DOE/OR/01033--T785

ANALYSIS OF OFFSITE DOSE CALCULATION METHODOLOGY FOR A NUCLEAR POWER REACTOR

by

Donna Smith Moser

A thesis submitted to the faculty of The University of North Carolina at Chapel Hill in partial fulfillment of the requirements for the degree of Masters of Sciences in the Department of Environmental Sciences and Engineering.

Chapel Hill

1995

Approved by:

James E. Watson

James E. Watson Jr., Ph.D.

Douglas Crawford-Brown, Ph.D.

Donald A. Tyndall, Ph.D.

DISCLAIMER

DONNA SMITH MOSER. Analysis of Offsite Dose Calculation Methodology for a Nuclear Power Reactor. (Under the direction of Dr. James E. Watson)

ABSTRACT

This technical study reviews the methodology for calculating offsite dose estimates as described in the offsite dose calculation manual (ODCM) for Pennsylvania Power and Light - Susquehanna Steam Electric Station (SSES). An evaluation of the SSES ODCM dose assessment methodology indicates that it conforms with methodology accepted by the US Nuclear Regulatory Commission (NRC). Using 1993 SSES effluent data, dose estimates are calculated according to SSES ODCM methodology and compared to the dose estimates calculated by PP&L. The results of this comparison reveal differences between the written methodology of SSES ODCM and the computer model used to produce the reported 1993 dose estimates. The 1993 SSES dose estimates are based on the axioms of Publication 2 of the International Commission of Radiological Protection (ICRP). SSES Dose estimates based on the axioms of ICRP Publication 26 and 30 reveal the total body dose estimates to be the most affected.

ACKNOWLEDGMENTS

First, I would like to thank the members of my committee for their assistance and guidance throughout this project. I greatly appreciate Dr. James Watson for his guidance, graciousness and tactful comments throughout this project and my graduate studies. I would also like to thank Dr. Don Tyndall for his technical and editorial suggestions. Furthermore, thanks to Dr. Crawford-Brown for his expertise and assistance in this project in regards to internal dose models.

I am grateful to Robert Barclay of Pennsylvania Power and Light for his patience and thoroughness in answering all of my questions and for responding rapidly to my many requests and inquires. Appreciation is also expressed to the US Nuclear Regulatory Commission for the opportunity to obtain a Master of Science Degree and to Thomas Richmond and Rose Etta Cox of Oak Ridge Institute of Science and Education for their assistance in administrating the US NRC Fellowship*.

I am particularly grateful to Ms. Emma York for her suggestions, comments, and tolerance with some of the details of this thesis. I am fortunate to have a colleague and friend as wonderful as Ms. York.

Special gratitude is expressed to my family for their support and patience, especially to my husband Eric for his constant encouragement and love. Above all things, thanks to God for His abundant love and the blessings given to me.

*This research was performed under appointment to the Nuclear Regulatory Commission Fellowship Program administered by Oak Ridge Institute for Science and Education for the U. S. Nuclear Regulatory Commission.

TABLE OF CONTENTS

LIST OF TABLES vii

LIST OF ABBREVIATIONS x

I. INTRODUCTION 1

- A. Regulations Governing Radioactive Effluents 1
- B. Definition and Purpose of ODCM 2
- C. Focus of Technical Review 3

II. NRC GUIDANCE 5

- A. 10 CFR - Appendix I 5
- B. Regulatory Guide 1.109 6
- C. NUREG-0133 7
- D. GASPAR and LADTAP Computer Dose Models 7

III. PP&L OFFSITE DOSE ASSESSMENT METHODOLOGY 9

- A. Individual Dose Calculations Due to Waterborne Effluents 9
    1. Potable Water Pathway Dose Equations 11
    2. Fish Ingestion Pathway Dose Equations 13
    3. Shoreline Exposure Pathway Dose Equations 15
- B. Individual Dose Calculations Due to Airborne Effluents 18
    1. Gamma and Beta Air Dose Equations for Noble Gas Exposure 20
    2. Total Body Dose Equations for Noble Gases Exposure 24
    3. Skin Dose Equations for Noble Gas Exposure 28
    4. Organ Dose Equations for Exposure to Airborne Radioactive Iodine and Particulate Material 32
        - a. The Inhalation Pathway Dose Factor 35
        - b. The Ground Plane Pathway Factor 36
        - c. The Grass-Cow-Milk Pathway Dose Factor 38
        - d. The Grass-Goat-Milk Pathway Dose Factor 40

e. The Grass-Cow-Meat Pathway Dose Factor 42
f. The Vegetation Pathway Dose Factor 44

IV. CALCULATION OF DOSE ESTIMATES FROM 1993 EFFLUENT DATA 47

A. 1993 Dose Estimates.. 48
B. Individual Dose Estimates Due to Waterborne Effluents..................... 50

1. Potable Water Pathway Dose Estimate 52
2. Fish Ingestion Pathway Dose Estimate 53
3. Shore Exposure Pathway Dose Estimate 55
4. Total Dose Estimate from Liquid Effluents............................ 56

C. Individual Dose Estimates Due to Airborne Effluents........................ 58

1. Gamma and Beta Air Dose Estimates from Noble Gas Exposure 64
2. Total Body Dose Estimate from Noble Gas Exposure................ 65
3. Skin Dose Estimate from Noble Gas Exposure........................ 67
4. Organ Dose Estimate from Airborne Iodine And Particulate Material Effluents.. 68

D. Summary of Manually Calculated Dose Estimates............................ 75

V. IDENTIFICATION OF KEY PARAMETERS AND ASSUMPTIONS............ 78

VI. CHANGES IN INTERNAL DOSE FACTORS...................................... 88

A. Differences Between Internal Dose Models of ICRP 2 and ICRP 30 89

1. Differences in Dosimetric Models of ICRP 2 and ICRP 30......... 90

a. ICRP 2 Dosimetric Models..................................... 90
b. ICRP 30 Dosimetric Models 94

2. Differences in Absorbed Effective Energy of ICRP 2 and ICRP 30 95

a. ICRP 2 Absorbed Effective Energy 95
b. ICRP 30 Absorbed Effective Energy........................... 96

3. Differences Between ICRP 2 and ICRP 30 in regards to the Attribution of Risk to Tissues of the Body 97

B. 1993 SSES Dose Estimate Based on ICRP 30 Dose Equivalents........... 100

1. Total Body Dose Estimate from Liquid Effluents..................... 104
2. Organ Dose Estimate from Liquid Effluents........................... 107
3. Total Body Dose Estimate from Noble Gas Exposure................ 109
4. Skin Dose Estimate from Noble Gases Exposure 111
5. Organ Dose Estimate from Iodine and Particulate Material Effluents.. 112
6. Summary of ICRP 30 Dose Calculations.............................. 117

VII. CONCLUSION 120

VIII. REFERENCES... 123

IV. APPENDICES 126

A. Examples of Dose Factors Given in SSES ODCM 126
B. Mixing Characteristics of The Susquehanna River and Atmospheric Dispersion Factors for SSES.. 131
C. Calculated Dose Estimates from Liquid Effluents 136
D. ICRP 30 Dosimetric Models for the Gastrointestinal Tract and Respiratory System.. 146

LIST OF TABLES

Table 2-1. Summary of Numerical Guides in 10 CFR 50 Appendix I 5

Table 3-1. Shore Width Factors 17

Table 3-2. Dose Factors For Exposure To A Semi-Finite Cloud Of Noble Gases 20

Table 3-3. Equation (29) Parameter Values for Mn-54 Exposure of Teen Kidney Via Grass-Cow-Milk Pathway 39

Table 3-4. Equation (30) Parameter Values for Mn-54 Exposure of Teen Kidney Via Grass-Goat-Milk Pathway 41

Table 3-5. Equation (31) Parameter Values for Mn-54 Exposure of Teen Kidney Via Grass-Cow-Meat Pathway 43

Table 3-6. Equation (32) Parameter Values for Mn-54 Exposure of Teen Kidney Via Vegetation Pathway 45

Table 4-1. Summary of Maximum Individual Offsite Dose Commitments to Members of the Public: 1/1/93 to 12/31/93 48

Table 4-2. Site Specific Parameter Used in Liquid Effluent Calculations 51

Table 4-3. Radioactive Airborne Effluents Released in 1993 59

Table 4-4. Atmospheric Dispersion Factors Used to Calculate the 1993 Dose Estimates from Airborne Radionuclide 63

Table 4-5. Teen lung Dose Factors for Inhalation, Ingestion and Ground Exposure Pathways from Radioactive Iodine and Particulate Material 70

Table 4-6. Teen Lung Dose Estimate from Exposure to Iodine and Particulate Material 71

Table 4-7. Child Lung Dose Estimate from Exposure to Iodine and Particulate Material 73

Table 4-8. Teen and Child Lung Dose Estimates Recalculate Using Tritium Dose Factors Based on a Quality Factor of 1.0 74

Table 4-9. Summary of Hand-Calculated Dose Estimates....75

Table 4-10. Summary of Revised Hand-Calculated Dose Estimates....77

Table 5-1. Percent of Total Dose from Each Radionuclide Specific to Pathway....79

Table 5-2. Maximum, Minimum and Nominal Site-Specific Values for Liquid Effluent Pathways....82

Table 5-3. Sensitivity Analysis of the Site-specific Parameters for the Liquid Pathway....83

Table 5-4. Observed Dilution of Peak Concentrations Released in Blowdown Line....85

Table 6-1. Quality Factors Recommended in ICRP 26....97

Table 6-2. Weighting Factors Recommended in ICRP 26....99

Table 6-3. Summary of Require Dose Estimates, Pathways Analyzed and Reg. Guide 1.109 Applicable Dose Factors....101

Table 6-4. Summary of Maximum Adult Offsite Dose Commitments to Members of the Public: 1/1/93 to 12/31/93....103

Table 6-5. Weighted CDEs for Ingestion of Mn-54....105

Table 6-6. Ingestion Dose Equivalents of ICRP 30 and the Adult Ingestion Dose Factors of Reg. Guide 1.109 For Total Body....106

Table 6-7. Adult-Total Body Dose Based on ICRP 30 Dose Equivalents....107

Table 6-8. ICRP 30 Ingestion CDEs and Reg. Guide 1.109 Ingestion Dose Factors for Adult Lower Large Intestine....108

Table 6-9. Adult GI-LLI Dose Based on ICRP 30 Dose Equivalents....109

Table 6-10. The Weighted Dose Equivalent Rates Given in ICRP 30 for Xenon-133....110

Table 6-11. Inhalation and Ingestion CDEs of ICRP 30 and Dose Factors of Reg. Guide 1.109 for the Lung of Maximum Adult....113

Table 6-12. Adult Lung Dose Based on ICRP 30 Dose Equivalents....115

Table 6-13. Recalculation of 1993 SSES Adult Lung Dose Estimate Using Tritium Dose Factors Based on a Quality Factor of 1.0 116

Table 6-14. Summary Of Dose Calculations Using ICRP Dose Equivalents 117

LIST OF ABBREVIATIONS

ADF	atmospheric dispersion factor
AEWDR	Annual Effluent and Waste Disposal Report
CDE	committed dose equivalent
DF	dose factor
GI-LLI	lower large intestine of gastrointestinal tract
ICRP	International Commission for Radiological Protection
I&PM	iodine and particulate materials
LCO	limiting conditions for operation
ODCM	Offsite Dose Calculation Manual
PP&L	Pennsylvania Power and Light
SSES	Susquehanna Steam Electric Station

I. INTRODUCTION

A. Regulations Governing Radioactive Effluents

Of primary concern in nuclear power reactor operation is the health and safety of the public and the environment. Protection of the health and safety of the public in the operation of nuclear reactors is augmented through the implementation of federal regulations administered under the auspices of the United States Nuclear Regulatory Commission (NRC). Title 10 part 50 of the Code of Federal Regulations (10 CFR 50), "Domestic Licensing of Production and Utilization Facilities," contains the federal regulations governing the production and utilization of nuclear energy (U.S. NRC 1993). As required by 10 CFR 50, each operating license for a nuclear power reactor must contain written technical specifications that set forth safety limits, limiting conditions for operations (LCO), and other regulatory requirements on facility operation for the protection of the health and safety of the public. Sections 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Materials in Effluents," and 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," require that each operation license for a nuclear power reactor include design objectives and technical specifications that are designed "to keep releases of radioactive materials to unrestricted areas during normal reactor operations . . . as low as reasonably achievable (ALARA)" (U.S. NRC 1993). To assist licensees in meeting the ALARA requirements of 10 CFR 50 sections 34a and 36a, numerical guides for design objectives and LCOs are provided in 10 CFR 50 - Appendix I. Compliance with the design objectives and LCOs of 10 CFR 50 - Appendix I "shall be demonstrated by calculational procedures based upon models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated" (U.S. NRC 1993). The methodology and parameters used in the calculation of offsite doses due to radioactive liquid and gaseous effluents are contained in a report referred to as an Offsite Dose Calculation Manual (ODCM) (U.S. NRC 1978a).

B. Definition and Purpose of ODCM

Radiological dose assessment is a quantitative process that characterizes the relationship between the environmental release of radioactive effluents and the potential effects on human health. The dose assessment process is a structured process that maps the progression of a radionuclide from its point of release to the environment, through various environmental pathways, resulting ultimately in exposure to man. Numerous mathematical models are required to describe this environmental transportation process and to describe the uptake, retention, and elimination of consumed radioactivity within the body for the purpose of estimating dose. The information and values used to quantify the modeling process from source term release to human exposure are referred to as model parameters.

The SSES ODCM describes the methodology and provides the model parameters used to calculate offsite doses due to radioactive liquid and airborne effluents. The SSES ODCM also contains the methodology and parameters needed to calculate the alarm/trip setpoints of the monitoring instrumentation for liquid and airborne effluents (PP&L 1993a). ODCMs are submitted with the site-specific technical specifications as part of the license application. If any changes are made to an ODCM, the revised ODCM is included in the Annual Effluent and Waste Disposal Report (AEWDR) (U.S. NRC 1978a). The AEWDR, required for NRC review, provides a summary of the radioactive liquid and gaseous effluents released during the year and concurrent meteorological conditions that are necessary for dose assessment calculations. In addition the AEWDR includes a summary of the offsite individual and population dose estimates for the calendar year (PP&L 1993a).

C. Focus of Technical Review

The purpose of this technical study is to review the methodology for estimating individual radiation exposure due to the release of radioactive materials in liquid and airborne effluents as describe in the ODCM for the Pennsylvania Power and Light (PP&L) Susquehanna Steam Electric Station (SSES). Review of the dose methodology set forth in the SSES ODCM is accomplished by the following:

1) Comparing the dose methodology of the SSES ODCM with the methodology currently accepted by the NRC for calculating individual dose estimates due to the release of effluent radioactive materials, and
2) Manually calculating the 1993 maximum individual offsite doses according to the dose assessment methodology given in the SSES ODCM and comparing these hand calculated dose estimates to the dose estimates reported in the 1993 AEWDR.

In addition, this review provides a limited examination of the sensitivity of site-specific parameters used to calculate the reported 1993 individual dose estimates. The site-specific parameters that demonstrate the most influence on the resulting dose estimates will be identified and their derivation will be discussed.

The internal dose factors accepted by the NRC to calculate the annual dose estimates from nuclear power reactors are primarily founded on the axioms and dose models given in Publication 2 of the International Commission on Radiological Protection (ICRP), Report of Committee II on Permissible Dose of Internal Radiation, (ICRP 1959 and U.S. NRC 1977b). This review also examines how the 1993 SSES dose estimates might change if they were calculated using internal dose factors derived from the more recent dose models of ICRP Publication 26, Recommendations of the International Commission on Radiological Protection, and Publication 30, Limits for Intakes of Radionuclides by Workers, (ICRP 1977 and 1978).

II. NRC GUIDANCE

A. 10 CFR 50 - Appendix I

10 CFR 50 sets forth the general requirements for a nuclear power reactor licensee application. Sections 34a and 36a of 10 CFR 50 require that the license application include design objectives and technical specifications to assure that released radioactive effluents are kept ALARA. To assist licensees and license applicants in meeting the ALARA requirements of sections 34a and 36a, numerical guides in the form of dose limits are given in 10 CFR 50 - Appendix I (U.S. NRC 1993). The quantities of radioactive liquid or airborne effluents released from each nuclear power reactor unit are limited to that amount that will not result in estimated annual doses in excess of the dose limits given in 10 CFR 50 - Appendix I. The dose limits given in 10 CFR 50 - Appendix I are specific to effluent type: liquid, gaseous, and airborne iodine and particulate materials (I&PM). The numerical guides given in 10 CFR 50, Appendix I per effluent type are outlined in Table 2-1 (U.S. NRC 1993).

Table 2-1. Summary of Numerical Guides in 10 CFR 50 Appendix I

Effluent	Target	Annual Dose Limit
Liquid	Total Body	3 mrem
Liquid	Organ	10 mrem
Gaseous	Gamma air-dose	10 mrad
Gaseous	Beta air-dose	20 mrad
Gaseous	Total Body	5 mrem
Gaseous	Skin	15 mrem
Iodine & Particulate	Organ	15 mrem

B. Regulatory Guide 1.109

A series of Regulatory Guides has been developed by the NRC to assist licensees and license applicants implement the numerical guidance of 10 CFR 50 - Appendix I. These regulatory guides provide "methods acceptable to the staff for the calculation of preoperational estimates of effluent releases, dispersion of the effluent in the atmosphere and different water bodies, and estimation of the associated radiation doses to man" (U.S. NRC 1977a). The fundamental federal guidance document for the calculation of offsite dose estimates is Regulatory Guide 1.109 (Reg. 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 (U.S. NRC 1977a). Reg. Guide 1.109 presents calculational models and suggested parameters for the calculation of annual radiation dose estimates from radioactive effluents.

In providing guidance to implement the design objective of 10 CFR 50 - Appendix I for liquid and airborne I&PM effluents, the NRC staff uses a maximum exposed individual approach. In this approach, the population surrounding nuclear facilities is divided into four age groups: adult, teen, child and infant (U.S. NRC 1977a). Each age group is represented by the maximum exposed individual of that age group. In Reg. Guide 1.109, the maximum exposed individual for the infant age group "is assumed to be a newborn, the maximum child is taken to be 4 years old, the maximum teen is taken to be 14 years old and the maximum adult is taken to be 17 years old" (U.S. NRC 1977a). Reg. Guide 1.109 contains precalculated internal dose factors specific to age group, radionuclide, target organ and method of uptake (i.e. inhalation and ingestion) that are used to estimate individual dose per unit radioactivity consumed. Anatomical data specific to the maximum exposed individual of each age group are used to derive the internal dose factors given in Reg. Guide 1.109. The required dose estimates of 10 CFR 50 - Appendix I for liquid and airborne I&PM effluents are calculated for each age group using the dose factors given in Reg. Guide 1.109 (U.S. NRC 1977c). The annual dose estimates reported in the AEWDR for liquid and airborne I&PM effluents releases are the highest calculated dose estimates of the four age groups. It should be noted that the maximum exposed individual approach is not used to estimate dose from noble gas exposure. Furthermore, noble gas dose estimates are calculated assuming exposure occurs at the site boundary (U.S. NRC 1978b).

C. NUREG-0133

NUREG-0133, Preparation of Radiological Effluent Technical Specification for Nuclear Power Plants, describes the methods acceptable to the NRC for the calculation of dose factors required for the preparation of radiological effluent technical specifications (U.S. NRC 1978a). In addition, NUREG-0133 includes methodology for "estimating radiation exposure due to the release of radioactive materials in effluents" (U.S. NRC 1978a). Both Reg. Guide 1.109 and NUREG-0133 serve as the primary benchmarks of this review.

D. GASPAR and LADTAP Computer Dose Models

Calculating the dose that an individual receives from reactor effluents requires numerous parameters and mathematical computations. In response to the numerous calculations that are necessary to perform a thorough radiological dose assessment, various computer modeling codes have been developed to calculate dose estimates. The computer modeling programs GASPAR and LADTAP II are used by PP&L to calculate dose estimates for SSES from airborne and liquid effluents (PP&L 1993a). The

GASPAR computer dose model calculates dose estimates resulting from exposure to noble gas effluents and airborne radioactive I&PM effluents (U.S. NRC 1987). The LADTAP II computer dose model calculates dose estimates resulting from exposure to liquid effluents (U.S. NRC 1986). Both GASPAR and LADTAP II were developed by Pacific Northwest Laboratories for the NRC. The dose methodology given in Reg. Guide 1.109 forms the basis for both GASPAR and LADTAP II computer dose models (U.S. NRC 1986 and 1987). In addition to Reg. Guide 1.109 and NUREG-0133, the user guides for both computer models were consulted in the review of the SSES ODCM.

III. PP&L OFFSITE DOSE ASSESSMENT METHODOLOGY

The dose assessment methodology detailed in the SSES ODCM follows the methodology and structure given in Reg. Guide 1.109 and NUREG-0133. The dose methodology given in the SSES ODCM is separated into individual dose due to waterborne effluents and individual dose due to airborne effluents. This section presents the dose methodology given in the SSES ODCM in comparison to the dose methodologies given in Reg. Guide 1.109 and NUREG-0133. Dose assessment methodology is first discussed for waterborne effluents and then for airborne effluents.

A. Individual Dose Calculations Due to Waterborne Effluents

Appendix A of Reg. Guide 1.109 outlines the methods acceptable to the NRC for calculating doses to man from liquid effluent pathways. Reg. Guide 1.109 first presents a general equation for calculating exposure to man from liquid effluents and then illustrates how this general equation is developed into equations specific to four principal exposure pathways in the aquatic environment: potable water ingestion, fish ingestion, shore exposure, and irrigated foods) (U.S NRC 1977c). The general equation given in Appendix A of Reg. Guide 1.109 (U.S. NRC 1977c) for calculating radiation dose due to radioactive liquid effluents is the following:

$$R_{aipj} = C_{ip} \times U_{ap} \times D_{aipj} \qquad (1)$$

Where

R_{aipj} = annual dose commitment to organ j of an individual of age group a from nuclide i via pathway p, in units of mrem/yr;

C_{ip} = concentration of nuclide i in the media of pathway p, in units of pCi/l, pCi/kg or pCi/m^3;

U_{ap} = exposure time or intake rate (usage) of the liquid effluent associated with pathway p for age group a, in hr/yr, l/yr or kg/yr (as appropriate); and

D_{aipj} = dose factor, specific to age group a, radionuclide i, pathway p, and organ j. It represents the dose due to the intake of a radionuclide in mrem/pCi, or from exposure to a given concentration of a radionuclide in sediment, in mrem-m^2/pCi-hr.

Reg. Guide 1.109 contains ingestion dose factors for 73 radionuclides that are specific to seven critical organs and four different age groups: adult, teen, child, and infant. In addition, Reg. Guide 1.109 provides dose factors for external exposure to a contaminated ground surface in units of mrem-m^2/pCi-hr. The dose factors given in Reg. Guide 1.109 for standing on contaminated ground are used to calculate skin and total body dose from external exposure to radioactivity that is deposited on a ground plane surface such as shore exposure from liquid effluents (U.S. NRC 1977a).

From this general equation, equations specific to different aquatic exposure pathways are developed. The different exposure pathways in the aquatic environment are ingestion of potable water, fish ingestion, shoreline exposure and foods irrigated with radioactive contaminated water. The SSES ODCM contains calculational procedures to estimate the dose contributions from the fish ingestion, potable water and shore exposure pathways. Because the irrigation pathway is not considered to exist at the SSES facility, calculational procedures to estimate dose from the irrigation pathway are not included in the SSES ODCM (PP&L 1993b).

1. Potable Water Pathway Dose Equations

The following equation is given in Reg. Guide 1.109 (U.S. NRC 1977a) to calculate the annual dose estimate from ingestion of potable water:

$$R_{apj} = 1100 \times \frac{M_p \times U_{ap}}{F} \times \sum_i Q_i \times D_{aipj} \times \exp(-\lambda_i t_p) \qquad (2)$$

Where

M_p = mixing ratio (reciprocal of the dilution factor) at the point of exposure (or the point of withdrawal of drinking water), dimensionless;

U_{ap} = usage factor that specifies the drinking water intake rate for an individual of age group *a* associated with the potable water pathway, in units of kg/yr or l/yr;

F = liquid effluent flow rate, in ft^3/sec;

Q_i = release rate of the nuclide *i*, in Ci/yr;

D_{aipj} = dose factor, specific to a age group *a*, radionuclide *i*, pathway *p* and organ *j*, in mrem/pCi;

λ_i = radioactive decay constant of nuclide *i*, in hr^{-1};

t_p = average transit time required for nuclides to reach the point of exposure, in hrs; and

1100 = factor to convert from (Ci/yr)/(ft^3/sec) to pCi/l;

(i.e., 1 Ci-sec/yr-ft^3 $\times$ 1E 12 pCi/Ci $\times$ 3.17×10^{-8} yr/sec $\times$ 0.03531 ft^3/l $\approx$ 1100 pCi/l).

The difference between equation (2), which is specific to the potable water pathway, and equation (1) is the expression $[1100 \times Q_i \times M_p / F \times \exp(-\lambda_i t_p)]$ which yields the concentration of nuclide i at the time the water is consumed, in pCi/l. This term expressing concentration is equivalent to the C_{ip} term of equation (1).

The following equation is given in the SSES ODCM (PP&L 1993a) to calculate the dose contribution from liquid effluents via the potable water pathway:

$$R_{aipj} = \sum_i \left[\frac{K_{aipj} \times \exp(-\lambda_i \times t_p) \times C_i \times V_i \times k}{DF_p \times F} \right] \tag{3}$$

The parameters of this equation that are similar to the potable water equation of Reg. Guide 1.109 are the decay due to transit time, $\exp[-\lambda_i \times t_p]$ (unitless), the minimum blowdown or flow rate of the liquid effluent F in ft^3/sec, and the dilution factor DF_p which is the reciprocal of the mixing ratio M_p (unitless). The parameters C_i, V_i, and k are the average concentration of radionuclide i in Ci/ml, the volume of radionuclide i released in gallons, and a conversion factor of 3.785×10^3 ml/gallon, respectively. Multiplying the C_i, V_i, and k parameters yields the radioactivity in curies that is released during a period of one year which is equivalent to the Q_i term of equation (2).

The potable water ingestion dose factor K_{aipj} of equation (3) differs from the D_{aipj} dose factor of Reg. Guide 1.109 in that it incorporates the usage factor U_{ap} of equation (2). As a result, the K_{aipj} dose factor is expressed in units of mrem-ft^3/Ci-sec instead of mrem/pCi. The potable water ingestion dose factors K_{aipj} used in equation (3) are given in Tables 5-2 a, b, c, and d of the SSES ODCM. A table of potable water dose factors taken from the SSES ODCM for the adult age group is included as Table A-1 in Appendix A of this review. The following computation illustrates the conversion from the Reg. Guide 1.109 dose factor for adult total body from ingestion of 1 pCi of tritium to the corresponding potable water dose factor of the SSES ODCM:

1.05×10^{-7} mrem/pCi	(Reg. Guide 1.109 dose factor)
$\times$ 730 l/yr	(average adult water usage)
$\times$ 0.03531 ft^3/l	(conversion factor from liters to cubic feet of water)
$\times 1 \times 10^{12}$ pCi/Ci	(conversion from picocuries^{-1} to curies^{-1})
$\times 3.17 \times 10^{-8}$ yr/sec	(conversion from years^{-1} to seconds^{-1})
$= 8.58 \times 10^{-2}$ mrem-ft^3/Ci-sec	

This number approximates the adult dose factor given in the SSES ODCM for potable water ingestion (i.e., 8.43×10^{-2} mrem-ft^3/pCi-sec) (PP&L 1993a).

2. Fish Ingestion Pathway Dose Equations

The following equation is given in Reg. Guide 1.109 (U.S. NRC 1977a) to calculate the annual dose estimate from the fish ingestion pathway:

$$R_{apj} = 1100 \frac{M_p \ U_{ap}}{F} \sum_i \left[Q_i \ B_{ip} \ D_{aipj} \ \exp(-\lambda_i t_p) \right] \tag{4}$$

With the exception of the bioaccumulation factor B_{ip}, all of the parameters have been defined previously. The bioaccumulation factor is the ratio of the radionuclide concentration in biota to the radionuclide concentration in water, expressed in liters per kilogram (pCi/kg in biota per pCi/l in water). The usage factor U_{ap} of equation (4) specifies the intake rate of aquatic foods for an individual of age group *a* in units of kg/yr.

The following equation is given in the SSES ODCM (PP&L 1993a) to calculate the dose contribution from liquid effluents via the fish ingestion pathway:

$$R_{aipj} = \sum_{i} \left[\frac{K_{aipj} \times C_i \times V_i \times k}{F} \right] \tag{5}$$

Similar to the potable water equation, the $C_i \times V_i \times k$ expression and the minimum blowdown or flow rate F of equation (5) are equivalent to the Q_i and F term of equation (4). The K_{aipj} parameter is the fish ingestion dose factor specific to radionuclide *i*, organ *j*, and age group *a*. Fish ingestion dose factors can be found in Tables 5-1 a, b, and c of the SSES ODCM. A table of fish ingestion dose factors taken from the SSES ODCM for the adult age group is included as Table A-2 in Appendix A of this review. The fish ingestion dose factor is expressed in units of mrem-ft^3/Ci-sec. The fish ingestion dose factor K_{aipj} is a combination of the ingestion dose factors D_{aipj} given in Reg. Guide 1.109, the usage value U_{ap}, the dilution factor DF_p, the decay due to transition time, $\exp(-\lambda_i \times t_p)$, and the bioaccumulation factor B_{ip}. The following computation illustrates the conversion from Reg. Guide 1.109 dose factor for adult total body from the ingestion of 1 pCi of tritium to the corresponding fish ingestion dose factor of the SSES ODCM:

1.05×10^{-7} mrem/pCi	(Reg. Guide 1.109 ingestion dose factor)
$\times$ 21 kg/yr	(average adult fish consumption)
$\div$ 15.9	(dilution factor)
$\times \exp[-6.44 \times 10^{-6}\ \text{hr}^{-1} \times 25\ \text{hr}]$	(radioactive decay due to transit time)
$\times\ 9.0 \times 10^{-1}$ l/kg	(bioaccumulation factor)

$\times\ 3.531 \times 10^{-2}$ ft^3/l (conversion from liters to cubic feet)

$\times\ 1 \times 10^{12}$ pCi/Ci (conversion from picocuries^{-1} to curies^{-1})

$\times\ 3.17 \times 10^{-8}$ yr/sec (conversion from years^{-1} to seconds^{-1})

$= 1.40 \times 10^{-4}$ mrem-ft^3/pCi-sec

This value approximates the adult dose factor given in the SSES ODCM for fish ingestion (i.e., 1.37×10^{-4} mrem-ft^3/pCi-sec) (PP&L 1993a).

3. Shoreline Exposure Pathway Dose Equations

The following equation is given in Reg. Guide 1.109 (U.S. NRC 1977a) to calculate the annual dose estimate from exposure to shoreline contaminated with radioactive materials from waterborne effluents:

$$R_{apj} = 110{,}000 \frac{U_{ap} M_p W}{F} \sum_i Q_i T_i D_{aipj} [\exp(-\lambda_i t_p)][1 - \exp(-\lambda_i t_b)] \qquad (6)$$

All of the parameters except for the shore-width factor W and the expression for accumulation of radioactivity in the soil, "$1.44 \times T_i \times [1\text{-}\exp(-\lambda_i \times t_b)]$", have been previously defined. The t_b term represents "the length of time the sediment is exposed to the contaminated water, nominally 15 years (approximate midpoint of facility operating life), in hours"(U.S. NRC 1977a). The 110,000 factor includes the 1100 conversion factor of equation (2) and (4) used in converting from (Ci/yr)/(ft^3/sec) to pCi/l, multiplied by the 1.44 value from the accumulation of radioactivity in soil expression, a transfer proportionality constant K_C, equal to 0.072 l/kg-hr, and appropriate conversion factors. The factor of 110,000 is derived as follows (U.S. NRC 1977a):

$$110{,}000 \approx 1100 \times K_C \times 1380$$

Where

$1100 =$ factor to convert from (Ci/yr)/(ft^3/sec) to pCi/l; (i.e., 1 Ci-sec/yr-ft^3 $\times$ 1E 12 pCi/Ci $\times$ 3.17 $\times$ 10^{-8} yr/sec $\times$ 0.03531 ft^3/l $\approx$ 1100 pCi/l);

$K_C =$ 0.072 l/kg-hr, an assumed transfer constant from water to sediment; and

$1380 \approx$ 40 kg/m^2 $\times$ 24 hr/day $\times$ 1.44 days

The transfer proportionality constant K_C accounts for the transfer of radioactivity from water to sediment. The value of K_C was empirically derived for several radionuclides from water and sediment samples taken over several years from the Columbia River in Washington and from the Tillamook Bay in Oregon (U.S. NRC 1977a). The mass per unit surface area is given as 40 kg/m^2.

The usage factor U_{ap} in equation (6) represents the exposure time to contaminated soil for an individual of age group *a* in units of hours per year. In equation (6), D_{aipj} is the external dose factor for standing on contaminated ground which is used to calculate the radiation dose from exposure to a given concentration of a radionuclide in sediment. External dose factors for standing on contaminated ground, expressed as a ratio of the dose rate (in mrem/hr) and the areal radionuclide concentration (in pCi/m^2), are given in Table E-6 of Reg. Guide 1.109 (U.S. NRC 1977a).

The shore-width factors *W* are derived from experimental data and "represent the fraction of the dose from an infinite plane source that is estimated for [specific] shoreline situations" (i.e., lake shoreline, river shoreline, etc.) (U.S. NRC 1977a).

Table 3-1 summarizes the derived shore width factors given in Appendix A of Reg. Guide 1.109 (U.S. NRC 1977a).

Table 3-1. Shore Width Factors

Exposure Situation	Shore Width Factor
Discharge canal bank	0.1
River Shoreline	0.2
Lake shore	0.3
Nominal ocean site	0.5
Tidal basin	1.0

Since the primary body of water near the Susquehanna Steam Electric Station is the Susquehanna River, the appropriate exposure situation is the river shoreline with a corresponding shore width factor of 0.2.

