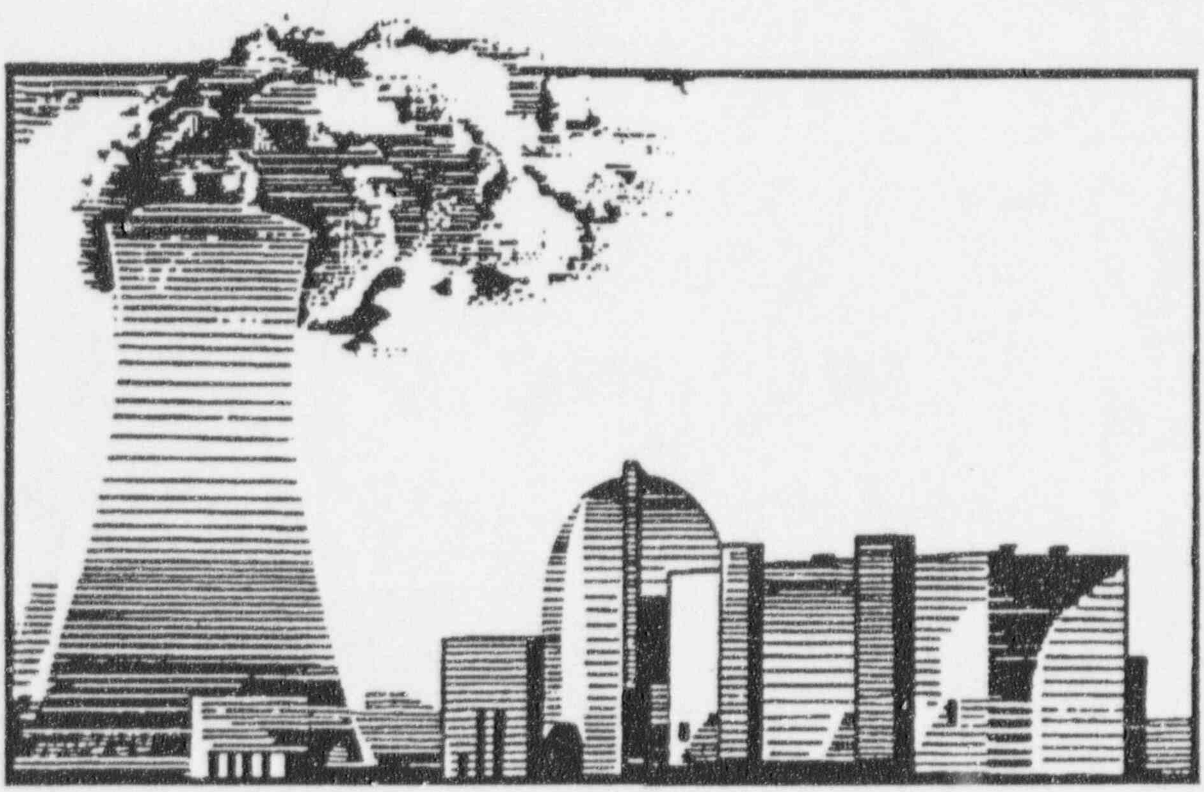
UNION ELECTRIC

Rev. 6

CALLAWAY PLANT
OFF-SITE DOSE CALCULATION MANUAL

October, 1996



RECORD OF REVISIONS

Rev. No. 2 **Date: May, 1991**

Section 2.4.2 - Changed gross alpha analysis frequency from "each batch" to a monthly composite as per Table 9.3-A, and the Callaway Plant NPDES permit (reissued March 15, 1991).

Rev. No. 3 **Date: June, 1993**

Deleted HF-RE-45 and LE-RE-59 as effluent monitors. Revised table numbering for consistency with those in Section 9.0, deleted redundant material, incorporated 1992 Land Use Census results, moved LLD description to Attachment 1, moved REC Bases to Attachment 2. Deleted reporting requirements for solid radwaste, which are described in APA-ZZ-01011, PROCESS CONTROL PROGRAM. Addressed compliance with 10 CFR 20.1301. Revised the dilution flow rate to allow values other than 5000 gpm, based on dilution flow monitor setpoint. Revised "MPC" terminology to "ECV". Added Action 46 to REC 9.2 to clarify actions for inoperable mid and high range WRGM Channels. Revised references to be consistent with the revised 10 CFR 20. Added Appendix A. Revised Action 41 of Rec 9.2 and the operability requirements of GT-RE-22/33. Incorporated the revised R_i values in Tables 3.2 and 3.3. Added Section 6.2 and Table 6.5.

Rev. No. 4 **Date: September, 1994**

Increased the minimum channels OPERABLE requirement of REC 9.2 for GT-RE-22 & 23 from 1 channel to 2 channels. Revised Action 41 and the Bases for REC 9.2 accordingly. Incorporated the operability requirements from Tech Spec 3.9.9 into the Action statement for clarity. (Refer to SOS 94-1176).

Rev. No. 5 **Date: February, 1995**

Removed the REMP station locations. Removed particulate nuclides with a half-life of less than 8 days from Tables 3.2-3.4 and removed C^{14}, P^{32}, Ni^{63}, Te^{125m}, and from Tables 2.1, 2.2, 3.2, 3.3, and 3.4. Changed the reporting frequency of the Semiannual Effluent Release Report from semiannual to annual. Removed the meat, milk and vegetable pathway dispersion parameters from Tables 6.1, 6.2, and 6.3, and clarified the applicability of the dispersion parameters and dose locations in Table 6.4. Relocated REC 9.1 and 9.2 to the FSAR. Revised footnotes 3 and 7 of Table 9.6-A to require additional sampling of the Unit Vent in the event of a reactor power transient, only if the Unit Vent noble gas activity increases by a factor of 3 or greater. Added Section 4.1.3.1.3 for determination of dose due to the on-site storage of low level radioactive waste.

Rev. No. 6 **Date September, 1996**

<u>Section 2:</u> Added dose factors (A_{it})for Ag^{110m}, Np^{237}, Pu^{238}, $Pu^{239/240}$, Pu^{241}, Am^{241}, Cm^{242}, and $Cm^{234/244}$ to Table 2.1, and Bioaccumulation Factors (Bf_i) for Ag, Pu, Am, and Cm to Table 2.2 due to a change in the liquid radwaste treatment process. Revised the description of the methodology for performing the 31 day dose projection in Section 2.5. Revised the maximum allowable background for HB-RE-18.

<u>Section 3:</u> Eliminated Y^{91m} and Tc^{99m} from Table 3.4 (Meat Pathway) due to a half-life of < 8 days. Substituted the phrase "more restrictive" in lieu of "lesser" in Section 3.2. Revised the definition of F_a in equation 3.1. Added description of use of samples to verify dose rates in Section 3.3.1.2. Augmented the definition of q_i in Section 3.3.2.1. Edited equations 3.13 and 3.14 and added equation 3.15 to clarify dose calculations. Revised the methodology for performing the 31 day dose projection in Section 3.4.

<u>Section 4:</u> Strengthened the discussion of the reevaluation of assumptions in Section 4.1.3.

<u>Section 6:</u> Added new table 6.6 to describe the selection and use of dispersion parameters during the preparation of the Annual Effluent Release Report. Updated Tables 6.1 and 6.2 to reference the 1995 Land Use Census. There were no changes in the receptor locations.

<u>Section 8:</u> Replaced the reference to HDP-ZZ-04500 to a more generic reference to the plant operating procedures, due to change in organizational structure and responsibilities.

RECORD OF REVISIONS

Section 9: (1) Eliminated 9.0.1 and 9.0.2 due to redundancy with Technical Specifications 3.0.1 and 3.0.2; (2) Revised Table 9.3-A to incorporate sampling and analysis requirements for TRU nuclides in liquid effluents; (3) Eliminated sampling of Fuel Building Exhaust from Table 9.6-A and the associated footnotes due to redundancy with Unit Vent sampling; revised the continuous sampling requirements for the gaseous batch release points consistent with plant design; revised the H^3 analysis frequency for Purges from weekly to "prior to each purge"; and, (4) Revised the air sampling station location criteria on Table 9.11-A and footnote # 1, and eliminated footnote #3 in order to be less generic and more descriptive of the parameters used in determining the station locations (see SOS 95-2280). Revised the location requirements for milk and vegetables. Revised description of use of baseline samples to trigger gamma isotopic analysis in footnote #4, revised requirement for location of downstream sample station in footnote #6. Revised Surveillance Requirement 9.10.2.1 to eliminate liquid effluents from the surveillance. (5) Revised REC 9.5 and REC 9.9 to eliminate exceptions for partially tested effluents being released in excess of the respective limit.

Section 11: Added reference 11.14.13.

Attachment 2: Revised the Bases for REC 9.10 to support the elimination of liquid effluents from Surveillance 9.10.2.1.

The remaining changes are editorial in nature and have no technical impact.

(This revision implements SOS's 95-2055, 96-0167, 96-0961, 95-2280, and 96-0986).

CALLAWAY PLANT

ADMINISTRATIVE PROCEDURE

APA-ZZ-01003

OFF-SITE DOSE CALCULATION MANUAL

RESPONSIBLE DEPARTMENT ______ HEALTH PHYSICS ______

WRITTEN BY ______ /c ______

PREPARED BY ______ CCG [illegible] ______

APPROVED BY ______ R Affdt ______

DOCUMENT CONTROL
ISSUED
OCT 23 1996
ACCOUNTABLE HOLDER
113

DATE ISSUED 10-23-96

This procedure contains the following:

Pages	1	through	88
Attachments	1	through	2
Tables		through	
Figures	1	through	1
Appendices	A	through	A
Checkoff Lists		through	

This procedure has ______ checkoff list(s) maintained in the mainframe computer.

TABLE OF CONTENTS

TABLE OF CONTENTS

FIGURES

TABLE OF CONTENTS

Section | Page Number

TABLES

RECORD OF REVISIONS

Rev. No. 0 **Date: March 1983**

Rev. No. 1 **Date: November, 1983**

Revised to support the current RETS submittal and to incorporate NRC Staff comments.

Rev. No. 2 **Date: March, 1984**

Revised to incorporate NRC Staff comments

Rev. No. 3 **Date: 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. No. 4 **Date: February, 1987**

Minor clarifications, incorporated 31-day projected dose methodology. Change in the utilization of areas within the Site Boundary.

Rev. No. 5 **Date: 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 TLD testing required by Reg. Guide 4.13, revised Tables 1, 2, 4 and 5 to add additional nuclides, deleted redundant material from Chapter 6.

Rev. No. 6 **Date: May, 1989**

Revised methodology for calculating maximum permissible liquid effluent discharge rates and liquid effluent discharge rates and liquid effluent monitor setpoints, provided methodology for calculating liquid effluent monitors response correction factors, provided an enhanced description of controls on liquid monitor background limits, provided additional liquid and gaseous dose conversion factors and bioaccumulation factors (Tables 1, 2, 4 & 5), provided description of the use of the setpoint required by Technical Specification 4.9.4.2 during Core Alterations, added discussion of gaseous and liquid monitor setpoint selection in the event that the sample contains no detectable activity, added minimum holdup requirements for Waste Gas Decay tanks, revised dispersion parameters and accompanying description per FSAR Change Notice 88-42.

APA-ZZ-01003
Rev. No. 0 **Date: August, 1989**

Radiological Effluent Technical Specifications were moved from the Callaway Plant Technical Specifications to Section 9.0, Radioactive Effluent Controls, of the ODCM as per NRC Generic Letter 89-01. At the same time, in order to formalize control of the entire ODCM, it was converted to APA-ZZ-01003, OFF-SITE DOSE CALCULATION MANUAL.

Rev. No. 1 **Date: October, 1990**

Revise Action 41 of Table 9.2-A to allow continued purging for 24 hours as per Amendment 20 to operating license, issued 4/10/87.

OFF-SITE DOSE CALCULATION MANUAL

1. PURPOSE AND SCOPE

The OFF-SITE DOSE CALCULATION MANUAL (ODCM) describes the methodology and parameters used in the calculation of off-site doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM also contains the Radioactive Effluent Controls and Radiological Environmental Monitoring Program required by Technical Specification 6.8.4, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Technical Specifications 6.9.1.6 and 6.9.1.7.

Compliance with the Radiological Effluent Controls limits demonstrates compliance with the limits of 10 CFR 20.1301. (Ref. 11.1.1, 11.2.1, 11.23.3)

2. LIQUID EFFLUENTS

2.1 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 Effluent Concentration Values (ECV) 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:

$$\frac{cf}{F + f} \leq C$$

Where:

C = The liquid effluent concentration value (ECV) implementing REC 9.3.1.1 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 related to the volumetric flow of the effluent line and directly related to the volumetric flow of the dilution stream plus the effluent steam, represents a value, which, if exceeded, would result in concentrations exceeding the values of 10 CFR Part 20 Appendix B, Table II, Column 2, 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 rate setpoint as measured prior to the release point, in volume per unit time. {If (F) is large compared to (f), then $F + f \cong F$}.

(Ref. 11.8.1)

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 limiting concentration which corresponds to the liquid radwaste effluent monitor setpoint is to be calculated using methodology from the expression above.

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

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

The alarm/trip setpoint calculations are based on the minimum dilution flow rate (corresponding to the dilution flow rate setpoint), the maximum effluent stream flow rate, and the actual isotopic analysis. Due to the possibility of a simultaneous 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 2.2. In the event the alarm/trip setpoint is reached, an evaluation will be performed using actual dilution and effluent flow values and actual isotopic analysis to ensure that REC 9.3.1.1 limits were not exceeded.

2.1.1 Continuous Liquid Effluent Monitors

The radiation detection monitor associated with continuous liquid effluent releases is (Ref. 11.6.1, 11.6.2):

Monitor I.D.	Description
BM-RE-52	Steam Generator Blowdown Discharge Monitor

The Steam Generator Blowdown discharge is 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 Table 9.3-A.

2.1.2 Radioactive Liquid Batch Release Effluent Monitors

The radiation monitor which is associated with the liquid effluent batch release system is (Ref. 11.6.4):

Monitor I.D.	Description
HB-RE-18	Liquid Radwaste Discharge Monitor

This effluent stream is normally considered to be radioactive. The sampling frequency, minimum analysis frequency, and the type of analysis performed are as per Table 9.3-A.

2.2 CALCULATION 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. 11.8.1)

2.2.1 Calculation of the ECV Sum

The isotopic concentration of the release(s) being considered must be determined. This is obtained from the analyses required per Table 9.3-A, and is used to calculate an ECV sum (ECVSUM):

$$ECVSUM = \left(\sum_{i = g,\, a,\, s,\, t,\, f} (C_i)/(ECV_i)\right) \qquad (2.3)$$

Where:

c = the concentration of each measured gamma emitting nuclide observed by gamma-ray spectroscopy of the waste sample.

C_a^* = the measured concentration of alpha emitting nuclides measured by gross alpha analysis of the monthly composite sample.

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

C_t = the measured concentration of H-3 in the waste sample.

C_f^* = the measured concentration of Fe-55 as determined by analysis of the quarterly composite sample.

ECV_g, ECV_s, ECV_a, ECV_f, ECV_t = are the limiting concentrations of the appropriate radionuclides from 10 CFR 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.

---

* Values for these concentrations are based on previous composite sample analyses as required by Table 9.3-A.

For the case ECVSUM ≤1, the monitor tank effluent concentration meets the limits of REC 9.3.1.1 without dilution and the effluent may be released at any desired flow rate. If ECVSUM > 1 then dilution is required to ensure compliance with REC 9.3.1.1 concentration limits. If simultaneous releases are occurring or are anticipated, an allocation fraction, N, must be applied so that available dilution flow may be apportioned among simultaneous discharge pathways. The value of N may be any value between 0 and 1 for a particular discharge point, provided that the sum total allocation fractions for all discharge points must be ≤1.

2.2.2 Calculation of the Maximum Permissible Liquid Effluent Discharge Flowrate

The maximum permissible liquid effluent discharge flowrate is calculated by:

$$f_{max} \leq (F + f_p)\,(SF)\,(N) \div (ECVSUM) \qquad (2.4)$$

Where:

f_{max} = maximum permissible liquid effluent discharge flowrate, in (gallons/minute);

f_p = the expected undiluted liquid effluent flowrate, in gpm.

N = the allocation fraction which apportions dilution flow among simultaneous discharge pathways (see discussion above)

SF = the safety factor; an administrative factor used to compensate for statistical fluctuations and errors of measurements. This factor also provides a margin of safety in the calculation of the maximum liquid effluent discharge flowrate (f_{max}). The value of SF should be ≤ 1.

F & ECVSUM, are previously defined.

The dilution water supply is furnished with a flow monitor which isolates the liquid effluent discharge if the dilution flow rate falls below its setpoint value.

In the event that f_{max} is less than f_p, then the value of f_{max} is substituted into the equation for f_p, and a new value of f_{max} is calculated. This substitution is performed for three iterations in order to calculate the correct value of f_{max}.

2.2.3 Calculation Of Liquid Effluent Monitor Setpoint

The liquid effluent monitors are NaI(Tl) based systems and respond primarily to gamma radiation. Accordingly, their setpoint is based on the total concentration of gamma emitting nuclides in the effluent:

$$c = BKG + (\Sigma (C_g) \div SF) = \mu Ci/ml \qquad (2.5)$$

Where:

c = the monitor setpoint as previously defined, in (μCi/ml);

BKG = the monitor background prior to discharge, in (μCi/ml);

C_g and SF are as previously defined.

The monitor's background is controlled at an appropriate limit to ensure adequate sensitivity. Utilizing the methodology of ANSI N13.10-1974 (Ref. 11.21), the background must be maintained at a value of less than or equal to 9E-6 μCi/ml (relative to Cs-137) in order to detect a change of 4E-7 μCi/ml of Cs-137. (Ref. 11.25).

In the event that there is no detectable gamma activity in the effluent or if the value of $(\Sigma(C_g) \div SF)$ is less than the background of the monitor, then the monitor setpoint will be set at twice the current background of the monitor.

As previously stated, the monitor's response is dependent on the gamma emitting radionuclide distribution of the effluent. Accordingly, a new database conversion factor is calculated for each release based upon the results of the gamma spectrometric analysis of the effluent sample and the measured response of the monitor to National Institute of Standards and Technology (NIST) traceable calibration sources:

$$DBCF_C = \left(\sum\left(C_g\right)\right) \div (\mathrm{CMR}) \times (\mathrm{ECF}) \qquad (2.6)$$

$DBCF_c$ = the monitor data base conversion factor which converts count rate into concentration (μCi/ml);

CMR = the calculated response of the radiation monitor to the liquid effluent;

ECF = the conversion factor for Cs-137, which converts count rate into concentration (μ Ci/ml).

C_g is as previously defined.

The new value of the $DBCF_c$ is calculated and entered into the monitor data base prior to each discharge. A more complete discussion of the derivation and calculation of the CMR is given in reference 11.14.7.

2.3 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 the 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 liquid effluent discharge is monitored prior to entering the dilution discharge line and will automatically be terminated if the pre-selected alarm/trip setpoint is exceeded. Concentrations are determined primarily from the gamma isotopic and H-3 analyses of the liquid batch sample. For gross alpha, Sr-89, Sr-90, & Fe-55, the measured concentration from the previous composite analysis is used. Composite samples are collected for each batch release. Monthly analysis for gross alpha and quarterly analyses for Sr-89, Sr-90, and Fe-55 are performed in accordance with Table 9.3-A. Doses from liquids discharged as continuous releases are calculated by utilizing the last measured values of samples required in accordance with Table 9.3-A.

2.4 DOSE DUE TO LIQUID EFFLUENTS

2.4.1 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. 11.8.3)

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. 11.7.1, 11.6.6), 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. 11.6.7)

2.4.2 Calculation Of Dose From Liquid Effluents

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. Dose is calculated for all radionuclides identified in liquid effluents released to UNRESTRICTED AREAS using the following expression (Ref. 11.8.3):

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

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 ℓ th time period over which $C_{i\ell}$ and F_ℓ are averaged for all liquid releases, in hours. Δt_ℓ corresponds to the actual duration of the release(s).

$C_{i\ell}$ = 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 Table 9.3-A, (in mrem/hr) per (μCi/ml). The calculation of the $A_{i\tau}$ values is detailed in Ref. 11.14.5 and are given in Table 2.1.

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

$$F_\ell = \frac{f_{max}}{(F + f_{max})\, 89.77}$$

Where:

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

F = average dilution flow

89.77 = site specific applicable factor for the mixing effect of the discharge structure. (Ref. 11.5.1)

The term $C_{i\ell}$ is the undiluted concentration of radioactive material in liquid waste at the common release point determined in accordance with REC 9.3.1.1, Table 9.3-A, "Radioactive Liquid Waste Sampling and Analysis Program". All dilution factors beyond the sample point(s) are included in the F_ℓ term.

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, Missouri, 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 potable 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. 11.6.6).

2.4.3 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 Lower Level of Detection (LLD) level for that nuclide. The "less than" values are not used in the dose calculations.

2.5 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 REC 9.3.1.1. and 9.4.1.1.

Projected doses due to liquid releases to UNRESTRICTED AREAS are determined each 31 days and are equal to the average of the previous 12 months. This may be modified as appropriate to account for changes in radwaste treatment which may have a significant effect on the projected doses.

2.6 DOSE FACTORS

The dose conversion factors provided in Table 2.1 were derived from the appropriate dose conversion factors of Regulatory Guide 1.109 and other sources as necessary (Ref. 11.14.5 and 11.14.12) Non-gamma emitting nuclides not listed in Table 9.3-A are not considered.

