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UC-610

Radiation Dose Assessments to Support Evaluations of Radiological Control Levels for Recycling or Reuse of Materials and Equipment

R. L. Hill
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July 1995

Prepared for
the U.S. Department of Energy
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Pacific Northwest Laboratory
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ABSTRACT

Pacific Northwest Laboratory (PNL) is providing Environmental Protection Support and Assistance to the U.S. Department of Energy (DOE), Office of Environmental Guidance, Air, Water, and Radiation Division (DOE/EH-232). As part of this effort, PNL is collecting data and conducting technical evaluations to support DOE analyses of the feasibility of developing radiological control levels for recycling or reuse of metals, concrete, or equipment containing residual radioactive contamination from DOE operations. The radiological control levels will be risk-based, as developed through a radiation exposure scenario and pathway analysis. The analysis will include evaluation of relevant radionuclides, potential mechanisms of exposure, and both health and non-health-related impacts. The main objective of this report is to develop a methodology for establishing radiological control levels for recycle or reuse.

This report provides the results of the radiation exposure scenario and pathway analyses for 42 key radionuclides generated during DOE operations that may be contained in metals or equipment considered for either recycling or reuse. The scenarios and information developed by the International Atomic Energy Agency (IAEA) in Safety Series No. 111-P-1.1 (1992), Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities, are used as the initial basis for this study. The analyses were performed for both selected worker populations at metal smelters and for the public downwind of a smelter facility. Doses to the public downwind were estimated using the U.S. Environmental Protection Agency's (EPA) CAP88-PC computer code with generic data on atmospheric dispersion and population density. Potential non-health-related effects of residual activity on electronics and on film were also analyzed.

As additional relevant data are collected, this initial generic assessment may need to be revised. Thus, this report is intended to serve as an intermediate product that will help to focus future technical support efforts for the DOE.

ACKNOWLEDGMENTS

The authors would like to thank all of those individuals who contributed to the development of this report. We greatly appreciate the guidance, assistance, technical review, and encouragement provided by Andrew Wallo III and Harold H. Peterson of the U.S. Department of Energy. We are grateful to the individuals from Pacific Northwest Laboratory, Argonne National Laboratory, Brookhaven National Laboratory, Savannah River Laboratory, DOE/EM-331, DOE/EH-411, and DOE/EH-232 who provided constructive suggestions and technical review. Finally, the authors are indebted to Sandra Snyder for final technical review and publication oversight; to Donald Hanley and James Weber for editorial assistance; and to Marianna Cross and Darla Kennedy for word processing assistance.

EXECUTIVE SUMMARY

Pacific Northwest Laboratory (PNL) is providing Environmental Protection Support and Assistance to the U.S. Department of Energy (DOE), Office of Environmental Guidance, Air, Water and Radiation Division (DOE/EH-232). As part of this effort, PNL is collecting data and conducting technical analyses to support joint efforts by the DOE, the U.S. Environmental Protection Agency (EPA), and the U.S. Nuclear Regulatory Commission (NRC) to develop radiological control levels for the recycling and reuse of scrap materials and equipment that contain residual radioactive contamination. The analysis includes determining relevant radionuclides, potential mechanisms of exposure, and methods to determine possible non-health-related impacts from residual radioactive contamination in materials considered for recycling or reuse.

The data and models contained in this report may be used by DOE (in coordination with other agencies) in selecting radiological control criteria for recycling that are as low as reasonably achievable (ALARA) and in supporting environmental regulations. Figure S.1 shows how such control criteria, once selected, might be applied in practice. To determine if recycling is the "preferred" action or approach for management of material, DOE has identified two criteria. The action must be 1) environmentally acceptable and cost-effective and/or 2) environmentally preferred. Under this approach, which is one of several being considered, some situations exist under which the direct costs may be higher for recycling than for burial. However, the indirect benefits of recycling (e.g., not having to mine, mill, refine, and refabricate the metal, thus avoiding the environmental impacts associated with these processes) could balance the higher economic costs of recycling.[a] Figure S.1 includes a logical framework for DOE-Only Recycle. However, this subject is not further addressed in this document. The focus of this document is unrestricted release.

(a) April 2, 1993, letter from Andrew Wallo (Director of the Air, Water, and Radiation Division of the U.S. Department of Energy Office of Environmental Guidance) to W.E. Kennedy, Jr. (Pacific Northwest Laboratory).