The following equation is given in the SSES ODCM (PP&L 1993a) to calculate the dose contribution from liquid effluents via the shore exposure pathway:

$$R_{aipj} = \sum_{i} \left[\frac{K_{aipj} \times C_i \times V_i \times k}{F} \right] \tag{7}$$

As before, the $C_i \times V_i \times k$ expression yields the amount of radionuclide i released in a year which is analogous to the Q_i term of equation (6). The flow rate or blowdown parameter F is also the same in this equation as in equation (6). The dose factor for the shore exposure pathway K_{aipj} of equation (7) equals the Reg. Guide 1.109 external dose factor for standing on contaminated ground D_{aipj} multiplied by the usage factor U_{ap}, the reciprocal of the dilution factor DF (i.e., the mixing ratio, M_p), the shore width factor W, the radiological half-life T_i, the transit decay expression, $\exp(-\lambda_i t_p)$, the expression for the accumulation of radioactivity in the soil, $(1 - \exp(-\lambda_i t_b))$, and the 110,000 conversion factor. As previously stated, the 110,000 factor incorporates the

1100 conversion factor used in converting from (Ci/yr)/(ft^3/sec) to pCi/l and the transfer proportionality constant K_C. A table of shore exposure dose factors taken from the SSES ODCM is included as Table A-3 in Appendix A of this review. The following computation illustrates the conversion from the Reg. Guide 1.109 external dose factor for standing on ground contaminated with sodium-24 to the corresponding adult dose factor for shore exposure that is given in the SSES ODCM:

2.50×10^{-8} mrem/pCi	(Reg. Guide 1.109 external dose factor)
$\times$ 12 hr/yr	(average adult exposure time to shore)
$\div$ 15.9	(dilution factor)
$\times$ 0.2	(width factor)
$\times \exp[-4.62 \times 10^{-2}\ hr^{-1} \times 1\ hr]$	(radioactive decay due to transit time)
$\times$ 0.625 days	(half-life of Na-24)
$\times [1-\exp(-4.62 \times 10^{-2} s \times 131400 s)]$	(radioactivity accumulation in soil)
$\times$ 100 l/m^2-day	(transfer constant)
$\times 3.531 \times 10^{-2}$ ft^3/l	(conversion from liters to cubic feet)
$\times 1 \times 10^{12}$ pCi/Ci	(conversion from picocuries^{-1} to curies^{-1})
$\times 3.17 \times 10^{-8}$ yr/sec	(conversion from years^{-1} to seconds^{-1})
$= 2.52 \times 10^{-4}$ mrem-ft^3/pCi-sec	

This value approximates the adult dose factor given in the SSES ODCM for shore exposure (i.e., 2.48×10^{-4} mrem-ft^3/pCi-sec) (PP&L 1993a).

B. Individual Dose Calculations Due to Airborne Effluents

As previously stated, 10 CFR 50 - Appendix I provides numerical guides in the form of dose limits to assist nuclear power reactor licensees in meeting the ALARA requirements of 10 CFR 50 for effluent releases. The dose limits given in Appendix I for airborne effluents are specific to gaseous effluents such as noble gases and radioactive iodine and particulate material (I&PM) effluents. Four dose limits are given in 10 CFR 50 - Appendix I that restrict the release of noble gas effluents to quantities that will not exceed (U.S. NRC 1993):

1) an annual air dose estimate of 10 mrads from noble gas gamma radiation per reactor unit;
2) an annual air dose estimate of 20 mrads from noble gas beta radiation per reactor unit;
3) an annual total body dose estimate of 5 mrem per reactor unit; and
4) an annual skin dose estimate of 15 mrem per reactor unit.

The dose limit given in 10 CFR 50 - Appendix I for airborne radioactive I&PM effluents limits the quantity released to that amount that will not result in an annual estimated dose of 15 mrem to any organ (U.S. NRC 1993).

Reg. Guide 1.109 and NUREG-0133 provide the methodology accepted by the NRC for calculating dose estimates from exposure to radioactive gaseous effluents and airborne I&PM effluents (U.S. NRC 1977a). The methodology given in the SSES ODCM to calculate individual dose estimates from airborne effluents will be presented in comparison with the methodology outlined in Reg. Guide 1.109 and NUREG-0133.

Before presenting the dose assessment methodology for exposure to noble gases, it is necessary to introduce the dose factors given in Reg. Guide 1.109 that are used to estimate dose from exposure to noble gas effluents. Table 3-2 lists the dose factors given in Reg. Guide 1.109 for exposure to a semi-infinite cloud of noble gases (U.S. NRC 1977a). All guidance documents presented in this review and the SSES ODCM incorporate the dose factors of Table 3-2 in their dose methodology.

Table 3-2. Dose Factors For Exposure To A Semi-Finite Cloud Of Noble Gases

Nuclide	β-Air ($DF\beta_i$)	β-Skin (DFS_i)	γ-Air ($DF\gamma_i$)	γ-Body (DFB_i)
Units	mrad·m³/pCi·yr	mrem·m³/pCi·yr	mrad·m³/pCi·yr	mrem·m³/pCi·yr
Kr-83m	2.88×10^{-4}	----	1.93×10^{-5}	7.56×10^{-8}
Kr-85m	1.97×10^{-3}	1.46×10^{-3}	1.23×10^{-3}	1.17×10^{-3}
Kr-85	1.95×10^{-3}	1.34×10^{-3}	1.72×10^{-5}	1.61×10^{-5}
Kr-87	1.03×10^{-2}	9.73×10^{-3}	6.17×10^{-3}	5.92×10^{-3}
Kr-88	2.93×10^{-3}	2.37×10^{-3}	1.52×10^{-2}	1.47×10^{-2}
Kr-89	1.06×10^{-2}	1.01×10^{-2}	1.73×10^{-2}	1.66×10^{-2}
Kr-90	7.83×10^{-3}	7.29×10^{-3}	1.63×10^{-2}	1.56×10^{-2}
Xe-131m	1.11×10^{-3}	4.76×10^{-4}	1.56×10^{-4}	9.15×10^{-5}
Xe-133m	1.48×10^{-3}	9.94×10^{-4}	3.27×10^{-4}	2.51×10^{-4}
Xe-133	1.05×10^{-3}	3.06×10^{-4}	3.53×10^{-4}	2.94×10^{-4}
Xe-135m	7.39×10^{-4}	7.11×10^{-4}	3.36×10^{-3}	3.12×10^{-3}
Xe-135	2.46×10^{-3}	1.86×10^{-3}	1.92×10^{-3}	1.81×10^{-3}
Xe-137	1.27×10^{-2}	1.22×10^{-2}	1.51×10^{-3}	1.42×10^{-3}
Xe-138	4.75×10^{-3}	4.13×10^{-3}	9.21×10^{-3}	8.83×10^{-3}
Ar-41	3.28×10^{-3}	2.69×10^{-3}	9.30×10^{-3}	8.84×10^{-3}

The β-air and the γ-air dose factors, $DF\beta_i$ and $DF\gamma_i$, given in Table 3-2 are used to calculate the air dose due to beta radiation and the air dose due to gamma radiation. The γ-Body dose factor DFB_i is used to calculate total body dose estimates from noble gas exposure. Equations estimating dose to the skin from exposure to noble gas use both the β-Skin dose factor DFS_i and the γ-air dose factor $DF\gamma_i$ (U.S. NRC 1977a).

1. Gamma and Beta Air Dose Equations for Noble Gas Exposure

The following equations are given in Reg. Guide 1.109 (U.S. NRC 1977a) to calculate the annual gamma air dose and the annual beta air dose estimates from noble gas effluents:

$$D^{\gamma}(r,\theta) = 3.17 \times 10^4 \sum_{i} Q_i \, [X/Q]^D(r,\theta) \;\; DF^{\gamma}{}_i \tag{8}$$

$$D^{\beta}(r,\theta) = 3.17 \times 10^4 \sum_{i} Q_i \, [X/Q]^D(r,\theta) \;\; DF^{\beta}{}_i \tag{9}$$

Where

$D^{\gamma}(r,\theta)$ and $D^{\beta}(r,\theta)$ = annual gamma and beta doses, respectively, in sector θ at distance r from the discharge point, in mrad/yr;

3.17×10^4 = number of pCi per Ci divided by the number of seconds per year;

Q_i = release rate of the radionuclide i, in Ci/yr;

$[X/Q]^D(r,\theta)$ = annual average gaseous dispersion factor at distance r in sector θ, in sec/m^3; and

$DF^{\gamma}{}_i$ and $DF^{\beta}{}_i$ = gamma and beta air dose factors for a uniform semi-infinite cloud of radionuclide i, in mrad-m^3/pCi-yr.

The following equations are given in NUREG-0133 (U.S. NRC 1978a) to calculate the annual gamma air dose and the annual beta air dose from noble gas effluents:

$$3.17 \times 10^{-8} \sum_{i} \; [M_i[(X/Q_v)Q_{iv} + (x/q)_v q_{iv}] + [B_i Q_{is} + b_i q_{is}]] \tag{10}$$

$$3.17 \times 10^{-8} \sum_{i} \; N_i[(X/Q_v)Q_{iv} + (x/q)_v q_{iv} + (X/Q_s)Q_{is} + (x/q)_s q_{is}] \tag{11}$$

Where

M_i and N_i = air dose factor due to gamma emissions and beta emissions, respectively, for each identified noble gas radionuclide, in mrad/yr per μCi/m^3;

$(X/Q)_v$ and $(X/Q)_s$ = highest calculated annual average relative concentration for any area at or beyond the unrestricted area boundary for long term releases, in sec/m^3;

$(x/q)_v$ and $(x/q)_s$ = relative concentration for areas at or beyond the unrestricted area boundary for short term releases, in sec/m^3;

Q_{iv} and Q_{is} = cumulative release of noble gas radionuclide *i*, in gaseous effluent per calendar year or quarter, for long term releases (greater than 500 hr/yr), in uCi;

q_{iv} and q_{is} = cumulative release of noble gas radionuclide *i*, in gaseous effluents per calendar year or quarter, for short term releases, in uCi;

B_i and b_i = constant for long term releases and short term releases, respectively, for each identified noble gas radionuclide accounting for the gamma radiation from the elevated finite plume, in mrad/yr per μCi/sec; and

3.17×10^{-2} = inverse of the number of seconds in a year.

In equations (10) and (11), the *v* subscript denotes vent releases, whereas the *s* subscript denotes releases from free-standing stacks. All gaseous effluents released from SSES are released from vents on top of the reactor and turbine buildings. NUREG-0133 states "if there are no free-standing stacks, the factors denoted by the subscript *s* need not be considered" (U.S. NRC 1978a). Therefore, the parameters that pertain to free-standing stacks are eliminated from equations (10) and (11). Also, the parameters specific to short term releases (i.e., releases less than 500 hours per year) are eliminated since all airborne releases at the SSES facility are continuous throughout the year (PP&L 1993a). Eliminating the unnecessary parameters reduces equations (10) and (11) to:

$$3.17 \times 10^{-8} \sum_i M_i (X/Q_v) Q_{iv} \tag{12}$$

$$3.17 \times 10^{-8} \sum_i N_i (X/Q_v) Q_{iv} \tag{13}$$

The following equations are given in the SSES ODCM (PP&L 1993a) to calculate the annual gamma air dose and beta air dose due to noble gas effluents:

$$D_g = 3.17 \times 10^{-8} \sum_i M_i (X/Q_v) Q_{iv} \tag{14}$$

$$D_b = 3.17 \times 10^{-8} \sum_i N_i (X/Q_v) Q_{iv} \tag{15}$$

The parameters of equations (14) and (15) are defined the same as the parameters of equations (12) and (13). Although the notation and parameter definitions differ slightly, the equations given in Reg. Guide 1.109, NUREG-0133, and PP&L ODCM to calculate gamma and beta air dose estimates from exposure to noble gases are fundamentally the same. The M_i and N_i parameters of equations (14) and (15) equal the $DF\gamma_i$ and $DF\beta_i$ parameters of equation (8) and (9) multiplied by 10^6 to convert from picocuries^{-1} to microcuries^{-1}.

The Reg. Guide $[X/Q]^D(r,\theta)$ parameter and the $(X/Q)_V$ parameter given in NUREG-0133 and SSES characterize the diffusion of radioactive effluents by atmospheric dispersion and physical removal mechanisms. Reg. Guide 1.109 refers to these parameters as dispersion factors; NUREG-0133 refers to them as annual average relative concentrations (U.S. NRC 1977a and 1978a). The 50 mile radius surrounding SSES is divided into 16 sectors distinguish by compass notation (i.e., North, Southwest, etc.). Atmospheric dispersion factors are calculated for each sector specific to a distance r from the initial release point. Both $[X/Q]^D(r,\theta)$ and $(X/Q)_V$ parameters represent the sector θ and the distance r from the initial release point with the highest calculated atmospheric dispersion factor. A later section of this review will address atmospheric dispersion factors in more detail.

2. Total Body Dose Equations for Noble Gases Exposure

The following equation is given in Reg. Guide 1.109 (U.S. NRC 1977a) to calculate the annual total body dose estimate from exposure to a semi-infinite cloud of noble gases:

$$D^T_\infty(r,\theta) = 3.17 \times 10^4 \; S_F \sum_i \; Q_i \, [X/Q]^D(r,\theta) \; DFB_i \qquad (16)$$

Where

$D^T_\infty(r,\theta)$ = annual total body dose factor due to immersion in a semi-infinite cloud at distance r in sector θ, in mrem/yr;

S_F = attenuation factor that accounts for the dose reduction due to shielding provided by residential structures (dimensionless);

DFB_i = total body dose factor for a semi-infinite cloud of radionuclide i due to gamma emissions which includes the attenuation of 5 g/cm^2 of tissue, in mrem-m^3/pCi-yr;

$[X/Q]^D(r,\theta)$ = annual average gaseous dispersion factor in sector θ at distance r from the release point, in sec/m^3;

Q_i = release rate of nuclide i, in Ci/yr; and

3.17×10^4 = number of pCi per Ci divided by the number of seconds per year.

The following equation is given in NUREG-0133 (U.S. NRC 1978a) to estimate the annual total body dose from exposure to noble gas effluents:

$$\sum_i \left[V_i Q_{is} + K_i ((X/Q)_v Q_{iv}) \right] \tag{17}$$

Where

V_i = constant for each identified noble gas radionuclide accounting for gamma radiation from elevated finite plume, in mrem/yr per uCi/sec;

Q_{is} and Q_{iv} = release rate of radionuclide i in gaseous effluents, in uCi/sec;

K_i = total body dose factor due to gamma emissions for noble gas radionuclide i, in mrem-m^3/uCi-yr; and

$(X/Q)_V$ = highest calculated annual average relative concentration for any area at or beyond the unrestricted area boundary, in sec/m^3.

In equation (17), the subscript s denotes releases from free-standing stacks and the v subscript denotes vent releases. As previously stated, all parameters denoted by the subscript s may be eliminated from equation (17) since all airborne effluents from SSES are released from vents on top of the reactor and turbine buildings. Therefore, equation (17) may be reduced to:

$$\sum_{i} \left[K_i \left((X/Q)_V \, Q_{iv}\right)\right] \qquad (18)$$

The K_i parameter of equation (18) equals the DFB_i parameter of equation (16) multiplied by 10^6 to convert from picocuries^{-1} to microcuries^{-1} and the $(X/Q)_V$ of parameter equation (18) equals the $[X/Q]^D(r,\theta)$ parameter of equation (16). Since the Q_{iv} parameter of equation (18) is defined as microcuries per second and the K_i parameter is in units mrem-m^3/uCi-yr, the 3.17×10^4 factor of equation (16) used in converting years^{-1} to seconds^{-1} and picocuries^{-1} to microcuries^{-1} is not needed in equation (18).

A discrepancy exists between equation (16), which is the equation given in Reg. Guide 1.109, and equation (17), the corresponding equation given in NUREG-0133. Equation (16) includes a structural shielding factor S_F that accounts for the shielding effects of gamma radiation such as the shielding provided by buildings during indoor residence (U.S. NRC 1977a). In contrast, equation (17) does not include a shielding factor. The approach taken in conservative dose assessment models assumes that a hypothetical man continually stands on a smooth infinite plane immersed in a cloud of noble gas. Residence time inside buildings will normally result in a reduced external dose rate compared to an external dose rate in an outdoor environment with no shielding. A shielding factor, defined as "the ratio of dose rate inside a building to the corresponding dose rate outdoors", is used in dose calculations to account for dose reduction due to shielding (U.S. NRC 1983). A structural shielding factor of 0.7 is recommended in Reg. Guide 1.109 (U.S. NRC 1977a). A structural shielding factor is not acknowledged in the equation given in NUREG-0133 for exposure to noble gases (U.S. NRC 1978a). The omission of the structural shielding factor from the equation given in NUREG-0133 is a conservative approach. Similar to Reg. Guide 1.109, the GASPAR computer dose model also utilizes a 0.7 shielding factor to estimate total body dose from noble gas exposure (U.S. NRC 1987). A later section of this review illustrating the manual recalculations of the 1993 reported dose estimates will demonstrate that the GASPAR computer program incorporates a 0.7 shielding factor in its dose model.

The following equation is given in the SSES ODCM (PP&L 1993a) to calculate the total body dose due to noble gas exposure:

$$D_g = 3.17 \times 10^{-8} \sum_{i} K_i \times (X/Q)_V \times Q_{iV} \qquad (19)$$

The K_i and $(X/Q)_V$ parameters of the equation (19) are defined the same as the K_i and $(X/Q)_V$ parameters of equations (17) and (18). The release rate parameter Q_{iV} of equation (19) is the radioactivity in curies released in one calendar year in lieu of curies per second, therefore a 3.17×10^{-8} year/sec conversion factor is needed in equation (19).

In contrast to Reg. Guide 1.109, the equation given in the SSES ODCM to calculate total body dose from noble gas exposure does not include a 0.7 shielding factor. Since the GASPAR computer dose model employed by PP&L uses a 0.7 shielding factor to calculate total body dose from noble gas exposure, the equation given in the SSES ODCM does not accurately describe the methodology used by PP&L to calculate the annual total body dose estimate for noble gas effluents reported in the AEWDR. It is important to note that this difference has been corrected. As of January 20, 1995 per ODCM Revision 3, the equation given in the SSES ODCM to estimate total body dose from noble gas exposure has been revised to include a 0.7 shielding factor.[1]

3. Skin Dose Equations for Noble Gas Exposure

The following equation is given in Reg. Guide 1.109 (U.S. NRC 1977a) to calculate the annual skin dose from noble gas exposure:

$$D^S_\infty(r,\theta) = 3.17 \times 10^4 \sum_i \left[(DFS_i + S_F \times 1.11 \times DF^\gamma_i) \times [X/Q]^D(r,\theta) \times Q_i\right] \quad (20)$$

Where

$D^S_\infty(r,\theta)$ = annual skin dose factor due to immersion in a semi-infinite cloud a distance r in sector θ, in mrem/yr;

DFS_i = beta skin dose factor for a semi-infinite cloud of radionuclide i which includes the attenuation by the outer 'dead' layer of the skin, in mrem-m^3/pCi-yr;

[1]Personal communication with Robert Barclay of Pennsylvania Power & Light Company, 18 May, 1995.

S_F = attenuation factor that accounts for the dose reduction due to shielding provided by residential structures (dimensionless);

1.11 = average ratio of tissue to air energy absorption coefficient;

DF^{γ}_i = gamma dose factor for a uniform semi-infinite cloud of radionuclide i, in mrad-m^3/pCi-yr;

$[X/Q]^D(r,\theta)$ = annual average gaseous dispersion factor in the sector at angle θ at the distance r from the release point, in sec/m^3;

Q_i = initial release rate of nuclide i, in Ci/yr; and

3.17×10^4 = number of pCi per Ci divided by the number of seconds per year.

The following equation is given in NUREG-0133 (U.S. NRC 1978a) to estimate annual skin dose from noble gas exposure:

$$\sum_i \left[(L_i\,(X/Q)_s + 1.1B_i)\,Q_{is} + (L_i + 1.1\,M_i)\,(X/Q)_v Q_{iv}\right] \tag{21}$$

Where

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

$(X/Q)_s$ and $(X/Q)_v$ = highest calculated annual average relative concentration for any area at or beyond the unrestricted area boundary in sec/m^3. The subscript s denotes release from free-standing stacks, the v subscript denotes vent releases;

B_i = constant for long term releases for each identified noble gas radionuclide counting for the gamma radiation from the elevated finite plume, in mrad/yr per μCi/m^3;

Q_{is} and Q_{iv} = release rate of radionuclides i in gaseous effluents, in uCi/sec. The subscript s denotes releases from free standing stack, the v subscript denotes vent releases; and

M_i = air dose factor due to gamma emissions for each identified noble gas radionuclide, in mrem-m^3/uCi-yr.

Eliminating those parameters of equation (21) that are associated with releases from free-standing stacks (i.e., subscript s) reduces equation (21) to:

$$\sum_{i} \left[(L_i + 1.1M_i)\,(X/Q)_v\,Q_{iv}\right] \qquad (22)$$

The L_i and M_i parameters of equation (22) are equivalent to the $DF\gamma_i$ and DFS_i parameters of equation (20) multiplied by 10^6 to convert from picocuries^{-1} to microcuries^{-1}. The $(X/Q)_v$ and Q_{iv} parameters of equation (22) are also analogous to the $[X/Q]^D(r,\theta)$ and Q_i parameters of equation (20).

Again, a discrepancy exists between equation (20), which is the equation given in Reg. Guide 1.109, and equation (22) which is derived from the equation given in NUREG-0133. Equation (20) incorporates a shielding factor multiplied by the gamma air dose factor to account for shielding of gamma radiation during indoor residence; however, equation (22) does not include a structural shielding factor.

The following equation is given in the SSES ODCM (PP&L 1993a) to estimate the annual skin dose due to noble gas exposure:

$$D_b = 3.17 \times 10^{-8} \, (L_i + 1.1M_i) \, (X/Q)_v \, Q_{iv} \tag{23}$$

The L_i, M_i, and $(X/Q)_v$, parameters of equation (23) are defined the same as the L_i, M_i, and $(X/Q)_v$ parameters of equations (21) and (22). The radioactive release parameter Q_{iv} of equation (23) is defined as the amount of radioactive material released per year; therefore, a 3.17×10^{-8} factor is included in equation (23) to convert years^{-1} to seconds^{-1}.

The equations given in the SSES ODCM to estimate dose from noble gas exposure are closely modeled after the noble gas exposure equations given in NUREG-0133. Although NUREG-0133 is an appropriate guidance document for PP&L to derive its methodology, the dose methodology given in NUREG-0133 to calculate skin dose from noble gas exposure is not consistent with Reg. Guide 1.109 or the GASPAR computer dose model employed by PP&L. Consistent with Reg. Guide 1.109, the GASPAR dose model uses a 0.7 structural shielding factor multiplied by the gamma-air dose factor to calculate skin dose from noble gas exposure. In contrast, the skin dose equation for noble gas exposure given in the SSES ODCM dose not include a 0.7 structural shielding factor. This difference between the GASPAR computer model and the noble gas skin dose equation given in the SSES ODCM has been corrected by PP&L. As of January 20, 1995 per ODCM Revision 3, the equation given in the SSES ODCM to estimate skin dose from noble gas exposure has been revised to include a 0.7 shielding factor multiplied by the gamma-air dose factor.[2]

[2]Personal communication with Robert Barclay of Pennsylvania Power & Light Company, 18 May, 1995.

4. Organ Dose Equations for Exposure to Airborne Radioactive Iodine and Particulate Material

The methodology for assessing dose to an individual from exposure to airborne radioactive I&PM effluents is presented in a simpler format in NUREG-0133 than in Reg. Guide 1.109. Furthermore, the dose methodology given in the SSES ODCM for exposure to airborne I&PM effluents directly follows the methodology given in NUREG-0133. Therefore the equations given in NUREG-0133 are presented in lieu of Reg. Guide 1.109 equations for comparison with the methodology given in the SSES ODCM.

Six different environmental pathways are considered by PP&L to calculate the dose an individual receives from exposure to radioactive I&PM airborne effluents:

1) inhalation,
2) ground plane exposure,
3) grass - cow - milk,
4) grass - cow - meat,
5) grass - goat - milk, and
6) vegetation.

Both NUREG-0133 and the SSES ODCM present a general equation that is used to estimate exposure from airborne radioactive I&PM effluents for all pathways. A precalculated dose factor R_{ip}, contained in the general equation, distinguishes the general equation for each pathway. The dose factor R_{ip} is derived by multiplying the appropriate dose factor given in Reg. Guide 1.109 by parameters that are characteristic of each airborne pathway. Equations are given in NUREG-0133 to derive R_{ip} dose factors specific to pathway, radionuclide, age group, and organ (U.S. NRC 1978a).

The R_{ip} dose factors used to estimate individual dose from airborne radioactive I&PM effluents are contained in Table 4 of the SSES ODCM (PP&L 1993a). The derivation of the dose factor R_{ip} will be illustrated below for each pathway.

The following equation given in NUREG-0133 (U.S. NRC 1978a) is used to calculate dose to an individual from airborne radioactive I&PM effluents:

$$3.17 \times 10^{-8} \sum_{ip} [R_{ip} (W_v Q_{iv} + w_v q_{iv})] \qquad (24)$$

Where

3.17×10^{-8} = inverse of the number of seconds in a year;

R_{ip} = dose factor for each identified radionuclide i specific to pathway p in mrem-m^2-sec per uCi-yr or mrem-m^3 per uCi-yr;

W_v and w_v = dispersion parameter for estimating the dose to an individual at the controlling location:

$W = (X/Q)$ for the inhalation pathway in sec/m^3 ,or

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

Q_{iv} and q_{iv} = release rate of radionuclides i, radioactive materials in particulate form, and radionuclides other than noble gases in gaseous effluents, in μCi;

The parameters denoted with small case notation, (i.e., w_v and q_v), signify short term releases equal to or less than 500 hours per year; the parameters denoted with upper case notation signify long term releases greater than 500 hours per year. The release of airborne radioactive effluents from SSES are continuous throughout the year, therefore, the small case parameters do not apply. Eliminating the parameters that are specific to short term releases reduces equation (24) to:

$$3.17 \times 10^{-8} \sum_{ip} [R_{ip}\, W_v\, Q_{iv}] \qquad (25)$$

The following equation is given in the SSES ODCM to calculate dose to an individual from airborne radioactive I&PM effluents:

$$D_c = 3.17 \times 10^{-8} \sum_{ip} [R_{ip}\, W_v\, Q_{iv}] \qquad (26)$$

The R_{ip}, W_v, and Q_{iv} parameters of equation (26) are defined the same as the R_{ip}, W_v, and Q_{iv} parameters of equation (25).

Regardless of the pathway, the radionuclide quantity released Q_{iv} for each radionuclide remains the same. The atmospheric dispersion factor W_v and the dose factor R_{ip} are specific to radionuclide and the pathway. The dose factor R_{ip} is also specific to the receiving age group and organ. The derivation and appropriate application of atmospheric dispersion factors will be addressed in more detail in a later section of this review.

The methodology given in NUREG-0133 to calculate the R_{ip} dose factors will be presented below for each pathway. In addition, the derivation of each pathway specific dose factor R_{ip} will be illustrated for the kidney of the teen age group from exposure to manganese-54. The teen age group, kidney and manganese-54 were arbitrarily chosen for illustrative purposes. Appendix A, Table A-4 of this review details the dose factors for manganese-54 that are given in Table 4 of the SSES ODCM specific to organ, age group and pathway (PP&L 1993a).

a. The Inhalation Pathway Dose Factor

The following equation is given in NUREG-0133 (U.S. NRC 1978a) to calculate the dose factor for the inhalation pathway:

$$R_{iajI} = 10^6 \ BR_a \ DFA_{ija} \qquad (27)$$

Where

R_{iajI} = dose factor for the inhalation pathway specific to radionuclide i, age group a and organ j, in mrem·m³/μCi-yr;

10^6 = a constant used in converting pCi to μCi;

BR_a = breathing rate of receptor of age group a, in m^3/yr; and

DFA_{ija} = inhalation dose factor given in Reg. Guide 1.109 Table E-7 through Table E-10 for age group a, radionuclide i and organ j, in mrem/pCi.

The breathing rates for different age categories given in Reg. Guide 1.109 are presented below (U.S. NRC 1977a);

Age Group	Breathing Rate (m^3/yr)
Infant	1400
Child	3700
Teen	8000
Adult	8000

Using the breathing rate corresponding to the teen age group, the calculation below illustrates the derivation of the inhalation pathway dose factor for the kidney of a teenager from manganese-54:

10^6 pCi/μCi	(conversion from μCi to pCi)
$\times$ 8000 m^3/yr	(breathing rate for teen age group)
$\times$ 1.59 $\times$ 10^{-6} mrem/pCi	(Reg. Guide 1.109 inhalation dose factor)
= 1.27 $\times$ 10^4 mrem-m^3/uCi-yr	

This value equals the corresponding dose factor value given in the SSES ODCM (i.e., 1.27 $\times$ 10^4 mrem-m^3/pCi-yr).

b. The Ground Plane Pathway Factor

The following equation is given in NUREG-0133 (U.S. NRC 1978a) to calculate the dose factor for the ground plane pathway:

$$R_{iajG} = 10^6 \times 8760\ (S_F)\ DFG_{iaj}\,[(1 - \exp^{-\lambda_i t})/\lambda_i] \qquad (28)$$

Where

R_{iajG} = dose factor for the ground plane pathway specific to radionuclide i, age group a and organ j, in m^2-mrem-sec per pCi-yr;

10^6 = a constant converting picocuries^{-1} to microcuries^{-1};

8760 = a constant converting hours^{-1} to years^{-1};

S_F = a dimensionless shielding factor to account for the effects of reduced dose due to shielding by building structures during indoor residence. A value of 0.7 is recommended in Reg. Guide 1.109;

DFG_i = external dose factor for standing on contaminated ground from Reg. Guide 1.109, Table E-6, in mrem-m^2/yr-pCi;

λ_i = decay constant for the radionuclide i, in sec^{-1}; and

t = exposure time in seconds. A value of 4.73×10^8 sec (15 years) is recommended in Reg. Guide 1.109.

The calculation below illustrates the derivation of the ground plane pathway dose factor for the kidney of a teenager from manganese-54:

10^6 pCi/μCi	(conversion from μCi to pCi)
$\times$ 8760 hr/yr	(conversion from hr^{-1} to yr^{-1})
$\times$ 0.7	(Dimensionless shielding factor)
$\times$ 5.80×10^{-9} mrem-m^2/pCi-hr	(Reg. Guide 1.109 external dose factor)
$\times$ [1-exp(-2.57×10^{-8} sec^{-1}	
$\times$ 4.73×10^8 sec)]	(radioactivity accumulation in soil)
$\div$ 2.57×10^{-8} sec^{-1}	(decay constant for Mn-54)
= 1.38×10^9 mrem-m^2-sec/μCi-yr	

This value is equivalent to the corresponding dose factor given in the SSES ODCM (i.e., 1.37×10^9 mrem-m^2-sec/μCi-yr).

c. The Grass-Cow-Milk Pathway Dose Factor

The following equation is given in NUREG-0133 (U.S. NRC 1978a) to calculate the dose factor for the grass-cow-milk pathway:

$$R_{aijCM} = 10^6 \frac{Q_F U_{ap}}{\lambda_i + \lambda_w} F_m(r) DFL_{iaj} \left[\frac{f_p f_s}{Y_p} + \frac{(1 - f_p f_s) \exp(-\lambda_i t_h)}{Y_s} \right] \exp(-\lambda_i t_f) \quad (29)$$

Where

R_{aijCM} = dose factor for radionuclide *i* specific to age group *a* and organ *j* for the grass-cow-milk pathway, in m^2-mrem-sec/pCi-yr;

10^6 = constant converting from picocuries^{-1} to microcuries^{-1};

Q_f = cow's consumption rate, in kg/day;

U_{ap} = usage factor for the receptor's milk consumption rate specific to age *a*, in l/yr;

F_m = stable element transfer coefficients for milk, in days/l;

r = fraction of deposited activity retained on cow's feed grass (dimensionless);

DFL_{iaj} = ingestion dose factor from Reg. Guide 1.109, Table E-11 through Table E-11 for radionuclide *i* specific to age group *a* and organ *j*, in mrem/pCi;

f_p = fraction of the year that the cow is on pasture (dimensionless);

f_s = fraction of the cow feed that is pasture grass while the cow is on pasture (dimensionless);

Y_p = agricultural productivity by unit area of pasture feed grass, in kg/m^2;

λ_i = decay constant for radionuclide *i*, in sec;

t_h = transport time from pasture, to harvest, to cow, to milk, to receptor, in sec;

Y_s = agricultural productivity by unit area of stored feed, in kg/m^2;

t_f = transport time from pasture to cow, to milk, to receptor, in sec; and

λ_w = decay constant for removal of activity on leaf and plant surfaces by weathering, 5.73×10^{-7} sec^{-1}.

Table 3-3 details the parameter values used in equation (29) to calculate the teen kidney dose from manganese-54 via the grass-cow-milk pathway and the corresponding reference sources.

Table 3-3. Equation (29) Parameter Values for Mn-54 Exposure of Teen Kidney Via Grass-Cow-Milk Pathway

Parameter	Value	Reference
Q_f	50 kg/day	Table E-3, Reg. Guide 1.109
U_{ap}	Infant - 330 liter/yr Child - 330 liter/yr Teen - 400 liter/yr Adult - 310 liter/yr	Table E-5, Reg. Guide 1.109
F_m (for Mn-54)	2.50×10^{-4} days/liter	Table E-1, Reg. Guide 1.109
r	radioiodines - 1.0* particulate - 0.2*	Table E-15, Reg. Guide 1.109
DFL_{ia}	1.76×10^{-6} mrem/pCi	Table E-13, Reg. Guide 1.109
f_p	0.60*	Appendix C, PP&L ODCM
f_s	0.42*	Appendix C. PP&L ODCM
Y_p	0.7 kg/m^2	Table E-15, Reg. Guide 1.109
λ_i	2.57×10^{-8} sec^{-1}	Table 5-3, PP&L ODCM
t_h	7.78×10^{6} sec	Table E-15, Reg. Guide 1.109
Y_s	2.0 kg/m^2	Table E-15, Reg. Guide 1.109
t_f	1.75×10^{5} sec	Table E-15, Reg. Guide 1.109
λ_w	5.73×10^{-7} sec^{-1}	NUREG-0133

*These values are dimensionless.

The calculation below illustrates the derivation of the grass-cow-milk pathway dose factor for the kidney of a teenager from manganese-54:

$$
\begin{array}{ll}
 & 10^6 \text{ pCi}/\mu\text{Ci} \\
\times & 50 \text{ kg/day} \\
\times & 400 \text{ l/yr} \\
\div & (2.57 \times 10^{-8} \text{ sec}^{-1} + 5.73 \times 10^{-7} \text{ sec}^{-1}) \\
\times & 2.50 \times 10^{-4} \text{ days/l} \\
\times & 0.2 \text{ (dimensionless)} \\
\times & 1.76 \times 10^{-6} \text{ mrem/pCi} \\
\times & [(0.6 \times 0.42 / 0.7)+((1-(0.6 \times 0.42)) \times \\
 & \exp(-2.57 \times 10^{-8} \text{ sec}^{-1} \times 7.78 \times 10^6 \text{ sec}) \div 2)] \\
\times & \exp(-2.57 \times 10^{-8} \text{ sec}^{-1} \times 1.73 \times 10^5 \text{ sec}) \\
= & 1.95 \times 10^6 \text{ mrem-m}^2\text{-sec}/\mu\text{Ci-yr}
\end{array}
$$

This value equals the corresponding dose factor given in the SSES ODCM, (i.e., 1.95×10^6 mrem-m^2-sec/μCi-yr).

d. The Grass-Goat-Milk Pathway Dose Factor

The following equation is given in NUREG-0133 (U.S. NRC 1978a) to calculate the dose factor for the grass-goat-milk pathway:

$$R_{aijGM} = 10^6 \frac{Q_F U_{ap}}{\lambda_i + \lambda_w} F_m(r) DFL_{iaj} \left[\frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s) \exp(-\lambda_i t_h)}{Y_s} \right] \exp(-\lambda_i t_f) \qquad (30)$$

All of the parameters of the equation (30) for calculating grass-goat-milk pathway dose factors are defined the same and have the same values as those in equation (29) except for the parameters Q_f, t_f, t_h, f_p, and f_s. These parameters are defined below and their values are given in Table 3-4.