TABLE 2.1

INGESTION DOSE COMMITMENT FACTOR ($A_{i\tau}$) 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
Na-24	4.07E+02	4.07E+02	4.07E+02	4.07E+02	4.07E+02	4.07E+02	4.07E+02
Cr-51	No Data	No Data	1.27E+00	7.62E-01	2.81E-01	1.69E+00	3.20E+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-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	2.27E+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	0
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
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	9.11E-03	1.31E-02	1.29E-01	No Data	2.36E-01	6.70E-03	0

TABLE 2.1 (Cont'd)

INGESTION DOSE COMMITMENT FACTOR ($A_{i\tau}$) FOR ADULT AGE GROUP

(mrem/hr) per (µCi/ml)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
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
Cd-109	No Data	5.54E+02	1.94E+01	No Data	5.31E+02	No Data	5.59E+03
Ag-110m	8.83E-01	8.17E-01	4.85E-01	No Data	1.61E+00	No Data	3.33E+02
Sn-113	5.66E+04	1.61E+03	3.26E+03	9.18E+02	No Data	No Data	1.69E+05
Sb-124	6.69E+00	1.26E-01	2.65E+00	1.62E-02	No Data	5.21E+00	1.90E+02
Sb-125	4.28E+00	4.78E-02	1.02E+00	4.35E-03	No Data	3.30E+00	4.71E+01
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	3.78E+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	1.09E+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
Eu-154	3.67E+01	4.52E+00	3.21E+00	No Data	2.16E+01	No Data	3.27E+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

TABLE 2.1 (Cont'd)

INGESTION DOSE COMMITMENT FACTOR ($A_{i\tau}$) FOR ADULT AGE GROUP

(mrem/hr) per (μCi/ml)

Nuclide	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Np-237	3.27E+04	2.84E+03	1.32E+03	No Data	9.85E+03	No Data	1.90E+03
Np-239	2.84E-02	2.80E-03	1.54E-03	No Data	8.72E-03	No Data	5.74E+02
Pu-238	5.69E+03	8.01E+02	1.43E+02	No Data	6.12E+02	No Data	6.11E+02
Pu-239*	6.58E+03	8.87E+02	1.60E+02	No Data	6.78E+02	No Data	5.67E+02
Pu-241	1.38E+02	7.06E+00	2.78E+00	No Data	1.28E+01	No Data	1.17E+01
Am-241	4.89E+04	1.72E+04	3.23E+03	No Data	2.43E+04	No Data	4.43E+03
Cm-242	1.23E+03	1.25E+03	8.19E+01	No Data	3.72E+02	No Data	4.73E+03
Cm-243**	3.82E+04	1.44E+04	2.24E+03	No Data	1.05E+04	No Data	4.67E+03

*Includes Pu-240 contribution

**Includes Cm-244 contribution

TABLE 2.2

BIOACCUMULATION FACTOR (Bf_i)[a]

(pCi/kg) per (pCi/liter)

Element	Bf_i Fish (Freshwater)
H	9.0 E - 01
Be	2.0 E + 00
*Na	1.0 E + 02
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
Ag	2.3 E + 00
Cd	2.0 E + 02
Sn	3.0 E + 03
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
Eu	2.5 E + 01
Hf	3.3 E + 00
W	1.2 E + 03
Np	1.0 E + 01
Pu	3.5 E + 00
Am	2.5 E + 01
Cm	2.5 E + 01

[a] Values from Regulatory Guide 1.109, Rev. 1, Table A-1 and References 11.14.4, 11.14.8, and 11.14.13.

3. GASEOUS EFFLUENTS

3.1 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 REC 9.6.1.1, at the SITE BOUNDARY location with the highest annual average X/Q value.

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. 11.6.3)

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. 11.8.5) Table 9.2-B provides the instrument surveillance requirements, such as calibration, source checking, functional testing, and channel checking.

3.1.1 Continuous Release Gaseous Effluent Monitors

The radiation detection monitors associated with continuous gaseous effluent releases are (Ref. 11.6.8, 11.6.9):

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 3.2. 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.

3.1.2 Batch Release Gaseous Monitors

The radiation monitors associated with batch release gaseous effluents are (Ref. 11.6.9, 11.6.10, 11.6.11):

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

The Containment Purge System continuously monitors 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.

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 REC 9.6.1.1 are not exceeded.

3.2 GASEOUS EFFLUENT MONITOR SETPOINTS

The alarm/trip setpoint for gaseous effluent monitors is determined based on the more restrictive of the total body dose rate (equation 3.1) and skin dose rate (equation 3.3), as calculated for the SITE BOUNDARY.

During core alterations, the setpoint for the Containment Purge Monitors, GT-RE-22 and GT-RE-33 is set at a value of less than or equal to 5E-3 μCi/cc, as required by Technical Specification 4.9.4.2. The actual setpoint value will be reduced according to the Instrument Loop Uncertainty Estimate (ILUE). This value will also be utilized in the event that there is no detectable noble gas activity in the containment atmosphere sample analyzed in accordance with REC 9.6.1.1. The full derivation of this value is discussed in reference 11.14.6.

3.2.1 Total Body Dose Rate Setpoint Calculations

To ensure that the limits of REC 9.6.1.1 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} = REC 9.6.1.1 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 REC 9.6.1.1.

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:

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

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 9.6-A. 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/m^3). Refer to Tables 6.1, 6.2 and 6.4.

K_i = the total body dose factor due to gamma emissions for each identified noble gas radionuclide, in (mrem/yr) per (µCi/m^3). (Table 3.1)

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

Q_i is calculated as the product of the ventilation path flow rate and the measured activity of the effluent stream as determined by sampling.

3.2.2 Skin Dose Rate Setpoint Calculation

To ensure that the limits of REC 9.6.1.1 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 alarm/trip setpoint based on the skin dose rate.

D_s = REC 9.6.1.1 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:

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

Where:

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

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).

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

3.3 CALCULATION OF DOSE AND DOSE RATE FROM GASEOUS EFFLUENTS

3.3.1 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. 11.8.6)

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

3.3.1.1 Noble Gases

The release rate limit for noble gases is determined according to the following general relationships (Ref. 11.8.6):

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

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

Where:

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

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

L_i, M_i, K_i, $\overline{(X/Q)}$, D_{tb} & D_s are as previously identified.

3.3.1.2 Radionuclides Other Than Noble Gases

The release rate limit for Iodine-131 and Iodine-133, for tritium, and for all radioactive materials in particulate form with half lives greater than 8 days is determined according to (Ref. 11.8.7):

$$D_o = \sum_i R_i \left[\overline{X/Q} \right] Q_i \leq 1500 \text{ mrem/yr} \qquad (3.7)$$

Where:

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

R_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 (μCi/m^3).

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

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

The dose parameter (R_i) includes the internal dosimetry of radionuclide, i, and the receptor's breathing rate, which are functions of the receptor's age. The child age group has been selected as the limiting age group. All radiodines are assumed to be released in elemental form (ref. 11.8.7).

R_i values were calculated according to (Ref. 11.8.8):

$$R_i = K' \text{ (BR) } DFA_i \qquad (3.8)$$

Where:

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

BR = The breathing rate. (Regulatory Guide 1.109, Table E-5).

DFA_i = The maximum organ inhalation dose factor for the ith radionuclide, in (mrem/pCi). The total body is considered as an organ in the selection of DFA_i. (Ref. 11.11.5 and 11.14.4)

The results of periodic tritium, iodine and particulate samples of the Unit Vent and Radwaste Vent are used to verify the dose rate limit was not exceeded for the period during which the samples or composite samples were obtained.

3.3.2 Dose Due to Gaseous Effluents

3.3.2.1 Noble Gases

The air dose at the SITE BOUNDARY due to noble gases is calculated according to the following methodology (Ref. 11.8.9):

During any calendar quarter, for gamma radiation:

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

During any calendar quarter, for beta radiation:

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

During any calendar year, for gamma radiation:

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

During any calendar year, for beta radiation:

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

Where:

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

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

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

q_i = The releases of noble gas radionuclides, i, in gaseous effluents, for all gaseous releases in (μCi). Releases are cumulative over the calendar quarter or year as appropriate. q_i is calculated as the product of the ventilation flow rate and the measured activity of the effluent stream as determined by sampling.

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

$\overline{X/Q}$ & M_i are as previously defined.

3.3.2.2 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:

$$\Sigma_j D_{I,j} \leq 7.5 \text{ mrem} \qquad (3.13)$$

During any calendar year:

$$\Sigma_j D_{I,j} \leq 15 \text{ mrem} \qquad (3.14)$$

For each pathway, j, (i.e., for inhalation, ground plane, meat, cow- milk, goat- milk, and vegetation) $D_{I,j}$ is calculated according to the expression:

$$D_{I,j} = 3.17E\text{-}8\ \Sigma_i R_{I,i,j} \{W_j q_i\} \qquad (3.15)$$

Where:

$D_{I,i,j}$ = Dose in mrem, to a MEMBER OF THE PUBLIC from radionuclides other than noble gases, from pathway $_j$, received by organ I (including total body).

$R_{I,i,j}$ = The dose factor for each identified radionuclide, i, in m^2(mrem/yr) per (µCi/sec) or (mrem/yr) per (µCi/m^3) as appropriate, for the pathway $_j$, and exposed organ I, appropriate to the age group of the critical MEMBER OF THE PUBLIC receptor.

W_j = $\left(\overline{X/Q}\right)$ for the inhalation and tritium pathways, in (sec/m^3). Refer to Tables 6.1, 6.2, and 6.4 for applicability.

W_j = $\left(\overline{D/Q}\right)$ for the food and ground plane pathways, in (meters^{-2}). Refer to Tables 6.1, 6.2 and 6.4 for applicability.

$\overline{(D/Q)}$ = the average relative deposition of the effluent at or beyond the SITE BOUNDARY, considering depletion of the plume during transport.

q_i = The releases of radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases, I, in gaseous effluents, for all gaseous releases in (µCi). Releases are cumulative over the calendar quarter or year as appropriate. q_i is calculated as the product of ventilation flow rate and the measured activity of the effluent stream as determined by sampling.

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

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

For the direction sectors with existing pathways within 5 miles from the site, the appropriate $R_{I,i,j}$ values are used. If no real pathway exists within 5 miles from the center of the building complex, the cow-milk $R_{I,i,j}$ 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 $R_{I,i,j}$ for an existing pathway within 5 miles is less than a cow-milk $R_{I,i,j}$ at 4.5 to 5.0 miles, then the value of the cow-milk $R_{I,i,j}$ at 4.5 to 5.0 miles is used. (Ref. 11.8.9)

Although the annual average relative concentration$\left(\overline{X/Q}\right)$ and the average relative deposition rate $\left(\overline{D/Q}\right)$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 $\left(\overline{X/Q}\right)$. (Ref. 11.8.9) The Total Body dose from ground plane deposition is added to the dose for each individual organ. (Ref. 11.11.3)

3.4 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. Normally, waste gases will be retained for at least 60 days prior to discharge. These systems will provide reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept ALARA.

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 REC 9.6.1.1, 9.7.1.1, and 9.8.1.1.

Projected doses (gamma air, beta air, and organ dose) due to gaseous effluents at or beyond the SITE BOUNDARY are determined each 31 days and are equal to the average of the previous 12 months. This may be modified as appropriate to account for changes in radwaste treatment which may have a significant effect on the projected doses.

3.5 DOSE FACTORS

The dose conversion factors provided in the following tables were derived from the appropriate dose conversion factors in Regulatory Guide 1.109 and other sources as necessary (Ref: 11.14.9 and 11.14.11). Per USNRC guidance, particulate nuclides with a half-life of less than 8 days are not considered (Ref: 11.24). Y-90, Nb-95, La-140, and Pr-144 are included because the parent half-life is greater than 8 days, and secular equilibrium is assumed. Non-gamma emitting nuclides not listed in Table 9.6-A are also not considered. **(CTSN 43121)**

TABLE 3.1

DOSE FACTOR FOR EXPOSURE TO A SEMI-INFINITE CLOUD OF NOBLE GASES

Radionuclide	Total Body Dose Factor K_i (mrem/yr) per (μCi/m^3)	Skin Dose Factor L_i (mrad/yr) per (μCi/m^3)	Gamma Air Dose Factor M_i (mrad/yr) per (μCi/m^3)	Beta Air Dose Factor Ni (mrad/yr per (μCi/m^3)
Kr-83m	7.56 E-02	-----	1.93 E+01	2.88 E+02
Kr-85m	1.17E+03	1.46E+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

TABLE 3.2

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES

OTHER THAN NOBLE GASES

Ground Plane Pathway

(m^2 mrem/yr) per (μCi/sec)

NUCLIDE	TOTAL BODY	SKIN
Be-7	2.24E+07	3.21E+07
Cr-51	4.66E+06	5.51E+06
Mn-54	1.39E+09	1.63E+09
Fe-59	2.73E+08	3.21E+08
Co-57	2.98E+08	4.37E+08
Co-58	3.79E+08	4.44E+08
Co-60	2.15E+10	2.53E+10
Zn-65	7.47E+08	8.59E+08
Rb-86	8.99E+06	1.03E+07
Sr-89	2.16E+04	2.51E+04
Y-90	5.36E+06	6.32E+06
Y-91	1.07E+06	1.21E+06
Zr-95	2.45E+08	2.84E+08
Nb-95	2.50E+08	2.94E+08
Ru-103	1.08E+08	1.26E+08
Ru-106	4.22E+08	5.07E+08
Ag-110m	3.44E+09	4.01E+09
Cd-109	3.76E+07	1.54E+08
Sn-113	1.43E+07	4.09E+07
Sb-124	8.74E+08	1.23E+09
Sb-125	3.57E+09	5.19E+09
Te-127m	9.17E+04	1.08E+05
Te-129m	1.98E+07	2.31E+07
I-130	5.51E+06	6.69E+06
I-131	1.72E+07	2.09E+07

TABLE 3.2

PATHWAY DOSE FACTORS (R_i) FOR RADIONUCLIDES

OTHER THAN NOBLE GASES

Ground Plane Pathway

(m^2 mrem/yr) per (μCi/sec)

NUCLIDE	TOTAL BODY	SKIN
I-132	1.25E+06	1.47E+06
I-133	2.45E+06	2.98E+06
I-134	4.47E+05	5.31E+05
I-135	2.53E+06	2.95E+06
Cs-134	6.85E+09	8.00E+09
Cs-136	1.51E+08	1.71E+08
Cs-137	1.03E+10	1.20E+10
Ba-140	2.05E+07	2.35E+07
La-140	1.47E+08	1.66E+08
Ce-141	1.37E+07	1.54E+07
Ce-144	6.96E+07	8.04E+07
Pr-144	4.35E+07	5.00E+07
Nd-147	8.39E+06	1.01E+07
Eu-154	2.21E+10	3.15E+10
Hf-181	1.97E+08	2.82E+08

TABLE 3.3

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

Inhalation Pathway

(mrem/yr) per (µCi/m³)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	ND	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03	1.12E+03
Be-7	8.47E+02	1.44E+03	9.25E+02	ND	ND	6.47E+04	2.55E+03
Cr-51	ND	ND	1.54E+02	8.55E+01	2.43E+01	1.70E+04	1.08E+03
Mn-54	ND	4.29E+04	9.51E+03	ND	1.00E+04	1.58E+06	2.29E+04
Fe-55	4.74E+04	2.52E+04	7.77E+03	ND	ND	1.11E+05	2.87E+03
Fe-59	2.07E+04	3.34E+04	1.67E+04	ND	ND	1.27E+06	7.07E+04
Co-57	ND	9.03E+02	1.07E+03	ND	ND	5.07E+05	1.32E+04
Co-58	ND	1.77E+03	3.16E+03	ND	ND	1.11E+06	3.44E+04
Co-60	ND	1.31E+04	2.26E+04	ND	ND	7.07E+06	9.62E+04
Zn-65	4.25E+04	1.13E+05	7.03E+04	ND	7.14E+04	9.95E+05	1.63E+04
Rb-86	ND	1.98E+05	1.14E+05	ND	ND	ND	7.99E+03
Sr-89	5.99E+05	ND	1.72E+04	ND	ND	2.16E+06	1.67E+05
Sr-90	1.01E+08	ND	6.44E+06	ND	ND	1.48E+07	3.43E+05
Y-90	4.11E+03	ND	1.11E+02	ND	ND	2.62E+05	2.68E+05
Y-91	9.14E+05	ND	2.44E+04	ND	ND	2.63E+06	1.84E+05
Zr-95	1.90E+05	4.18E+04	3.70E+04	ND	5.96E+04	2.23E+06	6.11E+04
Nb-95	2.35E+04	9.18E+03	6.55E+03	ND	8.62E+03	6.14E+05	3.70E+04
Ru-103	2.79E+03	ND	1.07E+03	ND	7.03E+03	6.62E+05	4.48E+04
Ru-106	1.36E+05	ND	1.69E+04	ND	1.84E+05	1.43E+07	4.29E+05
Ag-110m	1.69E+04	1.14E+04	9.14E+03	ND	2.12E+04	5.48E+06	1.00E+05
Cd-109	ND	5.48E+05	2.59E+04	ND	4.96E+05	1.05E+06	2.78E+04
Sn-113	1.13E+05	3.12E+03	8.62E+03	2.33E+03	ND	1.46E+06	2.26E+05
Sb-124	5.74E+04	7.40E+02	2.00E+04	1.26E+02	ND	3.24E+06	1.64E+05
Sb-125	9.84E+04	7.59E+02	2.07E+04	9.10E+01	ND	2.32E+06	4.03E+04
Te-127m	2.49E+04	8.55E+03	3.02E+03	6.07E+03	6.36E+04	1.48E+06	7.14E+04
Te-129m	1.92E+04	6.85E+03	3.04E+03	6.33E+03	5.03E+04	1.76E+06	1.82E+05
I-130	8.18E+03	1.64E+04	8.44E+03	1.85E+06	2.45E+04	ND	5.11E+03

TABLE 3.3 (Con't)

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

Inhalation Pathway

(mrem/yr) per (μCi/m³)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-131	4.81E+04	4.81E+04	2.73E+04	1.62E+07	7.88E+04	ND	2.84E+03
I-132	2.12E+03	4.07E+03	1.88E+03	1.94E+05	6.25E+03	ND	3.20E+03
I-133	1.66E+04	2.03E+04	7.70E+03	3.85E+06	3.38E+04	ND	5.48E+03
I-134	1.17E+03	2.16E+03	9.95E+02	5.07E+04	3.30E+03	ND	9.55E+02
I-135	4.92E+03	8.73E+03	4.14E+03	7.92E+05	1.34E+04	ND	4.44E+03
Cs-134	6.51E+05	1.01E+06	2.25E+05	ND	3.30E+05	1.21E+05	3.85E+03
Cs-136	6.51E+04	1.71E+05	1.16E+05	ND	9.55E+04	1.45E+04	4.18E+03
Cs-137	9.07E+05	8.25E+05	1.28E+05	ND	2.82E+05	1.04E+05	3.62E+03
Ba-140	7.40E+04	6.48E+01	4.33E+03	ND	2.11E+01	1.74E+06	1.02E+05
La-140	6.44E+02	2.25E+02	7.55E+01	ND	ND	1.53E+05	2.26E+05
Ce-141	3.92E+04	1.95E+04	2.90E+03	ND	8.55E+03	5.44E+05	5.66E+04
Ce-144	6.77E+06	2.12E+06	3.61E+05	ND	1.17E+06	1.20E+07	3.89E+05
Pr-143	1.85E+04	5.55E+03	9.14E+02	ND	3.00E+03	4.33E+05	9.73E+04
Pr-144	5.96E-02	1.85E-02	3.00E-03	ND	9.77E-03	1.57E+03	1.97E+02
Nd-147	1.08E+04	8.73E+03	6.81E+02	ND	4.81E+03	3.28E+05	8.21E+04
Eu-154	1.01E+07	9.21E+05	8.40E+05	ND	4.03E+06	6.14E+06	1.10E+05
Hf-181	2.78E+04	1.01E+05	1.25E+04	ND	2.05E+04	1.06E+06	6.62E+04

TABLE 3.3 (Cont'd)

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

Meat Pathway

(m^2 mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	ND	2.34E+02	2.34E+02	2.34E+02	2.34E+02	2.34E+02	2.34E+02
Be-7	7.38E+03	1.26E+04	8.07E+03	0.00E+00	1.23E+04	0.00E+00	7.00E+05
Cr-51	0.00E+00	0.00E+00	8.80E+03	4.88E+03	1.33E+03	8.92E+03	4.67E+05
Mn-54	0.00E+00	8.02E+06	2.14E+06	0.00E+00	2.25E+06	0.00E+00	6.73E+06
Fe-55	4.58E+08	2.43E+08	7.52E+07	0.00E+00	0.00E+00	1.37E+08	4.50E+07
Fe-59	3.77E+08	6.10E+08	3.04E+08	0.00E+00	0.00E+00	1.77E+08	6.35E+08
Co-57	0.00E+00	5.92E+06	1.20E+07	0.00E+00	0.00E+00	0.00E+00	4.85E+07
Co-58	0.00E+00	1.64E+07	5.03E+07	0.00E+00	0.00E+00	0.00E+00	9.59E+07
Co-60	0.00E+00	6.94E+07	2.05E+08	0.00E+00	0.00E+00	0.00E+00	3.84E+08
Zn-65	3.76E+08	1.00E+09	6.23E+08	0.00E+00	6.31E+08	0.00E+00	1.76E+08
Rb-86	0.00E+00	5.77E+08	3.55E+08	0.00E+00	0.00E+00	0.00E+00	3.71E+07
Sr-89	4.82E+08	0.00E+00	1.38E+07	0.00E+00	0.00E+00	0.00E+00	1.87E+07
Sr-90	1.04E+10	0.00E+00	2.64E+09	0.00E+00	0.00E+00	0.00E+00	1.40E+08
Y-90	1.93E+05	0.00E+00	5.16E+03	0.00E+00	0.00E+00	0.00E+00	5.49E+08
Y-91	1.80E+06	0.00E+00	4.82E+04	0.00E+00	0.00E+00	0.00E+00	2.40E+08
Zr-95	2.67E+06	5.86E+05	5.22E+05	0.00E+00	8.39E+05	0.00E+00	6.11E+08
Nb-95	4.26E+06	1.66E+06	1.18E+06	0.00E+00	1.56E+06	0.00E+00	3.07E+09
Ru-103	1.55E+08	0.00E+00	5.96E+07	0.00E+00	3.90E+08	0.00E+00	4.01E+09
Ru-106	4.44E+09	0.00E+00	5.54E+08	0.00E+00	6.00E+09	0.00E+00	6.91E+10
Ag-110m	8.40E+06	5.67E+06	4.53E+06	0.00E+00	1.06E+07	0.00E+00	6.75E+08
Cd-109	0.00E+00	1.91E+06	8.84E+04	0.00E+00	1.70E+06	0.00E+00	6.18E+06
Sn-113	2.18E+09	4.48E+07	1.24E+08	3.31E+09	0.00E+00	0.00E+00	1.54E+09
Sb-124	2.93E+07	3.80E+05	1.03E+07	6.46E+04	0.00E+00	1.62E+07	1.83E+08
Sb-125	2.85E+07	2.20E+05	5.97E+06	2.64E+04	0.00E+00	1.59E+07	6.81E+07
Te-127m	1.78E+09	4.78E+08	2.11E+08	4.25E+08	5.07E+09	0.00E+00	1.44E+09
Te-129m	1.79E+09	5.00E+08	2.78E+08	5.78E+08	5.26E+09	0.00E+00	2.19E+09
I-130	3.06E-06	6.18E-06	3.18E-06	6.80E-04	9.23E-06	0.00E+00	2.89E-06
I-131	1.66E+07	1.67E+07	9.47E+06	5.51E+09	2.74E+07	0.00E+00	1.48E+06
I-132	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

TABLE 3.3 (Cont'd)

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

Meat Pathway

(m^3 mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-133	5.70E-01	7.05E-01	2.67E-01	1.31E+02	1.17E+00	0.00E+00	2.84E-01
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	9.23E+08	1.51E+09	3.20E+08	0.00E+00	4.69E+08	1.68E+08	8.17E+06
Cs-136	1.62E+07	4.46E+07	2.89E+07	0.00E+00	2.38E+07	3.54E+06	1.57E+06
Cs-137	1.33E+09	1.28E+09	1.89E+08	0.00E+00	4.16E+08	1.50E+08	8.00E+06
Ba-140	4.39E+07	3.85E+04	2.56E+06	0.00E+00	1.25E+04	2.29E+04	2.22E+07
La-140	3.33E+02	1.17E+02	3.93E+01	0.00E+00	0.00E+00	0.00E+00	3.25E+06
Ce-141	2.22E+04	1.11E+04	1.65E+03	0.00E+00	4.86E+03	0.00E+00	1.38E+07
Ce-144	2.32E+06	7.27E+05	1.24E+05	0.00E+00	4.02E+05	0.00E+00	1.89E+08
Pr-143	3.34E+04	1.00E+04	1.66E+03	0.00E+00	5.43E+03	0.00E+00	3.61E+07
Pr-144	5.63E+02	1.74E+02	2.83E+01	0.00E+00	9.21E+01	0.00E+00	3.75E+05
Nd-147	1.17E+04	9.48E+03	7.34E+02	0.00E+00	5.20E+03	0.00E+00	1.50E+07
Eu-154	1.12E+07	1.01E+06	9.20E+05	0.00E+00	4.43E+06	0.00E+00	2.34E+08
Hf-181	4.77E+06	1.74E+07	2.15E+06	0.00E+00	3.53E+06	0.00E+00	6.41E+09

TABLE 3.3 (Cont'd)

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

Grass-Cow-Milk Pathway

(m^2 mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	0.00E+00	1.57E+03	1.57E+03	1.57E+03	1.57E+03	1.57E+03	1.57E+03
Be-7	7.50E+03	1.28E+04	8.20E+03	0.00E+00	1.25E+04	0.00E+00	7.12E+05
Cr-51	0.00E+00	0.00E+00	1.02E+05	5.66E+04	1.55E+04	1.03E+05	5.40E+06
Mn-54	0.00E+00	2.10E+07	5.59E+06	0.00E+00	5.89E+06	0.00E+00	1.76E+07
Fe-55	1.12E+08	5.94E+07	1.84E+07	0.00E+00	0.00E+00	3.36E+07	1.10E+07
Fe-59	1.20E+08	1.95E+08	9.70E+07	0.00E+00	0.00E+00	5.64E+07	2.03E+08
Co-57	0.00E+00	3.84E+06	7.78E+06	0.00E+00	0.00E+00	0.00E+00	3.15E+07
Co-58	0.00E+00	1.21E+07	3.72E+07	0.00E+00	0.00E+00	0.00E+00	7.08E+07
Co-60	0.00E+00	4.32E+07	1.27E+08	0.00E+00	0.00E+00	0.00E+00	2.39E+08
Zn-65	4.14E+09	1.10E+10	6.86E+09	0.00E+00	6.95E+09	0.00E+00	1.94E+09
Rb-86	0.00E+00	8.78E+09	5.40E+09	0.00E+00	0.00E+00	0.00E+00	5.65E+08
Sr-89	6.63E+09	0.00E+00	1.89E+08	0.00E+00	0.00E+00	0.00E+00	2.57E+08
Sr-90	1.12E+11	0.00E+00	2.84E+10	0.00E+00	0.00E+00	0.00E+00	1.51E+09
Y-90	3.38E+03	0.00E+00	9.05E+01	0.00E+00	0.00E+00	0.00E+00	9.62E+06
Y-91	3.91E+04	0.00E+00	1.04E+03	0.00E+00	0.00E+00	0.00E+00	5.20E+06
Zr-95	3.84E+03	8.43E+02	7.51E+02	0.00E+00	1.21E+03	0.00E+00	8.80E+05
Nb-95	3.72E+05	1.45E+05	1.03E+05	0.00E+00	1.36E+05	0.00E+00	2.68E+08
Ru-103	4.29E+03	0.00E+00	1.65E+03	0.00E+00	1.08E+04	0.00E+00	1.11E+05
Ru-106	9.25E+04	0.00E+00	1.15E+04	0.00E+00	1.25E+05	0.00E+00	1.44E+06
Ag-110m	2.09E+08	1.41E+08	1.13E+08	0.00E+00	2.63E+08	0.00E+00	1.68E+10
Cd-109	0.00E+00	3.86E+06	1.79E+05	0.00E+00	3.45E+06	0.00E+00	1.25E+07
Sn-113	6.11E+08	1.26E+07	3.48E+07	9.29E+08	0.00E+00	0.00E+00	4.32E+08
Sb-124	1.09E+08	1.41E+06	3.81E+07	2.40E+05	0.00E+00	6.03E+07	6.80E+08
Sb-125	8.71E+07	6.72E+05	1.83E+07	8.07E+04	0.00E+00	4.86E+07	2.08E+08
Te-127m	2.08E+08	5.61E+07	2.47E+07	4.98E+07	5.94E+08	0.00E+00	1.69E+08
Te-129m	2.72E+08	7.59E+07	4.22E+07	8.76E+07	7.98E+08	0.00E+00	3.31E+08

TABLE 3.3 (Cont'd)

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

Grass-Cow-Milk Pathway

(m^2 mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-130	1.73E+06	3.50E+06	1.80E+06	3.85E+08	5.23E+06	0.00E+00	1.64E+06
I-131	1.30E+09	1.31E+09	7.46E+08	4.34E+11	2.15E+09	0.00E+00	1.17E+08
I-132	6.92E-01	1.27E+00	5.85E-01	5.90E+01	1.95E+00	0.00E+00	1.50E+00
I-133	1.72E+07	2.13E+07	8.05E+06	3.95E+09	3.54E+07	0.00E+00	8.57E+06
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	5.41E+04	9.74E+04	4.61E+04	8.63E+06	1.49E+05	0.00E+00	7.42E+04
Cs-134	2.27E+10	3.72E+10	7.84E+09	0.00E+00	1.15E+10	4.14E+09	2.00E+08
Cs-136	1.01E+09	2.78E+09	1.80E+09	0.00E+00	1.48E+09	2.21E+08	9.78E+07
Cs-137	3.23E+10	3.09E+10	4.56E+09	0.00E+00	1.01E+10	3.62E+09	1.93E+08
Ba-140	1.17E+08	1.03E+05	6.84E+06	0.00E+00	3.34E+04	6.12E+04	5.94E+07
Ce-141	2.19E+04	1.09E+04	1.62E+03	0.00E+00	4.79E+03	0.00E+00	1.36E+07
La-140	1.78E+02	6.23E+01	2.10E+01	0.00E+00	0.00E+00	0.00E+00	1.74E+06
Ce-144	1.62E+06	5.09E+05	8.67E+04	0.00E+00	2.82E+05	0.00E+00	1.33E+08
Pr-143	7.19E+02	2.16E+02	3.57E+01	0.00E+00	1.17E+02	0.00E+00	7.76E+05
Pr-144	5.04E+00	1.56E+00	2.53E-01	0.00E+00	8.24E-01	0.00E+00	3.35E+03
Nd-147	4.45E+02	3.61E+02	2.79E+01	0.00E+00	1.98E+02	0.00E+00	5.71E+05
Eu-154	9.43E+04	8.48E+03	7.75E+03	0.00E+00	3.73E+04	0.00E+00	1.97E+06
Hf-181	6.44E+02	2.35E+03	2.91E+02	0.00E+00	4.76E+02	0.00E+00	8.66E+05

TABLE 3.3 (Cont'd)

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

Grass-Goat-M.''k Pathway

(m² mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	0.00E+00	3.20E+03	3.20E+03	3.20E+03	3.20E+03	3.20E+03	3.20E+03
Be-7	9.00E+02	1.53E+03	9.84E+02	0.00E+00	1.50E+03	0.00E+00	8.55E+04
Cr-51	0.00E+00	0.00E+00	1.22E+04	6.79E+03	1.85E+03	1.24E+04	6.48E+05
Mn-54	0.00E+00	2.52E+06	6.71E+05	0.00E+00	7.06E+05	0.00E+00	2.11E+06
Fe-55	1.45E+06	7.72E+05	2.39E+05	0.00E+00	0.00E+00	4.36E+05	1.43E+05
Fe-59	1.56E+06	2.53E+06	1.26E+06	0.00E+00	0.00E+00	7.34E+05	2.64E+06
Co-57	0.00E+00	4.61E+05	9.33E+05	0.00E+00	0.00E+00	0.00E+00	3.78E+06
Co-58	0.00E+00	1.46E+06	4.46E+06	0.00E+00	0.00E+00	0.00E+00	8.50E+06
Co-60	0.00E+00	5.19E+06	1.53E+07	0.00E+00	0.00E+00	0.00E+00	2.87E+07
Zn-65	4.97E+08	1.32E+09	8.23E+08	0.00E+00	8.34E+08	0.00E+00	2.32E+08
Rb-86	0.00E+00	1.05E+09	6.48E+08	0.00E+00	0.00E+00	0.00E+00	6.78E+07
Sr-89	1.39E+10	0.00E+00	3.97E+08	0.00E+00	0.00E+00	0.00E+00	5.39E+08
Sr-90	2.35E+11	0.00E+00	5.95E+10	0.00E+00	0.00E+00	0.00E+00	3.16E+09
Y-90	4.06E+02	0.00E+00	1.09E+01	0.00E+00	0.00E+00	0.00E+00	1.15E+06
Y-91	4.69E+03	0.00E+00	1.25E+02	0.00E+00	0.00E+00	0.00E+00	6.25E+05
Zr-95	4.60E+02	1.01E+02	9.01E+01	0.00E+00	1.45E+02	0.00E+00	1.06E+05
Nb-95	4.46E+04	1.74E+04	1.24E+04	0.00E+00	1.63E+04	0.00E+00	3.21E+07
Ru-103	5.14E+02	0.00E+00	1.98E+02	0.00E+00	1.29E+03	0.00E+00	1.33E+04
Ru-106	1.11E+04	0.00E+00	1.38E+03	0.00E+00	1.50E+04	0.00E+00	1.73E+05
Ag-110m	2.51E+07	1.69E+07	1.35E+07	0.00E+00	3.15E+07	0.00E+00	2.01E+09
Cd-109	0.00E+00	4.64E+05	2.15E+04	0.00E+00	4.14E+05	0.00E+00	1.50E+06
Sn-113	7.33E+07	1.51E+06	4.18E+06	1.11E+08	0.00E+00	0.00E+00	5.18E+07
Sb-124	1.30E+07	1.69E+05	4.57E+06	2.88E+04	0.00E+00	7.24E+06	8.16E+07
Sb-125	1.05E+07	8.06E+04	2.19E+06	9.68E+03	0.00E+00	5.83E+06	2.50E+07
Te-127m	2.50E+07	6.73E+06	2.97E+06	5.98E+06	7.13E+07	0.00E+00	2.02E+07
Te-129m	3.26E+07	9.10E+06	5.06E+06	1.05E+07	9.57E+07	0.00E+00	3.98E+07
I-130	2.08E+06	4.20E+06	2.16E+06	4.62E+08	6.27E+06	0.00E+00	1.96E+06
I-131	1.57E+09	1.57E+09	8.95E+08	5.21E+11	2.58E+09	0.00E+00	1.40E+08
I-132	8.30E-01	1.53E+00	7.02E-01	7.08E+01	2.34E+00	0.00E+00	1.80E+00

TABLE 3.3 (Cont'd)

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

Grass-Goat-Milk Pathway

(m^2 mrem/yr) per (μCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-133	2.06E+07	2.55E+07	9.66E+06	4.74E+09	4.25E+07	0.00E+00	1.03E+07
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	6.49E+04	1.17E+05	5.53E+04	1.04E+07	1.79E+05	0.00E+00	8.90E+04
Cs-134	6.80E+10	1.12E+11	2.35E+10	0.00E+00	3.46E+10	1.24E+10	6.01E+08
Cs-136	3.04E+09	8.35E+09	5.40E+09	0.00E+00	4.45E+09	6.63E+08	2.93E+08
Cs-137	9.68E+10	9.27E+10	1.37E+10	0.00E+00	3.02E+10	1.09E+10	5.80E+08
Ba-140	1.41E+07	1.23E+04	8.21E+05	0.00E+00	4.01E+03	7.35E+03	7.13E+06
La-140	2.14E+01	7.47E+00	2.52E+00	0.00E+00	0.00E+00	0.00E+00	2.08E+05
Ce-141	2.63E+03	1.31E+03	1.95E+02	0.00E+00	5.75E+02	0.00E+00	1.63E+06
Ce-144	1.95E+05	6.11E+04	1.04E+04	0.00E+00	3.38E+04	0.00E+00	1.59E+07
Pr-143	8.63E+01	2.59E+01	4.28E+00	0.00E+00	1.40E+01	0.00E+00	9.31E+04
Pr-144	6.05E-01	1.87E-01	3.04E-02	0.00E+00	9.89E-02	0.00E+00	4.03E+02
Nd-147	5.34E+01	4.33E+01	3.35E+00	0.00E+00	2.37E+01	0.00E+00	6.85E+04
Eu-154	1.13E+04	1.02E+03	9.29E+02	0.00E+00	4.47E+03	0.00E+00	2.37E+05
Hf-181	7.73E+01	2.81E+02	3.49E+01	0.00E+00	5.72E+01	0.00E+00	1.04E+05

TABLE 3.3 (Cont'd)

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

Vegetation Pathway

(m^3 mrem/yr) per (μCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	ND	4.01E+03	4.01E+03	4.01E+03	4.01E+03	4.01E+03	4.01E+03
Be-7	3.38E+05	5.76E+05	3.70E+05	0.00E+00	5.65E+05	0.00E+00	3.21E+07
Cr-51	0.00E+00	0.00E+00	1.17E+05	6.50E+04	1.78E+04	1.19E+05	6.21E+06
Mn-54	0.00E+00	6.65E+08	1.77E+08	0.00E+00	1.86E+08	0.00E+00	5.58E+08
Fe-55	8.01E+08	4.25E+08	1.32E+08	0.00E+00	0.00E+00	2.40E+08	7.87E+07
Fe-59	3.98E+08	6.43E+08	3.20E+08	0.00E+00	0.00E+00	1.87E+08	6.70E+08
Co-57	0.00E+00	2.99E+07	6.04E+07	0.00E+00	0.00E+00	0.00E+00	2.45E+08
Co-58	0.00E+00	6.44E+07	1.97E+08	0.00E+00	0.00E+00	0.00E+00	3.76E+08
Co-60	0.00E+00	3.78E+08	1.12E+09	0.00E+00	0.00E+00	0.00E+00	2.10E+09
Zn-65	8.13E+08	2.17E+09	1.35E+09	0.00E+00	1.36E+09	0.00E+00	3.80E+08
Rb-86	0.00E+00	4.52E+08	2.78E+08	0.00E+00	0.00E+00	0.00E+00	2.91E+07
Sr-89	3.60E+10	0.00E+00	1.03E+09	0.00E+00	0.00E+00	0.00E+00	1.39E+09
Sr-90	1.24E+12	0.00E+00	3.15E+11	0.00E+00	0.00E+00	0.00E+00	1.67E+10
Y-90	3.01E+06	0.00E+00	8.04E+04	0.00E+00	0.00E+00	0.00E+00	8.56E+09
Y-91	1.86E+07	0.00E+00	4.99E+05	0.00E+00	0.00E+00	0.00E+00	2.48E+09
Zr-95	3.86E+06	8.48E+05	7.55E+05	0.00E+00	1.21E+06	0.00E+00	8.85E+08
Nb-95	7.48E+05	2.91E+05	2.08E+05	0.00E+00	2.74E+05	0.00E+00	5.39E+08
Ru-103	1.53E+07	0.00E+00	5.90E+06	0.00E+00	3.86E+07	0.00E+00	3.97E+08
Ru-106	7.45E+08	0.00E+00	9.30E+07	0.00E+00	1.01E+09	0.00E+00	1.16E+10
Ag-110m	3.21E+07	2.17E+07	1.73E+07	0.00E+00	4.04E+07	0.00E+00	2.58E+09
Cd-109	0.00E+00	2.45E+08	1.14E+07	0.00E+00	2.18E+08	0.00E+00	7.94E+08
Sn-113	1.58E+09	3.25E+07	9.00E+07	2.40E+09	0.00E+00	0.00E+00	1.12E+09
Sb-124	3.52E+08	4.57E+06	1.23E+08	7.77E+05	0.00E+00	1.95E+08	2.20E+09
Sb-125	4.99E+08	3.85E+06	1.05E+08	4.63E+05	0.00E+00	2.78E+08	1.19E+09
Te-127m	1.32E+09	3.56E+08	1.57E+08	3.16E+08	3.77E+09	0.00E+00	1.07E+09
Te-129m	8.41E+08	2.35E+08	1.31E+08	2.71E+08	2.47E+09	0.00E+00	1.03E+09
I-130	6.16E+05	1.24E+06	6.41E+05	1.37E+08	1.86E+06	0.00E+00	5.82E+05
I-131	1.43E+08	1.44E+08	8.17E+07	4.76E+10	2.36E+08	0.00E+00	1.28E+07

TABLE 3.3 (Cont'd)

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

Vegetation Pathway

(m^2 mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-132	9.23E+01	1.70E+02	7.80E+01	7.87E+03	2.60E+02	0.00E+00	2.00E+02
I-133	3.53E+06	4.37E+06	1.65E+06	8.12E+08	7.28E+06	0.00E+00	1.76E+06
I-134	1.56E-04	2.89E-04	1.33E-04	6.65E-03	4.42E-04	0.00E+00	1.92E-04
I-135	6.26E+04	1.13E+05	5.33E+04	9.98E+06	1.73E+05	0.00E+00	8.59E+04
Cs-134	1.60E+10	2.63E+10	5.55E+09	0.00E+00	8.15E+09	2.93E+09	1.42E+08
Cs-136	8.24E+07	2.27E+08	1.47E+08	0.00E+00	1.21E+08	1.80E+07	7.96E+06
Cs-137	2.39E+10	2.29E+10	3.38E+09	0.00E+00	7.46E+09	2.68E+09	1.43E+08
Ba-140	2.77E+08	2.43E+05	1.62E+07	0.00E+00	7.90E+04	1.45E+05	1.40E+08
La-140	3.36E+04	1.18E+04	3.96E+03	0.00E+00	0.00E+00	0.00E+00	3.28E+08
Ce-141	6.56E+05	3.27E+05	4.86E+04	0.00E+00	1.43E+05	0.00E+00	4.08E+08
Ce-144	1.27E+08	3.98E+07	6.78E+06	0.00E+00	2.21E+07	0.00E+00	1.04E+10
Pr-143	1.46E+05	4.37E+04	7.23E+03	0.00E+00	2.37E+04	0.00E+00	1.57E+08
Pr-144	7.88E+03	2.44E+03	3.97E+02	0.00E+00	1.29E+03	0.00E+00	5.25E+06
Nd-147	7.15E+04	5.79E+04	4.48E+03	0.00E+00	3.18E+04	0.00E+00	9.17E+07
Eu-154	1.66E+08	1.50E+07	1.37E+07	0.00E+00	6.57E+07	0.00E+00	3.48E+09
Hf-181	4.90E+05	1.79E+06	2.21E+05	0.00E+00	3.63E+05	0.00E+00	6.59E+08