Q_f = goat's consumption rate, in kg/day;

t_f = transport from pasture to goat, to milk, to receptor, in seconds;

t_h = transport time from pasture, to harvest, to goat, to milk, to receptor, in seconds;

f_p = fraction of the year that the goat is on pasture (dimensionless); and

f_s = fraction of the goat feed that is pasture grass while the goat is on pasture (dimensionless).

Table 3-4. Equation (30) Parameter Values for Mn-54 Exposure of Teen Kidney Via Grass-Goat-Milk Pathway

Parameter	Value	Reference
Q_f	56 kg/day	Table E-3, Reg. Guide 1.109
t_h	7.78 E 06 sec	Table E-15, Reg. Guide 1.109
t_f	1.73 E 05 sec	Table E-15, Reg. Guide 1.109
f_p	0.60*	Appendix C, PP&L ODCM
f_s	0.75*	Appendix C. PP&L ODCM

*These values are dimensionless.

The calculation below yields the grass-goat-milk pathway dose factor for the kidney of a teenager from manganese-54:

10^6 pCi/μCi

$\times$ 6 kg/day

$\times$ 400 l/day

$\div$ $(2.57 \times 10^{-8}\ \text{sec}^{-1} + 5.73 \times 10^{-7}\ \text{sec}^{-1})$

$\times$ 2.50×10^{-4} days/liter

$\times$ 0.2 (dimensionless)

$\times$ 1.76×10^{-6} mrem/pCi

$\times$ $[(0.6 \times 0.75 / 0.7)+((1-(0.6 \times 0.75)) \times$

$\exp(-2.57 \times 10^{-8}\ \text{sec}^{-1} \times 7.78 \times 10^{6}\ \text{sec}) \div 2)]$

$\times$ $\exp(-2.57 \times 10^{-8}\ \text{sec}^{-1} \times 1.73 \times 10^{5}\ \text{sec})$

$=$ 3.05×10^{5} mrem-m^2-sec/μCi-yr

This value equals the corresponding dose factor given in the SSES ODCM, (i.e., 3.05×10^{5} mrem-m^2-sec/μCi-yr).

e. The Grass-Cow-Meat Pathway Dose Factor

The following equation is given in NUREG-0133 (U.S. NRC 1978a) to calculate the dose factor for the grass-cow-meat pathway:

$$R_{aijCT} = 10^6 \frac{Q_F U_{ap}}{\lambda_i + \lambda_w} F_f(r) DFL_{iaj} \left[\frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s)\exp(-\lambda_i t_h)}{Y_s} \right] \exp(-\lambda_i t_f) \qquad (31)$$

All of the parameters of equation (31) for calculating grass-cow-meat pathway dose factors are defined the same and have the same value as those in equation (29) except for the parameters U_{ap}, F_f, t_h, t_f, f_p, and f_s. These parameters are defined below and their values are given in Table 3-5.

U_{ap} = usage factor for the receptor's meat consumption rate for the grass-cow-meat pathway specific to age group a, in kg/yr;

F_f = stable element transfer coefficients for meat, in days/kg;

t_h = transport time from pasture, to harvest, to cow, to receptor, in seconds;

t_f = transport from pasture to cow, to receptor, in seconds;

f_p = fraction of the year beef cattle are on pasture (dimensionless); and

f_s = fraction of daily intake of beef cattle derived from pasture while on pasture (dimensionless).

Table 3-5. Equation (31) Parameter Values for Mn-54 Exposure of Teen Kidney Via Grass-Cow-Meat Pathway

Parameter	Value	Reference
U_{ap}	Infant - 0 kg/yr Child - 41 liter/yr Teen - 65 liter/yr Adult - 110 liter/yr	Table E-5, Reg. Guide 1.109
F_f(for Mn-54)	8.00×10^3 days/liter	Table E-1, Reg. Guide 1.109
t_h	7.78×10^6 sec	Table E-15, Reg. Guide 1.109
t_f	1.73×10^6 sec	Table E-15, Reg. Guide 1.109
f_p	0.60*	Appendix A, PP&L ODCM
f_s	0.55*	Appendix A, PP&L ODCM

*These values are dimensionless

The calculation below illustrates the derivation of the grass-cow-meat pathway dose factor for the kidney of a teenager from Mn-54:

$$10^6 \text{ pCi}/\mu\text{Ci}$$

$$\times \quad 50 \text{ kg/day}$$

$$
\begin{aligned}
\times\ & 65 \text{ kg/yr} \\
\div\ & (2.57 \times 10^{8} \text{ sec}^{-1} + 5.73 \times 10^{-7} \text{ sec}^{-1}) \\
\times\ & 8.00 \times 10^{-4} \text{ days/l} \\
\times\ & 0.2 \text{ (dimensionless)} \\
\times\ & 1.76 \times 10^{-6} \text{ mrem/pCi} \\
\times\ & [(0.6 \times 0.55 \,/\, 0.7) + ((1-(0.6 \times 0.55)) \times \\
& \exp(-2.57 \times 10^{-8} \text{ sec}^{-1} \times 7.78 \times 10^{6} \text{ sec}) \div 2)] \\
\times\ & \exp(-2.57 \times 10^{-8} \text{ sec}^{-1} \times 1.73 \times 10^{6} \text{ sec}) \\
=\ & 1.09 \times 10^{6} \text{ mrem-m}^{2}\text{-sec}/\mu\text{Ci-yr}
\end{aligned}
$$

This value equals the corresponding dose factor found in the SSES ODCM, (i.e., 1.09×10^{6} mrem-m^{2}-sec/μCi-yr).

f. The Vegetation Pathway Dose Factor

The equation given in Reg. Guide 1.109 and NUREG-0133 separates the vegetation consumed by man into leafy vegetation (fresh) and stored produce. Leafy vegetation differs from stored produce by the amount consumed U_{ap} and the time between harvest and consumption t_h. The following equation given in NUREG-0133 (U.S. NRC 1978a) is used to calculate the dose factor for the vegetation pathway:

$$
R_{aijV} = 10^{6} \left[\frac{r}{Y_V(\lambda_i + \lambda_w)} \right] DFL_{ia} \left[U_{La}\ f_L \exp^{-\lambda_i t_L} + U_{Sa}\ f_G \exp^{-\lambda_i t_h} \right] \qquad (32)
$$

All of the parameters of equation (32) for calculating the vegetation pathway dose factors are defined the same and have the same value as those in equation (29) except for the parameters U_{La}, U_{Sa}, f_l, f_g, t_l, t_h, and Y_v. These parameters are defined below and their values are given in Table 3-6.

U_{La} = usage factor for the consumption rate of fresh leafy vegetation specific to age group a, in kg/yr;

U_{Sa} = usage factor for the consumption rate of stored vegetation for age group a, in kg/y;.

f_l = fraction of the annual intake of fresh leafy vegetation grown locally (dimensionless);

f_g = fraction of the annual intake of stored vegetation grown locally (dimensionless);

t_l = average time between harvest of leafy vegetation and its consumption, in seconds;

t_h = average time between harvest of stored vegetation and its consumption, in seconds; and

Y_v = vegetation areal density, in kg/m^2.

Table 3-6. Equation (32) Parameter Values for Mn-54 Exposure of Teen Kidney Via Vegetation Pathway

Parameter	Value	Reference
U_{La}	Child - 26 kg/yr Teen - 42 kg/yr Adult - 64 kg/yr	Table E-5, Reg. Guide 1.109
U_{Sa}	Child - 520 kg/yr Teen - 630 kg/yr Adult - 520 kg/yr	Table E-1, Reg. Guide 1.109
f_l	0.33*	Appendix C, PP&L ODCM
f_g	0.76*	Appendix C, PP&L ODCM
t_l	8.64×10^4 sec	Table E-15, Reg. Guide 1.109
t_h	5.18×10^6 sec	Table E-15, Reg. Guide 1.109
Y_v	2.0 kg/m^2	Table E-15, Reg. Guide 1.109

*These values are dimensionless.

The calculation below illustrates the derivation of the vegetation pathway dose factor for the kidney of a teenager from manganese-54:

$$
\begin{array}{ll}
 & 10^{6}\ \text{pCi}/\mu\text{Ci} \\
\times & 0.2\ \text{(dimensionless)} \\
\times & 1.76 \times 10^{-6}\ \text{mrem/pCi} \\
\times & [(42\ \text{kg/yr} \times 0.33 \times \exp(-2.57 \times 10^{-8}\ \text{sec}^{-1} \times 8.64 \times 10^{4}\ \text{sec}) \\
 & +(630\ \text{kg/yr} \times 0.76 \times \exp(-2.57 \times 10^{-8}\ \text{sec}^{-1} \times 5.18 \times 10^{6}\ \text{sec}))] \\
\div & 2.0\ \text{kg/m}^{2} \\
\div & (2.57 \times 10^{-8}\ \text{sec}^{-1} + 5.73 \times 10^{-7}\ \text{sec}^{-1}) \\
= & 1.27 \times 10^{8}\ \text{mrem-m}^{2}\text{-sec}/\mu\text{Ci-yr}
\end{array}
$$

This value equals the corresponding dose factor given in the SSES ODCM, (i.e., 1.27×10^{8} mrem-m^{2}-sec/μCi-yr).

IV. CALCULATION OF DOSE ESTIMATES FROM 1993 EFFLUENT DATA

10 CFR 50, section 36a, "Technical Specifications on Effluents from Nuclear Power Reactors," states (U.S. NRC 1993):

> (2) Each licensee shall submit a report to the Commission annually that specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 12 months of operation, including any other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public . . . (U.S. NRC 1993).

As stated in the introduction, each license applicant for the operation of a nuclear power plant is responsible for developing site-specific technical specifications designed to protect the health and safety of the public (U.S. NRC 1993). Standard Technical Specifications have been provided by the NRC as guidance for license applicants in developing site-specific technical specifications. Standard Technical Specification 6.9.1.9 requires that radioactive effluent reports include a summary of the types and quantities of radioactive liquid and gaseous effluents released in a calendar year, a concurrent summary of the meteorological conditions, and the annual individual dose estimates that result from the radioactive effluents released. Standard Technical Specification 6.9.1.9 further requires that the assessment of radiation dose estimates "shall be performed in accordance with the ODCM" (U.S. NRC 1978b). Regulatory Guide 1.21, Measuring , Evaluating, and Reporting Radioactivity in Solid Waste and

Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, outlines methods acceptable to the NRC for "measuring, reporting and evaluating releases of radioactive materials in liquid and gaseous effluents" (U.S. NRC 1974).

A. 1993 Dose Estimates

The SSES Annual Effluent and Waste Disposal Report is the medium used by PP&L to meet the reporting requirements of 10 CFR 50 (PP&L 1993a). The AEWDR includes a summary of the types and quantities of radioactive effluents released in a calendar year, a summary of the meteorological data, and the annual offsite dose estimates. In addition, if any changes have been made to the ODCM, the revised ODCM is included in the AEWDR for NRC review. Table 4-1 summarizes the maximum individual offsite dose estimates reported in the 1993 AEWDR (PP&L 1993a).

Table 4-1. Summary of Maximum Individual Offsite Dose Commitments to Members of the Public: 1/1/93 to 12/31/93

Effluent	Age Group	Applicable Organ	Estimated Maximum Dose (mrem)	Location Dist. (mile)	Location Affected Sector	Percent of Limit	Limit (mrem)
Liquid	Teen	Total Body	6.93E-03	(1)		0.23	3
Liquid	Adult	GI-LLI	1.49E-02	(1)		0.15	10
Noble Gas	N/A	Air Dose (γ-mrad)	2.68E-03	1.03	WSW	0.03	10
Noble Gas	N/A	Air Dose (β-mrad)	7.96E-03	1.03	WSW	0.04	20
Noble Gas	N/A	Total Body	1.56E-03	1.03	WSW	0.03	5
Noble Gas	N/A	Skin	4.40E-03	1.03	WSW	0.03	15
I&PM	Teen	Lung	1.69E-02	1.10	WSW	0.12	15

(1) Doses from liquid effluents are estimated from fish ingestion and shoreline exposure at the site outfall and from the drinking water pathway at Danville, PA.

The required dose estimates of 10 CFR 50 - Appendix I are calculated by PP&L for specified organs of each age group using the computer programs GASPAR and LADTAP II (PP&L 1993a). The dose estimates of Table 4-1 represent the highest calculated organ and total body dose estimates of the four age groups. For example, the highest total body dose estimate for 1993 from liquid effluents is calculated to be that of the teen age group.

A direct method of analyzing the dose methodology given in the SSES ODCM would be to manually calculate the 1993 dose estimates according to the methodology given in the ODCM and then compare these hand-calculated dose estimates to the reported dose estimates of Table 4-1. Any discrepancies between the reported dose estimates of Table 4-1 and the hand-calculated dose estimates may reveal inadequacies of the ODCM to accurately describe the actual methodology used by PP&L to calculate annual reported dose estimates. Such discrepancies may also indicate differences between the methodology given in the ODCM and the GASPAR and LADTAP II computer dose models.

This section details the hand calculation of the 1993 dose estimates according to the methodology given in the SSES ODCM. The 1993 dose estimates were hand calculated using a computer spread-sheet program and the radioactive effluent data reported in the 1993 AEWDR. The calculations performed in review of the SSES ODCM dose assessment methodology are reduced to include only those calculations that will produce the dose estimates of Table 4-1. Therefore, only those model parameters and calculations that result in a dose estimate for the total body of a hypothetical teenager (teen total body) and for the lower large intestine of a hypothetical adult (adult GI-LLI) will be presented below for liquid effluent pathways. Similarly, dose calculations for airborne effluents will be restricted to those which produce the dose estimates comparable to Table 4-1.

B. Individual Dose Estimates Due to Waterborne Effluents

A hypothetical maximally exposed individual acquires an annual dose from waterborne radioactive effluents via three liquid pathways: potable water ingestion, fish ingestion, and shore exposure. Although the dose methodology for each pathway differs, the same radioactive release data and blowdown rates are used in calculating dose estimates for all liquid pathways. The types and quantities of liquid radionuclides released from SSES during 1993 are summarized per quarter year in Appendix C, Table C-1 (PP&L 1993a). Regulatory Guide 1.21 specifies quarterly reporting for effluent and waste disposal reports (U.S. NRC 1974). Dose estimates from liquid effluents are calculated per quarter year and then summed to obtain an annual dose.

Two parameters that distinguish the different liquid pathways are the dilution factor and the transit time. The dilution factors and transit times used by PP&L to calculate annual dose estimates were derived from tracer release studies conducted in 1985 of the mixing characteristics of the Susquehanna River (PP&L 1993a). A later section of this review gives a more detailed discussion of the 1985 tracer dye study.

PP&L uses a fixed dilution factor of 15.9 to calculate annual dose estimates for the fish ingestion and shore exposure pathways. The transit times used by PP&L for the fish ingestion and shore exposure pathways are 25 hours and 1 hour, respectively. Since the dilution factor and transit times for the fish ingestion and shore exposure pathways are considered constant, they are incorporated into the precalculated fish ingestion and shore exposure dose factors given in the SSES ODCM (PP&L 1993a). The derivation of these dose factors has been previously demonstrated.

In contrast to the fixed dilution factor and transit times for the fish ingestion and shore exposure pathways, the potable water pathway employs variable dilution factors and transit times which are a function of the river level at Danville, Pennsylvania. Water drawn from the Susquehanna River is used to provide drinking water for the downstream community of Danville, PA. The dilution factor and transit time used for the potable water pathway vary depending on river level at the Danville location. Dilution factors and transit times corresponding to river depths ranging from 1.5 to 22 feet were empirically derived from tracer dye measurements taken from the Susquehanna River at the Danville location. The dilution factors and transit times used by PP&L for the potable water pathway are given in Appendix B, Table B-1. Measurements of the river depth at Danville, PA are taken throughout the year and averaged per quarter year. As formerly stated, dose estimates from liquid effluents are calculated per quarter year and then summed to obtain a total annual dose estimate. The average river level per quarter year determines the dilution factor and transit time that is used to calculate the potable water dose estimate for each quarter, respectively (PP&L 1993a).

Table 4-2 lists the average river level at Danville, PA per quarter for 1993 and the corresponding dilution factors and transit times used to calculate quarterly dose estimates for the potable water pathway. Table 4-3 also contains the blowdown rates per quarter year that were used to calculate the quarterly dose estimates for all liquid pathways (PP&L 1993a).

Table 4-2. Site Specific Parameter Used in Liquid Effluent Calculations

Parameter	First Quarter	Second Quarter	Third Quarter	Fourth Quarter
Cooling Tower Blowdown (cfs)	15.6	18.4	15.4	14.1
Average Net River Level (ft)	7.4	10.1	3.5	6.7
Dilution Factor at Danville (2)	456.6	1285.3	280.9	378.8
Transit Time to Danville (hr)	23	13.5	45.9	26.2

(2) From Table B-1 of Appendix B (dimensionless)

1. Potable Water Pathway Dose Estimate

The following equation is given in the SSES ODCM (PP&L 1993a) to estimate dose from the potable water pathway:

$$R_{aipj} = \sum_{i} \left[\frac{K_{aipj} \times \exp(-\lambda_i \times t_p) \times C_i \times V_i \times k}{DF_p \times F} \right] \tag{3}$$

The parameters of equation (3) have been previously defined. The $C_i \times V_i \times k$ expression yields the amount of radionuclides, in curies, released during a period of time. Since the liquid effluent data and blowdown rate are reported per quarter year, all dose estimates will be calculated first for each quarter year and then summed to produce an annual dose. Therefore, the quarterly activity values of Table C-1 will be entered into equation (3) for the $C_i \times V_i \times k$ expression and the quarterly blowdown rates of Table 4-2 will be entered for the F parameter.

The potable water ingestion dose factor K_{aipj}, in units of mrem-ft^3/Ci-sec, is specific to age, organ, and radionuclide. The derivation of this parameter has been given previously. Tables 5-2a, b, c, and d of the ODCM list potable water ingestion dose factors for the adult, teen, child, and infant age groups, respectively.

An example of the potable water dose calculation for teen total body from first quarter release of tritium is demonstrated below:

5.95×10^{-2} mrem-ft^3/Ci-sec	(potable water teen total body dose factor for H-3)
$\times \exp(-6.44 \times 10^{-6}\ h^{-1} \times 23\ h)$	(decay due to transit time)
$\times$ 8.64 Ci	(1st quarter tritium release)
$\div$ 456.6	(1st quarter potable water dilution factor)
$\div$ 15.6 cfs	(1st quarter blowdown rate)
$= 7.22 \times 10^{-5}$ mrem	(1st quarter, potable water teen total body dose)

The calculation to estimate the adult GI-LLI dose from the potable water pathway differs only by the dose factor. The computation below illustrates this difference:

5.95×10^{-2} mrem-ft^3/Ci-sec	(potable water adult GI-LLI dose factor for H-3)
$\times$ exp(-6.44×10^{-6} h^{-1} $\times$ 23 h)	(decay due to transit time)
$\times$ 8.64 Ci	(1st quarter tritium release)
$\div$ 456.6	(1st quarter potable water dilution factor)
$\div$ 15.6 cfs	(1st quarter blowdown rate)
= 1.02×10^{-4} mrem	(first quarter, potable water adult GI-LLI dose)

The potable water dose estimates were calculated for each radionuclide and quarter year using a computer spread-sheet program. The results of these calculations for both teen total body and adult GI-LLI are given in Appendix C, Table C-2 and Table C-5, respectively. The dose estimates summed over all radionuclides and each quarter resulted in a total annual potable water ingestion dose of 7.41×10^{-4} mrem for teen total body and 1.16×10^{-3} mrem for adult GI-LLI.

2. Fish Ingestion Pathway Dose Estimate

The following equation is given in the SSES ODCM (PP&L 1993a) to estimate dose from the fish ingestion pathway:

$$R_{aipj} = \sum_i \left[\frac{K_{aipj} \times C_i \times V_i \times k}{F} \right] \qquad (5)$$

The parameters of equation (5) have been previously defined. As with the equation for the potable water pathway, the quarterly activity values given in Table C-1 as well as the blowdown rates from Table 4-2 will be entered into equation (5) for the $C_i \times V_i \times k$ expression and for the F parameter, respectively. The dose conversion factor K_{aipj} in this equation represents the fish ingestion dose factor specific to age, organ, and radionuclide in units of mrem-ft^3/Ci-sec. The derivation of the fish ingestion dose factor has been previously demonstrated. Tables 5-1 a, b, and c of the ODCM list fish ingestion dose factors by radionuclide and organ for the adult, teen, and child age groups respectively.

An example of the fish ingestion dose calculation for teen total body from the first quarter release of tritium is demonstrated below:

1.06×10^{-4} mrem-ft^3/Ci-sec	(fish ingestion teen total body dose factor for H-3)
$\times$ 8.64 Ci	(first quarter tritium release)
$\div$ 15.6 cfs	(first quarter blowdown rate)
$= 5.87 \times 10^{-5}$ mrem	(first quarter, fish ingestion teen total body dose)

The calculation to estimate the adult GI-LLI dose from the fish ingestion pathway differs only by the dose conversion factor. The computation below illustrates this difference:

1.37×10^{-4} mrem-ft^3/Ci-sec	(fish ingestion adult GI-LLI dose factor for H-3)
$\times$ 8.64 Ci	(first quarter tritium release)
$\div$ 15.6 cfs	(first quarter blowdown rate)
$= 7.59 \times 10^{-5}$ mrem	(first quarter fish ingestion adult GI-LLI dose)

Dose estimates for teen total body and adult GI-LLI from fish ingestion were calculated for each radionuclide and quarter year and the results are given in Appendix C, Table C-3 and Table C-6. The dose estimates summed over all radionuclides and each quarter resulted in a total annual fish ingestion dose of 3.57×10^{-3} mrem for teen total body and 1.34×10^{-2} mrem for adult GI-LLI.

3. Shore Exposure Pathway Dose Estimate

The following equation is given in the SSES ODCM (PP&L 1993a) to calculate the shore exposure pathway dose for each radionuclide:

$$R_{aipj} = \sum_{i} \left[\frac{K_{aipj} \times C_i \times V_i \times k}{F} \right] \qquad (7)$$

The parameters of equation (7) have been previously defined. The quarterly activity values of Table C-1 as well as the blowdown rates from Table 4-2 will be entered into equation (7) for the $C_i \times V_i \times k$ expression and F parameter, respectively. The dose conversion factor K_{aipj} in equation (7) represents the shore exposure dose factor, in units of mrem-ft^3/Ci-sec. The derivation of this dose factor has been previously demonstrated. Table 5-5 of the SSES ODCM lists shore exposure dose factors by radionuclide, organ and age group.

An example of the shore exposure dose calculation for teen total body from the first quarter release of manganese-54 is demonstrated below:

1.68×10^{-1} mrem-ft^3/Ci-sec	(Mn-54 teen total body dose factor, shore exposure)
$\times\ 3.73 \times 10^{-3}$ Ci	(first quarter Mn-54 release)

$\div$ 15.6 cfs (first quarter blowdown rate)

$= 4.02 \times 10^{-5}$ mrem (first quarter, shore exposure teen total body dose)

The calculation to estimate the adult GI-LLI dose from the shore exposure pathway differs only by the dose conversion factor. The computation below illustrates this difference:

3.01×10^{-2} mrem-ft^3/Ci-sec (shore exposure adult GI-LLI dose factor, Mn-54)

$\times$ 3.73×10^{-3} Ci (first quarter tritium release)

$\div$ 15.6 cfs (first quarter blowdown rate)

$= 7.20 \times 10^{-6}$ (first quarter, shore exposure adult GI-LLI dose)

Dose estimates for teen total body and adult GI-LLI from shore exposure were calculated for each radionuclide and quarter year and the results are given in Appendix C, Table C-4 and Table C-7. The doses summed over all radionuclides and each quarter year resulted in a total annual shore exposure dose of 3.03×10^{-3} mrem for teen total body and 5.43×10^{-4} mrem for adult GI-LLI.

4. Total Dose Estimate from Liquid Effluents

Summing the dose estimates calculated for all three liquid pathways yields a total annual dose estimate of 7.34×10^{-3} mrem for teen total body and 1.51×10^{-3} mrem for adult GI-LLI. Table C-8 of Appendix C summarizes the total dose results for teen total body and adult GI-LLI. The hand-calculated dose estimate for teen total body exceeds the reported dose estimate of Table 4-1 (6.96×10^{-3} mrem) by 5.5%. The adult GI-LLI dose estimate differs from the reported dose estimate (1.49×10^{-3} mrem) by 1.0%.

The small differences between the liquid dose estimates calculated manually and those reported in the 1993 AEWDR would indicate that the ODCM methodology agrees with the actual dose methodology employed by PP&L to calculate annual dose estimates from liquid effluents. However, this review has identified one difference in the dose estimates that were calculated according to the ODCM and those reported in the 1993 AEWDR. The methodology given in the SSES ODCM employs tritium dose factors that are approximately 1.7 times greater than the tritium dose factors used to calculate the reported dose estimates.

As previously demonstrated, the fish ingestion and potable water dose factors given in the SSES ODCM were derived from ingestion dose factors given in Reg. Guide 1.109 (PP&L 1993a). The ingestion dose factors of Reg. Guide 1.109 are based upon the internal dose models of Publication 2 of the ICRP (ICRP 2) (U.S. NRC 1977c). In ICRP 2, a quality factor of 1.7 is assigned to low energy beta radiation (i.e., energies less than 30 keV) such as the beta radiation that is emitted from tritium (ICRP 1959). Therefore, the tritium dose factors of Reg. Guide 1.109, and consequently those given in the SSES ODCM, are based upon a quality factor of 1.7.

Since the publication of ICRP 2 in 1959 the quality factor for low energy beta radiation has been reduced to 1.0 (U.S. NRC 1986). The tritium dose factors used by the LADTAP II computer dose model are derived using a quality factor of 1.0 instead of 1.7 (U.S. NRC 1986). Therefore, tritium dose estimates calculated according to the SSES ODCM are higher than tritium dose estimates calculated by LADTAP II by a factor 1.7. To illustrate, the adult dose factor given in Reg. Guide 1.109 for tritium ingestion is $1.05 \times 10\text{-}7$ mrem/pCi, whereas, the corresponding dose factor used by LADTAP II is $5.99 \times 10\text{-}8$ mrem/pCi (U.S. NRC 1977a and 1987). The adult dose factor given in Reg. Guide 1.109 for tritium ingestion is approximately 1.7 times greater than the LADTAP II dose factor. The LADTAP II computer code is used by PP&L to calculate the annual dose estimates reported in the AEWDR (PP&L 1993a).

The hand-calculated dose estimates for liquid effluents were recalculated using tritium dose factors based on a quality factor of 1.0 instead of 1.7; the recalculated dose estimates resulted in a teen total body dose estimate of 6.85×10^{-3} mrem and an adult GI-LLI dose estimate of 1.44×10^{-2} mrem. The revised hand-calculated dose estimates were obtained by dividing the tritium fish ingestion and potable water dose components of the total dose by 1.7. Table C-9 of Appendix C details the results of the revised hand calculations.

The difference between the revised teen total body dose estimate from liquid effluents and the corresponding reported dose estimate of Table 4-1 is reduced to 2%. The difference between the revised adult GI-LLI dose estimate and the reported dose estimate of Table 4-1, however, increases slightly to 3%. Differences in number round-off between the LADTAP II computer computations and the above hand-calculations are assumed to be partially responsible for the differences between the dose estimates. With the exception of the difference in the tritium dose factors between LADTAP and the ODCM which proved to be subtle, the variance between the hand-calculated dose estimates and the dose estimates reported in the 1993 AEWDR for liquid effluents is sufficiently small to conclude that the liquid dose assessment methodology given in the ODCM adequately describes PP&L's dose assessment methodology.

C. Individual Dose Estimates Due to Airborne Effluents

The SSES Technical Specifications that govern the release of radioactive airborne effluents include four dose limitations restricting the release of noble gas effluents and one dose limitation restricting the release of I&PM effluent releases. The dose limitations imposed by the SSES Technical Specifications are defined per each reactor unit. The Susquehanna Steam Electric Station consists of two boiling water reactor generating units (PP&L 1993). Therefore, dose estimates are calculated separately from the effluent releases of each reactor unit. The dose estimates from airborne effluents reported in the AEWDR represent the highest calculated dose estimates of the two reactor units. The effluent releases from reactor unit one for 1993 resulted in the highest dose estimates from airborne I&PM effluents, whereas, reactor unit two was the only reactor unit to release detectable levels of noble gases in 1993. Table 4-3 contains the types and quantities of released airborne radionuclides specific to each reactor unit that were used to compute the 1993 dose estimates.[3]

Table 4-3. Radioactive Airborne Effluents Released in 1993

Reactor Unit No.	Nuclide	Annual Release (Ci)
1	H-3	2.55 E 01
1	Mn-54	3.77 E-04
1	Co-58	2.74 E-05
1	Co-60	1.87 E-04
1	Zn-65	9.27 E-05
1	Sr-90	2.60E-07
2	Xe-133	7.68E+00

Meteorological conditions determine the fate of radionuclides contained in an effluent plume. In calculating dose estimates, the transportation and dilution of radionuclides in the atmosphere are characterized by precalculated, site-specific atmospheric dispersion factors (ADF), denoted *X/Q*. Multiplying the initial release of radioactivity by the appropriate ADF yields the radionuclide concentration at the point of human exposure (U.S. NRC 1977b).

Various atmospheric diffusion models exist which characterize the effluent plume as it travels from the release point. Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from

[3]Personal communication with Robert Barclay of Pennsylvania Power & Light Company, 31 January, 1995.

Light-Water-Cooled Reactors, (Reg. Guide 1.111) describes methods acceptable to the NRC for estimating the transport and dispersion of released airborne effluents (U.S. NRC 1977b). PP&L uses a computer software modeling program, XOQDOQ, to produce the site-specific ADFs used in calculating annual dose estimates. Developed in accordance with Reg. Guide 1.111, XOQDOQ is a straight-line air flow Gaussian plume model (PP&L 1993a).

The Gaussian diffusion model is based on a "horizontal wind field that can vary in time and space" (U.S. NRC 1977b). Meteorological information such as windspeed, wind direction, and atmospheric stability are used in the Gaussian plume model to derive ADFs. The NRC specifies that "meteorological information should be obtained to characterize transport processes . . . out to a distance of 50 miles" (U.S. NRC 1977b). The on-site meteorological program at SSES employs wind sensors that are mounted at 10 and 60 meter elevations on a 300 foot high tower located approximately 1000 feet to the southeast of the plant. Temperature sensors that measure the vertical temperature differential are located between the 10 and 60 meter elevations. The dew point and ambient temperature are also measured by sensors at the 10 meter elevation, while precipitation is measured at ground level. SSES also employs a back-up meteorological tower that is 10 meters high which provides alternate measurements of wind speed and wind direction. The meteorological information from these tower sensors is transmitted via telephone line data-link to the PP&L corporate computer in Allentown, Pennsylvania and is used to prepare summary reports, wind rose plots and dispersion estimates (PP&L 1993a).

Meteorological conditions, such as wind speed, wind direction and atmospheric stability, vary throughout the year. As a result, the dispersion of an effluent plume is not symmetrical about the facility. Regulatory Guide 1.21 prescribes that "at least 16 compass point sectors" (i.e., North, West, Southeast, etc.) should be used to classify windspeed and wind direction according to averaged hourly measurements (U.S. NRC

1974). Therefore, ADFs are generated by PP&L specific to 16 sectors surrounding SSES. In addition to these 16 sectors, ADFs are also calculated specific to five location types downwind of the initial release point. The five locations types are categorized as site boundary, resident, garden, dairy, and irrigation (PP&L 1993). Different location types address different dose calculations. For example, inert noble gases are considered non-depositing, thereby restricting the exposure pathway to inhalation or external exposure. It is conservatively assumed that exposure to a hypothetical person from noble gas effluents occurs at the site boundary location[4]. The table of ADFs given in the 1993 AEWDR is included in Appendix B as Table B-3. Atmospheric dispersion factors are listed in Table B-3 according to location type and sector (PP&L 1993a).

In addition to the different sectors and location types, ADFs are also radionuclide specific. Chemical and physical characteristics of different radionuclides will affect their removal and diffusion in the atmosphere. Removal mechanisms and, consequently, removal rates vary amongst different radionuclides (U.S. NRC 1977b). For a given sector and location type, one calculated dispersion factor does not adequately estimate the offsite effluent concentration for all radionuclides. Four different atmospheric dispersion factors, which represent various dispersion conditions of different radionuclides, are calculated for each sector and location type (PP&L 1993a). The four categories of dispersion factors are illustrated by the ADFs given in Table B-3 of Appendix B (PP&L 1993a). Classification of the four dispersion factors is governed by the amount of effluent removal resulting from radioactive decay and plume depletion (U.S. NRC 1987).

Radioactive decay and wet and dry deposition are the primary removal mechanisms that reduce the radioactive concentration of the effluent plume as it travels from its

[4]Personal communication with Robert Barclay of Pennsylvania Power & Light Company, 23 January, 1995.

initial release point (U.S. NRC 1977b). Radioactive decay depends on the half-life of the radionuclide. In Reg. Guide 1.111, a 2.26 day half-life is recommended to calculate ADFs for short lived noble gases while an 8 day half-life is used to calculate ADFs for radioactive iodines (U.S. NRC 1977b). In addition, GASPAR, the computer software developed for the NRC, uses ADFs based on 8 day half-life for inhalation dose calculations from all other radionuclides (U.S. NRC 1978a). Using these suggested half-lives to calculate ADFs in lieu of the actual half-life of each radionuclide reduces the number of ADFs generated thereby reducing the number of calculations.

Dry deposition is a continuous process which effectively removes most radionuclides by sorption onto ground surfaces. The removal rate for noble gases, tritium, carbon-14, and non-elemental lead, however, is so gradual that these radionuclides are considered non-depositing within 50 miles of the release point. Although wet deposition is a more efficient removal process, it occurs for only a few hours each year during periods of precipitation. Except for sites with well defined rainy seasons that directly correspond to grazing or harvesting periods, dry deposition is solely considered in deriving ADFs (U.S. NRC 1977b).