TABLE 3.4

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

Inhalation Pathway

(mrem/yr) per (μCi/m^3)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	ND	1.26E+03	1.26E+03	1.26E+03	1.26E+03	1.26E+03	1.26E+03
Be-7	4.27E+02	9.68E+02	4.70E+02	ND	ND	4.21E+04	5.35E+03
Cr-51	ND	ND	1.00E+02	5.95E+01	2.28E+01	1.44E+04	3.32E+03
Mn-54	ND	3.96E+04	6.30E+03	ND	9.84E+03	1.40E+06	7.74E+04
Fe-55	2.46E+04	1.70E+04	3.94E+03	ND	ND	7.21E+04	6.03E+03
Fe-59	1.18E+04	2.78E+04	1.06E+04	ND	ND	1.02E+06	1.88E+05
Co-57	ND	6.92E+02	6.71E+02	ND	ND	3.70E+05	3.14E+04
Co-58	ND	1.58E+03	2.07E+03	ND	ND	9.28E+05	1.06E+05
Co-60	ND	1.15E+04	1.48E+04	ND	ND	5.97E+06	2.85E+05
Zn-65	3.24E+04	1.03E+05	4.66E+04	ND	6.90E+04	8.64E+05	5.34E+04
Rb-86	ND	1.35E+05	5.90E+04	ND	ND	ND	1.66E+04
Sr-89	3.04E+05	ND	8.72E+03	ND	ND	1.40E+06	3.50E+05
Sr-90	9.92E+07	ND	6.10E+06	ND	ND	9.60E+06	7.22E+05
Y-90	2.09E+03	ND	5.61E+01	ND	ND	1.70E+05	5.06E+05
Y-91	4.62E+05	ND	1.24E+04	ND	ND	1.70E+06	3.85E+05
Zr-95	1.07E+05	3.44E+04	2.33E+04	ND	5.42E+04	1.77E+06	1.50E+05
Nb-95	1.41E+04	7.82E+03	4.21E+03	ND	7.74E+03	5.05E+05	1.04E+05
Ru-103	1.53E+03	ND	6.58E+02	ND	5.83E+03	5.05E+05	1.10E+05
Ru-106	6.91E+04	ND	8.72E+03	ND	1.34E+05	9.36E+06	9.12E+05
Ag-110m	1.08E+04	1.00E+04	5.94E+03	ND	1.97E+04	4.63E+06	3.02E+05
Cd-109	ND	3.67E+05	1.31E+04	ND	3.57E+05	6.83E+05	5.82E+04
Sn-113	5.72E+04	2.18E+03	4.39E+03	1.24E+03	ND	9.44E+05	1.18E+05
Sb-124	3.12E+04	5.89E+02	1.24E+04	7.55E+01	ND	2.48E+06	4.06E+05
Sb-125	5.34E+04	5.95E+02	1.26E+04	5.40E+01	ND	1.74E+06	1.01E+05
Te-127m	1.26E+04	5.77E+03	1.57E+03	3.29E+03	4.58E+04	9.60E+05	1.50E+05
Te-129m	9.76E+03	4.67E+03	1.58E+03	3.44E+03	3.66E+04	1.16E+06	3.83E+05
I-130	4.58E+03	1.34E+04	5.28E+03	1.14E+06	2.09E+04	ND	7.69E+03
I-131	2.52E+04	3.58E+04	2.05E+04	1.19E+07	6.13E+04	ND	6.28E+03

TABLE 3.4 (Cont'd)

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

Inhalation Pathway

(mrem/yr) per (μCi/m³)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-132	1.16E+03	3.26E+03	1.16E+03	1.14E+05	5.18E+03	ND	4.06E+02
I-133	8.64E+03	1.48E+04	4.52E+03	2.15E+06	2.58E+04	ND	8.88E+03
I-134	6.44E+02	1.73E+03	6.15E+02	2.98E+04	2.75E+03	ND	1.01E+00
I-135	2.68E+03	6.98E+03	2.57E+03	4.48E+05	1.11E+04	ND	5.25E+03
Cs-134	3.73E+05	8.48E+05	7.28E+05	ND	2.87E+05	9.76E+04	1.04E+04
Cs-136	3.90E+04	1.46E+05	1.10E+05	ND	8.56E+04	1.20E+04	1.17E+04
Cs-137	4.78E+05	6.21E+05	4.28E+05	ND	2.22E+05	7.52E+04	8.40E+03
Ba-140	3.90E+04	4.90E+01	2.57E+03	ND	1.67E+01	1.27E+06	2.18E+05
La-140	3.44E+02	1.74E+02	4.58E+01	ND	ND	1.36E+05	4.58E+05
Ce-141	1.99E+04	1.35E+04	1.53E+03	ND	6.26E+03	3.62E+05	1.20E+05
Ce-144	3.43E+06	1.43E+06	1.84E+05	ND	8.48E+05	7.78E+06	8.16E+05
Pr-143	9.36E+03	3.75E+03	4.64E+02	ND	2.16E+03	2.81E+05	2.00E+05
Pr-144	3.01E-02	1.25E-02	1.53E-03	ND	7.05E-03	1.02E+03	2.15E-08
Nd-147	5.27E+03	6.10E+03	3.65E+02	ND	3.56E+03	2.21E+05	1.73E+05
Eu-154	5.92E+06	7.28E+05	5.18E+05	ND	3.49E+06	4.67E+06	2.72E+05
Hf-181	1.41E+04	6.[illegible]2E+04	6.32E+03	ND	1.48E+04	6.85E+05	1.39E+05

TABLE 3.4 (Cont'd)

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

Meat Pathway

(m^2 mrem/yr) per (μCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	ND	3.25E+02	3.25E+02	3.25E+02	3.25E+02	3.25E+02	3.25E+02
Be-7	4.57E+03	1.04E+04	5.07E+03	ND	1.10E+04	ND	1.81E+06
Cr-51	ND	ND	7.04E+03	4.21E+03	1.55E+03	9.34E+03	1.77E+06
Mn-54	ND	9.17E+06	1.75E+06	ND	2.73E+06	ND	2.81E+07
Fe-55	2.93E+08	2.02E+08	4.72E+07	ND	ND	1.13E+08	1.16E+08
Fe-59	2.65E+08	6.24E+08	2.39E+08	ND	ND	1.74E+08	2.08E+09
Co-57	ND	5.63E+06	9.36E+06	ND	ND	ND	1.43E+08
Co-58	ND	1.82E+07	4.08E+07	ND	ND	ND	3.69E+08
Co-60	ND	7.51E+07	1.66E+08	ND	ND	ND	1.41E+09
Zn-65	3.56E+08	1.13E+09	5.11E+08	ND	7.57E+08	ND	7.13E+08
Rb-86	ND	4.87E+08	2.27E+08	ND	ND	ND	9.60E+07
Sr-89	3.01E+08	ND	8.65E+06	ND	ND	ND	4.83E+07
Sr-90	1.24E+10	ND	3.05E+09	ND	ND	ND	3.59E+08
Y-90	1.21E+05	ND	3.24E+03	ND	ND	ND	1.28E+09
Y-91	1.13E+06	ND	3.02E+04	ND	ND	ND	6.23E+08
Zr-95	1.87E+06	6.00E+05	4.06E+05	ND	9.42E+05	ND	1.90E+09
Nb-95	3.15E+06	1.75E+06	9.43E+05	ND	1.73E+06	ND	1.06E+10
Ru-103	1.05E+08	ND	4.53E+07	ND	4.01E+08	ND	1.23E+10
Ru-106	2.80E+09	ND	3.54E+08	ND	5.40E+09	ND	1.81E+11
Ag-110m	6.68E+06	6.18E+06	3.67E+06	ND	1.21E+07	ND	2.52E+09
Cd-109	ND	1.59E+06	5.55E+04	ND	1.52E+06	ND	1.60E+07
Sn-113	1.37E+09	3.88E+07	7.86E+07	2.22E+07	ND	ND	4.09E+09
Sb-124	1.98E+07	3.74E+05	7.84E+06	4.79E+04	ND	1.54E+07	5.61E+08
Sb-125	1.91E+07	2.13E+05	4.54E+06	1.94E+04	ND	1.47E+07	2.10E+08
Te-127m	1.11E+09	3.98E+08	1.36E+08	2.85E+08	4.53E+09	ND	3.74E+09
Te-129m	1.13E+09	4.23E+08	1.79E+08	3.89E+08	4.73E+09	ND	5.71E+09
I-130	2.12E-06	6.27E-06	2.47E-06	5.31E-04	9.78E-06	ND	5.40E-06
I-131	1.08E+07	1.54E+07	8.82E+06	5.04E+09	2.64E+07	ND	4.06E+06

TABLE 3.4 (Cont'd)

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

Meat Pathway

(m^2 mrem/yr) per (μCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-132	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	ND	0.00E+00
I-133	3.67E-01	6.39E-01	1.95E-01	9.38E+01	1.11E+00	ND	5.74E-01
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	ND	0.00E+00
I-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	ND	0.00E+00
Cs-134	6.57E+08	1.56E+09	1.28E+09	ND	5.06E+08	1.68E+08	2.74E+07
Cs-136	1.20E+07	4.76E+07	3.42E+07	ND	2.65E+07	3.63E+06	5.40E+06
Cs-137	8.71E+08	1.19E+09	7.81E+08	ND	4.04E+08	1.34E+08	2.31E+07
Ba-140	2.87E+07	3.61E+04	1.88E+06	ND	1.23E+04	2.07E+04	5.91E+07
La-140	2.21E+02	1.11E+02	2.94E+01	ND	ND	ND	8.18E+06
Ce-141	1.40E+04	9.49E+03	1.08E+03	ND	4.41E+03	ND	3.63E+07
Ce-144	1.46E+06	6.09E+05	7.82E+04	ND	3.61E+05	ND	4.92E+08
Pr-143	2.10E+04	8.40E+03	1.04E+03	ND	4.85E+03	ND	9.18E+07
Pr-144	3.52E+02	1.46E+02	1.79E+01	ND	8.24E+01	ND	5.06E-05
Nd-147	7.07E+03	8.17E+03	4.89E+02	ND	4.77E+03	ND	3.92E+07
Eu-154	8.02E+06	9.86E+05	7.01E+05	ND	4.72E+06	ND	7.14E+08
Hf-181	3.01E+06	1.46E+07	1.35E+06	ND	3.14E+06	ND	1.66E+10

TABLE 3.4 (Cont'd)

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

Grass-Cow-Milk Pathway

(m^2 mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	ND	7.63E+02	7.63E+02	7.63E+02	7.63E+02	7.63E+02	7.63E+02
Be-7	1.63E+03	3.72E+03	1.81E+03	ND	3.93E+03	ND	6.45E+05
Cr-51	ND	ND	2.86E+04	1.71E+04	6.30E+03	3.79E+04	7.19E+06
Mn-54	ND	8.42E+06	1.61E+06	ND	2.50E+06	ND	2.58E+07
Fe-55	2.51E+07	1.74E+07	4.05E+06	ND	ND	9.68E+06	9.96E+06
Fe-59	2.97E+07	6.98E+07	2.68E+07	ND	ND	1.95E+07	2.33E+08
Co-57	ND	1.28E+06	2.13E+06	ND	ND	ND	3.25E+07
Co-58	ND	4.72E+06	1.06E+07	ND	ND	ND	9.56E+07
Co-60	ND	1.64E+07	3.62E+07	ND	ND	ND	3.08E+08
Zn-65	1.37E+09	4.37E+09	1.97E+09	ND	2.92E+09	ND	2.75E+09
Rb-86	ND	2.60E+09	1.21E+09	ND	ND	ND	5.12E+08
Sr-89	1.45E+09	ND	4.17E+07	ND	ND	ND	2.33E+08
Sr-90	4.68E+10	ND	1.15E+10	ND	ND	ND	1.35E+09
Y-90	7.43E+02	ND	1.99E+01	ND	ND	ND	7.87E+06
Y-91	8.59E+03	ND	2.30E+02	ND	ND	ND	4.73E+06
Zr-95	9.44E+02	3.03E+02	2.05E+02	ND	4.75E+02	ND	9.59E+05
Nb-95	9.65E+04	5.37E+04	2.89E+04	ND	5.31E+04	ND	3.26E+08
Ru-103	1.02E+03	ND	4.39E+02	ND	3.89E+03	ND	1.19E+05
Ru-106	2.04E+04	ND	2.58E+03	ND	3.94E+04	ND	1.32E+06
Ag-110m	5.82E+07	5.39E+07	3.20E+07	ND	1.06E+08	ND	2.20E+10
Cd-109	ND	1.13E+06	3.95E+04	ND	1.08E+06	ND	1.14E+07
Sn-113	1.34E+08	3.81E+06	7.73E+06	2.18E+06	ND	ND	4.02E+08
Sb-124	2.57E+07	4.86E+05	1.02E+07	6.24E+04	ND	2.00E+07	7.31E+08
Sb-125	2.04E+07	2.28E+05	4.87E+06	2.08E+04	ND	1.58E+07	2.25E+08
Te-127m	4.58E+07	1.64E+07	5.58E+06	1.17E+07	1.86E+08	ND	1.54E+08
Te-129m	6.02E+07	2.25E+07	9.53E+06	2.07E+07	2.51E+08	ND	3.03E+08
I-130	4.21E+05	1.24E+06	4.91E+05	1.05E+08	1.94E+06	ND	1.07E+06
I-131	2.97E+08	4.25E+08	2.43E+08	1.39E+11	7.28E+08	ND	1.12E+08

TABLE 3.4 (Cont'd)

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

Grass-Cow-Milk Pathway

(m^2 mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-132	1.65E-01	4.42E-01	1.55E-01	1.55E+01	7.04E-01	ND	8.30E-02
I-133	3.88E+06	6.75E+06	2.06E+06	9.92E+08	1.18E+07	ND	6.07E+06
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	ND	0.00E+00
I-135	1.29E+04	3.37E+04	1.25E+04	2.23E+06	5.41E+04	ND	3.81E+04
Cs-134	5.65E+09	1.35E+10	1.10E+10	ND	4.35E+09	1.45E+09	2.35E+08
Cs-136	2.63E+08	1.04E+09	7.48E+08	ND	5.79E+08	7.93E+07	1.18E+08
Cs-137	7.38E+09	1.01E+10	6.61E+09	ND	3.43E+09	1.14E+09	1.95E+08
Ba-140	2.69E+07	3.38E+04	1.76E+06	ND	1.15E+04	1.93E+04	5.54E+07
La-140	4.14E+01	2.09E+01	5.51E+00	ND	ND	ND	1.53E+06
Ce-141	4.85E+03	3.28E+03	3.72E+02	ND	1.52E+03	ND	1.25E+07
Ce-144	3.58E+05	1.50E+05	1.92E+04	ND	8.87E+04	ND	1.21E+08
Pr-143	1.58E+02	6.34E+01	7.83E+00	ND	3.66E+01	ND	6.92E+05
Pr-144	1.10E+00	4.58E-01	5.61E-02	ND	2.58E-01	ND	1.59E-07
Nd-147	9.42E+01	1.09E+02	6.51E+00	ND	6.36E+01	ND	5.23E+05
Eu-154	2.37E+04	2.91E+03	2.07E+03	ND	1.39E+04	ND	2.11E+06
Hf-181	1.42E+02	6.92E+02	6.41E+01	ND	1.49E+02	ND	7.87E+05

TABLE 3.4 (Cont'd)

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

Grass-Goat-Milk Pathway

(m^2 mrem/yr) per (μCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	ND	1.56E+03	1.56E+03	1.56E+03	1.56E+03	1.56E+03	1.56E+03
Be-7	1.96E+02	4.47E+02	2.17E+02	ND	4.72E+02	ND	7.74E+04
Cr-51	ND	ND	3.43E+03	2.05E+03	7.56E+02	4.56E+03	8.63E+05
Mn-54	ND	1.01E+06	1.93E+05	ND	3.01E+05	ND	3.10E+06
Fe-55	3.27E+05	2.26E+05	5.26E+04	ND	ND	1.26E+05	1.30E+05
Fe-59	3.87E+05	9.08E+05	3.48E+05	ND	ND	2.54E+05	3.03E+06
Co-57	ND	1.54E+05	2.56E+05	ND	ND	ND	3.90E+06
Co-58	ND	5.66E+05	1.27E+06	ND	ND	ND	1.15E+07
Co-60	ND	1.97E+06	4.35E+06	ND	ND	ND	3.70E+07
Zn-65	1.65E+08	5.24E+08	2.37E+08	ND	3.51E+08	ND	3.30E+08
Rb-86	ND	3.12E+08	1.45E+08	ND	ND	ND	6.15E+07
Sr-89	3.05E+09	ND	8.75E+07	ND	ND	ND	4.89E+08
Sr-90	9.84E+10	ND	2.41E+10	ND	ND	ND	2.84E+09
Y-90	8.92E+01	ND	2.39E+00	ND	ND	ND	9.46E+05
Y-91	1.03E+03	ND	2.76E+01	ND	ND	ND	5.68E+05
Zr-95	1.13E+02	3.63E+01	2.46E+01	ND	5.70E+01	ND	1.15E+05
Nb-95	1.16E+04	6.45E+03	3.47E+03	ND	6.37E+03	ND	3.91E+07
Ru-103	1.22E+02	ND	5.27E+01	ND	4.67E+02	ND	1.43E+04
Ru-106	2.45E+03	ND	3.10E+02	ND	4.73E+03	ND	1.59E+05
Ag-110m	6.99E+06	6.47E+06	3.84E+06	ND	1.27E+07	ND	2.64E+09
Cd-109	ND	1.36E+05	4.74E+03	ND	1.30E+05	ND	1.37E+06
Sn-113	1.61E+07	4.58E+05	9.28E+05	2.62E+05	ND	ND	4.83E+07
Sb-124	3.09E+06	5.84E+04	1.23E+06	7.50E+03	ND	2.41E+06	8.78E+07
Sb-125	2.46E+06	2.74E+04	5.84E+05	2.50E+03	ND	1.89E+06	2.70E+07
Te-127m	5.50E+06	1.97E+06	6.70E+05	1.41E+06	2.23E+07	ND	1.84E+07
Te-129m	7.23E+06	2.70E+06	1.14E+06	2.48E+06	3.02E+07	ND	3.64E+07
I-130	5.05E+05	1.49E+06	5.88E+05	1.26E+08	2.32E+06	ND	1.28E+06
I-131	3.56E+08	5.09E+08	2.92E+08	1.67E+11	8.72E+08	ND	1.34E+08
I-132	1.98E-01	5.29E-01	1.85E-01	1.85E+01	8.43E-01	ND	9.95E-02

TABLE 3.4 (Cont'd)

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

Grass-Goat-Milk Pathway

(m^2 mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-133	4.65E+06	8.09E+06	2.47E+06	1.19E+09	1.41E+07	ND	7.27E+06
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	ND	0.00E+00
I-135	1.54E+04	4.04E+04	1.49E+04	2.67E+06	6.48E+04	ND	4.57E+04
Cs-134	1.70E+10	4.04E+10	3.30E+10	ND	1.31E+10	4.34E+09	7.07E+08
Cs-136	7.91E+08	3.12E+09	2.25E+09	ND	1.74E+09	2.38E+08	3.55E+08
Cs-137	2.22E+10	3.03E+10	1.99E+10	ND	1.03E+10	3.42E+09	5.87E+08
Ba-140	3.23E+06	4.06E+03	2.12E+05	ND	1.38E+03	2.32E+03	6.65E+06
La-140	4.97E+00	2.51E+00	6.62E-01	ND	ND	ND	1.84E+05
Ce-141	5.82E+02	3.94E+02	4.46E+01	ND	1.83E+02	ND	1.50E+06
Ce-144	4.30E+04	1.80E+04	2.31E+03	ND	1.07E+04	ND	1.45E+07
Pr-143	1.90E+01	7.61E+00	9.40E-01	ND	4.39E+00	ND	8.31E+04
Pr-144	1.33E-01	5.50E-02	6.74E-03	ND	3.10E-02	ND	1.91E-08
Nd-147	1.13E+01	1.31E+01	7.82E-01	ND	7.64E+00	ND	6.28E+04
Eu-154	2.84E+03	3.49E+02	2.49E+02	ND	1.67E+03	ND	2.53E+05
Hf-181	1.71E+01	8.31E+01	7.70E+00	ND	1.79E+01	ND	9.46E+04

TABLE 3.4 (Cont'd)

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

Vegetation Pathway

(m^2 mrem/yr) per (μCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	ND	2.26E+03	2.26E+03	2.26E+03	2.26E+03	2.26E+03	2.26E+03
Be-7	9.24E+04	2.11E+05	1.03E+05	ND	2.23E+05	ND	3.66E+07
Cr-51	ND	ND	4.64E+04	2.78E+04	1.02E+04	6.16E+04	1.17E+07
Mn-54	ND	3.13E+08	5.97E+07	ND	9.31E+07	ND	9.59E+08
Fe-55	2.10E+08	1.45E+08	3.38E+07	ND	ND	8.08E+07	8.31E+07
Fe-59	1.26E+08	2.96E+08	1.14E+08	ND	ND	8.28E+07	9.88E+08
Co-57	ND	1.17E+07	1.95E+07	ND	ND	ND	2.97E+08
Co-58	ND	3.07E+07	6.89E+07	ND	ND	ND	6.23E+08
Co-60	ND	1.67E+08	3.69E+08	ND	ND	ND	3.14E+09
Zn-65	3.17E+08	1.01E+09	4.56E+08	ND	6.75E+08	ND	6.36E+08
Rb-86	ND	2.19E+08	1.02E+08	ND	ND	ND	4.33E+07
Sr-89	9.97E+09	ND	2.86E+08	ND	ND	ND	1.60E+09
Sr-90	6.05E+11	ND	1.48E+11	ND	ND	ND	1.75E+10
Y-90	7.67E+05	ND	2.06E+04	ND	ND	ND	8.14E+09
Y-91	5.11E+06	ND	1.37E+05	ND	ND	ND	2.81E+09
Zr-95	1.17E+06	3.77E+05	2.55E+05	ND	5.91E+05	ND	1.19E+09
Nb-95	2.40E+05	1.34E+05	7.19E+04	ND	1.32E+05	ND	8.11E+08
Ru-103	4.77E+06	ND	2.06E+06	ND	1.82E+07	ND	5.57E+08
Ru-106	1.93E+08	ND	2.44E+07	ND	3.72E+08	ND	1.25E+10
Ag-110m	1.05E+07	9.75E+06	5.79E+06	ND	1.92E+07	ND	3.98E+09
Cd-109	0.00E+00	8.36E+07	2.92E+06	ND	8.00E+07	ND	8.43E+08
Sn-113	4.16E+08	1.18E+07	2.40E+07	6.75E+06	ND	ND	1.25E+09
Sb-124	1.04E+08	1.96E+06	4.11E+07	2.51E+05	ND	8.07E+07	2.94E+09
Sb-125	1.37E+08	1.53E+06	3.25E+07	1.39E+05	ND	1.05E+08	1.50E+09
Te-127m	3.49E+08	1.25E+08	4.26E+07	8.92E+07	1.42E+09	ND	1.17E+09
Te-129m	2.51E+08	9.38E+07	3.98E+07	8.64E+07	1.05E+09	ND	1.27E+09
I-130	3.93E+05	1.16E+06	4.57E+05	9.81E+07	1.81E+06	ND	9.97E+05
I-131	8.08E+07	1.16E+08	6.62E+07	3.79E+10	1.98E+08	ND	3.05E+07

TABLE 3.4 (Cont'd)

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

Vegetation Pathway

(m^2 mrem/yr) per (µCi/sec)

NUCLIDE	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
I-132	5.77E+01	1.54E+02	5.40E+01	5.40E+03	2.46E+02	ND	2.90E+01
I-133	2.09E+06	3.63E+06	1.11E+06	5.33E+08	6.33E+06	ND	3.26E+06
I-134	9.69E-05	2.63E-04	9.42E-05	4.56E-03	4.19E-04	ND	2.30E-07
I-135	3.90E+04	1.02E+05	3.77E+04	6.74E+06	1.64E+05	ND	1.15E+05
Cs-134	4.67E+09	1.11E+10	9.08E+09	ND	3.59E+09	1.19E+09	1.94E+08
Cs-136	4.27E+07	1.69E+08	1.21E+08	ND	9.38E+07	1.29E+07	1.91E+07
Cs-137	6.36E+09	8.70E+09	5.70E+09	ND	2.95E+09	9.81E+08	1.68E+08
Ba-140	1.29E+08	1.61E+05	8.42E+06	ND	5.49E+04	9.24E+04	2.65E+08
La-140	1.58E+04	7.98E+03	2.11E+03	ND	ND	ND	5.86E+08
Ce-141	1.97E+05	1.33E+05	1.51E+04	ND	6.19E+04	ND	5.10E+08
Ce-144	3.29E+07	1.38E+07	1.77E+06	ND	8.16E+06	ND	1.11E+10
Pr-143	6.26E+04	2.51E+04	3.10E+03	ND	1.45E+04	ND	2.74E+08
Pr-144	2.03E+03	8.43E+02	1.03E+02	ND	4.75E+02	ND	2.92E-04
Nd-147	3.33E+04	3.85E+04	2.31E+03	ND	2.25E+04	ND	1.85E+08
Eu-154	4.85E+07	5.97E+06	4.25E+06	ND	2.86E+07	ND	4.32E+09
Hf-181	1.40E+05	6.82E+05	6.32E+04	ND	1.47E+05	ND	7.76E+08

4. DOSE AND DOSE COMMITMENT FROM URANIUM FUEL CYCLE SOURCES

4.1 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 skin due to beta radiation from submersion in a could 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. 11.12.5)

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. 11.12.3).

There are no other Uranium Fuel Cycle Sources within 8km of the Callaway Plant.

4.1.1 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. 11.4 and 11.8.10). 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. 11.12.3 and 11.12.4).

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.

4.1.2 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 meteorological data for the location of the Nearest Resident with the highest value of X/Q.

The location of the Nearest Resident in each meteorological sector is determined from the Annual Land Use Census conducted in accordance with the Requirements of REC 9.12.1.1.

4.1.3 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. 11.7.2, 11.7.3, 11.13, and 11.13.1).

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. 11.5.5). Occupancy of areas within the SITE BOUNDARY is assumed to be averaged over a period of one year.

Any reevaluation of assumptions should consider only real receptors and real pathways using realistic assumption, and 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).

4.1.3.1 Total Dose to the Farmer from Gaseous Effluents

The Total Dose to the farmer from gaseous effluents is calculated for the adult age group using the methodology discussed in Section 3, utilizing concurrent meteorological data at the farmer's residence and historical meteorological data from Table 6.1 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, and already have the total occupancy of 1100 hours/year factored into their value (Ref. 11.5.6).

The residence of the current tenant is located at a distance of 3830 meters in the SE sector. The gaseous effluents dose at the farmer's residence is due to plume exposure from Noble Gases and the ground plane, inhalation, and ingestion pathways. For conservatism, it is acceptable to assume that all of the ingestion pathways exist at this location.

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.

4.1.3.1.1 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. 11.6.14, 11.6.15, 11.6.16 and 11.6.17).

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, which is 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. 11.6.17). The walls are of type 304 stainless steel and have an average thickness of .87 cm. (Ref. 11.14.1).

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. 11.17). The computer program ISOSHLD (Ref. 11.18, 11.19 and 11.20) will normally be utilized to perform this calculation.

4.1.3.1.2 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 within the Containment Building. 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 obscured by significant plant structures (Ref. 11.14.3).

The maximum direct radiation dose from the Unit to the farmer is thus approximately 9E-3 mrads per year, assuming a maximum occupancy of 1100 hours per year.

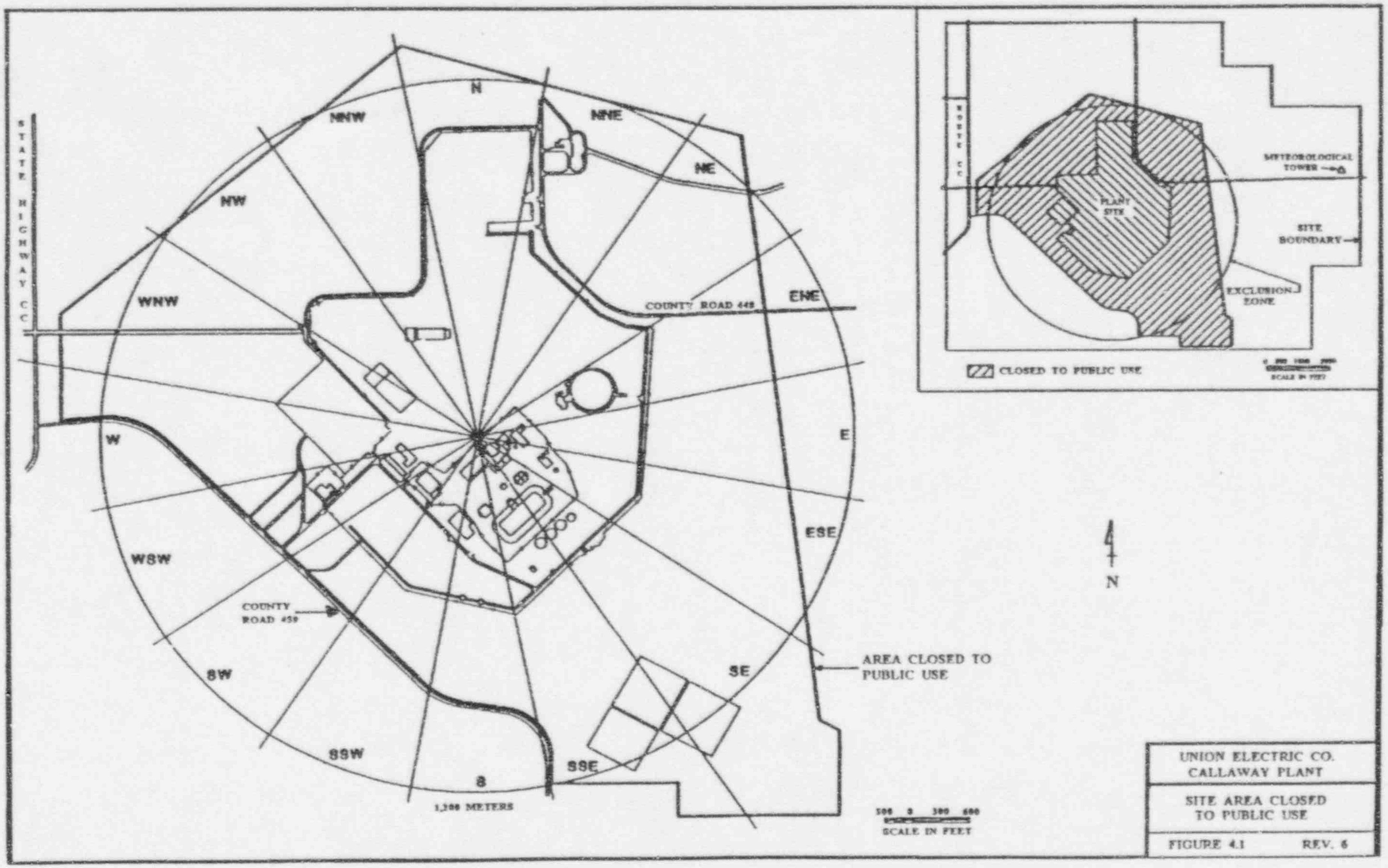
4.1.3.1.3 Direct Radiation Dose From On-Site Storage Of Low Level Radioactive Waste

The on-site storage area for radioactive wastes is located Plant Southwest of the radwaste building and consists of a concrete pad enclosed by a fence. The storage area is bounded on two sides by the radwaste building. The area is also partially bounded on a third side by the Discharge monitoring tanks dike system. The radioactive wastes are stored in this area using high integrity containers (HIC) inside Onsite Storage Containers (OSC) and LSA type storage containers. The HIC has the highest potential for containing significant amounts of radioactive material, and constitutes the only potentially significant source of direct radiation from on-site radioactive waste storage.

Direct radiation dose from the HICs 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.

The HICs typically are right circular cylinders approximately 1.7 meters in diameter and 1.8 meters in height. The HICs are stored inside OSCs which typically are constructed of concrete with additional shielding as necessary to minimize external doses. The individual parameters (e.g., dimensions, shielding material, etc.) for each OSC will be accounted for in the calculations.

The direct radiation dose from the On-Site Storage area is the summation of the individual calculated HIC doses based on the HIC isotopic contents and the OSC design parameters, considering buildup, attenuation, and shielding. Appropriate methodology for calculating the dose rate is given in Safety Analysis Calculations ZZ-293 and ZZ-310. The computer program MICROSHIELD (Ref. 11.24) will normally be utilized to perform this calculation.



UNION ELECTRIC CO.
CALLAWAY PLANT

SITE AREA CLOSED TO PUBLIC USE

FIGURE 4.1 REV. 6

5. RADIOLOGICAL ENVIRONMENTAL MONITORING

5.1 DESCRIPTION OF THE RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The Radiological Environmental 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 reasons. Efforts are made to obtain all required samples within the required time frame. Any deviation(s) in sampling frequency or location is documented in the Annual Radiological Environmental Operating Report.

Sampling, reporting, and analytical requirements are given in Tables 9.11-A, 9.11-B, and 9.11-C.

Airborne, waterborne, 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.

5.2 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. 11.16).

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. (Ref. 11.14.10)

6. DETERMINATION OF ANNUAL AVERAGE AND SHORT TERM ATMOSPHERIC DISPERSION PARAMETERS

6.1 ATMOSPHERIC DISPERSION PARAMETERS

The values presented in Table 6.1 and Table 6.2 were determined through the analysis of on-site meteorological 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 variable trajectory plume segment atmospheric transport model MESODIF-II (NUREG/-CR-0523) and the straight-line Gaussian dispersion model XOQDOQ (NUREG/CR2919) 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. 11.6.12).

The Unit Vent and Radwaste Building Vent releases are at elevations of 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. 11.5.2).

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. 11.6.13):

$$X = X_1 \left(d / d_1\right)^B \qquad (6.1)$$

Where:

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

X_1, X_2 = Atmospheric dispersion parameters at distance d_1 and d_2, respectively, from the source.

The distances d_1 and d_2 were selected such that they satisfy the relationship.

$$d_1 < d < d_2$$

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. 11.9.1 and 11.15.1):

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

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 6.3.

Short term dispersion estimates are not applicable to short term releases which are 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. 11.8.1).

6.1.3.1 The Determination of the Slope Factor (S)

The general approach employed by subroutine PURGE of XOQDOQ (Ref. 11.15.1) was utilized to produce values of the slope of the (X/Q) curves for both the Radwaste Building Vent and the Unit Vent. However, instead of using approximation procedures to produce the 15 percentile (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 that 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 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 11.15.1. The straight line connecting these points represents $(X/Q)_1$ values for intermittent releases, ranging in duration from one hour to 8760 hours. The slope (S) of the curve is expressed as:

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

or

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

6.1.4 Atmospheric Dispersion Parameters for Farming Areas within the SITE BOUNDARY

The dispersion parameters for farming areas within the SITE BOUNDARY are intended for a narrow scope application: That of calculating the dose to the current farmer from gaseous effluents while he conducts farming activities within the SITE BOUNDARY.

For the purpose of these calculations, it was assumed that all of the farmer's time, approximately 1100 hours per year, is spent on croplands within the SITE BOUNDARY, and that his time is divided evenly over all of the croplands. Fractional acreage/time - weighted dispersion parameters were calculated for each plot as described in reference 11.5.6. The weighted dispersion parameters for each plot were then summed (according to type) in order to produce a composite value of the dispersion parameters which are presented in Tables 6.1 and 6.2. These dispersion parameters therefore represent the distributed activities of the farmer within the SITE BOUNDARY and his estimated occupancy period.

6.2 ANNUAL METEOROLOGICAL DATA PROCESSING

The annual atmospheric dispersion parameters utilized in the calculation of doses for demonstration of compliance with the numerical dose objectives of 10 CFR 50, Appendix I, are determined using computer codes and models consistent with XOQDOQ (Ref. 11.15). These codes have been validated and verified by a qualified meteorologist prior to implementation. Multiple sensors are utilized to ensure 90% valid data recovery for the wind speed, wind direction, and ambient air temperature parameters as required by Regulatory Guide 1.23. The selection hierarchy is presented in Table 6.5.

TABLE 6.1

HIGHEST ANNUAL AVERAGE ATMOSPHERIC DISPERSION PARAMETERS

UNIT VENT

LOCATION (b)	SECTOR	DISTANCE (METERS)	X/Q	X/Q DECAYED/ UNDEPLETED	X/Q DECAYED/ DEPLETED	D/Q
			(sec/m^3)	(sec/m^3)	(sec/m^3)	(m^{-2})
SITE BOUNDARY(a)	NNW	2200	1.0E-6	9.9E-7	8.5E-7	4.3E-9
Nearest Residence (c) (d)	NNW	2864	6.8E-7	6.8E-7	5.7E-7	2.6E-9
Farmer's Residence(c)	SE	3830	2.5E-7	2.5E-7	2.1E-7	1.1E-9
Farming Areas within the Site Boundary (c) (e)	N/A	N/A	2.1E-7	2.1E-7	1.9E-7	1.1E-9

(a) Values given are from FSAR Table 2.3-82

(b) Data from 1995 Land Use Census

(c) Values derived from FSAR Table 2.3-83, using the methodology presented in Equation (6.1) (Ref. 11.5.6)

(d) All pathways are assumed to exist at the location of the nearest resident.

(e) These values were derived for a narrow scope application. Extreme caution should be exercised when determining their suitability for use in other applications.

Building Shape Parameter (C) = 0.5 (Ref. 11.5.3)

Vertical Height of Highest Adjacent Building (V) = 66.45 meters (Ref. 11.5.3)

TABLE 6.2
HIGHEST ANNUAL AVERAGE ATMOSPHERIC DISPERSION PARAMETERS

RADWASTE BUILDING VENT

LOCATION (b)	SECTOR	DISTANCE (METERS)	X/Q	X/Q DECAYED/ UNDEPLETED	X/Q DECAYED/ DEPLETED	D/Q
			(sec/m^3)	(sec/m^3)	(sec/m^3)	(m^{-2})
SITE BOUNDARY(a)	NNW	2200	1.3E-6	1.3E-6	1.1E-6	4.3E-9
Nearest Residence (c) (d)	NNW	2864	8.7E-7	8.7E-7	7.2E-7	2.6E-9
Farmer's Residence(c)	SE	3830	3.0E-7	3.0E-7	2.4E-7	1.1E-9
Farming Areas Within Site Boundary (c) (e)	N/A	N/A	2.9E-7	2.9E-7	2.6E-7	1.1E-9

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

(b) Data from 1995 Land Use Census

(c) Values derived from FSAR Table 2.3-81, using the methodology presented in Equation (6.1) (Ref. 11.5.6)

(d) All pathways are assumed to exist at the location of the nearest resident.

(e) These values were derived for a narrow scope application. Extreme caution should be exercised when determining their suitability for use in other applications

Building Shape Parameter (C) = 0.5 (Ref. 11.5.3)

Vertical Height of Highest Adjacent Building (V) = 19.96 meters (Ref. 11.5.3)

TABLE 6.3

SHORT TERM DISPERSION PARAMETERS (a) (c)

Location (b)	Sector	Distance	Slope Factor(s) Unit Vent	Slope Factor(s) Radwaste Building Vent
Site Boundary	S	1300	-.328	-.320
Nearest Residence (d)	NNW	2865	-.264	-.268

(a) Reference 11.5.3

(b) Data from 1995 Land Use Census

(c) Recirculation Factor = 1.0

(d) All pathways are assumed to exist at the location of the nearest resident.

TABLE 6.4

APPLICATION OF ATMOSPHERIC DISPERSION PARAMETERS

Dose Pathway	Dispersion Parameter	Controlling Age Group	Rec	Controlling Location
Noble Gas, Beta Air & Gamma Air	x/Q, decayed/undepleted (2.26 day halflife)	N/A	9.7	Site Boundary
Noble Gas, Total Body & Skin	x/Q, decayed/undepleted (2.26 day halflife)	N/A	9.6	Site Boundary
Inhalation	x/Q, decayed/depleted (8 day halflife)	Child	9.6 9.8	Nearest Resident Site Boundary
Ground Plane Deposition	D/Q	N/A	9.8	Nearest Resident
Ingestion pathways	D/Q*	Child	9.8	Nearest Resident

* For H-3, x/Q, decayed/depleted is used instead of D/Q (Ref. 11.11.1).

TABLE 6.5

METEOROLOGICAL DATA SELECTION HIEARCHY

Parameter	Primary	First Alternate	Second Alternate	Third Alternate
Wind Speed	10m Pri	10m Sec	60m Pri	90m Pri
Wind Direction	10m Pri	10m Sec	60m Pri	90m Pri
Air Temperature	10m Pri	10m Sec		
Wind Variability	10m Pri	10m Sec	60m Pri	90m Pri
Temp Different	60-10m Pri	90-10m Pri	90-60 Pri	
Dew Point	10m Pri			
Precipitation	1m Pri			

(a) Pri indicates primary tower

(b) Sec indicates secondary tower

Table 6.6

Application of Atmospheric Dispersion Parameters: Annual Effluent Release Report

Dose Pathway	Dispersion Parameter	Controlling Age Group	Dispersion Values	Controlling Location
Noble Gas, Beta Air & Gamma Air Dose	x/Q, decayed/undepleted (2.26 day halflife)	N/A	Concurrent	Site Boundary Nearest Resident
Noble Gas, Total Body & Skin Dose	x/Q, decayed/undepleted (2.26 day halflife)	N/A	Concurrent	Site Boundary Nearest Resident
			Concurrent Historical	Farmer's Residence Inside Site Boundary
Ground Plane Deposition Dose	D/Q	N/A	Concurrent	Site Boundary Nearest Resident
			Concurrent Historical	Farmer's Residence Inside Site Boundary
Inhalation Dose	x/Q, decayed/depleted (8 day halflife)	Child	Concurrent	Site Boundary Nearest Resident
		Adult	Concurrent Historical	Farmer's Residence Inside Site Boundary
Ingestion Dose Pathways	D/Q*	Child	Concurrent	Site Boundary Nearest Resident
		Adult	Concurrent Historical	Farmer's Residence Inside Site Boundary

* For H-3, x/Q, decayed/depleted is used instead of D/Q (Ref. 11.11.1).

7. REPORTING REQUIREMENTS

7.1 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (CTSN 2804)

Routine Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment.

The reports shall include the results of Land Use Census required by REC 9.12. It shall also include a listing of new locations for environmental monitoring identified by the Land Use Census pursuant to REC 9.12.1.

The Annual Radiological Environmental Operating Report shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the ODCM, as well as summarized tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action being taken if the specified program is not being performed as required by 9.13.1; reasons for not conducting the Radiological Environmental Monitoring Program as required by 9.11.1 and discussion of all deviations from the sampling schedule of Table 9.11-A, discussion of environmental sample measurements that exceed the reporting levels of Table 9.11-B, but are not the result of the plant effluents, pursuant to 9.11.1; and discussion of all analyses in which the LLD required by Table 9.11-C was not achievable.

7.2 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (CTSN 2805)

Routine Annual Radioactive Effluent Release Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases 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.

The Annual Radioactive Effluent Release Report shall include an annual summary of hourly meteorological data collected over the previous calendar year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distribution of wind speed, wind direction, and atmospheric stability*.

---

* One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

* In lieu of submission with the Annual Radioactive Effluent Release Report, Union Electric 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.

This report shall also include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. This report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Technical Specifications, Figures 5.1-3 and 5.1-4) during the report period using historical average atmospheric conditions. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. Assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Annual Radioactive Effluent Release Report shall include an assessment of radiation doses to the most likely 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 calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Doses to the MEMBER OF THE PUBLIC shall be calculated using the methodology and parameters of the ODCM.

The Annual Radioactive Effluent Release Reports shall 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 Annual Radioactive Effluent Release Reports shall include a summary description of any major changes made during the year to any Liquid or Gaseous Treatment Systems, pursuant to Section 10.1. It shall also include a listing of new locations for dose calculations identified by the Land Use Census pursuant to REC 9.12.1.

Reporting requirements for changes to Solid Waste Treatment Systems is addressed in APA-ZZ-01011, PROCESS CONTROL PROGRAM (PCP).

The Annual Radioactive Effluent Release Reports shall 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, and a description of the events leading to the liquid holdup tanks or gas storage tanks exceeding the limits of FSAR Section 16.11.1.1 OR 16.11.3.1.

The Annual Radioactive Effluent Release Reports shall include as part of or submitted concurrent with, a complete and legible copy of all revisions of the ODCM that occurred during the year pursuant of Technical Specification 6.14.