The four categories of ADFs used by PP&L are defined below (U.S. NRC 1987):

$(X/Q)_r$ = no radiological decay or plume depletion considered, in units of sec/m^3. This ADF is used for tritium, C-14, and other long-lived radionuclides that are non-depositing.

$(X/Q)_{dr}$ = radiological decay corresponding to a 2.26 day half-life and no plume depletion, in units of sec/m^3. This ADF is used for calculating inhalation doses from short lived radioiodines and external exposures from noble gases.

$(X/Q)_{ddr}$ = radiological decay corresponding to a 8 day half-life and plume depletion, in units of sec/m^3. This ADF is used for all other radionuclides other than tritium, C-14 and noble gases for evaluation of inhalation doses.

$(D/Q)_r$ = no radiological decay, in units of m^{-2}. This ADF relates the initially released activity to a ground concentration at the location of interest for 1 year of release and is used for all radionuclides other than tritium, C-14, and noble gases for ground exposure and ingestion pathways.

The $(X/Q)_r$, $(X/Q)_{dr}$, and $(X/Q)_{ddr}$ atmospheric dispersion factors multiplied by the amount of radioactivity released yields the radionuclide concentration in air at the point of exposure (i.e., Ci/yr × sec/m^3 × 3.17 × 10^{-8} yr/sec = Ci/m^3). The $(X/Q)_r$, $(X/Q)_{dr}$, and $(X/Q)_{ddr}$ atmospheric dispersion factors are used to calculate dose estimates for the inhalation pathway. The $(D/Q)_r$ atmospheric dispersion factor multiplied by the annual amount of radioactivity released yields the ground concentration per unit area which is necessary to calculate dose estimates via ingestion and external ground exposure (i.e., Ci/yr × m^{-2} = Ci/m^{-2} per year).

The location types and sectors that were used to calculate the reported 1993 dose estimates for noble gases and I&PM effluents are listed in the fifth column of Table 4-1. Table 4-4 below lists the ADFs taken from Appendix B, Table B-3 which correspond to the locations and sectors indicated in Table 4-1.

Table 4-4. Atmospheric Dispersion Factors Used to Calculate the 1993 Dose Estimates from Airborne Radionuclides

Effluent	distance (miles)	sector *	location type	$(X/Q)_r$ (sec/m^3)	$(X/Q)_{dr}$ (sec/m^3)	$(X/Q)_{ddr}$ (sec/m^3)	$(D/Q)_r$ (m^{-2})
Noble Gas	1.03	WSW	site boundary	1.418E-05	1.408E-05	1.236E-05	2.083E-08
I&PM	1.10	WSW	residence or garden	1.289E-05	1.280E-05	1.119E-05	1.858E-08

*WSW = west southwest

ADF values given in Table 4-4 will be used in the calculations below which estimate individual doses from airborne effluents. Close examination of the ADFs listed Table B-3 of Appendix B indicates that the ADFs given in Table 4-4 are the highest ADF values of those given in Table B-3. Thus, the ADF values of Table 4-4 will produce the most conservative dose estimates.

1. Gamma and Beta Air Dose Estimates from Noble Gas Exposure

The following equations are given in the SSES ODCM (PP&L 1993a) to calculate the gamma-air dose and the beta-air dose due to noble gas effluents:

$$D_g = 3.17 \times 10^{-8} \sum_i M_i (X/Q_v) Q_{iv} \tag{14}$$

$$D_b = 3.17 \times 10^{-8} \sum_i N_i (X/Q_v) Q_{iv} \tag{15}$$

The parameters in these equations have been previously defined. Xenon-133 is the only reported noble gas released from SSES during 1993. Calculations estimating the gamma air dose and beta air dose due to noble gas effluents are presented below. Tables 3-2, 4-3, and 4-4 provide the values for M_i and N_i, Q_{iv}, and $(X/Q)_v$ parameters, respectively.

Gamma air dose

3.17×10^{-8} years/sec	(inverse of seconds per year)
$\times\ 3.53 \times 10^{-8}$ mrad-m^3/pCi-yr	(γ-air dose factor for exposure to Xe-133)
$\times\ 1.408 \times 10^{-8}$ sec/m^3	(ADF for exposure to noble gas)

$\times$ 16.93 Ci/yr (Annual release of Xe-133)

$\times$ 1.00 $\times$ 10^{12} pCi/Ci (Conversion factor for pCi to Ci)

=2.67 $\times$ 10^{-3} mrad per year (Calculated annual γ-air dose from Xe-133)

Beta air dose

3.17 $\times$ 10^{-8} years/sec	(inverse of seconds per year)
$\times$ 1.05 $\times$ 10^{-3} mrad-m^3/pCi-yr	(β-air dose factor for exposure to Xe-133)
$\times$ 1.408 $\times$ 10^{-5} sec/m^3	(ADF for exposure to noble gas)
$\times$ 16.93 Ci/yr	(annual release of Xe-133)
$\times$ 1.00 $\times$ 10^{12} pCi/Ci	(conversion factor for pCi to Ci)
=7.93 $\times$ 10^{-3} mrad per year	(calculated annual β-air dose from Xe-133)

These calculated dose estimates for gamma air dose and beta air dose are equivalent to the dose estimates of Table 4-1, which are the dose estimates reported in the 1993 AEWDR (i.e., 2.68 $\times$ 10^{-3} mrad and 7.96 $\times$ 10^{-3} mrad, respectively). The similarity between the reported dose estimates and the hand-calculated dose estimates for the gamma-air and beta-air doses due to noble gas exposure indicates that the dose methodology given in the ODCM adequately describes the actual methodology used to produce the annual SSES dose estimates.

2. Total Body Dose Estimate from Noble Gas Exposure

The following equation is given in the SSES ODCM (PP&L 1993a) to calculate the total body dose due to exposure from noble gas effluents:

$$D_g = 3.17 \times 10^{-8} \sum_i K_i \times (X/Q)_v \times Q_{iv} \qquad (19)$$

As previously stated, a discrepancy was identified between equation (19) and the corresponding equation given in Reg. Guide 1.109. The equation given in Reg. Guide 1.109 to estimate the total body dose from noble gases includes a 0.7 shielding factor that accounts for the shielding of gamma radiation by residential structures (U.S. NRC 1977a); equation (19) does not include a structural shielding factor. However, the GASPAR computer model, which is used by PP&L to calculate total body dose estimates from noble gas exposure, includes a 0.7 structural shielding factor (U.S. NRC 1987). To illustrate this discrepancy, the following calculation estimates the total body dose according to equation (19) that results from exposure to the 1993 release of Xe-133.

3.17×10^{-8} years/sec	(inverse of seconds per year)
$\times$ 2.94×10^{-4} mrem-m^3/pCi-yr	(γ-body dose factor for exposure to Xe-133)
$\times$ 1.408×10^{-5} sec/m^3	(ADF for exposure to noble gas)
$\times$ 16.93 Ci/yr	(annual release of Xe-133)
$\times$ 1.00×10^{12} pCi/Ci	(conversion factor for pCi to Ci)
$=2.22 \times 10^{-3}$ mrem/year	(calculated annual total body dose from Xe-133)

This dose estimate is 41% greater than the total body dose estimate given in Table 4-1 for noble gas exposure (i.e., 1.56×10^{-3} mrem/yr). Multiplying 2.22×10^{-3} mrem/yr by 0.7 yields a total body dose estimate of 1.55×10^{-3} mrem/yr which is equivalent to the reported dose estimate of Table 4-1. The noble gas total body dose equation corrected to include a structural shielding factor S_F is given below:

$$D_g = 3.17 \times 10^{-8} \sum_{i} S_F * K_i * (X/Q)_v * Q_{iv} \qquad (33)$$

3. Skin Dose Estimate from Noble Gas Exposure

The following equation is given in the SSES ODCM (PP&L 1993a) to calculate skin dose from exposure to noble gas effluents:

$$D_b = 3.17 \times 10^{-8}\,(L_i + 1.1 M_i)\,(X/Q)_v\,Q_{iv} \qquad (23)$$

This equation was also previously shown to exclude the 0.7 shielding factor. The GASPAR computer dose model and the equation given in Reg. Guide 1.109 to calculate the skin dose from noble gas exposure incorporate a 0.7 shielding factor that is multiplied by the gamma-air dose factor, M_i (U.S. NRC 1977a and 1987). Equation (24) revised to include the structural shielding factor S_F may be given as:

$$D_b = 3.17 \times 10^{-8}\,(L_i + 1.1\,S_F\,M_i)\,(X/Q)_v\,Q_{iv} \qquad (34)$$

To illustrate the necessity of a 0.7 shielding factor, both equation (23) and equation (34) are used below to calculate the skin dose from noble gas exposure.

Equation (23) - Skin Dose from Noble Gas Exposure

3.17×10^{-8} years/sec	(inverse of seconds per year)
$\times$ [3.06×10^{-4} mrem-m^3/pCi-yr	(Xe-133 β-skin dose factor)
+ 1.1	(tissue to air absorption coefficient)
$\times$ 3.53×10^{-4} mrad-m^3/pCi-yr]	(Xe-133 γ-air dose factor)
$\times$ 1.408×10^{-5} sec/m^3	(ADF for exposure to noble gas)
$\times$ 16.93 Ci/yr	(annual release of Xe-133)
$\times$ 1.00×10^{12} pCi/Ci	(conversion factor for pCi to Ci)
= 5.25×10^{-3} mrem per year	(calculated annual skin dose)

Equation (34) - Skin Dose from Noble Gas Exposure

3.17×10^{-8} years/sec	(inverse of seconds per year)
$\times$ [3.06×10^{-4} mrem-m^3/pCi-yr	(Xe-133 β-skin dose factor)
+ 1.1×0.7 (shielding factor)	(tissue to air absorption coefficient)
$\times$ 3.53×10^{-4} mrad-m^3/pCi-yr]	(Xe-133 γ-air dose factor)
$\times$ 1.408×10^{-5} sec/m^3	(ADF for exposure to noble gas)
$\times$ 16.93 Ci/yr	(annual release of Xe-133)
$\times$ 1.00×10^{12} pCi/Ci	(conversion factor for pCi to Ci)
= 4.37×10^{-3} mrem per year	(calculated annual skin dose)

The skin dose that is calculated according to equation (23) differs from the reported 1993 skin dose estimate of Table 4-1 (i.e., 4.40×10^{-3} mrem) by 20%. By incorporating a 0.7 shielding factor multiplied by the gamma-air dose factor, the difference between the skin dose calculated according to equation (34) and the reported 1993 dose estimate is reduced to 0.6%.

4. Organ Dose Estimate from Airborne Iodine And Particulate Material Effluents

Six pathways are recognized in the SSES ODCM as possible pathways by which individuals may be exposed to airborne radioactive I&PM effluents. However, the estimated dose from I&PM effluents is not necessarily calculated using all six pathways. The ingestion pathways (i.e., vegetation, grass-cow-meat, grass-cow-milk, grass-goat-milk) are considered in dose calculations based on their existence within the monitoring location of the plant site. The inhalation and ground plane exposure pathways are assumed to exist at every location and thus are included in the dose calculation for I&PM effluents (U.S. NRC 1978a). Each year a study is conducted of the land surrounding the SSES site to identify which ingestion pathways are significant so that those pathways that are inconsequential to the estimated organ dose may be eliminated. This annual land use study collects information relating to the production and usage of produce, cow milk, goat milk etc. in the surrounding area of the facility in order to identify the existing ingestion pathways. For example, if the annual land use study reveals that goat milk production and usage within the monitoring area of the facility is insignificant, the grass-goat-milk dose pathway is not considered in calculating the dose estimate (PP&L 1993b). The inhalation, ground plane and vegetation pathways are identified in the 1993 AEWDR as the existing airborne pathways used to calculate the 1993 SSES dose estimate from exposure to radioactive I&PM effluents (PP&L 1993a).

The following equation is given in the SSES ODCM (PP&L 1993a) to estimate the organ dose from radioactive I&PM effluents released to the atmosphere:

$$D_c = 3.17 \times 10^{-8} \sum_{ip} [R_{ip}\, W_v\, Q_{iv}] \tag{26}$$

The parameters of this equation have been previously defined. The SSES 1993 organ dose estimate from I&PM effluents is calculated using the effluent data of reactor unit one. Table 4-3 contains the types and quantities of airborne radionuclides, Q_{iv}, that were released from reactor unit one during the 1993 calendar year. The atmospheric dispersion factors, W_v, used to calculate the 1993 SSES dose estimates are given in Table 4-4. The age group and organ which received the highest dose estimate from exposure to iodine and particulate material in 1993 was calculated to be the lung of a maximally exposed teenager (teen lung). Table 4-5 lists the inhalation, vegetation ingestion, and ground exposure dose factors R_i for teen lung that are given in the SSES ODCM for each radionuclide released from reactor unit one in 1993.

Table 4-5. Teen lung Dose Factors for Inhalation, Ingestion and Ground Exposure Pathways from Radioactive Iodine and Particulate Material

Radionuclide	Inhalation Pathway	Vegetation Ingestion Pathway	Ground Exposure Pathway
	$\frac{\text{mrem-m}^3}{\mu\text{Ci-yr}}$	$\frac{\text{mrem-m}^3\text{-sec}}{\mu\text{Ci-yr}}$	$\frac{\text{mrem-m}^3\text{-sec}}{\mu\text{Ci-yr}}$
H-3	1.27×10^3	$*2.18 \times 10^3$	0.00
Mn-54	1.98×10^6	0.00	1.37×10^9
Co-58	1.34×10^6	0.00	3.75×10^8
Co-60	8.72×10^6	0.00	2.13×10^{10}
Zn-65	1.24×10^6	0.00	7.40×10^8
Sr-90	1.65×10^7	0.00	0.00

*The units for the tritium dose factor is mrem-m^3/μCi-yr.

Using the parameter values from Tables 4-3, 4-4, and 4-5, an example calculation is provided below for each of the three pathways that were analyzed to calculate the 1993 SSES dose estimate for I&PM effluents.

Inhalation Pathway Dose Calculation for Tritium

25.5 Ci	(radionuclide quantity released, Table 4-3)
$\times$ 1.285 $\times$ 10^{-5} sec/m^3	(atmospheric dispersion factor, Table 4-4)
$\times$ 1.27 $\times$ 10^3 mrem-m^3/μCi-yr	(H-3 Inhalation pathway dose factor, Table 4-5)
$\times$ 1 $\times$ 10^6 μCi/Ci	(conversion from curies to picocuries)
$\times$ 3.17 $\times$ 10^{-8} yr/sec	(conversion from seconds to years)
=1.32 $\times$ 10^{-2} mrem	(dose estimate from inhalation of tritium)

Ingestion Pathway Dose Calculation for Tritium

25.5 Ci	(radionuclide quantity released, Table 4-3)
$\times$ 1.285 $\times$ 10^{-5} sec/m^3	(atmospheric dispersion factor, Table 4-4)

$\times$ 2.18 $\times$ 10^3 mrem-m^3/μCi-yr (H-3 vegetation ingestion dose factor, Table 4-5)

$\times$ 1 $\times$ 10^6 μCi/Ci (conversion from curies to picocuries)

$\times$ 3.17 $\times$ 10^{-8} yr/sec (conversion from seconds to years)

=2.26 $\times$ 10^{-2} mrem (dose estimate from vegetation ingestion of H-3)

Ground Exposure Pathway Dose Calculation for Manganese-54

3.77 $\times$ 10^{-4} Ci (radionuclide quantity released, Table 4-3)

$\times$ 1.858 $\times$ 10^{-8} m^{-3} (atmospheric dispersion factor, Table 4-4)

$\times$ 1.37 $\times$ 10^9 mrem-m^3-sec/μCi-yr (Mn-54 ground exposure dose factor, Table 4-5)

$\times$ 1 $\times$ 10^6 μCi/Ci (conversion from curies to picocuries)

$\times$ 3.17 $\times$ 10^{-8} yr/sec (conversion from seconds to years)

=3.04 $\times$ 10^{-4} mrem (dose estimate from ground exposure of Mn-54)

Using a computer spread-sheet program, dose estimate specific to radionuclide and pathway were calculated for teen lung and the results are given in Table 4-6.

Table 4-6. Teen Lung Dose Estimate from Exposure to Iodine and Particulate Material

Radionuclide	Dose from Inhalation (mrem)	Dose from Ingestion (mrem)	Dose from Ground Exposure (mrem)
H-3	1.32×10^{-2}	2.27×10^{-2}	0.00
Mn-54	2.65×10^{-4}	0.00	3.04×10^{-4}
Co-58	1.32×10^{-5}	0.00	6.11×10^{-6}
Co-60	5.80×10^{-4}	0.00	2.35×10^{-3}
Zn-65	4.09×10^{-5}	0.00	4.04×10^{-5}
Sr-90	1.52×10^{-6}	0.00	0.00
Subtotal	1.41×10^{-2}	2.27×10^{-2}	2.70×10^{-3}
Total Teen Lung Dose Estimate			3.96×10^{-2}

The calculated dose estimate given in Table 4-6 for teen lung does not equal the 1.69×10^{-4} mrem dose estimate reported in the 1993 AEWDR for teen lung. This discrepancy reveals an error in the teen lung dose estimate that was reported in the 1993 AEWDR. It was discovered that the dose contribution from the vegetation pathway was inadvertently omitted from the organ dose calculated by PP&L for exposure to airborne radioactive I&PM effluents.[5] Thus, the dose estimate given in the 1993 AEWDR for teen lung is the sum of dose contributions from the inhalation and ground exposure pathways only:

$$1.41 \times 10^{-2} \text{ mrem} + 2.70 \times 10^{-3} \text{ mrem} = 1.68 \times 10^{-2} \text{ mrem.}$$

Using the GASPAR computer program, the organ dose from exposure to radioactive iodine and particulate material has been recalculated by PP&L to incorporate the vegetation pathway as indicated in the 1993 AEWDR. The teen lung dose estimate corrected to include the vegetation pathway is calculated by the GASPAR program to be 3.98×10^{-2}. This revised teen lung dose estimate is within 1% of the hand-calculated teen lung dose estimate of Table 4-6. However, with the inclusion of the vegetation pathway, the highest organ dose estimate is no longer the lungs of the teen age group but the lungs of the child age group, 5.11×10^{-2} mrem. The 1995 AEWDR will contain a disclaimer notifying the NRC of the corrected dose estimates.[6] The corrected organ dose estimate from I&PM effluents is still much less than the regulatory limit of 15 mrem per year. The child dose estimate has also been calculated according to the methodology of the ODCM and the results are displayed in Table 4-7.

---

[5]Personal communication with Robert Barclay of Pennsylvania Power & Light Company, 23 January, 1995.
[6]Personal communication with Robert Barclay of Pennsylvania Power & Light Company, 23 January, 1995.

Table 4-7. Child Lung Dose Estimate from Exposure to Iodine and Particulate Material

Radionuclide	Dose from Inhalation (mrem)	Dose from Ingestion (mrem)	Dose from Ground Exposure (mrem)
H-3	1.17×10^{-2}	3.57×10^{-2}	0.00
Mn-54	2.12×10^{-4}	0.00	3.04×10^{-4}
Co-58	1.09×10^{-5}	0.00	6.11×10^{-6}
Co-60	4.70×10^{-4}	0.00	2.35×10^{-3}
Zn-65	3.28×10^{-5}	0.00	4.04×10^{-5}
Sr-90	1.37×10^{-6}	0.00	0.00
Subtotal	1.24×10^{-2}	3.57×10^{-2}	2.70×10^{-3}
Total Child Lung Dose Estimate			5.08×10^{-2}

The organ dose estimate given in Table 4-7 for child lung is within 1 percent of the child lung dose calculated using the GASPAR computer program. Agreement between the hand-calculated dose estimate for child lung and the dose estimate calculated using the GASPAR computer program indicates that the dose methodology given in the ODCM to estimate organ doses from I&PM effluents adequately describes the actual methodology used to produce annual SSES dose estimates.

Although the methodology given in the ODCM for exposure to radioactive I&PM effluents agrees with the GASPAR dose model, the tritium dose factors given in the ODCM and those used by the GASPAR dose model are not consistent with the corresponding tritium dose factors used by the LADTAP II dose model. The LADTAP II computer model used by SSES to calculated liquid dose estimates employs tritium dose factors that were derived using a 1.0 quality factor for beta radiation. The tritium dose factors used in the GASPAR computer code, however, are based on the 1.7 quality factor recommended in ICRP 2 for low energy beta radiation.

The GASPAR computer program employed by SSES to calculate annual dose estimates from airborne effluents has been superseded by GASPAR II. One difference between the original version of GASPAR and GASPAR II is the quality factor assigned to low energy beta radiation such as that emitted from tritium. As previously stated, the 1.7 quality factor originally recommended in ICRP 2 for low energy beta radiation is now taken to be 1.0. Consistent with LADTAP II, the tritium inhalation and ingestion dose factors of GASPAR II are based on a beta quality factor of 1.0. The tritium dose factors of the original version of GASPAR, on the other hand, remain based on a beta quality factor of 1.7 (U.S. NRC 1978). Therefore, the airborne dose estimates for tritium that are calculated using the GASPAR computer code are inconsistent with the tritium dose estimates that are calculated by LADTAP II. Furthermore, the tritium dose factors given in the SSES ODCM for both the liquid and airborne pathways are also based on a 1.7 beta quality factor.

The 1993 SSES dose estimates for I&PM effluents would be significantly affected by a 1.7 reduction in the tritium dose factors. The dose contribution from tritium constitutes 90 percent of the teen lung dose given in Table 4-6 and 93 percent of the child-lung dose given in Table 4-7. The teen lung and child lung dose estimates have been recalculated using tritium dose factors given in the ODCM that have been reduced by 1.7 to reflect a 1.0 beta quality factor and the results are given in Table 4-8.

Table 4-8. Teen and Child Lung Dose Estimates Recalculate Using Tritium Dose Factors Based on a Quality Factor of 1.0

Nuclide	Teen Lung Dose Estimate			Child Lung Dose Estimate		
	Inhalation Dose (mrem)	Ingestion Dose (mrem)	Ground Exposure Dose (mrem)	Inhalation Dose (mrem)	Ingestion Dose (mrem)	Ground Exposure Dose (mrem)
H-3	7.79×10^{-3}	1.34×10^{-2}	0.00	6.87×10^{-3}	2.10×10^{-2}	0.00
Mn-54	2.65×10^{-4}	0.00	3.04×10^{-4}	2.12×10^{-4}	0.00	3.04×10^{-4}
Co-58	1.32×10^{-5}	0.00	6.11×10^{-6}	1.09×10^{-5}	0.00	6.11×10^{-6}
Co-60	5.80×10^{-4}	0.00	2.35×10^{-3}	4.70×10^{-4}	0.00	2.35×10^{-3}
Zn-65	4.09×10^{-5}	0.00	4.04×10^{-5}	3.28×10^{-5}	0.00	4.04×10^{-5}
Sr-90	1.52×10^{-6}	0.00	0.00	1.37×10^{-6}	0.00	0.00
	8.69×10^{-3}	1.34×10^{-2}	2.70×10^{-3}	7.60×10^{-3}	2.10×10^{-2}	2.70×10^{-3}
		Total	2.48×10^{-2}		Total	3.13×10^{-2}

The dose estimates of Table 4-8 for the lungs of both teen and child are approximately 40 percent less than the lung dose estimates given in Table 4-6 and 4-7. PP&L should use tritium dose factors based on a beta quality factor of 1.0 instead of 1.7 to calculate airborne dose estimates since this would lead to lower dose estimates from airborne I&PM exposure and would be consistent with the LADTAP II dose code employed for liquid dose estimates.

D. Summary of Manually Calculated Dose Estimates

Table 4-9 summarizes the 1993 dose estimates that were calculated according to the methodology given in the ODCM for all effluent types. For comparison, Table 4-9 also lists the dose estimates given in Table 4-1, which are the dose estimates reported in the 1993 AEWDR.

Table 4-9. Summary of Hand-Calculated Dose Estimates

Effluent	Age Group	Applicable Organ	Manually Calculated 1993 Dose Estimates (mrem)	Dose Estimates Reported in the 1993 AEWDR (mrem)	Difference
Liquid	Teen	Total Body	7.34×10^{-3}	6.96×10^{-3}	5.5%
Liquid	Adult	GI-LLI	1.51×10^{-3}	1.49×10^{-2}	1.3%
Noble Gas	N/A	γ-air dose	2.67×10^{-3}	2.68×10^{-3}	0.3%
Noble Gas	N/A	β-air dose	7.93×10^{-3}	7.96×10^{-3}	0.4%
Noble Gas	N/A	Total body	2.22×10^{-3}	1.56×10^{-3}	42.3%
Noble Gas	N/A	Skin	5.25×10^{-3}	4.40×10^{-3}	19.3%
I&PM	Teen	Lung	3.96×10^{-2}	1.69×10^{-2}	134%
I&PM	Child	Lung	5.08×10^{-2}	5.11×10^{-2}	0.6%

Differences between the hand-calculated and reported dose estimates of Table 4-9 indicate that a number of differences exist between the methodology given in the SSES

ODCM and the dose models utilized by PP&L. As previously mentioned, the LADTAP II computer dose model used by PP&L to calculate liquid dose estimates differs from the methodology given in the ODCM with regards to the dose factors used for tritium ingestion. The tritium dose factors given in the ODCM are based on a 1.7 beta quality factor, whereas, the tritium dose factors used by the LADTAP II computer model are based on a 1.0 beta quality factor.

The similarity between the hand-calculated and reported gamma-air and beta-air dose estimates of Table 4-9 indicates that the ODCM methodology agrees with the actual methodology used by PP&L to calculate gamma-air and beta-air dose estimates from noble gas exposure. The differences between the hand-calculated and reported noble gas dose estimates of Table 4-9 for total body and skin are attributed to the 0.7 shielding factor that is omitted from the noble gas equations given in the ODCM. The GASPAR computer dose model and the equations given in Reg. Guide 1.109 to estimate total body and skin dose from noble gas exposure include a 0.7 shielding factor, whereas, the noble gas equations given in the ODCM for total body and skin do not.

The inadvertent omission of the vegetation pathway dose from the reported organ dose estimate for airborne I&PM effluents was revealed by the significant difference between the hand-calculated and reported teen lung dose estimate. The organ dose for I&PM effluents recalculated to include the vegetation pathway identified the highest organ dose to be the lungs of a maximum exposed child.

The hand-calculated dose estimates have been recalculated to include a 1.7 reduction in the tritium dose factors for the liquid dose estimate and to include a 0.7 shielding factor for the noble gas total body and skin dose equations. Results of the revised hand-calculated dose estimates are listed in Table 4-10. Table 4-10 also lists the teen lung and child lung organ dose estimates that were recalculated by PP&L using the GASPAR computer code.

Table 4-10. Summary of Revised Hand-Calculated Dose Estimates

Effluent	Age Group	Applicable Organ	Manually Calculated 1993 Dose Estimates (mrem)	Dose Estimates Reported in the 1993 AEWDR (mrem)	Difference
Liquid	Teen	Total Body	6.85×10^{-3}	6.96×10^{-3}	1.6%
Liquid	Adult	GI-LLI	1.44×10^{-3}	1.49×10^{-2}	3.3%
Noble Gas	N/A	γ-air dose	2.67×10^{-3}	2.68×10^{-3}	0.3%
Noble Gas	N/A	β-air dose	7.93×10^{-3}	7.96×10^{-3}	0.4%
Noble Gas	N/A	Total body	1.55×10^{-3}	1.56×10^{-3}	0.6%
Noble Gas	N/A	Skin	4.37×10^{-3}	4.40×10^{-3}	0.7%
I/P	Teen	Lung	3.96×10^{-2}	*3.98×10^{-2}	0.5%
I/P	Child	Lung	5.08×10^{-2}	*5.11×10^{-2}	0.6%

*Dose estimates recalculated by PP&L to include vegetation pathway

V. IDENTIFICATION OF KEY PARAMETERS AND ASSUMPTIONS

Dosimetric models used to calculate dose to an organ or body tissue employ numerous model parameters specific to different environmental pathways and radionuclides. Model parameters which have the greatest influence on the prediction of a dosimetric model are identified through a sensitivity analysis. The purpose of a sensitivity analysis is to eliminate model parameters from further review that contribute little to the final dose estimate and narrow the focus on the most sensitive parameters in the model (U.S. NRC 1985).

To limit the scope of this review, a sensitivity analysis will only be conducted for those parameters of the liquid pathways that are site-specific. Site-specific parameters are those parameters whose values are derived from information obtained at SSES. Values for the remaining parameters are provided by regulatory guidance or other reference sources and are derived from information and experimental data that is not indigenous to the facility site; these values are referred to as default values. A large number of parameters use default values, and this review does not include an analysis of default values or the source information from which the default values were derived. A complete sensitivity analysis would extend beyond the scope of this review.

The results of the sensitivity analysis will show that the dilution factor is the most sensitive site specific parameter for the liquid pathway equations. The derivation and selection of the dilution factors that were used to produce the 1993 dose estimates will then be examined. To reduce the number of calculations, the parameters of the liquid pathways will be examined using those radionuclides and pathways that significantly contribute to the liquid dose estimates of Table 4-1 which were the total body dose estimate of a teen (teen total body) and the lower large intestine of the adult age group (adult GI-LLI).

The 1993 SSES dose estimates from radioactive liquid effluents consist of dose contributions from various radionuclides summed over three different pathways (i.e., potable water, fish ingestion, and shore exposure). Table 5-1 illustrates the percent of total dose contributed by each radionuclide specific to the three liquid pathways for both teen total body and adult GI-LLI.

Table 5-1. Percent of Total Dose from Each Radionuclide Specific to Pathway

Teen - Total Body				Adult - GI-LLI			
Nuclide	Fish Ingestion Pathway	Potable Water Pathway	Shore Exposure Pathway	Nuclide	Fish Ingestion Pathway	Potable Water Pathway	Shore Exposure Pathway
H-3	6%	10%	0%	H-3	4%	7%	0%
Na-24	<0.3%	<0.3%	<0.3%	Na-24	<0.3%	<0.3%	<0.3%
Cr-51	<0.3%	<0.3%	<0.3%	Cr-51	<0.3%	<0.3%	<0.3%
Mn-54	5%	<0.3%	2%	Mn-54	42%	<0.3%	<0.3%
Fe-55	1%	<0.3%	<0.3%	Fe-55	1%	<0.3%	<0.3%
Fe-59	1%	<0.3%	<0.3%	Fe-59	3%	<0.3%	<0.3%
Co-58	<0.3%	<0.3%	<0.3%	Co-58	1%	<0.3%	<0.3%
Co-60	5%	<0.3%	39%	Co-60	21%	1%	3%
Zn-65	27%	<0.3%	<0.3%	Zn-65	18%	<0.3%	<0.3%
Sr-90	<0.3%	<0.3%	<0.3%	Sr-90	<0.3%	<0.3%	<0.3%
Sr-92	<0.3%	<0.3%	<0.3%	Sr-92	<0.3%	<0.3%	<0.3%
Mo-99	<0.3%	<0.3%	<0.3%	Mo-99	<0.3%	<0.3%	<0.3%
Tc-99m	<0.3%	<0.3%	<0.3%	Tc-99m	<0.3%	<0.3%	<0.3%
Ag-110	<0.3%	<0.3%	<0.3%	Ag-110	<0.3%	<0.3%	<0.3%
Te-131	<0.3%	<0.3%	<0.3%	Te-131	<0.3%	<0.3%	<0.3%
I-131	<0.3%	<0.3%	<0.3%	I-131	<0.3%	<0.3%	<0.3%
Cs-134	<0.3%	<0.3%	<0.3%	Cs-134	<0.3%	<0.3%	<0.3%
Cs-137	3%	<0.3%	<0.3%	Cs-137	<0.3%	<0.3%	<0.3%
Ce-141	<0.3%	<0.3%	<0.3%	Ce-141	<0.3%	<0.3%	<0.3%
Nd-147	<0.3%	<0.3%	<0.3%	Nd-147	<0.3%	<0.3%	<0.3%
% of Total	49%	10%	41%	% of Total	89%	8%	3%

Table 5-1 identifies tritium, manganese-54, cobalt-60, zinc-65 and cesium-137 as those radionuclides that are significant in contributing to the total body dose of a teenager. Radionuclides that contribute significantly to the adult GI-LLI dose are identified as tritium, manganese-54, iron-59, cobalt-60, and zinc-65. Table 5-1 also demonstrates that for different radionuclides, different pathways predominate. For example, cobalt-60 principally contributes to the total body dose of a teen via the shoreline exposure pathway, whereas zinc-65 contributes primarily via the potable water pathway. From this information, key radionuclides and pathways significant to the total dose are chosen to be included in the sensitivity analysis.

Sensitivity analysis is an analytical tool that gauges the influence each parameter has on the outcome of model predictions. The value of each parameter is varied by specific increments, holding all other parameters the same, and the degree of outcome fluctuation is observed (U.S. NRC 1985). With the exception of the exponential expression, the parameters of the liquid effluent models used by PP&L are arranged in a simple multiplicative function (i.e., $A \times B \times C$ etc.). Deviating each parameter of a multiplicative function in a sensitivity analysis by a fixed increment from its nominal value would not distinguish the relative sensitivity of each parameter. Without knowledge of possible parameter values, each parameter in a multiplicative function would appear equally sensitive. Comparing parameters by their range of possible values allows parameters in a multiplicative model to be ranked according to their sensitivity. The parameter sensitivity of mathematically simple models can be assessed by simply observing the expected range of each parameter in lieu of the model outcome. This is not possible for models that incorporate complex mathematical functions, such as logarithms and integral functions (U.S. NRC 1983).

Reg. Guide 1.109 (U.S. NRC 1977a) gives the following model equations for the three liquid pathways:

Potable Water

$$R_{apj} = 1100 \frac{U_{ap}}{DF_p \quad F} \sum_i Q_i \quad D_{aipj} \quad \exp(-\lambda_i t_p) \tag{2}$$

Fish Ingestion

$$R_{apj} = 1100 \frac{U_{ap}}{DF_p \quad F} \sum_i Q_i \quad B_{ip} \quad D_{aipj} \quad \exp(-\lambda_i t_p) \tag{4}$$

Shore Exposure

$$R_{apj} = 110{,}000 \frac{U_{ap} W}{DF_p \quad F} \sum_i Q_i T_i D_{aipj} [\exp(-\lambda_i t_p)][1 - \exp(\lambda_i t_b)] \tag{6}$$

The values given in Reg. Guide 1.109 for the usage factor U_{ap}, bioaccumulation factor B_{ip} and the dose conversion factor D_{aipj} are used by PP&L to calculate annual dose estimates. Values for the soil activity time constant t_b and the width factor W of the shore exposure pathway, are also taken from Reg. Guide 1.109. As previously stated, the intent of this review does not include a review of Reg. Guide 1.109 default values.

The dilution factor DF_p, the blow-down rate F, and the transit time t_p are site-specific values provided in the AEWDR. Table 5-2 presents maximum and minimum values for the site-specific parameters of the liquid pathways and the nominal value used to calculate the 1993 dose estimates.

Table 5-2. Maximum, Minimum and Nominal Site-Specific Values for Liquid Effluent Pathways

Parameter	Minimum value	Nominal value	Maximum value
Blow Down Rate for all Liquid Pathways			
F (cfs)	14.1	15.8	18.4
Fish Ingestion Pathway			
*DF_D	15.9	15.9	165.6
t_D (hours)	0.01	25.0	200.00
Shore Exposure Pathway			
*DF_D	15.9	15.9	165.6
t_D (hours)	0.01	1.00	200.0
Potable Water Pathway			
*DF_D	280.9	413.2	1285.3
t_D (hours)	0.01	24.7	200.0

*Dimensionless

The blow-down rate is reported by control room instrumentation. The value given in Table 5-2 for the minimum blow-down rate is the lowest quarterly blow-down rate reported in 1993; the maximum value is the highest reported quarterly blow-down rate. The nominal value was derived by averaging the blowdown rate during periods of release over all four quarters of 1993.

Dilution and transit times for all liquid pathways were determined by a tracer dye release study conducted in 1985 by the Sutron Corporation (Sutron 1985). The minimum and maximum dilution factors given in Table 5-2 for each liquid pathway are the minimum and maximum dilution values determined by the Sutron study. The transit times determined by the Sutron study for the potable water pathway range from 9.3 to 141.2 hours (Sutron 1985). A range of transit times for the fish ingestion and shore exposure pathways is not given in the information provided by PP&L; only nominal values are provided. Nonetheless, the minimum and maximum values for the transit time parameter were selected to differ by four orders of magnitude for all pathways to illustrate the small influence of this parameter in the sensitivity analysis.

Table 5-3 lists the index of sensitivity for each site-specific parameter respective to each relevant radionuclide for both teen total body and adult GI-LLI. The sensitivity index was obtained for each parameter by the following calculation (U.S. NRC 1985):

$$\text{Sensitivity index} = 1 - \text{Dose}_{min}/\text{Dose}_{max}$$

Dose_{min} and Dose_{max} for a specific parameter are the minimum and maximum dose values respectively calculated over the expected range of the parameter while all other parameters values remain the same. The closer the sensitivity index approaches 1 the greater the sensitivity of that parameter. Parameters with sensitivity indexes less than 0.01 are considered relatively insensitive (U.S. NRC 1985).

Table 5-3. Sensitivity Analysis of the Site-specific Parameters for Liquid Pathways

Sensitivity Index							
Teen Total Body							
Parameter	Fish Ingestion					Shore Exposure	
	H-3	Mn-54	Co-60	Zn-65	Cs-137	Mn-54	Co-60
DF_p	0.90	0.90	0.90	0.90	0.90	0.90	0.90
F	0.23	0.23	0.23	0.23	0.23	0.23	0.23
t_p	0.00	0.02	0.00	0.02	0.00	0.02	0.00
Adult GI-LLI							
Parameter	Fish Ingestion					Shore Exposure	
	H-3	Mn-54	Co-60	Zn-65	Fe-59	Co-60	
DF_p	0.90	0.90	0.90	0.90	0.90	0.90	
F	0.23	0.23	0.23	0.23	0.23	0.23	
t_p	0.00	0.02	0.00	0.02	0.12	0.00	

The sensitivity indices for the site specific parameters of the potable water pathway are shown in Table 5-3 for the fish ingestion pathway. Since the potable water equation is similar to the fish ingestion equation and since the potable water pathway contributes little to the total dose of both teen total body and adult GI-LLI, it was not included in Table 5-3. Table 5-3 reveals that the dilution factor displays the greatest influence on the outcome of the liquid effluent dose models.

The dilution factors used by PP&L to calculate liquid dose estimates were derived from a tracer dye study of the mixing characteristics of the Susquehanna River conducted by the Sutron Corporation in 1985. Regulatory Guide 1.113, Estimating Aquatic Dispersion of Effluents from Accidental and Routine Releases for the Purposes of Implementing Appendix I, describes calculational models and suggested methods acceptable to the NRC for characterizing the dispersion of liquid effluents into surface water bodies (U.S. NRC 1977d). A tracer release study is recognized in Reg. Guide 1.113 as an acceptable method of deriving aquatic dispersion factors that "can provide accurate predictions without need for a model" (U.S. NRC 1977d).

The dilution factor used by PP&L to estimate annual doses from the fish ingestion and shore exposure pathways is different from the dilution factors used for the potable water pathway. A fixed dilution factor of 15.9 is used by PP&L to calculate annual dose estimates for the fish ingestion and shore exposure pathways, whereas variable dilution factors are used for the potable water pathway. Table 5-4 presents the information obtained in the 1985 study that was used to derive the dilution factor for the fish ingestion and shore exposure pathways (Sutron 1985).

Table 5-4. Observed Dilution of Peak Concentrations Released in Blowdown Line

Cross Section No.	Station (miles)	Low Flow Peak Conc. (ppb)	Low Flow Mixing Ratio (%)	Medium Flow Peak Conc. (ppb)	Medium Flow Mixing Ratio (%)	High Flow Peak Conc. (ppb)	High Flow Mixing Ratio (%)
Blowdown Line	N/A	430	100.000	1570.00	100.000	2730.00	100.000
1	0.1	27.00	6.279	20.00	1.274	16.50	0.604
2	1.3	9.20	2.140	11.99	0.764	7.04	0.258
3	3.5	7.50	1.744	9.00	0.573	5.40	0.198
4	5.9	7.60	1.767	8.40	0.535	4.94	0.181
5	8.3	5.00	1.163	6.30	0.401	3.52	0.129
6	11.1	6.02	1.104	5.89	0.375	3.75	0.137
7	15.0	- -	- -	- -	- -	3.90	0.143
8	18.4	4.80	1.116	5.31	0.338	3.30	0.121
9	21.7	4.30	1.000	5.01	0.319	3.08	0.113
10	25.3	- -	- -	- -	- -	2.70	0.099
11	30.5	3.50	0.814	4.51	0.287	2.54	0.093
Danville Water Works	29.5	2.90	0.674	1.70	0.108	1.14	0.073
Merck Co.	32.0	2.85	0.663	1.55	0.099	0.80	0.051

The Sutron Study consisted of three dye release studies: a low flow study at a 1,950 cfm river discharge, a medium flow study at a 15,400 cfm river discharge, and a high flow study at a 46,550 cfm river discharge. For each study a tracer dye, Rhodamine WT, was injected into the blowdown line. Dye concentration samples were collected from the Susquehanna River at intervals downstream as far as Danville, Pennsylvania. In addition, velocity and river depth data were recorded with each sample taken. The amount of dye released into the blowdown line was normalized to an initial peak concentration in parts per billion (ppb) and varied for each flow study; the initial dye releases for the low, medium and high flow rates are given as normalized peak concentrations of 430, 1570, and 2730 ppb, respectively. The mixing ratio values of Table 5-4 are obtained by dividing the measured peak concentration at each cross-section station by the initial peak concentration in the blowdown line. An example of this for the initial mixing zone (cross-section station 1) and for low river flow conditions is: 27.00 ppb $\div$ 430 ppb $\times$ 100% = 6.279% (Sutron 1985).

Two conservative assumptions are made by PP&L in selecting the dilution factor used for the fish ingestion and shore exposure pathways. The first conservative assumption assumes that all aquatic food harvesting and shore exposure occur at the initial mixing zone, the station closest to the blowdown line (i.e., cross-section no. 1). The second supposition assumes a low river flow rate throughout the year. From Table 5-4, the mixing ratio that corresponds to a low river flow rate and initial mixing zone conditions is 6.279% or 0.06279. Inverting the mixing ratio of 0.06279 yields 15.9 which is the dilution factor used for both the fish ingestion and shore exposure pathways (PP&L 1993a).

Instead of selecting a fixed dilution factor based on conservative assumptions, the dilution factors used by PP&L for the potable water pathway are variable and are derived as a function of river depth at Danville, PA. Water drawn from the Susquehanna River is used to provide drinking water for the downstream community of Danville, PA. The Sutron Corporation has developed an empirical model of the dilution process at Danville, PA based on the dye concentrations that were measured at the Danville location for varying river levels. Appendix B, Table B-1 lists the dilution factors specific to river depth for the potable water pathway that were derived using the empirical dilution model developed by the Sutron Corporation (Sutron 1985).

Measurements of the river depth at Danville, PA are taken throughout the year and averaged per quarter year. Dose estimates for the potable water pathway are calculated per quarter year and then summed to obtain an annual dose estimate. The average river level per quarter year determines the dilution factor that is used to calculate the dose estimate for each quarter, respectively. For example, the average river level reported for the second quarter in 1993 is 10.1 ft; from Table B-1, this river level corresponds to a dilution factor of 1285.3 which is the dilution factor used to calculate the potable water dose estimate for the second quarter in 1993.

In contrast to the variable method of selecting dilution factors for the potable water pathway, PP&L takes a decidedly more conservative approach in selecting the dilution factor used for the fish ingestion and shore exposure pathways. The 15.9 dilution factor used by PP&L for the fish ingestion and shore exposure pathways is the most conservative dilution factor derived by the Sutron study. The dilution factors selected for the potable water pathway are based on measured depths of the Susquehanna River at the Danville location and are considered more representative of the actual dilution process. The tracer dye release study employed by SSES provides more realistic dilution values for all three liquid pathways than could be obtained from other calculation models given in Reg. Guide 1.113.

VI. CHANGES IN INTERNAL DOSE FACTORS

This section focuses on the internal dose factors used to calculate the 1993 SSES dose estimates from effluent releases. Reg. Guide 1.109 provides four different types of dose factors specific to various modes of exposure. The calculation of dose estimates from external exposure to radioactive noble gases and contaminated ground uses dose factors given in Reg. Guide 1.109 - Table B-1, "Dose Factors for Exposure to a Semi-Infinite Cloud of Noble Gases," and Table E-6, "External Factors for Standing on Contaminated Ground," respectively. The calculation of internal dose estimates from inhaled or ingested radionuclides for a hypothetical adult, teen, child, or infant uses dose factors given in Reg. Guide 1.109 Table E-7 through E-10, "Inhalation Dose Factors," and Table E-11 through E-14, "Ingestion Dose Factors."

As previously stated, the internal dose factors of Reg. Guide 1.109 are primarily derived from the dose models given in ICRP 2 (U.S. NRC 1977c). Since the publication of ICRP 2 in 1959, scientist and researchers have achieved a better understanding of "the effects of radiation on the body [and] . . . the uptake and retention of radioactive materials in body tissues" (ICRP 1978). In addition, improvements have been made regarding radioactive decay schemes (ICRP 1977). The recommendations and the biological and physical data of ICRP 2 have been superseded by ICRP Publication 26 (ICRP 26), Recommendation of the International Commission on Radiological Protection, adopted in 1977, and ICRP Publication 30 (ICRP 30), Limits for intakes of Radionuclides by Workers, adopted in 1978 (ICRP 1977 and 1978).

The NRC has responded to the new recommendations of ICRP 26 and ICRP 30 with the revision of title 10 part 20 of the Code of Federal Regulations (10 CFR 20), "Standards for Protection Against Radiation"(U.S. NRC 1993). The dose factors of Reg. Guide 1.109, however, remain based on the dose methodology of ICRP 2. This section examines how the revised dose models given in ICRP 30 might effect the calculation of annual individual dose estimates.

First, this section outlines differences between the internal dose models of ICRP 2 and ICRP 30 that are central to the calculation of internal dose factors. Second, using ICRP 30 dose equivalent values, the highest adult total body and organ dose estimate from liquid effluents, the highest adult organ dose estimate from airborne I&PM material effluents and the total body and skin dose estimates from noble gas effluents will be recalculated to reflect the revised dose models given in ICRP 30.

Although more recent publications have been issued by the ICRP which supersede data given in ICRP 30, this review will restrict its focus on the data and dose models given in ICRP 30 in relation to those of ICRP 2 and the possible effect that the ICRP 30 dose models would have on the annual SSES dose estimates. The objective of restricting this study to a comparison of ICRP 30 and ICRP 2 is to limit the scope of this review to those ICRP publications that are fundamental to current NRC regulations and guidance.

A. Differences Between Internal Dose Models of ICRP 2 and ICRP 30

Both ICRP 2 and ICRP 30 present methodologies for calculating dose to organs and body tissues from inhaled or ingested radioactive contaminants. The differences between ICRP 2 and ICRP 30 regarding internal dose calculations recognized in this review may be grouped into three categories:

1) differences in dosimetric models,

2) differences in absorbed effective energy, and

3) differences in regards to the attribution of risk to tissues of the body.

1. Differences in Dosimetric Models of ICRP 2 and ICRP 30

a. ICRP 2 Dosimetric Models

The dosimetric models given in ICRP 2 to describe the distribution and retention of a radionuclide after ingestion or inhalation are simple in comparison to those of ICRP 30. In ICRP 2 the distribution of an inhaled or ingested radioisotope is characterized by the following distribution fractions (ICRP 1959):

f_w = fraction of the radionuclide quaintly ingested that is retained in the critical organ. For ingestion of soluble compounds:

$$f_w = f_1 \times f'_2;$$

f_a = fraction of the radionuclide quantity inhaled that is retained in the critical organ. For inhalation of soluble compounds:

$$f_a = (0.25 + 0.50 \times f_1) f'_2;$$

f_1 = fraction of the radionuclide quantity passing from GI tract to blood; and

f'_2 = fraction of the radionuclide quantity passing from blood to critical organ.

Values for ingestion and inhalation distribution fractions specific to a radionuclide and critical organ are provided in Table 12, "Biological and Related Physical Constants", of ICRP 2 (ICRP 1959). In ICRP 2, it is assumed that 25% of inhaled radioactivity is exhaled, another 25% is deposited in the lower respiratory tract and the remaining 50% is assumed to be deposited in the upper respiratory tract and translocated via mucociliary clearance and ingestion to the gastrointestinal (GI) tract. Soluble radionuclides deposited in the lower respiratory region are assumed to be transferred directly to the blood. Therefore, the fraction denoting the amount of inhaled soluble radioactivity transferred to a critical organ is: $f_a = (0.25 + 0.50 \times f_1)f'_2$. The transfer of insoluble radionuclides is considered negligible and is not addressed by ICRP 2 except for the GI tract and the lung. For insoluble materials, f_a equals 0.62 for the GI tract and f_a equals 0.12 for the lung (ICRP 1959).

The distribution fractions given in ICRP 2 denote how much of the radioactivity initially ingested or inhaled reaches the critical organ. The intake rate of a radionuclide into a body organ and elimination of that radionuclide from the organ may be described by the following equation given in ICRP 2 (ICRP 1959):

$$\frac{d\,q(t)}{dt} + \lambda\, q(t) = P \qquad (35)$$

Where

$q(t)$ = radioactivity in the critical organ at time t, in μCi or Ci;

P = uptake rate of the radionuclide by the critical organ, in units of μCi/day or Ci/yr; and

λ = effective decay constant of the radionuclide in the given organ, in days^{-1} or years^{-1}.

The effective decay constant λ is derived from the following equations (ICRP 1959):

$$\lambda = 0.693 / T_{½,e}$$

$$T_{½,e} = (T_{½,p}^{-1} + T_{½,b}^{-1})^{-1}$$

Where

$T_{½,e}$ = effective half-life of the radionuclide in the body, in hours^{-1};

$T_{½,p}$ = physical half-life of the radionuclide, in hours^{-1}; and

$T_{½,b}$ = biological half-life of the radionuclide in the body, in hours^{-1}.

Table 12 of ICRP 2 also provides physical and biological half-lives for each radionuclide specific to relevant organs and body tissues. The integral solution for equation (35) may be given by Bernoulli's equation (Gellert 1975):

$$\frac{dq(t)}{dt} = -\lambda q(t) + y(t) \qquad (36)$$

Where the general solution is:

$$q(t) = \exp^{-\int \lambda\, dt} \left[\int y(t) \exp^{\int \lambda\, dt}\, dt + q(t) \right] \qquad (37)$$

Using the solution for Bernoulli's equation the integrated solution for equation (35) is illustrated below:

$$\frac{dq(t)}{dt} = -\lambda\, q(t) + P \qquad (35)$$

$$q(t) = \exp^{-\int \lambda\, dt} \left[\int P \exp^{\int \lambda\, dt}\, dt + q(0) \right] \qquad (37)$$

This solution may be simplified by assuming that the radioactivity taken into an organ over a period of time occurs instantaneously at time $t = 0$. The activity taken into an organ is equal to the radioactivity initially consumed A_0 multiplied by the appropriate inhalation or ingestion distribution fraction: $A_0 f$ (ICRP 1959). Therefore, at time $t = 0$, $q(t)$ would be set equal to $A_0 f$ and P would be set equal to zero. Given this assumption, the radioactivity in the critical organ at time t may be expressed as:

$$q(t) = A_0 f \exp^{-\lambda t} \qquad (38)$$

Internally deposited radionuclides continually irradiate body tissues for as long as they exist in the body. In order to estimate the dose that an organ receives from internally deposited radionuclides, it is necessary to calculate the number of nuclear transformations that occur during the time the activity is retained in the critical organ. The number of radioactive transformations that occur within a specified time period is determined by integrating equation (38) over a prescribed time period. Integrating the radioactivity retained in a body tissue over a specific time period after consumption incorporates the dose component that will result from future irradiation of the body tissue (U.S. NRC 1977a). A 50 year time period is recommended in ICRP 2, ICRP 30 and Reg. Guide 1.109 for evaluating the committed dose per unit intake (ICRP 1959 and ICRP 1978 and U.S. NRC 1977a).

The integral expression for calculating the number of transformation that result over a period of 50 years from internally deposited radionuclides may be given as:

$$\int_0^{50y} A_0 f e^{(-\lambda t)} dt = A_0 f \left[\frac{1 - e^{(-\lambda \times 50y)}}{\lambda} \right] \qquad (39)$$

b. ICRP 30 Dosimetric Models

The models given in ICRP 30 which characterize the distribution and "retention of a radionuclide in a body tissue following its ingestion or inhalation ... are based on the assumption that the body consists of a number of separate compartments" (ICRP 1978). For example, the respiratory system and the gastrointestinal (GI) tract are considered to be the initial compartments for the inhalation and ingestion routes; the body fluids are considered to be the transfer compartment and each organ or body tissue is assumed to consist of one or more compartments. The transfer of a radionuclide from one compartment to another is "governed by first order kinetics" (ICRP 1978).

Instead of the distribution fractions of ICRP 2, the dosimetric models introduced in ICRP 30 employ a series of differential equations to characterize the distribution of a radionuclide through the body following ingestion or inhalation. Separate dosimetric models are given in ICRP 30 for the gastrointestinal tract, respiratory system, and for bone tissues (ICRP 1978). The dosimetric models and differential equations given in ICRP 30 for the gastrointestinal tract and respiratory system are summarized in Appendix D of this review.

The integral solutions for the differential equations of the gastrointestinal tract and respiratory system are detailed in ICRP 30, "Appendix: Exact and Approximate Solutions of the Compartmental Models Used in This Report" (ICRP 1978). The integral solutions given in ICRP 30 are used to obtain the number of transformations U_S that occur with in a source organ S over a 50 year period following consumption of a radionuclide. For each radionuclide, ICRP 30 provides precalculated values of U_S per unit intake specific to source organ (ICRP 1978).

2. Differences in Absorbed Effective Energy of ICRP 2 and ICRP 30

a. ICRP 2 Absorbed Effective Energy

Every nuclear transformation that occurs within an organ tissue is associated with an amount of energy absorbed by the surrounding body tissues. The energy parameter given in ICRP 2 for estimating internal dose is referred to as the effective energy. The effective energy parameter for a radionuclide is the sum of energy of each radiation type emitted per transformation modified by an appropriate relative biological effectiveness (RBE) (ICRP 1959). The detrimental effect from radiation exposure is not the same for all radiation types for a given amount of energy. The RBE is used, when calculating effective dose, to account for the relative effectiveness of different radiation types in producing tissue damage (Shapiro 1981). In ICRP 2, the RBE is 1 for gammas, x-rays, betas positrons, and conversion electrons; if the energy of the beta, positron or electron is less than 0.03 MeV, the RBE is assumed to be 1.7. The RBE for alpha particles and recoil atoms is given in ICRP 2 to be 10 and 20, respectively (ICRP 1959).

As a result of the short range of alpha and beta radiation, all energy from alpha and beta radiation emitted from a radionuclide is assumed to be completely dissipated and absorbed in the organ the radionuclide is deposited. With photon radiation, however, there exists a finite probability that a photon will travel through a medium without interaction. The probability that a photon radiation will pass through an organ tissue without interaction is a function of the organ thickness (size) and the photon linear attenuation coefficient of the medium (Shapiro 1981).

Effective energy values specific to relevant critical organs are given in Table 5, "Effective Energies," of ICRP 2 for each radionuclide. The effective energy values given in ICRP 2 for a critical organ accounts for the fraction of photon energy that escapes the critical organ without interaction.

b. ICRP 30 Absorbed Effective Energy

Although the effective energy parameter defined in ICRP 2 recognizes that photons have a finite probability of passing through an organ tissue without interaction, the effective energy absorbed in one organ does not include the probable exposure of photon energy emanating from radionuclides deposited in surrounding organs. In ICRP 30, the effective energy term of ICRP 2 is replaced by the specific effective energy (SEE). The SEE is defined as "the energy (MeV), suitably modified [by a] quality factor, imparted per gram of a target tissue *T* as a consequence of the emission of a specified radiation *i* from a transformation occurring in source tissue *S*" (ICRP 1978). Mathematically, the SEE for an individual nuclide *j* is defined as (ICRP 1978):

$$SEE(T \leftarrow S)_j = \sum_i \left[\frac{Y_i E_i AF(T \leftarrow S)_i Q_i}{M_T}\right]_j \qquad (40)$$

Where

SEE = specific effective energy in MeV g-1 per transformation;

Y_i = yield of radiation of type *i* per transformation of radionuclide *j*;

E_i = average or unique energy of radiation *i* as appropriate, in MeV;

$AF(T \leftarrow S)_i$ = fraction of energy absorbed in target organ *T* per emission of radiation *i* in the source organ *S*;

Q_i = quality factor appropriate for radiation of type i; and

M_t = mass of the target organ T.

Similar to the effective energy term in ICRP 2, the SEE parameter assumes that the energy from beta and alpha radiation is absorbed within the source organ. Furthermore, the quality factor Q_i of equation (40) is analogous to RBE parameter given in ICRP 2. The ICRP recommends that the following approximate values of Q be used for both internal and external dose estimation (ICRP 1977):

Table 6-1. Quality Factors Recommended in ICRP 26

Radiation Type	Quality Factor
Beta particles, electron and all electromagnetic radiation including gamma radiation, x rays and bremsstrahlung	1
Fission neutrons emitted in spontaneous fission and for photons	10
Alpha particles from nuclear transformation, for heavy recoil particles and for fission fragments	20

Unlike the effective energy term of ICRP 2, the SEE parameter for a target organ includes the photon exposure component that arises from radionuclides deposited in both the target organ and in other source organs. Moreover, the SEE term is defined per unit mass of the target organ. For each radionuclide, ICRP 30 provides a matrix of precalculated SEE values specific to target and source organ.

3. Differences Between ICRP 2 and ICRP 30 in regards to the Attribution of Risk to Tissues of the Body

Most radioactive exposures of the body involve the irradiation of more than one body tissue. The biological effects resulting from a systematic internal exposure to a radionuclide are not the same for tissues types. Certain radionuclides, due to their chemical nature, are absorbed more readily into particular tissues of the body. Various tissue types differ in their sensitivity to radiation exposure; some tissue and cell types are more susceptible to radiation damage than other tissue types. Furthermore, certain organs are more essential to the health and survival of an individual than other organs (ICRP 1977).

Radiation protection is designed to limit the exposure of radiation to that amount that results in an acceptable risk of human detriment. The recommended individual dose limits of ICRP 2 are defined by the dose equivalent of the critical organ (ICRP 1959). In the case of multiple organ exposure, the critical organ is the irradiated organ that "is likely to be of greatest importance because of the dose that it receives, its sensitivity to radiation or the importance to health of any damage that results" (ICRP 1977). The critical organ approach taken in ICRP 2 to limit radiation exposure does not consider the possible detriment or the relative radiosensitivity of other irradiated organs.

In ICRP 26 and ICRP 30, a dose limitation system is introduced that "takes account of the total risk attributable to the exposure of all tissue irradiated" (ICRP 1977). The two-part dose limitation system recommended in ICRP 30 is designed to "prevent detrimental non-stochastic effects and to limit the probability of stochastic effects to levels deemed acceptable" (ICRP 1977). For protection against non-stochastic effects, ICRP currently recommends an annual dose equivalent limit of 15 mSv (1.5 rem) for the lens of the eye and 50 mSv (5 rem) for the skin for members of the public (ICRP 1991).

The stochastic dose limit "is based on the principle that the risk should be equal whether the whole body is irradiated uniformly or whether there is non-uniform irradiation" (ICRP 1977). In ICRP 30, specific organs and tissues are assigned weighting factors w_T that represent "the proportion of the stochastic risk resulting from tissue T to the total risk, when the whole body is irradiated" (ICRP 1977). Table 6-2 lists the values of w_T that are given in ICRP 26 (ICRP 1977).

Table 6-2. Weighting Factors Recommended in ICRP 26

Tissue	w_T
Gonads	0.25
Breast	0.15
Red Bone Marrow	0.12
Lung	0.12
Thyroid	0.03
Bone Surface	0.03
Remainder	0.30

For protection against stochastic effects, ICRP currently recommends a single stochastic dose effective limit for the public of 1 mSv/yr (0.1 rem) (ICRP 1991). Compliance with the recommended stochastic dose limit is achieved if the following condition is met (ICRP 1977 and 1991):

$$\Sigma\ w_T H_T \leq H_{wb,L}$$

Where

w_T = appropriate weighting factor for tissue T, dimensionless;

H_T = committed dose equivalent in tissue T, in sievert or mrem; and

$H_{wb,L}$ = recommended annual dose effective limit, in sievert or mrem.

The committed dose equivalent (CDE) of an organ multiplied by its appropriate weighting factor is termed the weighted committed dose equivalent. ICRP 30 provides weighted CDEs specific to various target organs for each radionuclide. The whole body dose equivalent from exposure to a given radionuclide is obtained by summing the weighted CDEs specific to target organ that are listed in ICRP 30 for that radionuclide (ICRP 1978).

A significant difference between ICRP 2 and ICRP 30 is the approach taken to calculate total body dose. In ICRP 2, the total body dose is calculated as though the total body is one large tissue mass with uniform radionuclide distribution. In ICRP 30, the total body dose is defined as the sum of weighted committed dose equivalents. The approach utilized in ICRP 30 to estimate the total body dose "reflects both the distribution of dose among the various organs and tissues of the body and their assumed relative sensitivities to stochastic effects" (U.S. EPA 1988).

B. 1993 SSES Dose Estimate Based on ICRP 30 Dose Equivalents

In response to the publication of ICRP 26 and ICRP 30, the NRC has revised 10 CFR 20 to agree with the principles and recommendations given in ICRP 26 and ICRP 30. It is reasonable to assume that other NRC regulations and regulatory guidance will also be modified to conform with 10 CFR 20 and the axioms of ICRP 26 and ICRP 30. This section addresses the question: how might the required annual dose estimates change if they are calculated using dose factors based on the dosimetric models of ICRP 30?

The numerical guidance given in 10 CFR 50 - Appendix I requires the calculation of seven dose estimates which are specific to different effluent types. The seven dose estimates that are required by 10 CFR 50 - Appendix I are summarized in Table 6-3.

Table 6-3. Summary of Require Dose Estimates, Pathways Analyzed and Applicable Reg. Guide 1.109 Dose Factors

Effluent	Required Annual Dose Estimate	Pathway Analyzed	Applicable Reg. Guide 1.109 Dose Factors
Liquid	Highest Total Body Dose of the Four Age Groups	Fish Ingestion	Table E-11 thru E-12, "Ingestion Dose Factors for Adult, Teen, Child & Infant"
		Potable Water	
		Shore Exposure	Table E-6, "External Dose Factors for Standing on Contaminated Ground"
Liquid	Highest Organ Dose of the Four Age Groups	Fish Ingestion	Table E-11 thru E-12, "Ingestion Dose Factors for Adult, Teen, Child & Infant"
		Potable Water	
		Shore Exposure	Table E-6," External Dose Factor for Standing on Contaminated Ground"
Noble Gases	γ-Air Dose	*γ-Air	Table B-1, "Dose Factors for Exposure to a Semi-Infinite Cloud of Noble Gases"
	β-Air Dose	*β-Air	
	Total Body	*γ-Body	
	Skin	*β-Skin and γ-Air	
Airborne Iodine and Particulate Material	Highest Organ Dose of the Four Age Groups	Inhalation Pathway	Table E-7 thru E-8, "Inhalation Dose Factors for Adult, Teen, Child & Infant"
		Ingestion Pathway	Table E-11 thru E-12, Ingestion Dose Factors for Adult, Teen, Child & Infant"
		Ground Exposure	Table E-6, "External Dose Factors for Standing on Contaminated Ground"

*Applicable noble gas dose factor

In addition, Table 6-3 lists the various effluent pathways analyzed and the applicable Reg. Guide 1.109 dose factors used to calculate the respective dose estimate.

For each radionuclide, the internal inhalation and ingestion dose factors of Reg. Guide 1.109 are given in mrem per picocuries for the total body and for the following organs: bone, liver, thyroid, kidney, lung and lower large intestine (GI-LLI) (U.S. NRC 1977a). As previously stated, the NRC takes a maximum individual approach in providing guidance for 10 CFR 50 - Appendix I implementation. The ingestion and inhalation dose factors given in Reg. Guide 1.109 are derived and tabulated specific to four age groups: adult, teen, child and infant (U.S. NRC 1977a).

ICRP 30 lists internal inhalation and ingestion CDE per unit activity for various target organs in sievert per becquerel. The total body dose equivalent is defined in ICRP 30 as the sum of weighted CDEs specific to target organ and body tissue (ICRP 1978). The inhalation and ingestion CDE values given in ICRP 30 may be easily converted from sieverts per becquerel to mrem per picocurie. A practical method of calculating 1993 dose estimates based on ICRP 30 axioms is to replace the internal dose factors of Reg. Guide 1.109 in the dose calculations with corresponding dose equivalent values of ICRP 30. However, all dose equivalent values given in ICRP 30 are derived using anatomical data for a 70-kg reference adult. Therefore, the CDE values given in ICRP 30 could not be used to calculate dose estimates for the infant, child or teen age group.

For a select number of radionuclides, the ICRP has developed age-dependent physical and biokinetic models which evaluate the dose commitments per unit intake for members of the public. ICRP Publication 56 (ICRP 56), Age-dependent Doses to Members of the Public from Intake of Radionuclides: Part 1, and ICRP Publication 67 (ICRP 67), Age-dependent Doses to Members of the Public from Intake of Radionuclides: Part 2 Ingestion Dose Coefficients, provide dose equivalents for a select number of radionuclides that are derived for six different ages: 3 months, 1 year, 5 years, 10 years, 15 years, and adult (more than 17 years) (ICRP 1990 and 1993).

ICRP 56 and ICRP 67 do not, however, provide age-specific dose equivalent values for all radionuclides. Neither ICRP 56 nor ICRP 67 lists age-specific ingestion dose equivalents for iron-55, iron-59 or manganese-54. Furthermore, with the exception of tritium, strontium-90, and iodine-131, age-specific inhalation dose equivalents are not provided in either ICRP 56 or ICRP 67 for the remaining airborne radionuclides that were released from SSES in 1993. The ICRP plans to provide additional inhalation and ingestion age-specific dose equivalent values for other radionuclides in future ICRP publications (ICRP 1990 and 1993).

Since age-specific dose equivalents are currently not available for all radionuclides, this section will focus on the dose estimates that were calculated in 1993 specific to the adult age group. Table 6-4 lists the adult total body dose estimate for liquid effluents that was calculated for 1993 by PP&L and the highest calculated adult organ dose estimates for both liquid and airborne I&PM effluents.[7] The dose estimates of Table 6-4 were calculated using Reg. Guide 1.109 dose factors that were derived from ICRP 2 dose models. The total body and skin dose estimates of Table 6-4 for noble gases are not age specific and, therefore, are the same as those given in Table 4-1.

Table 6-4. Summary of Maximum Adult Offsite Dose Commitments to Members of the Public: 1/1/93 to 12/31/93

Effluent	Age Group	Applicable Organ	Estimated Maximum Dose (mrem)
Liquid	Adult	Total Body	5.14×10^{-3}
Liquid	Adult	GI-LLI	1.49×10^{-2}
Noble Gas	N/A	Air Dose (γ-mrad)	2.68×10^{-3}
Noble Gas	N/A	Air Dose (β-mrad)	7.96×10^{-3}
Noble Gas	N/A	Total Body	1.56×10^{-3}
Noble Gas	N/A	Skin	4.40×10^{-3}
Iodine & Particulate	Adult	Lung	3.57×10^{-2}

The dose estimates listed in Table 6-4 will be recalculated below using dose equivalent values given in ICRP 30. Consistent with this review, all calculations will be restricted to those liquid and airborne radionuclides that significantly contribute to the dose estimates of Table 6-4. This restriction reduces the number of necessary computations. For the liquid effluent pathways, the significant radionuclides are identified as tritium, manganese-54, iron-55, iron-59, cobalt-58, cobalt-60, zinc-65 and cesium-137. The airborne radioactive iodine and particulate materials that significantly contribute to the adult lung dose are identified as iodine-131, tritium, manganese-54,

[7]Personal communication with Robert Barclay of Pennsylvania Power & Light Company, 17 April, 1995.

cobalt-58, cobalt-60, zinc-65 and strontium-90. The only identified noble gas responsible for the noble gas skin and total body dose is xenon-133.

Neither ICRP 2 nor ICRP 30 prescribes methodologies for estimating dose from external sources such as exposure to contaminated ground surfaces. Therefore, the external dose factors of Reg. Guide 1.109 for standing on contaminated ground will be used with the inhalation and ingestion dose equivalents of ICRP 30 to recalculate the 1993 dose estimates.

ICRP 30 lists dose equivalent rates for external exposure to a cloud of noble gas. The skin and the total body dose equivalent rates given in ICRP 30 for exposure to xenon-133 will be used to recalculate the 1993 SSES noble gas skin and total body dose estimates. Since the γ-air and β-air dose factors of Reg. Guide 1.109 do not pertain to tissue dose estimates, the 1993 SSES γ-air and β-air dose estimates are not affected by the revised dose models given in ICRP 30, and will not be recalculated.