Solid Waste reporting is addressed in APA-ZZ-01011, PROCESS CONTROL PROGRAM (PCP).

8. IMPLEMENTATION OF ODCM METHODOLOGY (CTSN 2791)

The ODCM provides the mathematical relationships used to implement the Radioactive Effluent Controls. For routine effluent release and dose assessment, computer codes are utilized to implement the ODCM methodologies. These codes are evaluated in accordance with the requirements of plant operating procedures to ensure that they produce results consistent with the methodologies presented in the ODCM. Procedures which implement the ODCM methodology are contained in the Plant Operating Manual.

9. RADIOACTIVE EFFLUENT CONTROLS (REC)

a. The terms in this section that appear in capitalized type are defined in Technical Specifications.

b. All frequency notations are per Table 1.1 of Technical Specifications.

9.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

REC 9.1 has been relocated to Section 16.3.3.6 of the FSAR.

The following should be used to cross-reference REC 9.1 surveillances to the appropriate section of the FSAR:

Monitor	REC Number	FSAR Section	Type
HB-RE-18	9.1.2.1-1.a	16.3.3.6.1-1.a	Liquid rad monitor
BM-RE-52	9.1.2.1-1.b	16.3.3.6.1-1.b	Liquid rad monitor
HB-FE-2017	9.1.2.1-2.a	16.3.3.6.1-2.a	Flow element
BM-FE-0054	9.1.2.1-2.b	16.3.3.6.1-2.b	Flow element
FE-DB-1006, 1101	9.1.2.1-2.c	16.3.3.6.1-2.c	Flow element

9.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

REC 9.2 has been relocated to Section 16.3.3.7B of the FSAR.

The following should be used to cross-reference REC 9.2 surveillances to the appropriate section of the FSAR:

Monitor	REC Number	FSAR Section	Type
Unit Vent			
GT-RE-21B	9.2.2.1-1.a	16.3.3.7b.1-1.a	Gas
GT-RE-21A & B	9.2.2.1-1.c	16.3.3.7b.1-1.c	Iodine sampler
GT-RE-21A & B	9.2.2.1-1.b	16.3.3.7b.1-1.b	Particulate sampler
GT-RE-21A & B	9.2.2.1-1.d	16.3.3.7b.1-1.d	Unit Vent flow rate
GT-RE-21A & B	9.2.2.1-1.e	16.3.3.7b.1-1.e	Particulate and Radioiodine Sampler flow rate Monitor
Containment Purge			
GT-RE-22 GT-RE-33	9.2.2.1-2.a	16.3.3.7b.1-2.a	Gas
GT-RE-22 GT-RE-33	9.2.2.1-2.c	16.3.3.7b.1-2.c	Iodine sampler
GT-RE-22 GT-RE-33	9.2.2.1-2.b	16.3.3.7b.1-2.b	Particulate sampler
GT-RE-22 GT-RE-33	9.2.2.1-2.d	16.3.3.7b.1-2.d	Containment purge flow rate
GT-RE-22 GT-RE-33	9.2.2.1-2.e	16.3.3.7b.1-2.e	Particulate and Radioactive Sampler flow rate Monitor
Radwaste Building Ventilation			
GH-RE-10B	9.2.2.1-3.a	16.3.3.7b.1-3.a	Gas
GH-RE-10A & B	9.2.2.1-3.c	16.3.3.7b.1-3.c	Iodine sampler
GH-RE-10A & B	9.2.2.1-3.b	16.3.3.7b.1-3.b	Particulate sampler
GH-RE-10A & B	9.2.2.1-3.d	16.3.3.7b.1-3.d	Radwaste Building Vent Flow rate
GH-RE-10A & B	9.2.2.1-3.e	16.3.3.7b.1-3.e	Particulate and Radioactive Sampler flow rate Monitor

9.3 LIQUID EFFLUENTS CONCENTRATION

9.3.1 Controls (CTSN 41834)

9.3.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Technical Specifications, Figure 5.1-4) shall be limited to the concentration specified in 10 CFR Part 20.1-20.601, 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 2×10^{-4} microCurie/ml total activity.)

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.3.2 Surveillance Requirements

9.3.2.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 9.3-A.

9.3.2.2 The results of the radioactivity analysis shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of REC 9.3.1.1.

TABLE 9.3-A

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

1. Discharge Monitor Tanks (Batch Release)[2]			
SAMPLING FREQUENCY(7)	**MINIMUM ANALYSIS FREQUENCY**	**TYPE OF ACTIVITY ANALYSIS**	**LLD (1) (μCi/ml)**
Prior to Each Batch	Prior to Each Batch	Principal Gamma Emitters (3) I-131 Dissolved and Entrained Gases (Gamma Emitters) H-3	5E-7 1E-6 1E-5 1E-5
	Monthly Composite (4)	Gross Alpha	1E-7
	Quarterly Composite (4)	Sr-89, Sr-90 Fe-55 Np-237 Pu-238 Pu-239/240 Pu-241 Am-241 Cm-242 Cm-243/244	5E-8 1E-6 5E-9 5E-9 5E-9 5E-7 5E-9 5E-9 5E-9

2. Steam Generator Blowdown (Continuous Release)[5]			
SAMPLING FREQUENCY(7)	**MINIMUM ANALYSIS FREQUENCY**	**TYPE OF ACTIVITY ANALYSIS**	**LLD (1) (μCi/ml)**
Daily Grab Sample (6)	Daily	Principal Gamma Emitters (3) I-131 Dissolved and Entrained Gases (Gamma Emitters) H-3	5E-7 1E-6 1E-5 1E-5
	Monthly Composite (4)	Gross Alpha	1E-7
	Quarterly Composite (4)	Sr-89, Sr-90 Fe-55	5E-8 1E-6

TABLE 9.3-A (Cont'd)

TABLE NOTATIONS

(1) The LLD is described in Attachment 1.

(2) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed a method described in the ODCM to assure representative sampling.

(3) The principal gamma emitters for which the LLD control applies include the following radionuclides: Mn-54, Fe-59, Co-58, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report pursuant to Technical Specification 6.9.1.7, in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(4) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specium that is representative of the liquids released. Prior to analysis, all samples taken for the composite shall be thoroughly mixed in order for the composite samples to be representative of the effluent release.

(5) A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

(6) Samples shall be taken at the initiation of effluent flow and at least once per 24 hours thereafter while the release is occurring. To be representative of the liquid effluent, the sample volume shall be proportioned to the effluent stream discharge volume. The ratio of sample volume to effluent discharge volume shall be maintained constant for all samples taken for the composite sample.

(7) Samples shall be representative of the effluent release.

9.4 DOSE FROM LIQUID EFFLUENTS

9.4.1 Controls (CTSN 41834)

9.4.1.1 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Technical Specifications, Figure 5.1-4) shall be limited:

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

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

APPLICABILITY: At all times.

ACTION: (CTSN 1161)

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Technical Specification 6.9.2, Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include: (2) the results of radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141, Clean Drinking Water Act.*

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.4.2 Surveillance Requirements

9.4.2.1 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

* The requirements of ACTION a.(1) and (2) are applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge. In the case of river-sited plants this is 3 miles downstream only.

9.5 LIQUID RADWASTE TREATMENT SYSTEM

9.5.1 Controls **(CTSN 41834)**

9.5.1.1 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Technical Specifications, Figure 5.1-4) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION: **(CTSN 1161)**

a. With radioactive liquid waste being discharged in excess of the above limits and the Liquid Radwaste Treatment Systems are not being fully utilized, prepare and submit to the Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report that includes the following information:

1) Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability.

2) Action(s) taken to restore the inoperable equipment to OPERABLE status, and

3) Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.5.2 Surveillance Requirements

9.5.2.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

9.5.2.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting REC 9.3.1.1 and 9.4.1.1.

9.6 GASEOUS EFFLUENTS DOSE RATE

9.6.1 Controls (CTSN 41834)

9.6.1.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Technical Specifications, Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/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 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.6.2 Surveillance Requirements

9.6.2.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

9.6.2.2 The dose rate due to Iodine-131 and 133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 9.6-A.

TABLE 9.6-A

RADIOACTIVE GASEOUS EFFLUENTS SAMPLING AND ANALYSIS PROGRAM

1. Waste Gas Decay Tank			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Prior to each release- grab sample	Prior to each tank	Principal Gamma Emitters- particulate, iodine, noble gas (2)	1E-4
Continuous	See footnote 8		

2. Containment Purge or Vent			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Prior to each release- grab sample	Prior to each release	Principal Gamma Emitters- particulate, iodine, noble gas (2) H-3(oxide)	1E-4 1E-6
Continuous	See footnote 8		

3. Unit Vent (3)			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Monthly- grab sample (3)(4)	Monthly (3) (4)	Principal Gamma Emitters- particulate, iodine, noble gas (2) H-3(oxide)	1E-4 1E-6
Continuous (6)	Weekly (7)	I-131 I-133 Principal Gamma Emitters- particulate nuclides only (2)	1E-12 1E-10 1E-11
	Monthly Composite	Gross Alpha	1E-11
	Quarterly Composite	Sr-89, Sr-90	1E-11

4. Radwaste Building Vent			
SAMPLING FREQUENCY (9)	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LLD (1) (μCi/ml)
Monthly- grab sample	Monthly	Principal Gamma Emitters- particulate, iodine, noble gas (2)	1E-4
Continuous (6)	Weekly (7)	I-131 I-133 Principal Gamma Emitters- particulate nuclides only (2)	1E-12 1E-10 1E-11
	Monthly Composite	Gross Alpha	1E-11
	Quarterly Composite	Sr -89, Sr-90	1E-11

TABLE 9.6-A (Cont'd)

TABLE NOTATIONS

(1) The LLD is described in Attachment 1.

(2) The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, I-131, Cs-134, Cs-137, Ce-141, and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Any nuclide which is identified in the sample and which is also listed in the ODCM gaseous effluents dose factor tables, shall be analyzed and reported in the Annual Effluent Release Report.

(3) If the Unit Vent noble gas monitor (GT-RE-21B) shows that the effluent activity has increased (relative to the pre-transient activity) by more than a factor of 3 following a reactor shutdown, startup, or a thermal power change which exceeds 15% of the rated thermal power within a 1 hour period, samples shall be obtained and analyzed for noble gas, particulates and iodines. This sampling shall continue to be performed at least once per 24 hours for a period of 7 days or until the Unit Vent noble gas monitor no longer indicates a factor of 3 increase in Unit Vent noble gas activity, whichever comes first.

(4) Tritium grab samples shall be taken and analyzed at least once per 24 hours when the refueling canal is flooded.

(5) Deleted

(6) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with REC 9.6.1.1, 9.7.1.1, and 9.8.1.1.

(7) Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or removal from the sampler. When sampling is performed in accordance with footnote 3 (above), then the LLD may be increased by a factor of 10.

(8) Continuous sampling of this batch release pathway is included in the continuous sampling performed for the corresponding continuous release pathway.

(9) Samples shall be representative of the effluent release.

9.7 DOSE - NOBLE GASES

9.7.1 Controis **(CTSN 41834)**

9.7.1.1 The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see Technical Specifications Figure 5.1-3) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION: **(CTSN 1161)**

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.7.2 Surveillance Requirements

9.7.2.1 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

9.8 DOSE - IODINE-131 AND 133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

9.8.1 Controls (CTSN 41834)

9.8.1.1 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, from each unit, to areas at and beyond the SITE BOUNDARY (see Technical Specifications, Figure 5.1-3) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION: (CTSN 1161)

a. With the calculated dose from the release of Iodine-131 and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Technical Specification 6.9.[illegible], a Special Report that identifies the cause(s) for exceeding the limits and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.8.2 Surveillance Requirements

9.8.2.1 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131 and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

9.9 GASEOUS RADWASTE TREATMENT SYSTEM

9.9.1 Controls **(CTSN 41834)**

9.9.1.1 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 (see Figure Technical Specification's 5.1-3) 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 a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times

ACTION:

a. With radioactive gaseous waste being discharged in excess of the above limits, and the Gaseous Radwaste Treatment Systems are not being fully utilized, prepare and submit to the Commission within 30 days, pursuant to Technical Specifications 6.9.2, a Special Report that includes the following information:

1) Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

2) Action(s) taken to restore the inoperable equipment to OPERABLE status, and

3) Summary description of action(s) taken to prevent a recurrence.

b. The provision of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.9.2 Surveillance Requirements

9.9.2.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

9.9.2.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM and the WASTE GAS HOLDUP SYSTEMS shall be considered OPERABLE by meeting REC 9.6.1.1 and 9.7.1.1 or 9.8.1.1.

9.10 TOTAL DOSE

9.10.1 Controls **(CTSN 41834)**

9.10.1.1 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 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in gaseous effluents exceeding twice the limits of REC 9.4.1.1b, 9.7.1.1a, 9.7.1.1b, 9.8.1.1a, or 9.8.1.1b, calculations should be made including direct radiation contributions from the units and from outside storage tanks to determine whether the above limits of REC 9.10.1.1 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent release to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.2203, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.10.2 Surveillance Requirements

9.10.2.1 Cumulative dose contributions from gaseous effluents shall be determined in accordance with REC 9.7.2.1, and 9.8.2.1, and in accordance with the methodology and parameters in the ODCM.

9.10.2.2 Cumulative dose contributions from direct radiation from the units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirements is applicable only under conditions set forth in ACTION a. of REC 9.10.1.1.

9.11 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

9.11.1 Controls (CTSN 41834)

9.11.1.1 The Radiological Environment Monitoring Program shall be conducted as specified in Table 9.11-A.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 9.11-A, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Technical Specification 6.9.1.6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 9.11-B when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of REC 9.4.1.1, 9.7.1.1, or 9.8.1.1. When more than one of the radionuclides in Table 9.11-B are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 9.11-B are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of REC 9.4.1.1, 9.7.1.1 or 9.8.1.1. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report, required by Technical Specification 6.9.1.6.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 9.11-A, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program**. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In the next Annual Radiological Environmental Operating Report include the revised figure(s) and tables reflecting the new sample location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of new location(s) for obtaining samples.

d. When LLDs specified in Table 9.11-C are unachievable due to uncontrollable circumstances, (such as background fluctuations, unavailable small sample sizes, the presence of interfering nuclides, etc.) the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report.

e. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

* The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

** Excluding short term or temporary unavailability.

9.11.2 Surveillance Requirements

9.11.2.1 The radiological environmental monitoring samples shall be collected pursuant to Table 9.11-A and shall be analyzed pursuant to the requirements of Table 9.11-A and the detection capabilities required by Table 9.11-C.

TABLE 9.11-A

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation (2)	Forty routine monitoring stations with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of sixteen stations, one in each meteorological sector in the general area of the SITE BOUNDARY; An outer ring of stations, one in each meteorological sector in the 6- to 8-km (3 to 5 mile) range from the site; and Eight stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly	Gamma dose quarterly
2. Airborne Radioiodine and Particulates	Samples from five locations; Three samples from close to the SITE BOUNDARY locations, in different sectors, with high calculated annual average ground level D/Qs. One sample from the vicinity of a community located near the plant with a high calculated annual average ground level D/Q. One sample from a locartion in the vicinity of Fulton, MO. (Ref SOS 95-2280).	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Canister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change; (4) and gamma isotopic analysis (5) of composite (by location) quarterly.

TABLE 9.11-A (Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne			
a. Surface (6)	One sample upstream One sample downstream	Composite sample over 1-month period (7).	Gamma isotopic(5) and tritium analysis monthly
b. Drinking	One sample of each of one to three of the nearest water supplies within 10 miles downstream that could be affected by its discharge. One sample from a control location.	Composite sample over 2-week period(7) 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(8). Composite for gross beta and gamma isotopic analyses(5) 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, the program will be revised to include this pathway (Ref. 11.6.6).

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
c. Sediment from shoreline	One sample from downstream area with existing or potential recreational value	Semiannually	Gamma isotopic analysis(5) semiannually
4. Ingestion			
a. Milk	Samples from milking animals in three different meteorological sectors within 5 km (3 mile) distance having the highest dose potential. If there are none, then one sample from milking animals in each of three different meteorological sectors between 5 to 8 km (3 to 5 mile) distance where doses are calculated to be greater than 1 mrem per yr. One sample from milking animals at a control location, 15 to 30 km (10 to 20 mile) distance and in one of the least prevalent wind directions.	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic(5) and I-131 analysis semimonthly when animals are on pasture; monthly at other times

Due to the lack of milking animals which satisfy these requirements, the milk pathway is currently not included as 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.

TABLE 9.11-A (Continued)

TABLE NOTATIONS

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Cont'd)			
b. Fish	One sample of each commercially and recreationally important species in vicinity of plant discharge area. One sample of same species in areas not influenced by plant discharge.	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis(5) on edible portions
c. Food Products	One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest (9)(10)	Gamma isotopic analysis (5) on edible portion

As there are no areas irrigated by water in which liquid plant wastes have been discharged within 50 miles downstream of the discharge point, this sample type is not currently included as part of the Callaway Plant Radiological Environmental Monitoring Program. Should future irrigation water intakes be constructed within 10 river miles downstream of the discharge point, the program will be revised to include this sample type (Ref. 11.7.4 and 11.7.5).

	Samples of three different kinds of broad leaf vegetation if available 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 (5) and I-131 analysis
	One sample of each of the similar broad leaf vegetation grown 15 to 30 km (10 to 20 mile) distant in one of the least prevalent wind directions if milk sampling is not performed	Monthly when available	Gamma isotopic (5) and I-131 analysis

TABLE 9.11-A (Continued)

TABLE NOTATIONS

(1) Specific parameters of distance and direction sector from the centerline of one unit, and additional description where pertinent, shall be provided for each and every sample location in Table 9.11-A in a table and figure(s) in the appropriate plant procedure. 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 malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 6.9.1.6. **(CTSN 2804)**

It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable specific 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. Submit in the next Annual Radiological Environmental Operating Report documentation for a change including the revised figure(s) and table reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the selection of the new location(s) for obtaining samples.

The selection of sample locations should consider accessibility of the sample site, availability of power, wind direction frequency, sector population, equipment security, and the presence of potentially adverse environmental conditions (such as unusually dusty conditions, etc.).

(2) 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 purposes 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 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.

(3) (Deleted)

(4) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than an established baseline activity level, gamma isotopic analysis shall be performed on the individual samples. **(CTSN 43303)**

(5) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

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

(7) In this program, composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

(8) Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

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

(10) 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 9.11-B

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

	REPORTING LEVELS				
ANALYSIS	WATER (pCi/ℓ)[a]	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)[b]	MILK (pCi/ℓ)[a]	FOOD PRODUCTS (pCi/kg, wet)[b]
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	

(a) Multiply the values in this table by 1E-9 to convert to units of µCi/ml.
(b) Multiply the values in this table by 1E-9 to convert to units of µCi/g.

---

* For drinking water samples. This is 40 CFR Part 141 value. For surface water samples, a value of 30,000 pCi/ℓ may be used.

** Total activity, parent plus daughter activity.

TABLE 9.11-C

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

LOWER LIMIT OF DETECTION (LLD) (1),(2), (3)

ANALYSIS	SURFACE WATER (pCi/ℓ)[a]	DRINKING WATER (pCi/ℓ)[a]	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)[b]	MILK (pCi/ℓ)[a]	FOOD PRODUCTS (pCi/kg, wet)[b]	SEDIMENT (pCi/kg, dry)[b]
Gross Beta	4	4	0.01				
H-3	3000	2000					
Mn-54	15	15		130			
Fe-59	30	30		260			
Co-58,60	15	15		130			
Zr-Nb-95**	15	15					
I-131	1000	1	0.07		1	60	
Cs-134	15	15	0.05	130	15	60	150
Cs-137	18	18	0.06	150	18	80	180
Ba-La-140**	15	15			15		

(a) Multiply the values in this table by 1E-9 to convert to units of μCi/ml.
(b) Multiply the values in this table by 1E-9 to convert to units of μCi/g.

** Total activity, parent plus daughter activity.

TABLE 9.11-C (Continued)

TABLE NOTATIONS

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

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13, Revision 1, July 1977.

(3) The LLD is described in Attachment 1.

9.12 RADIOLOGICAL ENVIRONMENTAL MONITORING LAND USE CENSUS

9.12.1 Controls (CTSN 41835)

9.12.1.1 A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than $50m^2$ ($500\ ft^2$) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in REC 9.8.2.1, identify the new location(s) in the next Annual Radioactive Effluent Release Report, pursuant to Technical Specification 6.9.1.7.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with REC 9.11.1.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program except for vegetation samples which shall be added to the program before the next growing season. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this Land Use Census was conducted. In the next Annual Radiological Environmental Operating Report include the revised figure(s) and tables reflecting the new sample location(s) with information supporting the change in sample location.

c. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.12.2 Surveillance Requirements

9.12.2.1 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information which will provide the best results, such as, but not limited to, door-to-door survey, aerial survey, or by consulting local agriculture authorities and/or residents. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 6.9.1.6.

---

* Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each to two different direction sectors with the highest predicted D/Q's in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 9.11-A, Part 4.c shall be followed, including analysis of control samples.

9.13 RADIOLOGICAL ENVIRONMENTAL MONITORING INTERLABORATORY COMPARISON PROGRAM

9.13.1 Controls **(CTSN 41835)**

9.13.1.1 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the USNRC.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in The Annual Radiological Environmental Operating Report pursuant to Technical Specification 6.9.1.6.

b. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

9.13.2 Surveillance Requirements

9.13.2.1 The Interlaboratory Comparison Program shall be described in the plant procedures. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 6.9.1.6.

10. ADMINISTRATIVE CONTROLS

10.1 MAJOR CHANGES TO LIQUID AND GASEOUS RADWASTE TREATMENT SYSTEMS

10.1.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid and gaseous):

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the On-Site Review Committee (ORC). The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components and process involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents, to the actual releases for the period prior to when the changes are to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the ORC.

b. Shall become effective upon review and approval by the ORC and in accordance with Technical Specification 6.5.3.1.

10.2 CHANGES TO THE OFFSITE DOSE CALCULATION MANUAL (ODCM) **(CTSN 2815)**

10.2.1 All changes to the ODCM shall be completed pursuant to Technical Specification 6.14 and approved as per APA-ZZ-00101, "Preparation, Review, Approval and Control of Procedures".

10.2.1.1 All changes shall be approved by the ORC prior to implementation.

10.2.2 Cross Disciplinary Review for each revision of the ODCM must include, as a minimum, the Health Physics, Quality Assurance, and Licensing and Fuels Radiological Engineering Departments.

10.2.3 A complete and legible copy of each revision of the ODCM that became effective during the last annual period shall be submitted as a part of, or concurrent with that years Annual Radioactive Effluent Release Report pursuant to Technical Specification 6.14.

11. REFERENCES

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

11.1.1 Statements of Consideration, Federal Register, Vol. 56, No. 98, Tuesday, May 21, 1991, Subpart D, page 23374.

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

11.2.1 10 CFR 50.36 a (b)

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

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

11.4.1 Section 6.8.1

11.4.2 Section 6.8.4f

11.5 COMMUNICATIONS

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

11.5.2 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.

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

11.5.4 Reference Deleted

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

11.5.6 Calculation ZZ-67, "Annual Average Atmospheric Dispersion Parameters", April 1989.

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

11.6.1 Section 11.5.2.2.3.1

11.6.2 Section 11.5.2.2.3.4

11.6.3 Section 11.5.2.1.2

11.6.4 Section 11.5.2.2.3.2

11.6.5 Section 11.5.2.2.3.3

11.6.6 Section 11.2.3.3.4

11.6.7 Section 11.2.3.4.3

11.6.8 Section 11.5.2.3.3.1

11.6.9 Section 11.5.2.3.3.2

11.6.10 Section 11.5.2.3.2.3

11.6.11 Section 11.5.2.3.2.2

11.6.12 Section 2.3.5

11.6.13 Section 2.3.5.2.1.2

11.6.14 Section 9.2.6

11.6.15 Section 9.2.7.2.1

11.6.16	Section 6.3.2.2
11.6.17	Table 11.1-6
11.6.18	Deleted
11.6.19	Deleted
11.6.20	Deleted
11.6.21	Deleted
11.6.22	Table 2.3-68
11.7	Union Electric Company Callaway Plant Environmental Report, Operating License Stage.
11.7.1	Table 2.1-19
11.7.2	Section 2.1.2.3
11.7.3	Section 2.1.3.3.4
11.7.4	Section 5.2.4.1
11.7.5	Table 2.1-19
11.8	U.S. Nuclear Regulatory Commission, Preparation of Radiological Effluent Technical Specification for Nuclear Power Plants", USNRC NUREG-0133, Washington, D. C. 20555, October 1978.
11.8.1	Pages AA-1 through AA-3
11.8.2	Section 5.3.1.3
11.8.3	Section 4.3
11.8.4	Section 5.3.1.5
11.8.5	Section 5.1.1
11.8.6	Section 5.1.2
11.8.7	Section 5.2.1
11.8.8	Section 5.2.1.1
11.8.9	Section 5.3.1
11.8.10	Section 3.8
11.8.11	Section 3.3
11.9	U.S. Nuclear Regulatory Commission, "XOQDOQ, Program For the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations", USNRC NUREG-0324, Washington, D. C. 20555.
11.9.1	Pages 19-20 Subroutine PURGE
11.10	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. D. 20555, July, 1977.
11.10.1	Section c.1.b
11.10.2	Figures 7 through 10
11.10.3	Section c.4

11.11 Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purposes of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, October 1977.

11.11.1 Appendix C, Section 3.a

11.11.2 Appendix E, Table E-15

11.11.3 Appendix C, Section 1

11.11.4 Appendix E, Table E-11

11.11.5 Appendix E, Table E-9

11.12 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.

11.12.1 Section I, Page 2

11.12.2 Section IV, Page 8

11.12.3 Section IV, Page 9

11.12.4 Section III, Page 6

11.12.5 Section III, Page 8

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

11.13.1 Exhibit A

11.14 <u>MISCELLANEOUS REFERENCES</u>

11.14.1 Drawing Number M-109-0007-06, Revision 5

11.14.2 Callaway Plant Annual Environmental Operating Report (updated annually)

11.14.3 UE Safety Analysis Calculation 87-001-00

11.14.4 Calculation ZZ-48, "Calculation of Inhalation and Ingestion Dose Commitment Factors for the Adult and Child", January, 1988

11.14.5 HPCI 89-02, "Calculation of ODCM Dose Commitment Factors", March, 1989

11.14.6 HPCI 87-04, "Calculation of the Limiting Setpoint for the Containment Purge Exhaust Monitors, GT-RE-22 and GT-RE-33", March, 1987

11.14.7 HPCI 88-10, "Methodology for Calculating the Response of Gross NaI(Tl) Monitors to Liquid Effluent Streams", June, 1988

11.14.8 Calculation ZZ-57, "Dose Factors for Eu-154", January, 1989

11.14.9 Calculation ZZ-78, Rev. 2, "ODCM Gaseous Pathway Dose Factors for Adult Age Group", July, 1992.

11.14.10 HPCI 88-08, "Performance Testing of the Environment TLD System at Callaway Plant", August, 1989.

11.14.11 Calculation ZZ-250, Rev. 0, "ODCM Gaseous Pathway Dose Factors for Child Age Group and Ground Plane Dose Factors", September, 1992.

11.14.12 UOTH 83-58, "Documentation of ODCM Dose Factors and Parameters", February, 1983.

11.14.13 Calculation HPCI 95-004 (Rev. 0), "Calculation of Liquid Effluent Dose Commitment factors (A_{it}) for the Adult Age Group", June, 1996.

11.15 U. S. Nuclear Regulatory Commission, "XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations", USNRC NUREG/CR-2919, September 1982, Washington, D. C. 20555

11.15.1 Section 4, "Subroutine PURGE", pages 27 and 28

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

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

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

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

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

11.21 ANSI N13.10-1974 , "Specification & Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents"; September, 1974

11.22 Nuclear Regulatory Commission Generic Letter 89-01, "Guidance for the Implementation of Programmatic Controls for RETS in the Administrative Controls Section of Technical Specifications and the Relocation of Procedural Details of Current RETS to the Offsite Dose Calculation Manual or Process Control Program", January 1989

11.23 NRC Answers to 10 CFR 20 Implementation Questions

11.23.1 Letter, F. J. Congel to J. F. Schmidt, dated December 9, 1991.

11.23.2 Internal USNRC memo, F. J. Congel to V. L. Miller, et al, dated April 17, 1992.

11.23.3 Letter, F. J. Congel to J. F. Schmidt, dated April 23, 1992.

11.23.4 Letter, F. J. Congel to J. F. Schmidt, dated September 14, 1992.

11.23.5 Letter, F. J. Congel to J. F. Schmidt, dated June 8, 1993.

11.24 USNRC Inspection Report 50-483/92002(DRSS) Section 5, page 5.

11.25 HPCI 96-005, "Calculation of Maximum Background Value for HB-RE-18".

11.26 EGG-PHY-9703, "Technical Evaluation Report for the evaluation of ODCM Revison 0 (May, 1990) Callaway Plant, Unit 1", transmitted via letter, Samual J. Collins (USNRC) to D. F. Schnell (UE), dated July 12, 1996.

LOWER LIMIT OF DETECTION (LLD)

A detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Curie, L.A. "Limits for Qualitative Detection and Qualitative Determination - Application to Radiochemistry", Anal. Chem. 40, 586-93 (1986), and Hartwell, J.K., "Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

The LLD is defined, for purposes of these controls, 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 only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation.

$$LLD = \frac{4.56\ S_b}{E \times V \times 2.22E6 \times Y \times \exp(-\lambda\Delta t}$$

Where:

LLD	=	the "a priori" lower limit of detection (microCuries 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 (counts per minute),
E	=	the counting efficiency (counts per disintegration),
V	=	the sample size (units of mass or volume),
2.22E6	=	the number of disintegrations per minute per microCurie,
Y	=	the fractional radiochemical yield, when applicable,
λ	=	the radioactive decay constant for the particular radionuclide (sec^{-1}), and
Δt	=	the elapsed time between the midpoint of the sample collection period, and the time of counting (sec), for effluent samples, or
Δ	=	the elapsed time between the end of the sample collection period, and the time of counting (sec), for environmental samples.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as 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 shall be performed in such a manner that the stated LLD's will be achieved under routine conditions.

The definition of Δt applies only to the calculation of the LLD. A more rigorous treatment of the buildup and decay during the sample collection and/or counting period(s) may be applied to actual sample analysis if desired.

BASES FOR RADIOLOGICAL EFFLUENT CONTROLS

The BASES presented below summarize the reasons for the specified Radiological Effluent Control, but in accordance with 10 CFR 50.36 are not part of these controls.

REC 9.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Refer to FSAR CN #94-51

REC 9.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Refer to FSAR CN #94-51

REC 9.3 LIQUID EFFLUENTS CONCENTRATION

This section is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in Appendix B, Table II, Column 2 to 10 CFR 20.1-20.601. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.1301 to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLD's).

BASES FOR RADIOLOGICAL EFFLUENT CONTROLS

REC 9.4 DOSE FROM LIQUID EFFLUENTS

This section is provided to implement the requirements of Sections II.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable".

Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I which specify that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculations of Annual Doses to Man from Routine Releases of Reactor Effluents with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic and Dispersion of Effluents from accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", April 1977.

REC 9.5 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This section implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

BASES FOR RADIOLOGICAL EFFLUENT CONTROLS

REC 9.6 GASEOUS EFFLUENTS DOSE RATE

This section is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The dose rate limits are the doses associated with the concentrations of 10 CFR Part 20.1-20.601, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the dose limits specified in 10 CFR Part 20 10 CFR 20.1301. For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLD's).

The requirement for additional sampling of the Unit Vent following a reactor power transient is provided to ensure that the licensee is aware of and properly accounts for any increases in the release of gaseous effluents due to spiking which may occur as a result of the power transient. Monitoring the Unit Vent for increased noble gas activity is appropriate because it is the release point for any increased activity which may result from the power transient. Since the escape rate coefficients for the noble gas nuclides is equal to or greater than the escape rate coefficient for iodine and the particulate nuclides[4,5] , it is reasonable to assume that the RCS spiking behavior of the noble gas nuclides is similar to that of the particulate and iodine nuclides. Considering the effects of iodine and particulate partitioning, plateout on plant and ventilation system surfaces, and the 99% efficiency of the Unit Vent HEPA filters and charcoal absorbers, it is reasonable to assume that the relative concentrations of the noble gas nuclides will be much greater than those of the iodine and particulate nuclides. Therefore, an increase in the iodine and particulate RCS activity is not an appropriate indicator of an increase in the Unit Vent activity, and it is appropriate to monitor the Unit Vent effluent activity as opposed to the RCS activity as an indicator of the need to perform post-transient sampling.. In addition, it is appropriate to monitor the noble gas activity due to its relatively greater concentration in the Unit Vent.

---

[4] Cohen, Paul, Water Coolant Technology of Power Reactors, Table 5.19, page 198. American Nuclear Society. 1980.
[5] NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents", Silberberg, M., editor, USNRC; Figure 4.3, page 4.22. June, 1981.

BASES FOR RADIOLOGICAL EFFLUENT CONTROLS

REC 9.7 DOSE - NOBLE GASES

This section is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable".

The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways in unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases on Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors", Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

REC 9.8 DOSE - IODINE-131, & 133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This section is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the release of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as reasonably achievable". The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in 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, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluent sin Routine Releases from Light-Water Cooled Reactors", Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate controls for Iodine-131, and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition of radionuclides onto grassy areas where milk animals and meat-producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

BASES FOR RADIOLOGICAL EFFLUENT CONTROLS

REC 9.9 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This control implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

REC 9.10 TOTAL DOSE

This REC is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20.1301. The control requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and the radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or any organ except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and from outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits.

For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR 20.2203, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in REC 9.3.1.1 and 9.6.1.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

BASES FOR RADIOLOGICAL EFFLUENT CONTROLS

There are three defined effluent release categories: 1.) Releases directly to the hydrosphere; 2.) noble gas releases to the atmosphere; and, 3.) radioiodine and particulate releases to the atmosphere. For each effluent release category, it is assumed in the dose calculations that an individual with the highest dose potential is the receptor. In general, the adult is considered to be the critical age group for liquid effluents, and the child age group is the most limiting for radioiodines and particulates in gaseous effluents. Thus, it is highly unlikely or impossible for the same individual to simultaneously receive the highest dose via all three effluent categories. For most reactor sites, it is also unlikely that all different potential dose pathways would contribute to the dose to a single real individual. Since it is difficult or impossible to continually determine actual food use patterns and critical age groups, for calculational purposes, assumptions are made which tend to maximize doses. Any refinements in the assumptions would have the effect of reducing the estimated dose. For radionuclides released to the hydrosphere, the degree of overestimation in most situations is such that no individual will receive a significant dose. These conservative assumptions generally result in an overestimation of dose by one or two orders of magnitude. Since these assumptions are reflected in the Radiological Effluent Controls limiting radionuclide releases to design objective individual doses, no offsite individual is likely to actually receive a significant dose. Since the doses from 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 need to be considered in determining potential compliance to 40 CFR 190 [1].

REC. 9.11 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this REC provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLD's). The LLD's required by Table 9.11-C are considered optimum for routine environmental measurements in industrial laboratories.

---

[1] NUREG- 0543, "Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR 190)", Congel, F. J., Office of Nuclear Reactor Regulation, USNRC. January, 1980. pp. 5- 8.

BASES FOR RADIOLOGICAL EFFLUENT CONTROLS

REC 9.12 RADIOLOGICAL ENVIRONMENTAL MONITORING LAND USE CENSUS

This REC is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program given in the ODCM are made if required by the results of this census. Information that will provide the best results, such as door-to-door survey, aerial survey, or consulting with local agricultural authorities, shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

REC 9.13 RADIOLOGICAL ENVIRONMENTAL MONITORING INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purpose of Section IV.B.2 of Appendix I to 10 CFR Part 50.

SUMMARY REVIEW OF RADIOLOGICAL EFFLUENT TECH SPECS POTENTIALLY AFFECTED BY THE IMPLEMENTATION OF THE REVISED 10CFR20

The following is a summary review of the current Tech Specs that are potentially affected by the implementation of the revised 10CFR20. In general, the potential impact is due to changes in the Effluent Concentration Values (ECV's) in 10CFR20, Appendix B, Table 2, Columns 1 and 2 (formerly MPC's), and 10CFR20.1601.

This summary is not intended to review those changes that may be necessary as a result of the eventual issuance of the Generic Letter.

The NRR staff has stated that the current level of effluent controls is sufficient to protect the health and safety of the public, and further restrictions resulting from the revision to Appendix B, Table 2, were unintentional. They are currently preparing a Generic Letter that will provide guidance for submitting Tech Spec changes that will return to the current level of control. This is currently anticipated during late 1993. Those who implement the revised rule prior to January 1, 1994, will have to do so under the requirements of 10CFR20.1008, which basically requires that the more restrictive requirement (Tech Specs or 10CFR20) be implemented.

DEFINITIONS OF RESTRICTED AREA & MEMBER OF THE PUBLIC, AND TECH SPEC 5.1.2, SITE BOUNDARY FOR GASEOUS EFFLUENTS

The definition of Restricted Area has not changed significantly from that in the former rule. The definition of the Member of the Public in the revised rule is significantly different from that in the Callaway Plant Technical Specifications (TS 1.17). There is no corresponding definition of Controlled Area in the former rule.

The Callaway Plant was licensed to operate with a Restricted Area as defined in the FSAR and shown on the figures in TS 5.1.4 and in the ODCM. Since the requirements have not been revised, there is no compelling reason to change the Callaway Plant Restricted Area from its current boundaries.

In addition, the NRC's backfit analysis[1], performed pursuant to 10 CFR 50.109, concludes that the revisions to 10CFR20 apply primarily to operational procedures and should cause no modifications in facility design. Since the plant siting and the location and size of the Restricted Area are considered to be a part of the facility design, it is clearly not the intent of the NRC that revisions to 10CFR20 would require changes to the Restricted Area for currently licensed facilities.

There is also no requirement for the existence of a Controlled Area as defined in the revised rule [2], therefore it is not necessary that one be created at Callaway.

---

[1] "Final Backfit Analysis for the Revision of 10CFR20, "Standards for Protection Against Radiation"", USNRC, office of Nuclear Regulatory Research, Division of Regulatory Applications. August, 1990. (Available USNRC Public Documents Review.)

[2] Refer to Question 26(a) (4th set).

The definition of the Member of the Public is significantly different in the revised rule relative to that provided in TS 1.17 and in 40 CFR 190. The revised rule defines the Member of the Public as anyone who is not in the Restricted Area. The Tech Specs and 40 CFR 190 generally define the Member of the Public as anyone who is not occupationally associated with plant operations, and also recognizes that the Member of the Public may, at times, be within the Restricted Area. The major difference is that pursuant to the revised rule, the Member of the Public receives dose against the occupational dose limits of 10 CFR 20.1201 once inside the Restricted Area, but the Tech Spec definition would limit the dose within the Restricted Area to the limits of 10 CFR 20.1301. Since the limit provided in 20.1301 is much lower than that of 20.1201, the continued use of the more restrictive 40 CFR 190 and Tech Spec 1.17 definitions for the Member of the Public is appropriate and is required pursuant to 10 CFR 20.1008(c).

A more thorough and detailed analysis of the definitions of the Member of the Public found in 10 CFR 20, 40 CFR 190, and Tech Spec 1.17, focusing on the applicability of Occupational Vs. Non-occupational dose limits, indicates a confusing and inconsistent array of definitions and dose limit applicability. For conservatism and simplicity, Union Electric has defined occupational dose as dose received while working with or around radioactive materials. This definition is more restrictive than the definition in 10 CFR 20 in that the more restrictive dose limits of 10 CFR 20.1301 are applied to Members of the Public within the Restricted Area, instead of the less restrictive limits of 10 CFR 20.1201. It is more restrictive than the Tech Spec definition in that delivery persons, service technicians, and others who may enter the site to perform non-radiological work activities are also limited to the more restrictive dose limits of 10 CFR 20.1301.

There are no changes recommended for those definitions and maps relative to the Restricted Area, Site Boundary, and dose to the Member of the Public.

TECH SPEC 6.8.4.E.2, LIQUID EFFLUENT RELEASE RATE LIMITS (REC 9.3)

On December 1, 1992, Union Electric Co. provided notification[3] of intent to implement the revised 10 CFR 20, Parts 20.1001- 20.2401 and associated appendices, pursuant to 10 CFR 20.1008(a). The revised rule was fully implemented on January 1, 1993. The following provides clarification with respect to compliance to 10 CFR 20.1001- 20.2401 and Callaway Plant Technical Specifications 6.8.4.e (2) and 6.8.4.e (7).

Union Electric implemented the use of the revised Appendix B, Table 2 values concurrent with the implementation of the revised rule. Technical Specification 6.8.4.e (2) requires that the concentration of radioactive material in liquid discharges not exceed the values of 10 CFR 20, Appendix B, Table II, Column 2. The NRC had indicated via the revision to 10 CFR 50.72 that the concentration values have nominally decreased by a factor of 10, and the NRC staff had stated on numerous occasions that they considered the values in the revised rule to be more restrictive than the those in the old rule. This was frequently referred to as an "implicit" change to the Technical Specifications.

10 CFR 20.1008 (a) requires that if the revised rule is implemented prior to January 1, 1994, then "the licensee shall implement all provisions of these sections,.... and shall provide written notification..... that the licensee is adopting early implementation (of the revised rule) and associated appendices." 10 CFR 20.1008 (b) requires that once implemented, "the applicable section of (the revised rule) shall be used in lieu of any section (of the old rule) that is cited in license conditions or technical specifications." It further states, "if the requirements of (the revised rule) are more restrictive than the existing license condition, then the licensee shall comply with (the revised rule).

[3] ULNRC 92-2729, D. F. Schnell to A. Bert Davis, dated December 1, 1992.

Additionally, the NRC had clarified the applicability of the revised Appendix B values to the Technical Specification instantaneous release rate limits via their formal response to three separate licensee questions. Question # 18 states[4] that the Tech Spec instantaneous release rate limit is based on the old Part 20 concentrations, and asks if changes are required in the Tech Specs and ODCM as a result of the revised rule. The NRC replies " the instantaneous release rates for liquid effluents, to the extent that they directly reference Appendix B concentration values, will need to be changed. The corresponding bases and certain alarm set-points will have to be changed by license amendment."

Question # 23 asks[5] if computer data bases that use the old Appendix B values must be revised to the use the new values. The NRC simply answers, "Yes".

Question # 22 states[6] that many alarm set-points are based on 10 CFR 20 Appendix B concentrations, and asks if they will have to be changed. The NRC answers that the alarm set-points of liquid effluent monitors are likely to require change, since they are based on 10 CFR 20 Appendix B concentrations, as required by Tech Specs. Because Appendix B concentration values differ for many radio nuclides between the old and new versions of Part 20 these set points may have to be changed. This is analogous to a restriction in flow rate, and the NRC cites the reduction in Appendix B concentrations as the root cause of the change.

Based on the preceding information, Union Electric implemented the use of the revised Appendix B values concurrent with the implementation of the revised rule on January 1, 1993. Because there were no values in the revised Appendix B for dissolved and entrained noble gases in liquid effluents, the old value of 2E-4 uCi/ml was used pending regulatory guidance.

The Callaway Plant Technical Specifications contain, in Section 6.8.4.e, several specifications which provide appropriate limits on the maximum quarterly and annual whole body and organ dose to the Member of the Public from the discharge of liquid and gaseous radioactive effluents. Compliance with these specifications demonstrates compliance with the limits of 10 CFR 50, Appendix I, and 40 CFR 190 and, as stated in the supplemental information to the revised rule[7], demonstrates compliance with the 100 mrem/yr dose limit of 10 CFR 20.1301.