1. Total Body Dose Estimate from Liquid Effluents

In ICRP 30, the total body dose equivalent for a radionuclide is defined as the sum of CDEs that have been modified by an appropriate weighting factor, w_t. Weighted CDEs are given in ICRP 30 specific to various target organs for each radionuclide. Table 6-5 lists the weighted CDEs given in ICRP 30 for ingestion of manganese-54 (ICRP 1978).

Table 6-5. Weighted CDEs for Ingestion of Mn-54

Target Organ	Weighted CDE (Sv/Bq)
Gonads	2.40E-10
Breast	4.20×10^{-11}
R. Marrow	5.90×10^{-11}
Lungs	2.70×10^{-11}
SI Wall	5.90×10^{-11}
ULI Wall	8.10×10^{-11}
LLI Wall	1.30×10^{-10}
Liver	6.00×10^{-11}
Remainder	3.00×10^{-11}
Total Body Dose Equivalent	7.28×10^{-10}

The weighted CDE for other organs and body tissues not listed in Table 6-5 are collectively represented by the weighted CDE given for "Remainder" (ICRP 1978). The sum of weighted CDEs of Table 6-5 yields an adult total body dose equivalent of 7.28×10^{-10} Sv/Bq. This value may be converted to units of mrem/pCi and substituted for the adult total body dose factor given in Reg. Guide 1.109 for ingestion of manganese-54:

$$7.28 \times 10^{-10} \text{ Sv/Bq} \times 0.0370 \text{ Bq/pCi} \times 100000 \text{ mrem/Sv} = 2.70 \times 10^{-6} \text{ mrem/pCi}.$$

As previously stated, the liquid radioactive effluents which significantly contribute to the adult total body dose estimate of Table 6-4 are identified as tritium, manganese-54, iron-55, iron-59, cobalt-60, zinc-65, and cesium-137. Table 6-6 displays the adult total body dose equivalents obtained from the weighted CDEs listed in ICRP 30 for each of the aforementioned radionuclides (ICRP 1978). For comparison purposes, Table 6-6 also includes the corresponding total body dose factors given in Reg. Guide

1.109 that were used to calculate the total body dose estimate listed in Table 6-4 for liquid effluents (U.S. NRC 1977a).

Table 6-6. Ingestion Dose Equivalents of ICRP 30 and the Adult Ingestion Dose Factors of Reg. Guide 1.109 For Total Body

Radionuclide	ICRP 30 Dose Equivalent (mrem/pCi)	Reg. Guide 1.109 Dose Factor (mrem/pCi)
H-3	6.30×10^{-8}	*5.99×10^{-8}
Mn-54	2.70×10^{-6}	8.72×10^{-7}
Fe-55	5.15×10^{-7}	4.43×10^{-7}
Fe-59	6.63×10^{-6}	3.91×10^{-6}
[1]Co-60	9.96×10^{-6}	4.72×10^{-6}
[2]Co-60	2.58×10^{-5}	4.72×10^{-6}
Zn-65	1.44×10^{-5}	6.96×10^{-6}
Cs-137	5.05×10^{-5}	7.14×10^{-5}

*The LADTAP computer model uses tritium dose factors based on a beta quality factor of 1.0 to calculate dose estimates in lieu of the ingestion dose factor given in Reg. Guide 1.109 (i.e., 1.05×10^{-7} mrem/pCi), which is based on a 1.7 quality factor (U.S. NRC 1977a and 1986).

The distribution of cobalt through the gastrointestinal-tract varies depending on its chemical composition. Two sets of weighted CDEs are listed in ICRP 30 for different compound types of cobalt-60 (ICRP 1978). Since the chemical composition of the radionuclides that were released from SSES is unknown, Table 6-6 includes both total body dose equivalents given in ICRP 30 for cobalt-60.

With the exception of cesium-137, the ICRP 30 dose equivalent values of Table 6-6 are 1.1 to 5.5 times greater than the corresponding Reg. Guide 1.109 dose factors. Therefore, the adult total body dose estimate calculated using ICRP 30 dose equivalent is expected to be higher than the adult total body dose estimate listed in Table 6-4, which is based on Reg. Guide 1.109 dose factors that were derived from ICRP 2 dose models.

Using the dose equivalent values of Table 6-6, the adult total body dose from liquid effluents has been recalculated and the results are presented in Table 6-7. Table 6-7 displays the dose contribution per radionuclide and liquid pathway.

Table 6-7. Adult-Total Body Dose Based on ICRP 30 Dose Equivalents

Nuclide	Fish Ingestion (mrem)	Potable Water (mrem)	Shore Exposure (mrem)	Total over Pathways (mrem)
H-3	3.70×10^{-4}	6.29×10^{-4}	0.00	9.99×10^{-4}
Mn-54	1.23×10^{-3}	3.88×10^{-6}	2.33×10^{-5}	1.26×10^{-3}
Fe-55	5.78×10^{-5}	6.15×10^{-9}	0.00	5.78×10^{-5}
Fe-59	7.49×10^{-5}	8.91×10^{-7}	4.60×10^{-7}	7.62×10^{-5}
[1]Co-60	8.10×10^{-4}	2.32×10^{-5}	5.15×10^{-4}	1.35×10^{-3}
Zn-65	4.05×10^{-3}	3.58×10^{-6}	1.55×10^{-6}	4.06×10^{-3}
Cs-137	3.05×10^{-4}	2.23×10^{-7}	4.57×10^{-7}	3.05×10^{-4}
Subtotal	6.90×10^{-3}	6.61×10^{-4}	5.40×10^{-4}	8.10×10^{-3}
			Total Dose	8.10×10^{-3}

The [1]cobalt-60 dose equivalent given in Table 6-6 was used to calculate the adult total body dose estimate of Table 6-7. Using the [2]cobalt-60 dose equivalent value to calculate total body dose increases the ICRP 30 dose estimate to 9.42×10^{-3} mrem. The two adult total body dose estimates calculated using ICRP 30 dose equivalent values are 58 and 83 percent greater than the original adult total body dose estimate of Table 6-4 (i.e., 6.96×10^{-3} mrem).

2. Organ Dose Estimate from Liquid Effluents

The highest calculated adult organ dose from liquid effluents released in 1993 is the dose to the lower large intestine of the adult age group (PP&L 1993a). The CDEs given in ICRP 30 for the lower large intestine may be converted into units of mrem/pCi and substituted for the corresponding adult GI-LLI dose factors of Reg. Guide 1.109.

Both ICRP 30 dose equivalent values and Reg. Guide 1.109 dose factors are listed in Table 6-8 for those radionuclides significant to the adult GI-LLI dose estimate (ICRP 1978 and U.S. NRC 1977a).

Table 6-8. ICRP 30 Ingestion CDEs and Reg. Guide 1.109 Ingestion Dose Factors for Adult Lower Large Intestine

Radionuclide	ICRP 30 Dose Equivalent (mrem/pCi)	Reg. Guide 1.109 Dose Factor (mrem/pCi)	Difference
H-3	6.29×10^{-8}	*5.99×10^{-8}	(+) 5.0%
Mn-54	8.14×10^{-6}	1.40×10^{-5}	(-) 41.9%
Fe-55	1.11×10^{-6}	1.09×10^{-6}	(+) 1.8%
Fe-59	3.11×10^{-5}	3.40×10^{-5}	(-) 8.5%
[1]Co-58	1.22×10^{-5}	1.51×10^{-5}	(-) 19.4%
[2]Co-58	1.48×10^{-5}	1.51×10^{-5}	(-) 2.0%
[1]Co-60	4.07×10^{-5}	4.02×10^{-5}	(+) 1.2%
[2]Co-60	5.18×10^{-5}	4.02×10^{-5}	(+) 28.9%
Zn-65	1.85×10^{-5}	9.70×10^{-5}	(-) 90.7%

*The LADTAP computer model uses tritium dose factors based on a beta quality factor of 1.0 in lieu of the dose factor given in Reg. Guide 1.109 (i.e., 1.05×10^{-7} mrem/pCi) which is based on a 1.7 quality factor (U.S. NRC 1977a and 1986).

Table 6-8 includes the two CDEs listed in ICRP 30 for the different compound types of cobalt-58 and cobalt-60. The differences between the CDEs listed in ICRP 30 for manganese-54, [1]cobalt-58, and zinc-65 are 19 to 90.7 percent less than the corresponding Reg. Guide 1.109 dose factors. The ICRP 30 CDE for [2]cobalt-60 is 28.9 percent greater than the cobalt ingestion dose factor given in Reg. Guide 1.109. All remaining differences are less than 8.5%.

Using the dose equivalent values of Table 6-8, the adult GI-LLI dose estimate has been recalculated and the result is given in Table 6-9.

Table 6-9. Adult GI-LLI Dose Based on ICRP 30 Dose Equivalents

Nuclide	Fish Ingestion (mrem)	Potable Water (mrem)	Shore Exposure (mrem)	Total for Nuclide (mrem)
H-3	3.70×10^{-4}	6.29×10^{-4}	0.00	9.99×10^{-4}
Mn-54	3.71×10^{-3}	1.17×10^{-5}	2.33×10^{-5}	3.75×10^{-3}
Fe-55	1.24×10^{-4}	1.33×10^{-8}	0.00	1.24×10^{-4}
Fe-59	3.51×10^{-4}	4.18×10^{-6}	4.60×10^{-7}	3.56×10^{-4}
[1]Co-58	6.54×10^{-5}	2.04×10^{-6}	6.02×10^{-7}	6.80×10^{-5}
[1]Co-60	3.31×10^{-3}	9.49×10^{-5}	5.15×10^{-4}	3.92×10^{-3}
Zn-65	5.21×10^{-3}	4.61×10^{-6}	1.55×10^{-6}	5.21×10^{-3}
Subtotal	1.31×10^{-2}	7.46×10^{-4}	5.41×10^{-4}	1.44×10^{-2}
			Total Dose	1.44×10^{-2}

The adult GI-LLI dose estimate given in Table 6-9 is calculated using the [1]cobalt-58 and [1]cobalt-60 dose equivalent values of Table 6-8. Using the [2]cobalt-58 and [2]cobalt-60 dose equivalent values, the ICRP 30 dose estimate increases to 1.54×10^{-2} mrem. Both dose estimates for adult GI-LLI calculated using ICRP 30 dose equivalent values are within 3.4 percent of the adult GI-LLI dose estimate given in Table 6-4 (i.e., 1.49×10^{-2} mrem).

3. Total Body Dose Estimate from Noble Gas Exposure

The γ-body dose factors for noble gas exposure are given in Reg. Guide 1.109 in units of mrem-m^3/pCi-yr. In ICRP 30, the weighted dose equivalent rates for exposure to noble gases are given in units of Sv-m^3/Bq-hr. The total body dose equivalent rate for noble gases is obtained by summing the weighted dose equivalent rates given in ICRP 30 for each organ. Table 6-10 lists the weighted dose equivalent rates given in ICRP 30 for exposure to xenon-133 (ICRP 1978).

Table 6-10. The Weighted Dose Equivalent Rates Given in ICRP 30 for Xenon-133

Target Organ or Tissue	Dose Equivalent ($Sv\text{-}m^3/Bq\text{-}hr$)
Gonads	1.60×10^{-12}
Breast	8.40×10^{-13}
R. Marrow	1.30×10^{-13}
Lungs	5.80×10^{-13}
Thyroid	2.10×10^{-13}
Bone Surface	3.50×10^{-1}
ST Wall	2.30×10^{-13}
Kidneys	2.60×10^{-13}
Liver	2.50×10^{-13}
Adrenals	2.30×10^{-13}
Brain	2.40×10^{-13}
Total Body Dose Equivalent	6.09×10^{-12}

Converting the total body dose equivalent of Table 6-10 into units of mrem-m^3/pCi-yr yields 1.97×10^{-4} mrem-m^3/pCi-yr. The γ-body dose factor given in Reg. Guide 1.109 for xenon-133 (i.e., 2.94×10^{-4} mrem-m^3/pCi-yr) is greater than the total body dose equivalent rate of ICRP 30 by a factor of 1.5 (U.S. NRC 1977a). Using the total body dose equivalent given in Table 6-10, the total body dose from exposure to xenon-133 is recalculated to be 1.04×10^{-3} mrem. This calculation is illustrated below.

3.17×10^{-8} yr/sec	(inverse of seconds per year)
$\times\ 1.97 \times 10^{-4}$ mrem-m^3/pCi-yr	(γ-body dose factor for exposure to Xe-133)
$\times\ 1.408 \times 10^{-5}$ sec/m^3	(ADF for exposure to noble gas)
$\times$ 16.93 Ci/yr	(Annual release of Xe-133)
$\times\ 1.00 \times 10^{12}$ pCi/Ci	(Conversion factor for pCi to Ci)
$\times$ 0.7	(structural shielding factor)
$= 1.04 \times 10^{-3}$ mrem per year	(Calculated total body dose from Xe-133)

As would be expected, the original total body dose estimate of Table 6-4 for noble gas effluents (i.e., 1.56×10^{-3} mrem) is greater than the ICRP 30 total body dose estimate given above by a factor of 1.5.

4. Skin Dose Estimate from Noble Gases Exposure

In Reg. Guide 1.109, the skin dose estimate from exposure to noble gases is calculated using the β-skin dose factor listed in Reg. Guide 1.109 in addition to the γ-air dose factor multiplied by the tissue-to-air dose ratio of 1.1 and a shielding factor of 0.7 (U.S. NRC 1977a). A single Reg. Guide 1.109 skin dose factor (DF) for exposure to noble gas may be given as:

$$\beta\text{-skin DF}\left\{\frac{\text{mrem-m}^3}{\text{pCi-yr}}\right\} + \gamma\text{-air DF}\left\{\frac{\text{mrad-m}^3}{\text{pCi-yr}}\right\} * 1.1 * 0.7 = \text{skin DF}\left\{\frac{\text{mrem-m}^3}{\text{pCi-yr}}\right\}$$

For xenon-133, a Reg. Guide 1.109 (U.S. NRC 1977a) skin dose factor is calculated to be (U.S. NRC 1977a):

$$3.04 \times 10^{-4}\,\frac{\text{mrem-m}^3}{\text{pCi-yr}} + 3.53 \times 10^{-4}\,\frac{\text{mrad-m}^3}{\text{pCi-yr}} * 1.1 * 0.7 \;=\; 5.76 \times 10^{-4}\,\frac{\text{mrem-m}^3}{\text{pCi-yr}}$$

ICRP 30 gives a dose equivalent rate for skin from exposure to xenon-133 in units of Sv-m^3/Bq-hr, which is analogous to the above skin dose factor of Reg. Guide 1.109 for xenon-133. The ICRP 30 skin dose equivalent rate for xenon-133, converted to units of mrem-m^3/pCi-yr, is 6.16 E-04 mrem-m^3/pCi-yr (ICRP 1978). This value is 7 percent higher than the skin dose factor obtained from Reg. Guide 1.109. Therefore, the noble gas skin dose calculated using the ICRP 30 dose equivalent rate is expected to be slightly higher:

3.17×10^{-8} years/sec	(inverse of seconds per year)
$\times$ 6.16×10^{-4} mrem-m^3/pCi-yr	(ICRP 30 skin dose equivalent rate for Xe-133)
$\times$ 1.408×10^{-5} sec/m^3	(ADF for exposure to noble gas)
$\times$ 16.93 Ci/yr	(Annual release of Xe-133)
$\times$ 1.00×10^{12} pCi/Ci	(Conversion factor for pCi to Ci)
$= 4.65 \times 10^{-3}$ mrem per year	(Calculated annual skin dose)

The skin dose estimate calculated using the ICRP 30 dose equivalent rate for xenon-133 is 6 percent higher than the original noble gas skin dose of Table 6-4 (i.e., 4.40×10^{-3} mrem).

5. Organ Dose Estimate from Iodine and Particulate Material Effluents

The organ of the adult age group that received the highest dose in 1993 was calculated to be the lungs. The airborne pathways that were analyzed to calculate the adult lung dose estimate of Table 6-4 were identified to be the inhalation, vegetable ingestion, and ground exposure pathways (PP&L 1993a). Table 6-11 lists the ICRP 30 inhalation and ingestion CDEs and corresponding Reg. Guide 1.109 dose factors specific to the lungs of an adult for the airborne radionuclides that were released from reactor unit one in 1993 (ICRP 1978 and U.S. NRC 1977a).

Table 6-11. Inhalation and Ingestion CDEs of ICRP 30 and Dose Factors of Reg. Guide 1.109 for the Lung of Maximum Adult

Nuclide	Inhalation		Ingestion	
	ICRP 30 Dose Equivalent (mrem/pCi)	Reg. Guide 1.109 Dose Factor (mrem/pCi)	ICRP 30 Dose Equivalent (mrem/pCi)	Reg. Guide 1.109 Dose Factor (mrem/pCi)
H-3	9.44×10^{-8}	1.58×10^{-7}	6.29×10^{-8}	1.05E-07
[1]Mn-54	4.44×10^{-6}	1.75×10^{-4}	8.51×10^{-7}	--
[2]Mn-54	2.48×10^{-5}	1.75×10^{-4}	--	--
[1]Co-58	2.92×10^{-5}	1.16×10^{-4}	1.52×10^{-6}	--
[2]Co-58	5.92×10^{-5}	1.16×10^{-4}	--	--
[1]Co-60	1.33×10^{-4}	7.46×10^{-4}	3.22×10^{-6}	--
[2]Co-60	1.26×10^{-3}	7.46×10^{-4}	1.85×10^{-5}	--
Zn-65	7.77×10^{-5}	1.08×10^{-4}	1.15×10^{-5}	--
Sr-90	1.07×10^{-2}	1.20×10^{-3}	0.00E+00	--

Table 6-11 includes both CDE values given in ICRP 30 for the different compound types of manganese-54, cobalt-58, and cobalt-60 which effect lung clearance rates and the transfer rate of cobalt-60 through the gastrointestinal tract.

The inhalation dose factors given in Reg. Guide 1.109 for tritium "include an increase of 50 percent to account for the additional amount of this isotope absorbed through the skin" (U.S. NRC 1977a). Unlike the inhalation dose factors given in Reg. Guide 1.109 for tritium, the derivation of the ICRP 30 CDEs for tritium inhalation does not include a 50 percent increase for skin absorption. The ICRP, however, concurs that the tritium dose equivalents given in ICRP 30 should be increased 50 percent when estimating dose to account for absorption of tritium through the skin (ICRP 1978). Therefore, the ICRP 30 dose equivalent listed in Table 6-11 for tritium inhalation has been increased 50 percent to correlate with the tritium inhalation dose factor of Reg. Guide 1.109:

	$6.29 \times 10\text{-}8$ mrem/pCi	(ICRP 30 CDE for tritium inhalation)
$\times$	1.5	(50 percent increase for skin absorption)
$=$	$9.44 \times 10\text{-}8$ mrem/pCi.	

With the exception of strontium-90 and [2]cobalt-60, the inhalation dose factors of Reg. Guide 1.109 given in Table 6-11 are 1.4 to 39 times greater than the corresponding inhalation CDE values of ICRP 30. The inhalation dose factors given in Reg. Guide 1.109 for strontium-90 and [2]cobalt-60 are less than the corresponding CDEs of ICRP 30 by a factor of 8.9 and 1.7, respectively.

Both the inhalation and ingestion adult dose factors given in Reg. Guide 1.109 for tritium are greater than the corresponding tritium CDEs given in ICRP 30 by a factor of 1.7. This 1.7 difference is attributed to the 1.7 beta quality factor used to derive the tritium dose factors of Reg. Guide 1.109. As previously stated, PP&L employs the original version of the GASPAR computer dose model to calculate annual organ dose estimates from exposure to airborne I&PM effluents. The GASPAR computer model uses the tritium dose factors given in Reg. Guide 1.109 to calculate tritium dose estimates. Consequently, the tritium dose factors given in Reg. Guide 1.109 and those used by GASPAR are based upon the 1.7 quality factor recommended in ICRP 2 for low energy beta radiation. In contrast, the beta quality factor used to derive the tritium CDEs of ICRP 30 is 1.0. Dividing the Reg. Guide 1.109 tritium dose factors listed in Table 6-11 by 1.7 results in an inhalation dose factor of 9.29×10^{-8} mrem and an ingestion dose factor of 6.17×10^{-8} mrem. The Reg. Guide 1.109 tritium dose factors reduced by 1.7 are within 2 percent of the ICRP 30 tritium CDEs listed in Table 6-11.

Using the ICRP 30 dose equivalents listed in Table 6-11, the adult lung dose of Table 6-4 has been recalculated and the results are given in Table 6-12.

Table 6-12. Adult Lung Dose Based on ICRP 30 Dose Equivalents

Nuclide	Ingestion Pathway (mrem)	Inhalation Pathway (mrem)	Ground Pathway (mrem)	Total over Pathways (mrem)
H-3	7.78×10^{-3}	1.14×10^{-2}	0.00E+00	1.66×10^{-2}
[1]Mn-54	4.76×10^{-6}	1.16×10^{-5}	3.04×10^{-4}	3.20×10^{-4}
[1]Co-58	2.30×10^{-6}	8.66×10^{-7}	6.11×10^{-6}	9.27×10^{-6}
[1]Co-60	7.08×10^{-5}	2.51×10^{-5}	2.35×10^{-3}	2.44×10^{-3}
Zn-65	2.05×10^{-5}	3.67×10^{-5}	4.04×10^{-5}	9.75×10^{-5}
Sr-90	7.93×10^{-6}	0.00	0.00	7.93×10^{-6}
Subtotal	7.98×10^{-3}	1.15×10^{-2}	2.70×10^{-3}	2.21×10^{-2}
			Total Dose	2.21×10^{-2}

The adult lung dose estimate of Table 6-12 was calculated using the [1]manganese-54, [1]cobalt-58 and [1]cobalt-60 dose equivalent values given in Table 6-11. Using the [2]manganese-54, [2]cobalt-58, and [2]cobalt-60 dose equivalent values of Table 6-11 increases the calculated dose estimate to 2.29×10^{-2} mrem. The two adult lung dose estimates calculated using ICRP 30 dose equivalent values are 34 and 38 percent less than the adult lung dose estimate given in Table 6-4 (i.e., 3.57×10^{-2} mrem).

The tritium dose component constitutes 84 and 87 percent of the 2.21×10^{-2} mrem and 2.29×10^{-8} mrem adult lung dose estimates, respectively, that were calculated using ICRP 30 CDEs. Therefore, the differences between the adult lung dose estimate of Table 6-4 and the two adult lung dose estimates calculated using ICRP 30 CDEs are primarily attributed to the differences between the tritium dose factors of Reg. Guide 1.109 and the corresponding tritium CDEs of ICRP 30. To reiterate, the GASPAR computer code utilizes the tritium adult dose factors given in Reg. Guide 1.109 which are greater than the tritium CDEs listed in ICRP 30 by a factor of 1.7. The difference between the tritium dose factors of Reg. Guide 1.109 and the tritium CDEs of ICRP 30 results from the different beta quality factors used for low energy beta radiation; a 1.7 quality factor is used to calculate the tritium dose factors of

Reg. Guide 1.109, whereas, the tritium CDEs of ICRP 30 are derived using a 1.0 quality factor.

It is important to note that the revised GASPAR II computer dose model, consistent with the LADTAP II computer dose model employed by PP&L, uses tritium dose factors based on a 1.0 quality factor to calculate tritium dose estimates (U.S. NRC 1978a). If PP&L had employed the GASPAR II computer model to calculate the reported 1993 dose estimates instead of the original GASPAR version, the 1993 estimated dose for adult lung would be equivalent to the adult lung dose estimates that were calculated using ICRP 30 CDEs. To illustrate, the adult lung dose estimate of Table 6-4 has been recalculated to reflect a 1.7 reduction in the Reg. Guide 1.109 dose factors for tritium and the result is given in Table 6-13.

Table 6-13. Recalculation of 1993 SSES Adult Lung Dose Estimate Using Tritium Dose Factors Based on a Quality Factor of 1.0

Radionuclide	Dose from Inhalation (mrem)	Dose from Ingestion (mrem)	Dose from Ground Exposure (mrem)
H-3	7.73×10^{-2}	1.12×10^{-2}	0.00
Mn-54	1.88×10^{-5}	0.00	3.04×10^{-4}
Co-58	9.12×10^{-6}	0.00	6.11×10^{-6}
Co-60	3.97×10^{-4}	0.00	2.35×10^{-3}
Zn-65	2.85×10^{-5}	0.00	4.04×10^{-5}
Sr-90	8.78×10^{-7}	0.00	0.00
Subtotal	8.18×10^{-3}	1.12×10^{-2}	2.70×10^{-3}
Subtotal over all pathways			2.20×10^{-2}

The adult lung dose estimate of Table 6-13 differs from the ICRP 30 dose estimates for adult lung by less than 4 percent.

6. Summary of ICRP 30 Dose Calculations

The 1993 SSES dose estimates recalculated using dose equivalent values given in ICRP 30 are summarized in Table 6-14. Table 6-14 also includes the 1993 SSES dose estimates of Table 6-4 that were calculated by PP&L using the GASPAR and LADTAP II computer dose models.

Table 6-14. Summary Of Dose Calculations Using ICRP Dose Equivalents

Effluent	Age Group	Applicable Organ	1993 SSES Dose Estimates	*1993 Dose Estimates Based on ICRP 30 Dose Equivalent		Difference
Liquid	Adult	Total Body	5.14×10^{-3}	8.10×10^{-3}	9.42×10^{-3}	58 - 83%
Liquid	Adult	GI-LLI	1.49×10^{-2}	1.44×10^{-2}	1.54×10^{-2}	3.4%
Noble Gas	N/A	γ-air dose	2.68×10^{-3}	N/A	N/A	
Noble Gas	N/A	β-air dose	7.96×10^{-3}	N/A	N/A	
Noble Gas	N/A	Total body	1.56×10^{-3}	1.04×10^{-3}	--	50%
Noble Gas	N/A	Skin	4.40×10^{-3}	4.65×10^{-3}	--	5.6%
I&PM	Adult	Lung	3.57×10^{-2}	2.21×10^{-2}	2.29×10^{-2}	34 - 36%
**I&PM	Adult	Lung	2.20×10^{-2}	2.21×10^{-2}	2.29×10^{-2}	.5 - 4%

*The first and second columns of ICRP 30 dose estimates were calculated using the lowest and highest CDE values, respectively, given in ICRP 30 for manganese-54, cobalt-58, or cobalt-60.

**Original dose estimate recalculated using tritium dose factors of Reg. Guide 1.109 that have been reduced by 1.7.

The differences between the original 1993 dose estimates calculated by SSES and those calculated using ICRP 30 CDEs indicate that the total body dose estimates for liquid and noble gas effluents and the organ dose for I&PM effluents are most effected by the revised dose models of ICRP 30. As previously stated, the 34 and 36 percent differences between the original SSES dose estimate for adult lung and the corresponding ICRP 30 dose estimates are attributed to the different tritium quality factors used to calculate organ dose estimates for airborne I&PM effluents. The tritium dose factors given in Reg. Guide 1.109 and those used by the GASPAR computer code are based on a beta quality factor of 1.7, whereas, the tritium CDEs given in ICRP 30 are derived using a beta quality factor of 1.0. The adult lung dose estimate that was recalculated to reflect a 1.7 reduction in the Reg. Guide 1.109 dose factors for tritium is also included in Table 6-14. The differences between this adult lung dose estimate and the corresponding dose estimates that were calculated using ICRP 30 CDEs are less than 4 percent. The remaining differences between the ICRP 30 organ dose estimates and the original 1993 adult organ dose estimates calculated by PP&L are less than 6 percent.

The ICRP 30 total body dose estimates for liquid effluents are 58 and 83 percent greater than the original total body dose estimate calculated using the LADTAP II computer model. As previously stated, the internal dose factors listed in Reg. Guide 1.109 are based on the dose models of ICRP 2 (U.S. NRC 1977c). The methodology given in ICRP 30 to calculate total body dose differs significantly from the methodology given in ICRP 2 and that adopted by Reg. Guide 1.109. The total body dose factors of Reg. Guide 1.109 were calculated as though the total body is one large tissue mass with uniform radionuclide distribution. ICRP 26 and ICRP 30, on the other hand, define the total body dose as a weighted sum of CDEs given to various organs and body tissues. In contrast to ICRP 2, the approach introduced in ICRP 26 to estimate whole body dose considers the relative radiation sensitivities of different tissue types and the non-uniform distribution of radionuclides in the body (U.S. EPA 1988).

Unlike the total body dose estimate for liquid effluents, the ICRP 30 total body dose estimate for noble gas exposure is 50 percent less than the noble gas total body dose estimate calculated using the GASPAR computer model. In Reg. Guide 1.109, the total body dose factor from external exposure to noble gases is computed at a depth of 5 cm into the body. In ICRP 30, the total body dose from external exposure to noble gases is also a weighted sum of dose equivalent rates given to different internal organ and body tissues. The dosimetric models employed to derive the committed dose equivalent rates for noble gas exposure "takes into account the shielding of body organs by overlying tissue" (U.S. EPA 1988).

VII. CONCLUSION

The methodology given in the SSES ODCM to estimate individual total body and organ doses from exposure to radioactive liquid and airborne I&PM effluents conforms to U.S. NRC guidance documents, Reg. Guide 1.109 and NUREG-0133. The equations given in Reg. Guide 1.109 to calculate total body and skin dose from noble gas exposure include a 0.7 structural shielding factor to account for the shielding of gamma radiation by residential structures, whereas the noble gas equations given in NUREG-0133 do not. Consistent with Reg. Guide 1.109, the GASPAR computer dose model employed by PP&L uses a 0.7 shielding factor to calculate the reported noble gas dose estimates for total body and skin. However, the noble gas equations given in the SSES ODCM to calculate total body and skin dose estimates do not include a structural shielding factor. As of January 20, 1995 per ODCM Revision 3, the equations given in the SSES ODCM to estimate total body and skin dose from noble gas exposure have been revised to include a 0.7 shielding factor.

The hand-calculated dose estimates for 1993 effluent releases compared to the dose estimates reported in the 1993 AEWDR illustrate the 0.7 shielding factor discrepancy between the ODCM and the GASPAR computer code for the noble gas total body and skin dose estimates. The hand-calculated dose estimates also revealed that the vegetation pathway was inadvertently omitted from the reported organ dose estimate for airborne I&PM effluents. The organ dose estimate recalculated by GASPAR to include the vegetation pathway identified the lung of a child to be the highest organ dose in lieu of the originally reported teen lung dose estimate. The organ dose estimate for airborne I&PM effluents recalculated to include the vegetation pathway is only 0.34

percent of the 15 mrem dose limit. The other hand calculated dose estimates for 1993 were shown to agree with the corresponding dose estimates reported in the 1993 AEWDR and were less than 0.23 percent of the respective dose limits.

The tritium dose factors given in the ODCM and those used by the GASPAR computer program to calculate airborne dose estimates are not consistent with the tritium dose factors that are used by the LADTAP II computer program to calculate liquid dose estimates. The LADTAP II computer code employs tritium dose factors based on a beta quality factor of 1.0 whereas the tritium dose factors of the GASPAR computer code remained based on the 1.7 quality factor recommended in ICRP 2. The 1993 organ dose estimate from airborne I&PM effluents was shown to be significantly reduced by a 1.7 reduction in the tritium dose factors. It is recommended that PP&L use tritium dose factors based on a beta quality factor of 1.0 instead of 1.7 to calculate dose estimates from airborne I&PM effluents since this would lead to lower dose estimates and would be consistent with the LADTAP II computer dose model.

A sensitivity analysis performed on the site-specific parameters of the liquid dose model revealed the dilution factor to be the most sensitive liquid site-specific parameter and the transit time to be the least sensitive. The tracer dye release study employed by SSES to calculate dilution factors and transit times is a method accepted by the U.S. NRC which provides more realistic values than could be obtained from calculation models given in Reg. Guide 1.113.

The dose estimates most affected by the revised dose models given in ICRP 30 are the total body dose estimates from liquid and noble gas effluents. The ICRP 30 total body dose estimates for liquid and noble gas effluents differ by more than 50 percent from the total body estimates that were calculated by PP&L in 1993, using currently accepted methodology which is based on ICRP 2. The individual organ dose estimates calculated using ICRP 30 CDE values differ by less than 6 percent from those calculated using Reg. Guide 1.109 dose factors, assuming the tritium dose factors are based upon a beta quality factor of 1.0. All 1993 SSES dose estimates recalculated using ICRP 30 CDEs were less than 0.32 percent of the dose limits given in 10 CFR 50 - Appendix I.

VIII. REFERENCES

Gellert, W.; Kustner, H.; Hellwich, M.; Kastner, H. The VNR concise encyclopedia of mathematics. Atlanta: Van Nostrand Reinhold Company; 1977.

International Commission of Radiological Protection. Report of committee II on permissible dose for internal radiation. Oxford: Pergamon Press; ICRP Publication 2; 1959.

International Commission of Radiological Protection. Report of the task group on reference man. Oxford: Pergamon Press; ICRP Publication 23; 1975

International Commission of Radiological Protection. Recommendation of the International Commission on Radiological Protection. Oxford: Pergamon Press; ICRP Publication 26; 1977.

International Commission of Radiological Protection. Limits for intakes of radionuclides by workers. Oxford: Pergamon Press; ICRP Publication 30, Part 1; 1978.

International Commission of Radiological Protection. Radionuclide transformations energy and intensity of emissions. Oxford: Pergamon Press; ICRP Publication 38; 1978b.

International Commission of Radiological Protection. Age-dependent doses to members of the public from intakes of radionuclide: Part 1. Oxford: Pergamon Press; ICRP Publication 56; 1990.

International Commission of Radiological Protection. 1990 recommendations of the international commission on radiological protection. Oxford: Pergamon Press; ICRP Publication 60; 1991.

International Commission of Radiological Protection. Age-dependent doses to members of the public from intakes of radionuclide: Part 2 ingestion dose coefficients. Oxford: Pergamon Press; ICRP Publication 67; 1994.

Pennsylvania Power and Light Company. Susquehanna steam electric station annual effluent and waste disposal report. Pennsylvania, PA; 1993a.

Pennsylvania Power and Light Company. Susquehanna steam electric station 1993 land use census. Pennsylvania, PA; 1993b.

Sharpiro, J. Radiation protection: A guide for scientists and physicians. 2nd ed. Cambridge: Havard University Press; 1981.

Sutron Corporation. Study of travel times and mixing characteristics for the Susquehanna steam electric station: Final report. Herndon, Virginia; 1985.

U.S. Environmental Protection Agency. Limiting values of Radionuclide intake and air concentrations and dose conversion factors for inhalation, submersion, and ingestion. Washington, DC.: U.S. Government Printing Office; Federal Guidance Report No. 11; 1988.