However, compliance with the dose rate limits of Specifications 6.8.4.e items (2) and (7) with respect to the implementation of the revised rule is less clear, as there is no longer a regulatory basis for these Specifications. These Specifications formerly implemented the requirements of 10 CFR 20.106, which provided annual average concentration limits on liquid and gaseous effluents, and specifically referenced the limits of Appendix B, Table II, Columns 1 and 2.

---

[4] Letter, F. J. Conjel (USNRC) to J. F. Schmitt (NUMARC), dated December 9, 1991. page 16 of Enclosure 1.

[5] ibid, page 14 of Enclosure 1.

[6] USNRC Memorandum, F. J. Conjel to V. L. Miller, et al, dated April 17, 1992. page 13 of Enclosure 1.

[7] Federal Register, Vol. 58, No. 98, Tuesday, May 21, 1991. pages 23360-23474.

Unlike the former rule, the values in the revised Appendix B, Table 2, Columns 1 and 2 do not of themselves constitute a limit on the release rate of radioactive effluents, but rather, as discussed in 10 CFR 20.1302 (b)(2)(i), merely provide one means of demonstrating compliance with the annual dose limit of 10 CFR 20.1301. Since there is no release rate limit provided in the revised rule, the subject Specifications are therefore license conditions. 10 CFR 20.1008 (c) requires that any existing license condition that is more restrictive than the revised rule remain in force until there is a technical specification change. Additionally, since the values in the revised Appendix B, Table 2 are not limits as was the case with 20.106, there is no corresponding provision in the new rule to 20.106. 10 CFR 20.1008(e) requires that if a license condition cites a provision in the old rule for which there is no corresponding provision in the new rule, then the license condition remains in force until there is a technical specification change.

The values of Appendix B, Table 2, Columns 1 and 2 of the revised rule did not change in a uniform fashion, i.e., certain nuclides numerically decreased in value whereas others numerically increased in value. Furthermore, the values did not change by a consistent amount, varying by as much as a factor of 20 with respect to the corresponding nuclide in the former rule. This inconsistency is clearly evident for those nuclides which are commonly associated with nuclear power plant effluents. In addition, the bases for the revised values is the dosimetry system of ICRP 26 [8] and ICRP 30 [9]. This is inconsistent with the bases for the dose limits of 10 CFR 50, Appendix I and 40 CFR 190, and the dose calculational methodologies of Regulatory Guide 1.109, which are largely based on the dosimetry system of ICRP 2 [10].

Since the values of the revised Appendix B, Table 2, Columns 1 and 2 did not uniformly increase or decrease in value, it is not possible to determine whether Appendix B, Table II of the former rule or Appendix B, Table 2 of the revised rule provides, *in toto,* the more conservative values for implementation of the subject license conditions. It is clear, however, that the bases for the revised Appendix B, Table 2 values are inconsistent with the bases of 10 CFR 50, Appendix I and 40 CFR 190, and Regulatory Guide 1.109. Furthermore, the operational history of the Callaway Plant demonstrates that the use of the 10 CFR 20.1- 20.601, Appendix B, Table II values is appropriate to maintain compliance with the requirements of 10 CFR 50, Appendix I and 40 CFR 190, which, in turn, demonstrates compliance with the 100 mrem/yr dose limit of 10 CFR 20.1301. The concentration limits of the old Appendix B, Table II were based on a dose of 500 mrem/yr, which, when expressed as a dose rate, is equal to .057 mrem/hr. Compliance with the requirements of Technical Specifications 6.8.4.e (2) and (7) using 10CFR 20.106, Appendix B, Table II values is conservative with respect to the 2 mrem/hr limit of 10CFR20.1301(a)(2). Additionally, Technical Specifications 6.4.8.e(2) and (7) specifically require the use of Appendix B, Table II to 10CFR20.1- 20.601, since there is no corresponding provision in the revised rule.

Thus, 10 CFR 20.1008 (c) and (e) require the continued use of the values provided in Appendix B, Table II to 10 CFR 20.1- 20.601 for the implementation of Technical Specifications 6.8.4.e, items (2) and (7).

Although the 2 mrem/hr limit of 10 CFR 20.1301(a)(2) was referenced in the preceding discussion, it is important to note that the regulation specifically states that this limit is applicable to **external** sources. Since, for the Callaway Plant, the only dose pathway to man from the discharge of liquid radioactive effluent is through the consumption of fish, there are no **external** dose pathways, and therefore the requirements of 10 CFR 20.1301(a)(2) are satisfied *apriori.*

---

[8] International Commission on Radiation Protection, Publication 26, "Recommendations of the International Commission on Radiation Protection" , Annals of the ICRP, Volume 1, No. 2, 1977.

[9] International Commission on Radiation Protection, Publication 30, "Limits for Intakes of Radionuclides by Workers", Annals of the ICRP, Volume 2, No. 3/4, 1979.

[10] International Commission on Radiation Protection, Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation", 1960.

Union Electric re instituted the use of the values in Appendix B, Table II, Columns 1 and 2, to 10 CFR 20.1- 20.601 for Technical Specifications 6.8.4.e, items (2) and (7) pursuant to the requirements of 10 CFR 20.1008(c) and (e), on May 4, 1993.

This position was affirmed by the USNRC on June 30, 1993[11].

EFFLUENT CONCENTRATION VALUE FOR GROSS ALPHA IN LIQUID EFFLUENTS

There are two values in the revised Appendix B for unknown mixtures in liquid effluents: 2E-9 and 1E-6 uCi/ml. The less restrictive value is appropriate if it is known that certain nuclides are "not present". The appropriate value for gross alpha in liquid effluents at the Callaway Plant from Appendix B, Table 2, Column 2 is 1E-6 uCi/ml.

The value of 1E-6 uci/ml in Appendix B, Table 2, Column 2 only applies to an unknown mixture of nuclides where those listed opposite the value are known to be "not present". These nuclides are Fe-60, Sr-90, Cd-113m, Cd-113, In-115, I-129, Cs-134, Sm-147, Gd-148, Gd-152, Hg-194 (organic), Bi-210m, Ra-223, Ra-224, Ra-225, Ac-225, Th-228, Th-230, U-233, U-234, U-235, U-236, U-238, U-nat., Cm-242, Cf-248, Es-254, Fm-257, and Md-258. The other nuclides listed in the immediately preceding values for unknown mixtures in gaseous effluents do not apply, since they specifically apply to gaseous effluents as indicated by the designation of applicable lung clearance classifications for each of the nuclides listed. The NRC's response to Question # 71 reiterates that ingestion ALI's do not have lung clearance classifications, which is also consistent with ICRP 30 and all other industry standards. Additionally, several of those listed in the list for liquid effluents also appear in the list of nuclides given for airborne activity, which indicates that only those specifically listed with the liquid effluent value apply.

Of those nuclides listed for unknown mixtures in liquid effluents, only Ra-224, Th-228, U-234, U-235, U-236, U-238, and Cm-242 are LWR produced alpha emitting nuclides. Sr-90, Cd-113m, I-129, and Cs-134 are also LWR produced, but are beta or beta/gamma emitters, and are not determined via a gross alpha analysis. The remainder of the nuclides in the list are not LWR produced.

The phrase "not present" is not defined in the revised 10 CFR 20, however there is a large body of information which can be applied to determine the meaning of "not present". The former rule, in footnote 5 to Appendix B, stated that a nuclide may be considered to be "not present" if it constitutes less than 10% of the total activity, provided that the aggregate of all such "not present" nuclides does not exceed 25% of the total activity. The use of the "ten percent rule" is consistent with the basis of the revised rule, the NRC's response to questions regarding the meaning of "not present", and the current ICRP guidance as shown below:

a. The revised rule is based on the dosimetry and methodology of ICRP 30 [12], which in paragraph 3.1.3, describes the use of the current ten percent rule.
b. The NRC's response to Question # 146 [13] clearly indicates that the ten-percent rule is applicable to Appendix B.

---

[11] Letter, Thomas E. Murley, Director, NRR, USNRC, to Thomas E. Tipton, NUMARC, dated June 30, 1993.

[12] ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers", in Annals of the ICRP, Volume 2, Number 3/4, 1979.

[13] Letter, Frank J. Congel, Director, DRPEP, USNRC, to John F. Schmidt, NUMARC, dated September 14, 1992 (commonly referred to as the 4th set of Q&A)

c. The current ICRP recommendations on the release of radioactive materials to the environment[14], and the updated recommendations[15] to ICRP 30 continue to propagate the ten-percent rule, and apply it to offsite dose as well as dose to radiation workers.

It is therefore clear that the ten-percent rule continues to apply to the values in Appendix B of the revised rule.

Callaway Plant liquid effluents have been analyzed for transuranic nuclides (TRU) on two separate occasions, during the second and third quarters of 1987. In each instance, TRU nuclides were not detectable, with an MDA of 1E-8, uCi/ml, which is a factor of 10 below the gross alpha LLD of 1E-7 uCi/ml.

The concentration of the TRU nuclides can be inferred through the use of a tracer nuclide, such as Ce-144. Ce-144 is particularly well suited for this purpose in that it is a fission product, can be measured by gamma ray spectroscopy, and is chemically similar to the TRU nuclides. Based on published ORIGEN code calculations[16] of a representative LWR, and assuming a 90 day decay, the ratio of the nuclides of interest to Ce-144 is:

Ra-224/Ce-144	1.45E-9
Th-228/Ce-144	1.45E-9
U-234/Ce-144	1.14E-6
U-235/Ce-144	1.75E-8
U-236/Ce-144	2.58E-7
U-238/Ce-144	3.24E-7
Cm-242/Ce-144	2.66E-2

Vollique, et al [17], found the Cm-242/Ce-144 ratio to be 6.5E-3, which is consistent with the above value. Based on the above, it can be seen that Cm-242 is the only nuclide with a significant Ce-144 ratio.

Based on the data contained in the Semiannual Effluent Release Reports for the period January, 1989-July, 1992, Ce 144 accounted for less than 0.3% of the total fission and activation product activity in liquid effluents, and less than 5E-6% of the total activity discharged in liquid effluents during the same period. Therefore, the maximum activity that could have been discharged of each of the above listed nuclides is much less than 10%. Accordingly, these nuclides are "not present".

TECH SPEC 6.8.4.E.4, DOSE FROM LIQUID EFFLUENTS (REC 9.4), & TECH SPEC 6.8.4.E.5, LIQUID RADWASTE TREATMENT SYSTEM (REC 9.5)

These specifications are derived from 10CFR50, Appendix I, and are not affected by the revised rule. Doses are calculated in accordance with Regulatory Guide 1.109 which has not been revised. No changes are anticipated for these specifications.

---

[14] ICRP Publication 56, "Age-dependent doses to Members of the Public from the Intake of Radionuclides: Part 1", Annals of the ICRP, Volume 20, Number 2, 1989.

[15] ICRP Publication 61, "Annual Limits on Intake of Radionuclides by Workers Based on the 1990 Recommendations", Annals of the ICRP, volume 21, Number 4, 1991.

[16] Light Water Reactor Nuclear Fuel Cycle, Wymer, Raymond G. and Vondra, Benedict L., editors, Table 6, pages 70- 71 and Table 7, page 72. CRC Press, 1981.

[17] Vollique, P. G., et al, "Solubility of Transuranic Nuclides in Aerosols in Two Ginna Steam Generator Work Environments". Proceedings of the Twenty-First Midyear Topical Meeting of the Health Physics Society, Pages 251-260. 1987.

FSAR 16.11.1.1, CURIE CONTENT OF OUTDOOR LIQUID STORAGE TANKS

The purpose of this specification is to limit the activity in the nearest receiving waters, excluding tritium and entrained noble gases, to the concentrations in 10CFR20, Appendix B, Table 2, Column 2.

The effect of accidental contamination of the nearest ground water discharge locations due to accidental rupture of tanks containing radioactive liquids was performed as detailed in FSAR Section 2.4.13.3. It was assumed that the liquid contents of a ruptured tank would immediately merge with the ground water 5 feet below plant grade and travel directly from the tank to the nearest down-gradient well (Well 23). The results of the calculation show that, with the exception of H-3 and Sr-90, the radio nuclide concentrations found in ground water after a tank rupture will be below the original 10CFR20, Appendix B, Table II, Column 2 values by the time the contaminated ground water reaches the nearest stream tributaries. The dilution capability of the streams is sufficient to reduce the concentration of H-3 and Sr-90 below the original Appendix B values. All computed concentrations at Well 23 were below the Appendix B limits for unrestricted areas.

Tables I and II list the curie contents of the primary spent resin storage tank and refueling water storage tank used in the FSAR calculations. These values were adjusted to reflect a total tank curie content of 150 Curies, the limit identified in Tech Spec 16.11.1.1 (Even though the spent resin storage tank is not an outdoor tank, the data was used for this calculation since it is expected to have the highest curie contents for Sr-90, Cs-137 and Co-60 and the postulated accident assumes that all liquid released immediately merges with the ground water.) The resultant peak concentrations at the discharge point at Logan Creek were calculated using the normalized values then compared to the revised Appendix B effluent concentration values (ECV). All calculated concentrations at the discharge point were less than the applicable ECV.

Based on the above calculation, the existing Tech Spec limit of 150 Curies is conservative in comparison to the revised 10CFR20, Appendix B values and is therefore still applicable.

TABLE I

A. Curie Content of Radionuclides in the Primary Spent Resin Storage Tank

NUCLIDE	Ci* (in tank)	Ci (normalized to 150 Ci, total)
Mn-54	2.91E+01	8.17E-01
Co-58	6.10E+02	1.71E+01
Co-60	2.56E+02	7.19E+00
Sr-89	9.80E+00	2.75E-01
Sr-90	1.35E+00	3.79E-02
Nb-95	3.00E+00	8.42E-02
Zr-95	2.12E+00	5.95E-02
I-131	1.17E+03	3.28E+01
Cs-134	1.78E+03	5.00E+01
Cs-137	1.48E+03	4.15E+01
Ba-140	1.63E+00	4.58E-02
TOTAL	5.343E+03	149.91

* Values are from FSAR Table 2.4-28.

B. Peak Concentrations of Radionuclides at the Logan Creek Discharge Point

NUCLIDE	μCi/ml* (original calc)	μCi/ml (based on 150 Ci total)	ECV	%ECV
Mn-54	3.1E-22	8.7E-24	3E-05	3E-17%
Co-60	3.6E-23	1.0E-24	3E-06	3E-20%
Sr-90	1.2E-05	3.4E-07	5E-07	67.4%
Cs-137	5.5E-06	1.5E-07	1E-06	15.4%

* Values are from FSAR Table 2.4-30.

TABLE II

A. Curie Content of Radionuclides in Refueling Water Storage Tank

NUCLIDE	Ci* (in tank)	Ci (normalized to 150 Ci, total)
Mn-54	6.99E-06	2.19E-02
Co-58	3.36E-04	1.05E+00
Co-60	4.58E-05	1.43E-01
Sr-89	5.92E-05	1.85E-01
Sr-90	1.92E-06	6.02E-03
Nb-95	1.31E-06	4.10E-03
Zr-95	1.25E-06	3.92E-03
I-131	2.34E-02	7.33E+01
Cs-134	1.39E-02	4.35E+01
Cs-137	1.01E-02	3.16E+01
Ba-140	2.56E-05	8.02E-02
TOTAL	4.788E-02	149.9

* Values are from FSAR Table 2.4-28.

B. Peak Concentrations of Radionuclides at Logan Creek Discharge Point

NUCLIDE	μCi/ml* (original calc)	μCi/ml (based on 150 Ci total)	ECV	%ECV
Co-60	1.1E-30	3.4E-27	3E-06	1E-19%
Sr-90	2.5E-13	7.8E-10	5E-07	0.16%
Cs-137	8.4E-13	2.6E-09	1E-06	0.26%

* Values are from FSAR Table 2.4-30.

TECH SPEC 6.8.4.E.7, DOSE RATE LIMIT FOR GASEOUS EFFLUENTS (REC 9.6)

This specification provides a gaseous effluent dose rate limit conforming to the ECV's in 10CFR20, Appendix B, Table 2, Column 1. For the nuclides of interest to Callaway, the revised ECV's are numerically greater, therefore the current REC is more restrictive than the dose rates conforming to the revised Appendix B values. 10CFR20.1008 requires the implementation of the more restrictive of the requirements of 10CFR20, technical specifications, or any special license conditions. The current REC represents the more restrictive requirement and will be implemented without revision.

The former rule, in 20.106(a), limited the amount of radioactivity released in effluents to the concentrations specified in Appendix B, Table 2, averaged over a period of one year. Although not specified as a limit, this corresponded to an annual whole body dose limit of 500 mRem to the Member of the Public. The former rule did not specify a dose rate limit.

The revised rule, in 20.1301, specifies two limits on radioactivity in effluents: An annual dose limit of 100 mRem, TEDE (20.1301(a)(1)) and a dose rate limit of 2 mRem/h TEDE (20.1301(a)(2)). Note that the revised rule does not specify limits on concentration as did the former rule but does allow licensees to utilize the concentration values in Appendix B, Table 2 to demonstrate compliance with the limits of 20.1301 (20.1302(b)(2)). Note that 20.1302(b)(2)(i) describes these as "annual average concentrations" as opposed to instantaneous limits. Measurements and calculation means are also allowed (20.1302(b)(1)).

Radiological Effluent Control (REC) 9.6 is required by Technical Specification 6.8.4.e.7 to contain:

> "Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10CFR20, Appendix B, Table 2, Column 1."

The bases for this Control state that its purpose is to ensure that the dose at any time from gaseous effluents is within the annual dose limit of 10CFR20, which is the dose associated with the concentrations of 10CFR20, Appendix B, Table 2, Column 1. Additionally, this Control provides assurance that the release of gaseous effluents will not result in the exposure of a Member of the Public to annual average concentrations in excess of the values of 10CFR20, Appendix B, Table 2, Column 1. Note that in each case, the bases references an annual dose limit but makes no reference to a dose rate limit.

The REC establishes a release rate limit of 500 mRem/y that is equal to approximately 0.06 mRem/h, well below the dose rate limit of 2 mRem/h specified in 20.1301(a)(2), and is therefore more restrictive.

The preamble to the revised rule states that demonstration of compliance with the limits of 40CFR190 and with 10CFR50, Appendix I is sufficient to demonstrate compliance with the 100 mRem dose limit of 20.1301(a)(1). Other Controls are provided as required Technical Specification 6.8.4.e (items 8,9, and 10) which ensure that the limits of 40CFR190 and 10CFR50, Appendix I are not exceeded.

The Bases for this Control reference the concentration values of 10CFR20, Appendix B, Table 2, Column 1 as a basis for the specified dose rate limits. These values were derived using ICRP 30 calculation methodology and the dose and dose rate values they represent are the Total Effective Dose Equivalent (TEDE) which is the summation of the external and internal dose components. Compliance with the Control is demonstrated through calculation methodologies and parameters as established in Regulatory Guide 1.109 and NUREG 0133, which are based on the ICRP 2 maximum organ methodology, and thus cannot be used to calculate effluent doses and dose rates that correspond to the concentration values specified in the revised 10CFR20, Appendix B, Table 2, Column 1.

The table below compares the numerical value of the former and revised Appendix B values for those nuclides most commonly reported in the Callaway Plant's gaseous effluents:

10CFR20 APPENDIX B CONCENTRATION VALUES

Nuclide	Former Rule	Revised Rule	New/Old
Kr-85	3E-7 μCi/ml	7E-7 μCi/ml	2.3
Xe-133	3E-7	5E-7	1.7
Xe-135	1E-7	7E-8	0.7
I-131	1E-10	2E-10	2.0
I-133	4E-10	1E-9	2.5
Co-58	2E-9	1E-9	0.5
Co-60	3E-10	5E-11	0.2

Of these, Xe-133 accounts for greater than 90% of the total activity released from the Callaway Plant in gaseous effluents for the past three years (1989-1991). The concentration value for Xe-133 actually increased in the revise rule, as did that for Kr-85 and both iodine nuclides. Although the Co-58 and Co-60 values did decrease in the revised rule, they are relatively insignificant contributors to the whole body and organ dose from gaseous effluents discharged from the Callaway Plant as summarized below.

GASEOUS EFFLUENT ACTIVITY PROFILE 1989 - 1991

Nuclide	Fraction of Total Activity Released	Ratio of Appendix B Concentration Values
Noble Gases:		
Xe-133	0.92	1.7
Xe-135	0.04	0.7
Xe-133m	0.01	2.0
Kr-85m	0.01	1.0
Kr-85	0.01	2.3
Particulates and Iodines:		
I-131	0.72	2.0
I-133	0.11	2.5
Co-58	0.03	0.5
Co-60	0.14	0.2

The NRC states, in their response to Question 19, that until 10CFR50, Appendix I is changed, licensees must continue to show compliance with Tech Specs in terms of organ and whole body doses as per Regulatory Guide 1.109. The response to Question 21 states that Regulatory Guide 1.109 will not be revised at this time, thus Regulatory Guide 1.109 methodology continues to be utilized to show compliance with Tech Specs. [illegible] the dose calculation methodology has not been revised, it would be more conservative to continue to utiliz[illegible] current REC values vice dose rate limits calculated from the revised 10CFR20, Appendix B values.

Refer to the discussion of T/S 6.8.4.e.2 (REC 9.3) for additional details.

TECH SPEC 6.8.4.E.6,	GASEOUS RADWASTE TREATMENT SYSTEM OPERABILITY (REC 9.9)
TECH SPEC 6.8.4.E.8,	DOSE FROM NOBLE GASES (REC 9.7)
TECH SPEC 6.8.4.E.9,	DOSE FROM IODINES AND PARTICULATES IN GASEOUS EFFLUENTS (REC 9.8)
TECH SPEC 6.8.4.E.10,	TOTAL DOSE FROM THE URANIUM FUEL CYCLE (REC 9.10)

These specifications are derived from 10CFR50, Appendix I and 40CFR190 and are not affected by the revised rule. Doses continue to be calculated in accordance with Regulatory Guide 1.109 which has not been revised. No changes are anticipated for these specifications.