U.S. Nuclear Regulatory Commission. Measuring, evaluating, and reporting radioactivity in solid wastes and releases of radioactive materials in liquid and gaseous effluents from light-water-cooled nuclear power plants. Washington, DC: U.S. Government Printing Office; Regulatory Guide 1.21; 1974.

U.S. Nuclear Regulatory Commission. 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. Washington, DC: U.S. Government Printing Office; Regulatory Guide 1.109; 1977a.

U.S. Nuclear Regulatory Commission. Methods for estimating atmospheric transport and dispersion of gaseous effluents in routine releases from light-water-cooled reactors. Washington, DC: U.S. Government Printing Office; Regulatory Guide 1.111; 1977b.

U.S. Nuclear Regulatory Commission. Age-specific radiation dose commitment factors for a one-year chronic intake. Washington, DC: U.S. Government Printing Office; NUREG-0172; 1977c.

U.S. Nuclear Regulatory Commission. Estimating aquatic dispersion of effluents from accidental and routine reactor releases for the purpose of implementing appendix I. Washington, DC: U.S. Government Printing Office; Regulatory Guide 1.113; 1977d.

U.S. Nuclear Regulatory Commission. Preparation of radiological effluent technical specification for nuclear power plants. Washington, DC: U.S. Government Printing Office; Prepared for Division of Site Safety and Environmental Analysis, Office of Nuclear Reactor Regulation; NUREG-0133; 1978a.

U.S. Nuclear Regulatory Commission. Draft radiological effluent technical specification for PWR's. Washington, DC: U.S. Government Printing Office; NUREG-0472; 1978b.

U.S. Nuclear Regulatory Commission. Radiological assessment: A textbook on environmental dose analysis. Washington, DC: U.S. Government Printing Office; NUREG/CR-3332; 1983.

U.S. Nuclear Regulatory Commission. LADTAP II - technical reference and user guide. Washington, DC: U.S. Government Printing Office; NUREG/CR-4013; 1986.

U.S. Nuclear Regulatory Commission. GASPAR II - technical reference and user guide. Washington, DC: U.S. Government Printing Office; NUREG/CR-4653; 1987.

U.S. Nuclear Regulatory Commission. Code of federal regulations. Washington, DC: U.S. Government Printing Office; 10 CFR Part 20; 1993.

APPENDIX A :

EXAMPLES OF DOSE FACTORS GIVEN IN THE SSES ODCM

Table A-1: Dose Factors for Potable Water Pathway (PP&L 1993a)

Dose Factors for Potable Water Pathway: Maximum Hypothetical Adult (page 1 of 2)
Dose Factor Units: mrem-ft^3/Ci-sec
Location: Danville Receiver/ VARIABLE DILUTION

Usage (Uap) (kg/yr: WATER) =			730				
Isotope	Bone	Liver	T. Body	Thyroid	Kidney	Lung	GI-LLI
H-3	*0.00E+00	8.43E-02	8.43E-02	8.43E-02	8.43E-02	8.43E-02	8.43E-02
C-14	2.28	4.56E-01	4.56E-01	4.56E-01	4.56E-01	4.56E-01	4.56E-01
Na-24	1.37E+00	1.37E+00	1.37E+00	1.37E+00	1.37E+00	1.37E+00	1.37E+00
P-32	1.55E+02	9.64E+00	5.99E+00	0.00E+00	0.00E+00	0.00E+00	1.74E+01
Cr-51	0.00E+00	0.00E+00	2.14E-03	1.28E-03	4.71E-04	2.83E-03	5.37E-01
Mn-54	0.00E+00	3.67E+00	7.00E+01	0.00E+00	1.09E+00	0.00E+00	1.12E+01
Mn-56	0.00E+00	9.23E-02	1.64E-02	0.00E+00	1.17E-01	0.00E+00	2.95E+00
Fe-55	2.21E+00	1.53E+00	3.56E-01	0.00E+00	0.00E+00	8.51E-01	8.75E-01
Fe-59	3.49E+00	8.19E+00	3.14E+00	0.00E+00	0.00E+00	2.29E+00	2.73E+01
Co-58	0.00E+00	5.98E-01	1.34E+00	0.00E+00	0.00E+00	0.00E+00	1.21E+01
Co-60	0.00E+00	1.72E+00	3.79E+00	0.00E+00	0.00E+00	0.00E+00	3.23E+01
Ni-63	1.04E+02	7.24E+00	3.50E+00	0.00E+00	0.00E+00	0.00E+00	1.51E+00
Ni-65	4.24E-01	5.51E-02	2.51E-02	0.00E+00	0.00E+00	0.00E+00	1.40E+00
Cu-64	0.00E+00	6.69E-02	3.14E-02	0.00E+00	1.69E-01	0.00E+00	5.70E+00
Zn-65	3.89E+00	1.24E+01	5.59E+00	0.00E+00	8.27E+00	0.00E+00	7.79E+00
Zn-69	8.27E-03	1.58E-02	1.10E-03	0.00E+00	1.03E+02	0.00E+00	2.38E-03
Br-83	0.00E+00	0.00E+00	3.23E-02	0.00E+00	0.00E+00	0.00E+00	4.65E-02
Br-84	0.00E+00	0.00E+00	4.18E-02	0.00E+00	0.00E+00	0.00E+00	3.28E-07
Br-85	0.00E+00	0.00E+00	1.72E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-86	0.00E+00	1.69E+01	7.89E+00	0.00E+00	0.00E+00	0.00E+00	3.34E+00
Rb-88	0.00E+00	4.86E-02	2.58E-02	0.00E+00	0.00E+00	0.00E+00	6.71E-13
Rb-89	0.00E+00	3.22E-02	2.26E-02	0.00E+00	0.00E+00	0.00E+00	1.87E-15
Sr-89	2.47E+02	0.00E+00	7.10E+00	0.00E+00	0.00E+00	0.00E+00	3.97E+01
Sr-90	6.09E+03	0.00E+00	1.49E+03	0.00E+00	0.00E+00	0.00E+00	1.76E+02
Sr-91	4.55E+00	0.00E+00	1.84E-01	0.00E+00	0.00E+00	0.00E+00	2.17E+01
Sr-92	1.73E+00	0.00E+00	7.47E-02	0.00E+00	0.00E+00	0.00E+00	3.42E+01
Y-90	7.72E-03	0.00E+00	2.07E-04	0.00E+00	0.00E+00	0.00E+00	8.19E+01
Y-91m	7.30E-05	0.00E+00	2.83E-06	0.00E+00	0.00E+00	0.00E+00	2.14E-04
Y-91	1.13E-01	0.00E+00	3.03E-03	0.00E+00	0.00E+00	0.00E+00	6.23E+01
Y-92	6.79E-04	0.00E+00	1.98E-05	0.00E+00	0.00E+00	0.00E+00	1.19E+01
Y-93	2.15E-03	0.00E+00	5.94E-05	0.00E+00	0.00E+00	0.00E+00	6.83E+01
Zr-95	2.44E-02	7.83E-03	5.30E-03	0.00E+00	1.23E-02	0.00E+00	2.48E+01
Zr-97	1.35E-03	2.72E-04	1.24E-04	0.00E+00	4.11E-04	0.00E+00	8.43E+01
Nb-95	4.99E-03	2.78E-03	1.49E-03	0.00E+00	2.75E-03	0.00E+00	1.69E+01
Mo-99	0.00E+00	3.46E+00	6.58E-01	0.00E+00	7.84E+00	0.00E+00	8.02E+00

*0.00E+00 = 0.00 × 10^0

Table A-2: Dose Factors for Fish Ingestion Pathway (PP&L 1993a)

Dose Factors for Fish Pathway: Maximum Hypothetical Adult (page 1 of 2)
Dose Factor Units: mrem-ft^3/Ci-sec
Location: Outfall / FIXED DILUTION

Usage (Uap) (kg/yr: FISH) =			21				
Dilution (1/Mp: FISH) =			15.9				
Transit time (tf) hrs =			25				
Isotope	Bone	Liver	T. Body	Thyroid	Kidney	Lung	GI-LLI
H-3	*0.00E+00	1.37E-04	1.37E-04	1.37E-04	1.37E-04	1.37E-04	1.37E-04
C-14	1.90E+01	3.80E+00	3.80E+00	3.80E+00	3.80E+00	3.80E+00	3.80E+00
Na-24	7.78E-02	7.78E-02	7.78E-02	7.78E-02	7.78E-02	7.78E-02	7.78E-02
P-32	2.67E+04	1.66E+03	1.03E+03	0.00E+00	0.00E+00	0.00E+00	3.00E+03
Cr-51	0.00E+00	0.00E+00	7.53E-04	4.50E-04	1.66E-04	9.99E-04	1.89E-01
Mn-54	0.00E+00	2.65E+00	5.06E-01	0.00E+00	7.89E-01	0.00E+00	8.12E+00
Mn-56	0.00E+00	8.06E-05	1.43E-05	0.00E+00	1.02E-04	0.00E+00	2.57E-03
Fe-55	3.99E-01	2.76E-01	6.43E-02	0.00E+00	0.00E+00	1.54E-01	1.58E-01
Fe-59	6.20E-01	1.46E+00	5.59E-01	0.00E+00	0.00E+00	4.07E-01	4.86E+00
Co-58	0.00E+00	5.36E-02	1.20E-01	0.00E+00	0.00E+00	0.00E+00	1.09E+00
Co-60	0.00E+00	1.55E-01	3.43E-01	0.00E+00	0.00E+00	0.00E+00	2.92E+00
Ni-63	1.89E+01	1.31E+00	6.33E-01	0.00E+00	0.00E+00	0.00E+00	2.73E-01
Ni-65	7.91E-05	1.03E-05	4.69E-06	0.00E+00	0.00E+00	0.00E+00	2.61E-04
Cu-64	0.00E+00	1.55E-03	7.26E-04	0.00E+00	3.90E-03	0.00E+00	1.32E-01
Zn-65	1.40E+01	4.46E+01	2.02E+01	0.00E+00	2.98E+01	0.00E+00	2.81E+01
Zn-69	2.35E-10	4.49E-10	3.13E-11	0.00E+00	2.92E-10	0.00E+00	6.75E-11
Br-83	0.00E+00	0.00E+00	1.74E-05	0.00E+00	0.00E+00	0.00E+00	2.51E-05
Br-84	0.00E+00	0.00E+00	2.14E-16	0.00E+00	0.00E+00	0.00E+00	1.68E-21
Br-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-86	0.00E+00	5.90E+01	2.75E+01	0.00E+00	0.00E+00	0.00E+00	1.16E+01
Rb-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-89	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	1.32E+01	0.00E+00	3.80E-01	0.00E+00	0.00E+00	0.00E+00	2.12E+00
Sr-90	3.30E+02	0.00E+00	8.11E+01	0.00E+00	0.00E+00	0.00E+00	9.54E+00
Sr-91	3.99E-02	0.00E+00	1.61E-03	0.00E+00	0.00E+00	0.00E+00	1.90E-01
Sr-92	1.57E-04	0.00E+00	6.77E-06	0.00E+00	0.00E+00	0.00E+00	3.10E-03
Y-90	2.67E-04	0.00E+00	7.15E-06	0.00E+00	0.00E+00	0.00E+00	2.83E+00
Y-91m	2.84E-15	0.00E+00	1.10E-16	0.00E+00	0.00E+00	0.00E+00	8.34E-15
Y-91	5.06E-03	0.00E+00	1.35E-04	0.00E+00	0.00E+00	0.00E+00	2.78E+00
Y-92	2.30E-07	0.00E+00	6.71E-09	0.00E+00	0.00E+00	0.00E+00	4.02E-03
Y-93	1.75E-05	0.00E+00	4.83E-07	0.00E+00	0.00E+00	0.00E+00	5.55E-01
Zr-95	1.44E-04	4.62E-05	3.13E-05	0.00E+00	7.25E-05	0.00E+00	1.46E-01
Zr-97	2.89E-06	5.83E-07	2.67E-07	0.00E+00	8.80E-07	0.00E+00	1.81E-01
Nb-95	2.66E-01	1.48E-01	7.94E-02	0.00E+00	1.46E-01	0.00E+00	8.97E+02
Mo-99	0.00E+00	4.82E-02	9.16E-03	0.00E+00	1.09E-01	0.00E+00	1.12E-01

*0.00E+00 = 0.00×10^0

Table A-3: Dose Factors for Shore Exposure Pathway (PP&L 1993a)

Dose Factor for Shore Exposure Pathway, All Age Groups (Page 1 of 2)
Dose Factor Units: mrem-ft^3/Ci-sec
Location: Edge of Initial Mixing Zone / FIXED DILUTION

Dilution (1/Mp: SHORE) =		15.9				
Transit time (ts) hrs =		1				
Sediment dep. time(tb) hrs =		131400				
	Adult		Teen		Child	
Usage(Uap) (hr/yr)	12		67		14	
Isotope	T. Body	Skin	T. Body	Skin	T. Body	Skin
H-3	*0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	2.48E-04	2.87E-04	1.38E-03	1.60E-03	2.89E-04	3.35E-04
P-32	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	1.01E-04	1.19E-04	5.64E-04	6.67E-04	1.18E-04	1.39E-04
Mn-54	3.01E-02	3.53E-02	1.68E-01	1.97E-01	3.51E-02	4.12E-02
Mn-56	1.50E-05	1.77E-05	8.37E-05	9.90E-05	1.75E-05	2.07E-05
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	5.92E-03	6.96E-03	3.31E-02	3.89E-02	6.91E-03	8.12E-03
Co-58	8.23E-03	9.64E-03	4.59E-02	5.38E-02	6.90E-03	1.12E-02
Co-60	4.68E-01	5.50E-01	2.61E+00	3.07E+00	5.45E-01	6.42E-01
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ni-65	4.90E-06	5.69E-06	2.74E-05	3.18E-05	5.72E-06	6.64E-06
Cu-64	1.25E-05	1.41E-05	6.97E-05	7.90E-05	1.46E-05	1.65E-05
Zn-65	1.62E-02	1.87E-02	9.06E-02	1.04E-01	1.89E-02	2.18E-02
Zn-69	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-83	7.92E-08	1.15E-07	4.42E-07	6.42E-07	9.24E-08	1.34E-07
Br-84	1.20E-06	1.39E-06	6.67E-06	7.79E-06	1.39E-06	1.63E-06
Br-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-86	1.95E-04	2.23E-04	1.09E-03	1.24E-03	2.27E-04	2.60E-04
Rb-88	6.99E-08	7.99E-08	3.90E-07	4.46E-07	8.16E-08	9.32E-08
Rb-89	1.82E-07	2.18E-07	1.02E-06	1.22E-06	2.12E-07	2.55E-07
Sr-89	4.70E-07	5.45E-07	6.20E-06	3.04E-06	5.48E-07	6.36E-07
Sr-91	4.34E-05	5.07E-05	2.42E-04	2.83E-04	5.06E-05	5.92E-05
Sr-92	1.31E-05	1.45E-05	7.29E-05	8.11E-05	1.52E-05	1.69E-05
Y-90	9.65E-08	1.14E-07	5.39E-07	6.37E-07	1.13E-07	1.33E-07
Y-91m	9.47E-07	1.10E-06	2.90E-06	6.12E-06	1.10E-06	1.28E-06
Y-91	2.33E-05	2.62E-04	1.30E-04	1.46E-04	2.72E-05	3.06E-05
Y-92	3.22E-06	3.83E-06	1.80E-05	1.80E-05	2.14E-05	4.46E-06
Y-93	3.72E-06	5.09E-06	2.08E-05	2.84E-05	4.34E-06	5.94E-06
Zr-95	5.31E-03	6.16E-03	2.97E-02	3.44E-02	6.20E-03	7.19E-03
Zr-97	6.17E-05	7.18E-05	3.45E-04	4.01E-04	7.20E-05	8.38E-05
Nb-95	2.97E-03	2.49E-03	1.66E-02	1.95E-02	3.46E-03	4.07E-03

*0.00E+00 = 0.00 × 10^0

Table A-4: Maximum Pathway Dose Factors for Iodine and Particulate Material Effluents (PP&L 1993a)

Airborne pathways and tritium ingestion: units are mrem/yr/uCi/m^3
Deposition pathways: units are mrem-m^2/yr/uCi/sec
Isotope: Manganese-54

Age Group	Bone	Liver	T. Body	Thyroid	Kidney	Lung	GI-LLI	Skin
Ground Pathway								
Adult	*1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.61E+09
Teen	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.61E+09
Child	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.61E+09
Infant	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.37E+09	1.61E+09
Goat Milk Pathway								
Adult	0.00E+00	6.14E+05	1.17E+05	0.00E+00	1.83E+02	0.00E+00	1.88E+06	N/A
Teen	0.00E+00	1.02E+06	2.03E+05	0.00E+00	3.05E+05	0.00E+00	2.10E+06	N/A
Child	0.00E+00	1.53E+06	4.07E+05	0.00E+00	4.29E+05	0.00E+00	1.28E+06	N/A
Infant	0.00E+00	2.84E+06	6.45E+05	0.00E+00	6.30E+05	0.00E+00	1.04E+06	N/A
Cow Milk Pathway								
Adult	0.00E+00	3.92E+06	7.49E+05	0.00E+00	1.17E+06	0.00E+00	1.20E+07	N/A
Teen	0.00E+00	6.54E+06	1.30E+06	0.00E+00	1.95E+06	0.00E+00	1.34E+07	N/A
Child	0.00E+00	9.78E+06	2.61E+06	0.00E+00	2.74E+06	0.00E+00	8.21E+06	N/A
Infant	0.00E+00	1.82E+07	4.12E+06	0.00E+00	4.03E+06	0.00E+00	6.68E+06	N/A
Cow Meat Pathway								
Adult	0.00E+00	4.79E+06	9.14E+05	0.00E+00	1.43E+06	0.00E+00	1.47E+07	N/A
Teen	0.00E+00	3.65E+06	7.25E+05	0.00E+00	1.09E+06	0.00E+00	7.49E+06	N/A
Child	0.00E+00	4.18E+06	1.11E+06	0.00E+00	1.17E+06	0.00E+00	3.51E+06	N/A
Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	N/A
Vegetable Pathway								
Adult	0.00E+00	2.80E+08	5.35E+07	0.00E+00	8.34E+07	0.00E+00	8.58E+08	N/A
Teen	0.00E+00	4.27E+08	8.46E+07	0.00E+00	1.27E+08	0.00E+00	8.75E+08	N/A
Child	0.00E+00	6.34E+08	1.69E+08	0.00E+00	1.78E+08	0.00E+00	5.32E+08	N/A
Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	N/A
Total Ingestion								
Adult	0.00E+00	2.90E+08	5.53E+07	0.00E+00	8.62E+07	0.00E+00	8.87E+08	N/A
Teen	0.00E+00	4.38E+08	8.69E+07	0.00E+00	1.31E+08	0.00E+00	8.98E+08	N/A
Child	0.00E+00	6.49E+08	1.73E+08	0.00E+00	1.82E+08	0.00E+00	5.45E+08	N/A
Infant	0.00E+00	2.10E+07	4.77E+06	0.00E+00	4.66E+06	0.00E+00	7.73E+06	N/A
Inhalation Pathway								
Adult	0.00E+00	3.96E+04	6.30E+03	0.00E+00	9.84E+03	1.40E+06	7.74E+04	N/A
Teen	0.00E+00	5.11E+04	8.40E+03	0.00E+00	1.27E+04	1.98E+06	6.68E+04	N/A
Child	0.00E+00	4.29E+04	9.51E+03	0.00E+00	1.00E+04	1.58E+06	2.29E+04	N/A
Infant	0.00E+00	2.53E+04	4.98E+03	0.00E+00	4.98E+03	1.00E+06	7.06E+03	N/A

*1.37E+09 = 1.37×10^9

APPENDIX B:

MIXING CHARATERISTICS OF THE SUSQUEHANNA RIVER AND ATMOSPHERIC DISPERSION FACTORS FOR SSES

Table B-1: Dilution Factors and Transit Times for SSES Effluents at Danville, PA (PP&L 1993a)

River Depth Meas. at Env. Lab (feet)	River Discharge	Leading Edge (hours)	Maximum Normalized Concentration	Dilution Factor
1.5	500	68.7	7.33×10^{-3}	136.4
1.6	530	67.8	7.14×10^{-3}	140.1
1.8	600	66.3	6.79×10^{-3}	147.3
2.0	670	64.8	6.43×10^{-3}	155.5
2.2	730	63.3	6.08×10^{-3}	164.5
2.4	780	61.8	5.75×10^{-3}	173.9
2.5	825	61.1	5.59×10^{-3}	179.1
2.6	870	60.3	5.42×10^{-3}	184.5
2.8	930	58.8	5.11×10^{-3}	195.7
3.0	1000	57.2	4.80×10^{-3}	208.3
3.2	1200	52.7	3.99×10^{-3}	250.6
3.4	1400	48.2	3.43×10^{-3}	291.5
3.5	1500	45.9	3.56×10^{-3}	280.9
3.6	1600	43.5	3.69×10^{-3}	271.0
3.8	1800	39.0	3.99×10^{-3}	250.6
4.0	2000	35.5	3.99×10^{-3}	250.6
4.2	2280	35.2	3.93×10^{-3}	254.5
4.4	2560	34.7	3.86×10^{-3}	259.1
4.5	2730	34.5	3.83×10^{-3}	261.4
4.6	2900	34.2	3.79×10^{-3}	263.9
4.8	3300	33.7	3.70×10^{-3}	270.3
5.0	3700	33.0	3.60×10^{-3}	277.8
5.2	4140	32.3	3.52×10^{-3}	284.1
5.4	4580	31.7	3.42×10^{-3}	292.4
5.5	4820	31.4	3.37×10^{-3}	297.2
5.6	5060	31.0	3.31×10^{-3}	302.1
5.8	5580	30.2	3.20×10^{-3}	312.5
6.0	6100	29.5	3.09×10^{-3}	323.6
6.2	6780	28.5	2.95×10^{-3}	339.0
6.4	7460	27.5	2.82×10^{-3}	354.6
6.5	7890	26.9	2.73×10^{-3}	366.3
6.6	8320	26.2	2.64×10^{-3}	378.8

Table B-1: Dilution Factors and Transit Times for SSES Effluents at Danville, PA (continued)

River Depth Meas. at Env. Lab (feet)	River Discharge	Leading Edge (hours)	Maximum Normalized Concentration	Dilution Factor
6.8	9360	24.7	2.42×10^{-3}	413.2
7.0	10400	23.0	2.19×10^{-3}	456.6
7.5	12500	20.0	1.70×10^{-3}	588.2
8.0	14900	16.5	1.15×10^{-3}	869.6
8.5	17500	15.3	1.02×10^{-3}	980.4
9.0	20700	14.7	9.33×10^{-4}	1071.8
9.5	24000	14.2	8.52×10^{-4}	1173.7
10.0	27000	13.5	7.78×10^{-4}	1285.3
10.5	30100	13.0	7.28×10^{-4}	1373.6
11.0	34570	12.2	6.38×10^{-4}	1567.4
11.5	38730	11.3	5.57×10^{-4}	1795.3
12.0	42530	10.7	4.86×10^{-4}	2057.6
12.5	46490	10.0	4.17×10^{-4}	2398.1
13.0	50630	10.0	3.85×10^{-4}	2597.4
13.5	54940	10.0	3.53×10^{-4}	2832.9
14.0	59430	9.8	3.26×10^{-4}	3067.5
14.5	64090	9.8	3.02×10^{-4}	3311.3
15.0	68930	9.8	2.81×10^{-4}	3558.7
15.5	74030	9.8	2.63×10^{-4}	3802.3
16.0	79130	9.8	2.45×10^{-4}	4081.6
16.5	84580	9.8	2.30×10^{-4}	4347.8
17.0	90030	9.7	2.15×10^{-4}	4651.2
17.5	95830	9.7	2.03×10^{-4}	4926.1
18.0	101630	9.7	1.91×10^{-4}	5235.6
18.5	107780	9.7	1.81×10^{-4}	5540.2
19.0	113930	9.7	1.70×10^{-4}	5882.4
19.5	120430	9.6	1.62×10^{-4}	6192.0
20.0	126930	9.5	1.53×10^{-4}	6535.9
20.5	133780	9.5	1.46×10^{-4}	6872.9
21.0	140630	9.5	1.38×10^{-4}	7246.4
21.5	147830	9.4	1.32×10^{-4}	7604.6
22.0	155030	9.3	1.25×10^{-4}	8000.0

Table B-2: 1993 SSES Atmospheric Dispersion Factors (PP&L 1993a)

Location Type	Sector	Distance (miles)	X/Q No decay Undepleted (sec/m)	X/Q 2.26 day decay Undepleted (sec/m)	X/Q 8.00 day decay Depleted (sec/m)	D/Q (m^{-2})
Site Boundary	S	0.34	*8.397E-06	8.383E-06	7.829E-06	4.033E-08
Site Boundary	SSW	0.41	1.173E-05	1.170E-05	1.082E-05	4.707E-08
Site Boundary	SW	0.83	8.230E-06	8.189E-06	7.279E-06	1.826E-08
Site Boundary	WSW	1.03	1.418E-05	1.408E-05	1.236E-05	2.083E-08
Site Boundary	W	1.02	6.406E-06	6.364E-06	5.591E-06	1.066E-08
Site Boundary	WNW	0.62	6.660E-06	6.635E-06	5.995E-06	1.549E-08
Site Boundary	NW	0.65	6.172E-06	6.148E-06	5.539E-06	1.702E-08
Site Boundary	NNW	0.59	5.119E-06	5.101E-06	4.623E-06	1.360E-08
Site Boundary	N	0.59	5.408E-06	5.390E-06	4.883E-06	1.709E-08
Site Boundary	NNE	0.78	4.406E-06	4.388E-06	3.912E-06	1.508E-08
Site Boundary	NE	0.61	5.343E-06	5.328E-06	4.813E-06	3.055E-08
Site Boundary	ENE	0.53	3.588E-06	3.581E-06	3.262E-06	2.885E-08
Site Boundary	E	0.53	1.928E-06	1.925E-06	1.754E-06	1.520E-08
Site Boundary	ESE	0.51	1.809E-06	1.806E-06	1.648E-06	1.460E-08
Site Boundary	SE	0.42	3.064E-06	3.060E-06	2.827E-06	2.125E-08
Site Boundary	SSE	0.34	5.027E-06	5.020E-06	4.687E-06	3.534E-08
Residence	N	1.30	1.673E-06	1.661E-06	1.434E-06	4.396E-09
Residence	NNE	1.00	3.279E-06	3.261E-06	2.866E-06	1.050E-08
Residence	NE	2.30	8.391E-07	8.304E-07	6.835E-07	3.447E-09
Residence	ENE	2.10	5.208E-07	5.167E-07	4.283E-07	3.104E-09
Residence	E	1.40	4.058E-07	4.038E-07	3.461E-07	2.641E-09
Residence	ESE	0.50	1.881E-06	1.878E-06	1.717E-06	1.525E-08
Residence	SE	0.40	3.264E-06	3.259E-06	3.018E-06	2.268E-08
Residence	ESE	0.60	2.304E-06	2.299E-06	2.078E-06	1.512E-08
Residence	S	1.00	1.785E-06	1.777E-06	1.561E-06	7.253E-09
Residence	SSW	0.90	3.885E-06	3.867E-06	3.420E-06	1.356E-08
Residence	SW	1.50	3.472E-06	3.441E-06	2.941E-06	6.532E-09
Residence	WSW	1.10	1.289E-05	1.280E-05	1.119E-05	1.858E-08
Residence	W	1.20	4.952E-06	4.914E-06	4.270E-06	7.831E-09
Residence	WNW	0.80	4.568E-06	4.545E-07	4.049E-06	1.032E-08
Residence	NW	0.80	4.845E-06	4.822E-06	4.296E-06	1.291E-08
Residence	NNW	0.60	4.945E-06	4.927E-06	4.458E-06	1.310E-08
Residence	NE	0.70	4.460E-06	4.446E-06	3.986E-06	2.489E-08
Residence	NNE	0.90	3.701E-06	3.683E-06	3.258E-06	1.222E-08

*8.397E-06 = 8.397×10^{-6}

Table B-2: 1993 SSES Atmospheric Dispersion Factors (continued)

Location Type	Sector	Distance (miles)	X/Q No decay Undepleted (sec/m)	X/Q 2.26 day decay Undepleted (sec/m)	X/Q 8.00 day decay Depleted (sec/m)	D/Q (m^{-2})
Garden	N	1.30	1.673E-06	1.661E-06	1.434E-06	4.396E-09
Garden	NNE	1.10	2.839E-06	2.823E-06	2.465E-06	8.855E-09
Garden	NE	2.30	8.391E-07	8.304E-07	6.835E-07	3.447E-09
Garden	ENE	2.40	4.391E-07	4.351E-07	3.562E-07	2.521E-09
Garden	E	1.40	4.058E-07	4.038E-07	3.461E-07	2.641E-09
Garden	ESE	2.50	1.288E-08	1.279E-07	1.041E-07	7.627E-10
Garden	SE	0.60	1.770E-06	1.766E-06	1.597E-06	1.173E-08
Garden	SSE	0.80	1.567E-06	1.563E-06	1.391E-06	9.769E-09
Garden	S	1.10	1.536E-06	1.528E-06	1.333E-06	6.088E-09
Garden	SSW	1.20	2.513E-06	2.498E-06	2.168E-06	8.159E-09
Garden	SW	1.90	2.457E-06	2.429E-06	2.038E-06	4.271E-09
Garden	WSW	1.10	1.289E-05	1.280E-05	1.119E-05	1.858E-08
Garden	W	1.20	4.952E-06	4.914E-06	4.270E-06	7.831E-09
Garden	WNW	1.30	2.162E-06	2.145E-06	1.853E-06	4.260E-09
Garden	NW	0.90	3.939E-06	3.918E-06	3.467E-06	1.019E-08
Garden	NNW	4.00	3.112E-07	3.037E-07	2.372E-07	4.686E-10
Dairy	E	4.50	6.166E-08	6.067E-08	4.641E-08	2.909E-10
Dairy	E	4.60	5.969E-08	5.871E-08	4.479E-08	2.796E-10
Dairy	ESE	2.70	1.123E-07	1.114E-07	9.003E-08	6.526E-10
Dairy	ESE	4.10	4.891E-08	4.831E-08	3.730E-08	2.550E-10
Dairy	ESE	4.70	3.631E-08	3.580E-08	2.719E-08	1.822E-10
Dairy	SE	2.60	1.818E-07	1.801E-08	1.462E-07	8.576E-10
Dairy	S	3.90	1.731E-07	1.700E-07	1.326E-07	4.735E-10
Dairy	SSW	3.00	5.934E-07	5.842E-07	4.693E-07	1.482E-09
Dairy	SSW	3.10	5.524E-07	5.436E-07	4.352E-07	1.365E-09
Dairy	SSW	3.50	4.228E-07	4.152E-07	3.282E-07	1.004E-09
Dairy	SSW	3.80	3.595E-07	3.525E-07	2.762E-07	8.302E-10
Dairy	SSW	14.00	2.048E-08	1.903E-08	1.365E-08	3.253E-11
Dairy	WSW	2.00	5.797E-06	5.725E-06	4.783E-06	6.779E-09
Dairy	W	5.00	4.867E-07	4.712E-07	3.595E-07	4.255E-10
Dairy	WNW	1.80	1.343E-06	1.328E-07	1.119E-06	2.389E-10
Dairy	NNW	4.20	2.909E-07	2.835E-07	2.203E-07	4.293E-10
Irrigation	SW	3.30	1.010E-06	9.900E-07	7.890E-07	1.401E-09
Irrigation	WSW	8.30	6.000E-07	5.693E-07	4.100E-07	3.490E-10

APPENDIX C:

CALCULATED DOSE ESTIMATES FROM LIQUID EFFLUENTS

Table C-1. 1993 SSES Waterborne Effluents (Ci) (PP&L 1993a)

Nuclide Released	First Quarter	Second Quarter	Third Quarter	Fourth Quarter	Annual Release
H-3	*8.64E+00	9.42E+00	2.95E+01	2.03E+01	6.79E+01
F-18	0.00E+00	1.22E-04	1.01E-04	3.42E-07	5.65E-07
Na-24	0.00E+00	7.44E-04	0.00E+00	0.00E+00	7.44E-07
Cr-51	1.33E-04	1.07E-03	1.22E-03	2.76E-04	2.70E-03
Mn-54	3.73E-03	3.56E-03	3.26E-03	1.81E-03	1.24E-02
Fe-55	9.87E-04	1.28E-02	0.00E+00	0.00E+00	1.38E-02
Fe-59	4.12E-04	4.52E-04	3.48E-04	5.79E-05	1.27E-03
Co-58	4.07E-05	3.95E-05	3.23E-04	6.69E-04	1.07E-03
Co-60	1.61E-03	1.52E-03	2.50E-03	1.06E-02	1.62E-02
Zn-65	2.27E-05	1.90E-06	9.21E-04	4.81E-04	1.43E-03
As-76	0.00E+00	1.14E-05	2.14E-05	0.00E+00	3.28E-05
Sr-89	8.82E-06	0.00E+00	0.00E+00	0.00E+00	8.82E-06
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	0.00E+00	0.00E+00	7.11E-07	3.42E-07	1.05E-06
Mo-99	0.00E+00	1.30E-05	0.00E+00	0.00E+00	1.30E-05
Tc-99m	0.00E+00	4.30E-06	0.00E+00	0.00E+00	4.30E-06
Ag-110	0.00E+00	2.10E-06	3.15E-04	1.25E-05	3.30E-04
Te-131	0.00E+00	0.00E+00	4.01E-12	0.00E+00	4.01E-12
I-131	0.00E+00	5.08E-07	0.00E+00	0.00E+00	5.08E-07
Cs-134	0.00E+00	0.00E+00	0.00E+00	6.86E-07	6.86E-07
Cs-137	8.42E-07	3.44E-07	1.48E-06	2.64E-05	2.91E-05
Ce-141	0.00E+00	0.00E+00	5.58E-07	0.00E+00	5.58E-07
Nd-147	1.07E-06	0.00E+00	0.00E+00	0.00E+00	1.07E-06
W-187	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

*8.64 E+00 = 8.64×10^{0}

Table C-2. Dose Estimates for Teen Total Body from Potable Water Pathway

Nuclide	Teen total Body Dose Factor (mrem-ft^3 / uCi-sec)	Decay Constant (hours^{-1})	First Quarter (mrem)	Second Quarter (mrem)	Third Quarter (mrem)	Fourth Quarter (mrem)	Annual Dose (mrem)
H-3	*5.95E-02	6.44E-06	7.22E-05	2.37E-05	4.06E-04	2.26E-04	7.28E-04
F-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	1.29E+00	4.62E-02	0.00E+00	2.18E-11	0.00E+00	0.00E+00	2.18E-11
Cr-51	2.02E-03	1.04E-03	3.68E-11	9.01E-11	5.43E-10	1.02E-10	7.72E-10
Mn54	6.56E-01	9.24E-05	3.43E-07	9.86E-08	4.92E-07	2.22E-07	1.16E-06
Fe-55	3.51E-01	2.69E-01	1.00E-10	5.03E-09	0.00E+00	0.00E+00	5.13E-09
Fe-59	2.97E+00	6.47E-04	1.69E-07	5.63E-08	2.32E-07	3.17E-08	4.89E-07
Co-58	1.26E+00	4.08E-04	7.13E-09	2.09E-09	9.23E-08	1.56E-07	2.58E-07
Co-60	3.55E+00	1.50E-05	8.02E-07	2.28E-07	2.05E-06	7.04E-06	1.01E-05
Zn-65	5.23E+00	1.18E-04	1.66E-08	4.20E-10	1.11E-06	4.70E-07	1.59E-06
As-76	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	7.07E+00	5.71E-04	8.64E-09	0.00E+00	0.00E+00	0.00E+00	8.64E-09
Sr-90	1.15E+03	2.77E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	7.29E-02	2.56E-01	0.00E+00	0.00E+00	9.45E-17	5.70E-15	5.80E-15
Mo-99	6.45E-01	1.05E-02	0.00E+00	3.08E-10	0.00E+00	0.00E+00	3.08E-10
Tc-99m	6.73E-03	1.15E-01	0.00E+00	2.59E-13	0.00E+00	0.00E+00	2.59E-13
Ag-110	6.62E-02	1.16E-04	0.00E+00	5.87E-12	4.79E-09	1.54E-10	4.96E-09
Te-131	4.89E-03	2.31E-02	0.00E+00	0.00E+00	1.57E-18	0.00E+00	1.57E-18
I-131	2.47E+00	3.59E-03	0.00E+00	5.05E-11	0.00E+00	0.00E+00	5.05E-11
Cs-134	5.13E+01	3.84E-05	0.00E+00	0.00E+00	0.00E+00	6.58E-09	6.58E-09
Cs-137	2.91E+01	2.62E-06	3.44E-09	4.23E-10	9.95E-09	1.44E-07	1.58E-07
Ce-141	5.72E-04	8.89E-04	0.00E+00	0.00E+00	7.08E-14	0.00E+00	7.08E-14
Nd-147	3.43E-04	2.63E-03	4.85E-14	0.00E+00	0.00E+00	0.00E+00	4.85E-14
W-187	2.34E-02	2.91E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
						SUM	7.41E-04

*5.95E-02 = 5.95×10^{-2}

Table C-3. Dose Estimates for Teen Total Body from Fish Ingestion Pathway

Nuclide	Teen Total Body Dose Factor (mrem-ft³ / uCi-sec)	First Quarter (mrem)	Second Quarter (mrem)	Third Quarter (mrem)	Fourth Quarter (mrem)	Annual Dose (mrem)
H-3	*1.06E-04	5.87E-05	5.43E-05	2.03E-04	1.53E-04	4.69E-04
F-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	8.02E-02	0.00E+00	3.24E-09	0.00E+00	0.00E+00	3.24E-09
Cr-51	7.76E-04	6.62E-09	4.51E-08	6.15E-08	1.52E-08	1.28E-07
Mn54	5.17E-01	1.24E-04	1.00E-04	1.09E-04	6.64E-05	3.99E-04
Fe-55	6.91E-02	4.37E-06	4.81E-05	0.00E+00	0.00E+00	5.24E-05
Fe-59	5.76E-01	1.52E-05	1.41E-05	1.30E-05	2.37E-06	4.47E-05
Co-58	1.23E-01	3.21E-07	2.64E-07	2.58E-06	5.84E-06	9.00E-06
Co-60	3.50E-01	3.61E-05	2.89E-05	5.68E-05	2.63E-04	3.85E-04
Zn-65	2.06E+01	3.00E-05	2.13E-06	1.23E-03	7.03E-04	1.97E-03
As-76	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	4.12E-01	2.33E-07	0.00E+00	0.00E+00	0.00E+00	2.33E-07
Sr-90	6.81E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	7.21E-06	0.00E+00	0.00E+00	3.33E-13	1.75E-13	5.08E-13
Mo-99	9.79E-03	0.00E+00	6.92E-09	0.00E+00	0.00E+00	6.92E-09
Tc-99m	1.12E-05	0.00E+00	2.62E-12	0.00E+00	0.00E+00	2.62E-12
Ag-110	3.00E-04	0.00E+00	3.42E-11	6.14E-09	2.66E-10	6.44E-09
Te-131	3.46E-21	0.00E+00	0.00E+00	9.01E-34	0.00E+00	9.01E-34
I-131	6.68E-02	0.00E+00	1.84E-09	0.00E+00	0.00E+00	1.84E-09
Cs-134	2.02E+02	0.00E+00	0.00E+00	0.00E+00	9.83E-06	9.83E-06
Cs-137	1.15E+02	6.21E-06	2.15E-06	1.11E-05	2.15E-04	2.35E-04
Ce-141	1.10E-06	0.00E+00	0.00E+00	3.99E-14	0.00E+00	3.99E-14
Nd-147	1.58E-05	1.08E-12	0.00E+00	0.00E+00	0.00E+00	1.08E-12
W-187	2.68E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
					SUM	3.57E-03

*1.06E-04 = 1.06×10^{-4}

Table C-4. Dose Estimates from Teen Total Body from Shore Exposure Pathway

Nuclide	Teen Total Body Dose Factor $\left(\frac{\text{mrem-ft}^3}{\text{uCi-sec}}\right)$	First Quarter (mrem)	Second Quarter (mrem)	Third Quarter (mrem)	Fourth Quarter (mrem)	Annual Dose (mrem)
H-3	*0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
F-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	1.38E-03	0.00E+00	5.58E-11	0.00E+00	0.00E+00	5.58E-11
Cr-51	5.64E-04	4.81E-09	3.28E-08	4.47E-08	1.10E-08	9.33E-08
Mn54	1.68E-01	4.02E-05	3.25E-05	3.56E-05	2.16E-05	1.30E-04
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	3.31E-02	8.74E-07	8.13E-07	7.48E-07	1.36E-07	2.57E-06
Co-58	4.59E-02	1.20E-07	9.85E-08	9.63E-07	2.18E-06	3.36E-06
Co-60	2.61E+00	2.69E-04	2.16E-04	4.24E-04	1.96E-03	2.87E-03
Zn-65	9.06E-02	1.32E-07	9.36E-09	5.42E-06	3.09E-06	8.65E-06
As-76	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	2.62E-06	1.48E-12	0.00E+00	0.00E+00	0.00E+00	1.48E-12
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	7.29E-05	0.00E+00	0.00E+00	3.37E-12	1.77E-12	5.13E-12
Mo-99	4.79E-04	0.00E+00	3.38E-10	0.00E+00	0.00E+00	3.38E-10
Tc-99m	1.99E-05	0.00E+00	4.65E-12	0.00E+00	0.00E+00	4.65E-12
Ag-110	4.17E-01	0.00E+00	4.76E-08	8.53E-06	3.70E-07	8.95E-06
Te-131	6.74E-07	0.00E+00	0.00E+00	1.76E-19	0.00E+00	1.76E-19
I-131	2.08E-03	0.00E+00	5.74E-11	0.00E+00	0.00E+00	5.74E-11
Cs-134	8.32E-01	0.00E+00	0.00E+00	0.00E+00	4.05E-08	4.05E-08
Cs-137	1.25E+00	6.75E-08	2.34E-08	1.20E-07	2.34E-06	2.55E-06
Ce-141	1.66E-03	0.00E+00	0.00E+00	6.01E-11	0.00E+00	6.01E-11
Nd-147	1.02E-03	7.00E-11	0.00E+00	0.00E+00	0.00E+00	7.00E-11
W-187	2.77E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
					SUM	3.03E-03

*0.00E+00 = 0.00×10^0

Table C-5. Dose Estimate for Adult GI-LLI from Potable Water Pathway

Nuclide	Adult GI-LLI Dose Factor (mrem-ft³ / uCi-sec)	Decay Constant (hours^{-1})	First Quarter (mrem)	Second Quarter (mrem)	Third Quarter (mrem)	Fourth Quarter (mrem)	Annual Dose (mrem)
H-3	*8.43E-02	6.44E-06	1.02E-04	3.36E-05	5.75E-04	3.20E-04	1.03E-03
F-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	1.37E+00	4.62E-02	0.00E+00	2.31E-11	0.00E+00	0.00E+00	2.31E-11
Cr-51	5.37E-01	1.04E-03	9.79E-09	2.40E-08	1.44E-07	2.70E-08	2.05E-07
Mn54	1.12E+01	9.24E-05	5.85E-06	1.68E-06	8.40E-06	3.79E-06	1.97E-05
Fe-55	8.75E-01	2.69E-01	2.49E-10	1.25E-08	0.00E+00	0.00E+00	1.28E-08
Fe-59	2.73E+01	6.47E-04	1.56E-06	5.17E-07	2.13E-06	2.91E-07	4.50E-06
Co-58	1.21E+01	4.08E-04	6.85E-08	2.01E-08	8.87E-07	1.50E-06	2.47E-06
Co-60	3.23E+01	1.50E-05	7.30E-06	2.08E-06	1.87E-05	6.41E-05	9.21E-05
Zn-65	7.79E+00	1.18E-04	2.48E-08	6.25E-10	1.65E-06	6.99E-07	2.37E-06
As-76	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	3.97E+01	5.71E-04	4.85E-08	0.00E+00	0.00E+00	0.00E+00	4.85E-08
Sr-90	1.76E+02	2.77E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	3.42E+01	2.56E-01	0.00E+00	0.00E+00	4.43E-14	2.68E-12	2.72E-12
Mo-99	8.02E+00	1.05E-02	0.00E+00	3.83E-09	0.00E+00	0.00E+00	3.83E-09
Tc-99m	3.32E-01	1.15E-01	0.00E+00	1.28E-11	0.00E+00	0.00E+00	1.28E-11
Ag-110	4.85E+01	1.16E-04	0.00E+00	4.30E-09	3.51E-06	1.13E-07	3.63E-06
Te-131	2.24E-03	2.31E-02	0.00E+00	0.00E+00	7.19E-19	0.00E+00	7.19E-19
I-131	1.26E+00	3.59E-03	0.00E+00	2.58E-11	0.00E+00	0.00E+00	2.58E-11
Cs-134	2.08E+00	3.84E-05	0.00E+00	0.00E+00	0.00E+00	2.67E-10	2.67E-10
Cs-137	1.69E+00	2.62E-06	2.00E-10	2.46E-11	5.78E-10	8.35E-09	9.16E-09
Ce-141	1.94E+01	8.89E-04	0.00E+00	0.00E+00	2.40E-09	0.00E+00	2.40E-09
Nd-147	2.80E+01	2.63E-03	3.96E-09	0.00E+00	0.00E+00	0.00E+00	3.96E-09
W-187	2.26E+01	2.91E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
						SUM	1.16E-03

*8.43E-02 = 8.43×10^{-2}

Table C-6. Dose Estimates for Adult GI-LLI from Fish Ingestion Pathway

Nuclide	Adult GI-LLI Dose Factor $\left[\frac{\text{mrem-ft}^3}{\text{uCi-sec}}\right]$	First Quarter (mrem)	Second Quarter (mrem)	Third Quarter (mrem)	Fourth Quarter (mrem)	Annual Dose (mrem)
H-3	*1.37E-04	7.59E-05	7.01E-05	2.62E-04	1.97E-04	6.06E-04
F-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	7.78E-02	0.00E+00	3.15E-09	0.00E+00	0.00E+00	3.15E-09
Cr-51	1.89E-01	1.61E-06	1.10E-05	1.50E-05	3.70E-06	3.13E-05
Mn54	8.12E+00	1.94E-03	1.57E-03	1.72E-03	1.04E-03	6.27E-03
Fe-55	1.58E-01	1.00E-05	1.10E-04	0.00E+00	0.00E+00	1.20E-04
Fe-59	4.86E+00	1.28E-04	1.19E-04	1.10E-04	2.00E-05	3.78E-04
Co-58	1.09E+00	2.84E-06	2.34E-06	2.29E-05	5.17E-05	7.98E-05
Co-60	2.92E+00	3.01E-04	2.41E-04	4.74E-04	2.20E-03	3.21E-03
Zn-65	2.81E+01	4.09E-05	2.90E-06	1.68E-03	9.59E-04	2.68E-03
As-76	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	2.12E+00	1.20E-06	0.00E+00	0.00E+00	0.00E+00	1.20E-06
Sr-90	9.54E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	3.10E-03	0.00E+00	0.00E+00	1.43E-10	7.52E-11	2.18E-10
Mo-99	1.12E-01	0.00E+00	7.91E-08	0.00E+00	0.00E+00	7.91E-08
Tc-99m	5.06E-04	0.00E+00	1.18E-10	0.00E+00	0.00E+00	1.18E-10
Ag-110	2.01E-01	0.00E+00	2.29E-08	4.11E-06	1.78E-07	4.31E-06
Te-131	1.53E-21	0.00E+00	0.00E+00	3.98E-34	0.00E+00	3.98E-34
I-131	3.13E-02	0.00E+00	8.64E-10	0.00E+00	0.00E+00	8.64E-10
Cs-134	7.52E+00	0.00E+00	0.00E+00	0.00E+00	3.66E-07	3.66E-07
Cs-137	6.13E+00	3.31E-07	1.15E-07	5.89E-07	1.15E-05	1.25E-05
Ce-141	3.44E-02	0.00E+00	0.00E+00	1.25E-09	0.00E+00	1.25E-09
Nd-147	1.19E+00	8.16E-08	0.00E+00	0.00E+00	0.00E+00	8.16E-08
W-187	2.38E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
					SUM	1.34E-02

*1.37E-04 = 1.37×10^{-4}

Table C-7. Dose Estimates for Adult GI-LLI from Shore Exposure Pathway

Nuclide	Adult GI-LLI Dose Factor $\left[\frac{\text{mrem-ft}^3}{\text{uCi-sec}}\right]$	First Quarter (mrem)	Second Quarter (mrem)	Third Quarter (mrem)	Fourth Quarter (mrem)	Annual Dose (mrem)
H-3	*0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
F-18	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	2.48E-04	0.00E+00	1.00E-11	0.00E+00	0.00E+00	1.00E-11
Cr-51	1.01E-04	8.61E-10	5.87E-09	8.00E-09	1.98E-09	1.67E-08
Mn54	3.01E-02	7.20E-06	5.82E-06	6.37E-06	3.86E-06	2.33E-05
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	5.92E-03	1.56E-07	1.45E-07	1.34E-07	2.43E-08	4.60E-07
Co-58	8.23E-03	2.15E-08	1.77E-08	1.73E-07	3.90E-07	6.02E-07
Co-60	4.68E-01	4.83E-05	3.87E-05	7.60E-05	3.52E-04	5.15E-04
Zn-65	1.62E-02	2.36E-08	1.67E-09	9.69E-07	5.53E-07	1.55E-06
As-76	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	4.70E-07	2.66E-13	0.00E+00	0.00E+00	0.00E+00	2.66E-13
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	1.31E-05	0.00E+00	0.00E+00	6.05E-13	3.18E-13	9.23E-13
Mo-99	8.59E-05	0.00E+00	6.07E-11	0.00E+00	0.00E+00	6.07E-11
Tc-99m	3.65E-06	0.00E+00	8.53E-13	0.00E+00	0.00E+00	8.53E-13
Ag-110	7.47E-02	0.00E+00	8.53E-09	1.53E-06	6.62E-08	1.60E-06
Te-131	1.21E-07	0.00E+00	0.00E+00	3.15E-20	0.00E+00	3.15E-20
I-131	3.72E-04	0.00E+00	1.03E-11	0.00E+00	0.00E+00	1.03E-11
Cs-134	1.49E-01	0.00E+00	0.00E+00	0.00E+00	7.25E-09	7.25E-09
Cs-137	2.24E-01	1.21E-08	4.19E-09	2.15E-08	4.19E-07	4.57E-07
Ce-141	2.97E-04	0.00E+00	0.00E+00	1.08E-11	0.00E+00	1.08E-11
Nd-147	1.82E-04	1.25E-11	0.00E+00	0.00E+00	0.00E+00	1.25E-11
W-187	4.96E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
					SUM	5.43E-04

*0.00E+00 = 0.00×10^0

Table C-8. Dose Contribution from Each Pathway Specific to Radionuclide

Teen - Total Body Dose Estimate (mrem)				Adult - GI-LLI Dose Estimate (mrem)			
Nuclide	Potable Water Pathway	Fish Ingestion Pathway	Shore Exposure Pathway	Nuclide	Potable Water Pathway	Fish Ingestion Pathway	Shore Exposure Pathway
H-3	*7.28E-04	4.69E-04	0.00E+00	H-3	1.03E-03	6.06E-04	0.00E+00
F-18	0.00E+00	0.00E+00	0.00E+00	F-18	0.00E+00	0.00E+00	0.00E+00
Na-24	2.18E-11	3.24E-09	5.58E-11	Na-24	2.31E-11	3.15E-09	1.00E-11
Cr-51	7.72E-10	1.28E-07	9.33E-08	Cr-51	2.05E-07	3.13E-05	1.67E-08
Mn54	1.16E-06	3.99E-04	1.30E-04	Mn54	1.97E-05	6.27E-03	2.33E-05
Fe-55	5.13E-09	5.24E-05	0.00E+00	Fe-55	1.28E-08	1.20E-04	0.00E+00
Fe-59	4.89E-07	4.47E-05	2.57E-06	Fe-59	4.50E-06	3.78E-04	4.60E-07
Co-58	2.58E-07	9.00E-06	3.36E-06	Co-58	2.47E-06	7.98E-05	6.02E-07
Co-60	1.01E-05	3.85E-04	2.87E-03	Co-60	9.21E-05	3.21E-03	5.15E-04
Zn-65	1.59E-06	1.97E-03	8.65E-06	Zn-65	2.37E-06	2.68E-03	1.55E-06
As-76	0.00E+00	0.00E+00	0.00E+00	As-76	0.00E+00	0.00E+00	0.00E+00
Sr-89	8.64E-09	2.33E-07	1.48E-12	Sr-89	4.85E-08	1.20E-06	2.66E-13
Sr-90	0.00E+00	0.00E+00	0.00E+00	Sr-90	0.00E+00	0.00E+00	0.00E+00
Sr-92	5.80E-15	5.08E-13	5.13E-12	Sr-92	2.72E-12	2.18E-10	9.23E-13
Mo-99	3.08E-10	6.92E-09	3.38E-10	Mo-99	3.83E-09	7.91E-08	6.07E-11
Tc-99m	2.59E-13	2.62E-12	4.65E-12	Tc-99m	1.28E-11	1.18E-10	8.53E-13
Ag-110	4.96E-09	6.44E-09	8.95E-06	Ag-110	3.63E-06	4.31E-06	1.60E-06
Te-131	1.57E-18	9.01E-34	1.76E-19	Te-131	7.19E-19	3.98E-34	3.15E-20
I-131	5.05E-11	1.84E-09	5.74E-11	I-131	2.58E-11	8.64E-10	1.03E-11
Cs-134	6.58E-09	9.83E-06	4.05E-08	Cs-134	2.67E-10	3.66E-07	7.25E-09
Cs-137	1.58E-07	2.35E-04	2.55E-06	Cs-137	9.16E-09	1.25E-05	4.57E-07
Ce-141	7.08E-14	3.99E-14	6.01E-11	Ce-141	2.40E-09	1.25E-09	1.08E-11
Nd-147	4.85E-14	1.08E-12	7.00E-11	Nd-147	3.96E-09	8.16E-08	1.25E-11
W-187	0.00E+00	0.00E+00	0.00E+00	W-187	0.00E+00	0.00E+00	0.00E+00
Subtotal	7.41E-04	3.57E-03	3.03E-03	Subtotal	1.16E-03	1.34E-02	5.43E-04
	Total Dose		7.34E-03		Total Dose		1.51E-02

*7.28E-04 = 7.28×10^{-4}

Table C-9. Dose Contribution from Each Pathway Specific to Radionuclide (Tritium Dose Calculated Using Quality Factor Of 1.0)

Teen - Total Body Dose Estimate (mrem)				Adult - GI-LLI Dose Estimate (mrem)			
Nuclide	Potable Water Pathway	Fish Ingestion Pathway	Shore Exposure Pathway	Nuclide	Potable Water Pathway	Fish Ingestion Pathway	Shore Exposure Pathway
H-3	*4.28E-04	2.76E-04	0.00E+00	H-3	6.06E-04	3.56E-04	0.00E+00
F-18	0.00E+00	0.00E+00	0.00E+00	F-18	0.00E+00	0.00E+00	0.00E+00
Na-24	2.18E-11	3.24E-09	5.58E-11	Na-24	2.31E-11	3.15E-09	1.00E-11
Cr-51	7.72E-10	1.28E-07	9.33E-08	Cr-51	2.05E-07	3.13E-05	1.67E-08
Mn54	1.16E-06	3.99E-04	1.30E-04	Mn54	1.97E-05	6.27E-03	2.33E-05
Fe-55	5.13E-09	5.24E-05	0.00E+00	Fe-55	1.28E-08	1.20E-04	0.00E+00
Fe-59	4.89E-07	4.47E-05	2.57E-06	Fe-59	4.50E-06	3.78E-04	4.60E-07
Co-58	2.58E-07	9.00E-06	3.36E-06	Co-58	2.47E-06	7.98E-05	6.02E-07
Co-60	1.01E-05	3.85E-04	2.87E-03	Co-60	9.21E-05	3.21E-03	5.15E-04
Zn-65	1.59E-06	1.97E-03	8.65E-06	Zn-65	2.37E-06	2.68E-03	1.55E-06
As-76	0.00E+00	0.00E+00	0.00E+00	As-76	0.00E+00	0.00E+00	0.00E+00
Sr-89	8.64E-09	2.33E-07	1.48E-12	Sr-89	4.85E-08	1.20E-06	2.66E-13
Sr-90	0.00E+00	0.00E+00	0.00E+00	Sr-90	0.00E+00	0.00E+00	0.00E+00
Sr-92	5.80E-15	5.08E-13	5.13E-12	Sr-92	2.72E-12	2.18E-10	9.23E-13
Mo-99	3.08E-10	6.92E-09	3.38E-10	Mo-99	3.83E-09	7.91E-08	6.07E-11
Tc-99m	2.59E-13	2.62E-12	4.65E-12	Tc-99m	1.28E-11	1.18E-10	8.53E-13
Ag-110	4.96E-09	6.44E-09	8.95E-06	Ag-110	3.63E-06	4.31E-06	1.60E-06
Te-131	1.57E-18	9.01E-34	1.76E-19	Te-131	7.19E-19	3.98E-34	3.15E-20
I-131	5.05E-11	1.84E-09	5.74E-11	I-131	2.58E-11	8.64E-10	1.03E-11
Cs-134	6.58E-09	9.83E-06	4.05E-08	Cs-134	2.67E-10	3.66E-07	7.25E-09
Cs-137	1.58E-07	2.35E-04	2.55E-06	Cs-137	9.16E-09	1.25E-05	4.57E-07
Ce-141	7.08E-14	3.99E-14	6.01E-11	Ce-141	2.40E-09	1.25E-09	1.08E-11
Nd-147	4.85E-14	1.08E-12	7.00E-11	Nd-147	3.96E-09	8.16E-08	1.25E-11
W-187	0.00E+00	0.00E+00	0.00E+00	W-187	0.00E+00	0.00E+00	0.00E+00
Subtotal	4.42E-04	3.38E-03	3.03E-03	Subtotal	7.31E-04	1.32E-02	5.43E-04
	Total Dose		6.85E-03		Total Dose		1.44E-02

*4.28E-04 = 4.28×10^{-4}

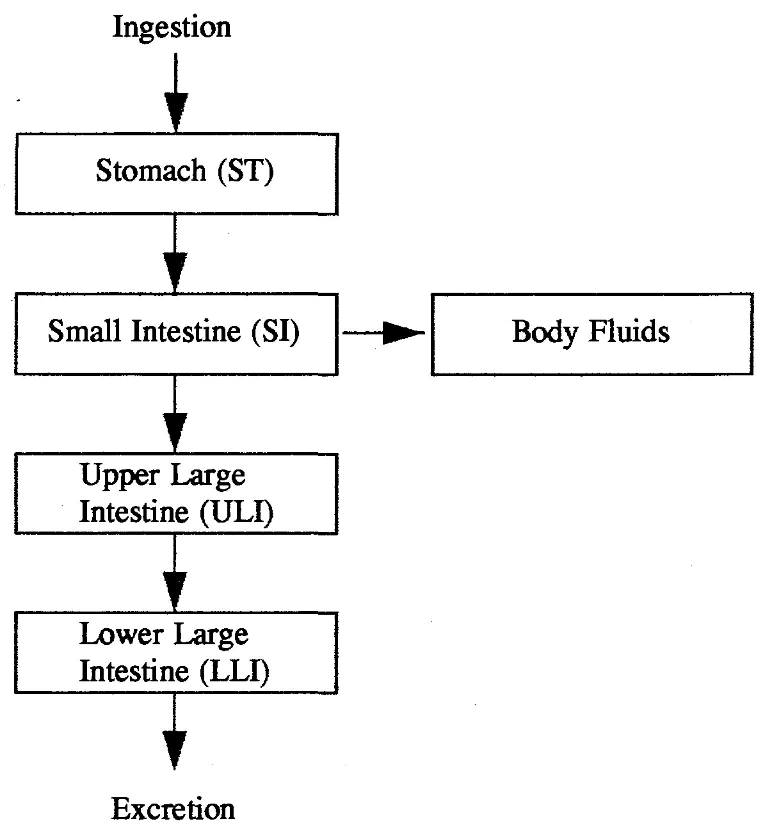
APPENDIX D:

ICRP 30 DOSIMETRIC MODELS FOR THE GASTROINTESTINAL TRACT AND RESPIRATORY SYSTEM

D.1 ICRP 30 Dosimetric Model for the Gastrointestinal Tract (ICRP 1978)

The dosimetric model given in ICRP 30 for the gastrointestinal (GI) tract divides the GI tract into four compartments as shown in Figure D-1 (ICRP 1978).

Fig. D-1. Compartmental model used to describe the kinetics of radionuclides in the gastrointestinal tract.



The clearance of ingested radioactive material from the different compartments of the GI tract is described by a series of first order differential equations (ICRP 1978):

$$\frac{d}{dt}q_{ST}(t) = -\lambda_{ST}q_{ST}(t) - \lambda_R q_{ST}(t) + I(t)$$

$$\frac{d}{dt}q_{SI}(t) = -\lambda_{SI}q_{SI}(t) - \lambda_R q_{SI}(t) - \lambda_B q_{SI}(t) + \lambda_{ST}q_{ST}(t)$$

$$\frac{d}{dt}q_{ULI}(t) = -\lambda_{ULI}q_{ULI}(t) - \lambda_R q_{ULI}(t) + \lambda_{SI}q_{SI}(t)$$

$$\frac{d}{dt}q_{LLI}(t) = -\lambda_{LLI}q_{LLI}(t) - \lambda_R q_{LLI}(t) + \lambda_{ULI}q_{ULI}(t)$$

Where

$q(t)$ = amount of ingested radioactivity in the respective compartment at time t;
λ_{ST} = biological clearance rate from stomach to small intestine compartment;
λ_{SI} = biological clearance rate from small intestine to upper large intestine;
λ_B = biological clearance rate from small intestine to body fluids;
λ_{ULI} = biological clearance rate upper large intestine to lower large intestine;
λ_{LLI} = biological clearance rate of lower large intestine via excretion.
λ_R = radioactive decay constant; and
$I(t)$ = rate of ingestion of activity of the radionuclide at time t.

These differential equations describe the translocation of radioactivity in and out of the respective compartments of the gastrointestinal tract. The $\lambda q(t)$ expression represents the rate of transfer from one compartment to another. Table D-1 contains the biological clearance rates recommended in ICRP 30 for each compartment of the GI tract (ICRP 1978).

Table D-1. Biological Clearance Rate for Gastrointestinal Tract

Section of GI Tract	Mean Residence Time (day)	Biological Clearance Rates (days^{-1})
Stomach (ST)	1/24	24
Small Intestine (SI)	4/24	6
Upper Large Intestine (ULI)	13/24	1.8
Lower Large Intestine (LLI)	24/24	1

The value of λ_B is derived from f_1, "the fraction of a stable element reaching the body fluids following ingestion" (ICRP 1978):

$$f_1 = \lambda_B \div (\lambda_{SI} + \lambda_B)$$

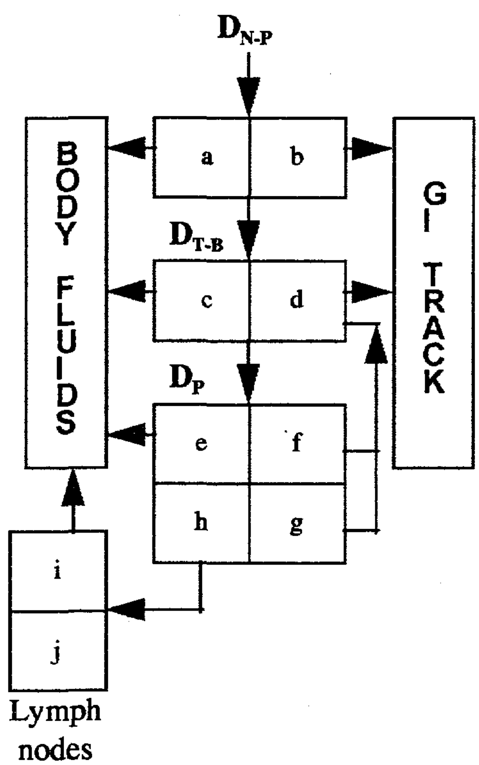
$$\therefore \lambda_B = f_1 \lambda_{SI} \div (1 - f_1)$$

Relevant metabolic data for each radioactive element are provided in ICRP 30. Values of f_1 are given in the metabolic data in addition to other data concerning the distribution and retention of specific elements. The distribution of certain radionuclides through the GI-tract may vary depending on the chemical composition of the radionuclide. Since the translocation rate of certain radionulcides may be affected by their chemical composition, different values of f_1 are given in ICRP 30 for different compound types of these radionuclides. For example, all "organically complexed compounds and...all inorganic compounds" of cobalt-60 except oxides and hydroxides are assigned a value of 0.3 for f_1; for "oxides and hydroxides of cobalt and for all other inorganic compounds", f_1 is given as 0.05 (ICRP 1978).

D.2 ICRP 30 Dosimetric Model for the Respiratory System (ICRP 1978)

The respiratory system described in ICRP 30 is separated into three distinct regions: the nasal passage (N-P), the trachea and bronchial tree (T-B) and the pulmonary parenchyma (P) (ICRP 1978). Three deposition fractions, D_{N-T}, D_{T-B}, and D_P denote the fraction of inhaled radioactive material that initially deposits in the N-P, T-B, and P regions, respectively. The three regions are further divided into two or four compartments as shown in figure D-2 (ICRP 1978).

Fig. D-2. Mathematical Model used to describe clearance from respiratory system.



The different compartments represent different clearance pathways for the respective region. For example, compartments a, c, and e are associated with absorption into the body fluids whereas compartments b, d, f, and g are associated with particle translocation to the GI tract. Each compartment is associated with a clearance half-time T and removal fraction F.

The clearance rate of inhaled radionuclides from the lung will vary depending on the chemical characteristics of radionuclides. For this reason, inhaled radionuclides are "classified as D, W, or Y which refer to their retention in the pulmonary region" (ICRP 1978) Radionuclide with clearance half-time less than 10 days are classified as D, clearance half-times ranging from 10 to 100 days are classified as W and clearance half-times greater than 100 days are classified as Y. Table D-2 contains clearance times and removal fraction associated with each compartment classified by clearance type, D, W, or Y.

Table D-2. Clearance Times And Removal Fraction for The Dosimetric Model of the Respiratory System

Region	Compartment	Class					
		D		W		Y	
		T day	F	T day	F	T day	F
N-P	a	0.01	0.5	0.01	0.1	0.01	0.01
(D_{N-P} = 0.30)	b	0.01	0.5	0.4	0.9	0.40	0.99
T-B	c	0.01	0.95	0.01	0.5	0.01	0.01
(D_{T-B} = 0.30)	d	0.2	0.05	0.2	0.5	0.2	0.99
	e	0.5	0.8	50	0.15	500	0.05
P	f	n.a.	n.a.	1.0	0.4	1.0	0.4
(D_P = 0.30)	g	n.a.	n.a.	50	0.4	500	0.4
	h	0.5	0.2	50	0.05	500	0.15
L	i	0.5	1.0	50	1.0	1000	0.9
	j	n.a.	n.a.	n.a.	n.a.	∞	0.1

The differential equations listed below characterize the clearance of inhaled radioactive material from the different compartments of the lungs:

$$\frac{d}{dt}q_a(t) = I(t)\, D_{N\text{-}P}\, F_a - \lambda_a q_a(t) - \lambda_R q_a(t)$$

$$\frac{d}{dt}q_b(t) = I(t)\, D_{N\text{-}P}\, F_b - \lambda_b\, q_a(t) - \lambda_R q_b(t)$$

$$\frac{d}{dt}q_c(t) = I(t)\, D_{T\text{-}P}\, F_c - \lambda_c q_c(t) - \lambda_R q_c(t)$$

$$\frac{d}{dt}q_d(t) = I(t)\, D_{T\text{-}B}\, F_d - \lambda_f q_f(t) - \lambda_g q_g(t) - \lambda_d q_d(t) - \lambda_R q_d(t)$$

$$\frac{d}{dt}q_e(t) = I(t)\, D_P\, F_e - \lambda_e q_e(t) - \lambda_R q_e(t)$$

$$\frac{d}{dt}q_f(t) = I(t)\, D_P\, F_f - \lambda_f q_f(t) - \lambda_R q_f(t)$$

$$\frac{d}{dt}q_g(t) = I(t)\, D_P\, F_g - \lambda_g q_g(t) - \lambda_R q_g(t)$$

$$\frac{d}{dt}q_h(t) = I(t)\, D_P\, F_h - \lambda_h q_h(t) - \lambda_R q_h(t)$$

$$\frac{d}{dt}q_i(t) = F_i \lambda_h q_h(t) - \lambda_i q_i(t) - \lambda_R q_i(t)$$

$$\frac{d}{dt}q_j(t) = F_j \lambda_h q_h(t) - \lambda_R q_j(t)$$

Where

$q_a(t)$, $q_b(t)$, etc. = activity of an inhaled radionuclide in compartment a, b, etc. at time t;

$I(t)$ = rate of inhalation of activity of the radionuclide;

λ_a to λ_i = biological clearance rates of compartments a to i;

λ_R = radioactive decay constant of the radionuclide; and

F_a to F_j = fractions of material entering the various regions of the lung associated with the various compartments contained therein.