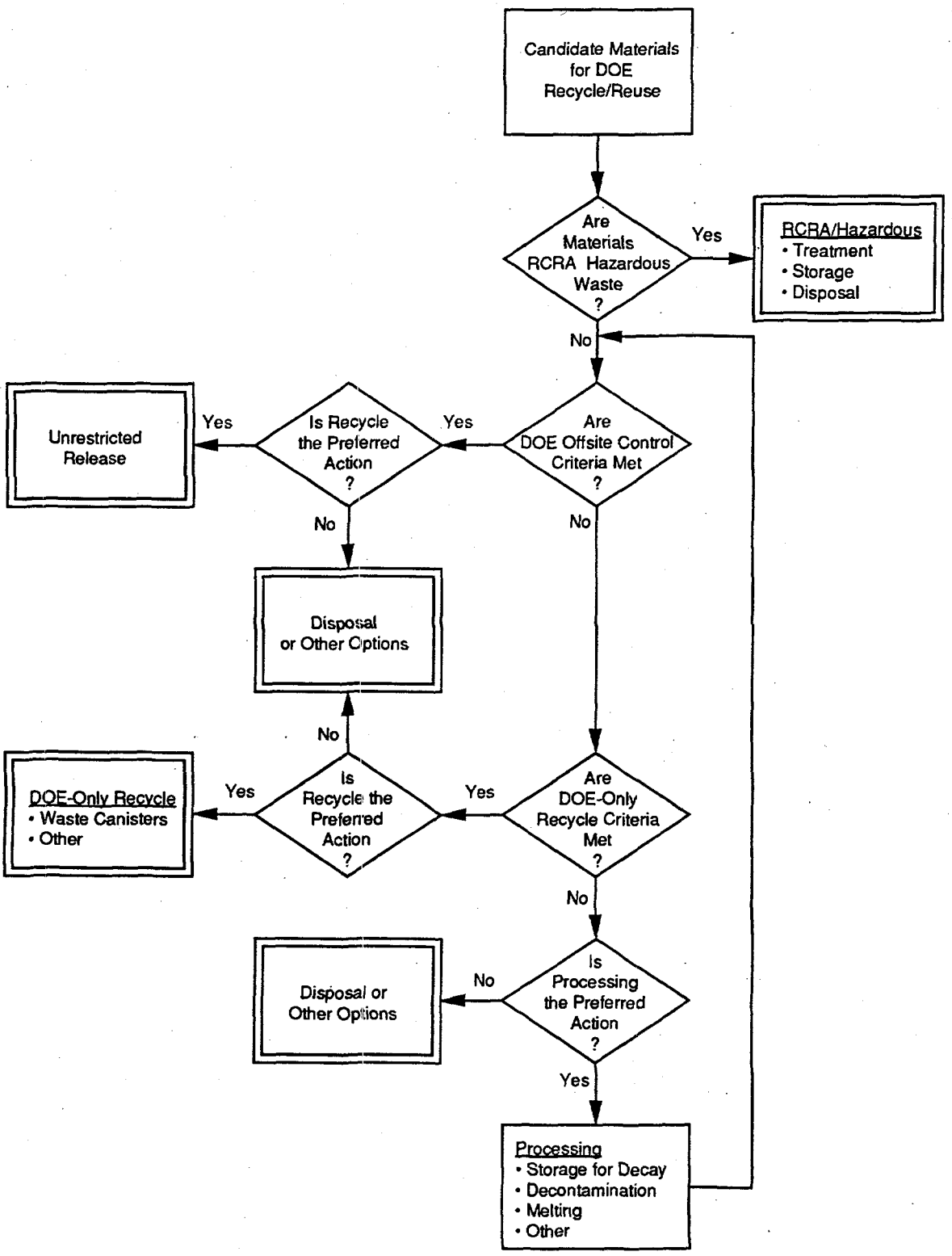


FIGURE S.1. Decision Process for Evaluating Potential for Recycling and Reuse

The results described in this report are based on generic exposure scenarios and pathway analyses using 42 radionuclides determined to be potentially present as residual contamination in metals or equipment from DOE operations. The scenarios and information developed by the International Atomic Energy Agency (IAEA) in Safety Series No. 111-P-1.1 (IAEA 1992), Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities, were considered in developing the initial radiation exposure pathway and scenario analysis.

Although several public dose limits were considered, the control criteria in this report are based on 1) a dose of 1 mrem/yr to a worker in a smelter or an individual who uses consumer products made from recycled materials, 2) a dose of 0.1 mrem/yr to an individual downwind from a smelter used to process recycled metals, or 3) non-health impacts associated with potential radiation effects on electronics or film.

Table S.1 summarizes the limiting concentrations based on individual radiation dose for residual contamination in (or on) recycled materials. For the radionuclides in Table S.1, doses to smelter workers or users of consumer products provided the most restrictive (smallest) derived residual concentrations. This table shows the radiological control levels for bulk materials, in units of pCi/g, for steel, aluminum, copper, and concrete. The table also lists control levels for surface contamination in units of pCi/cm^2.

Doses to individual members of the public downwind of a smelter were estimated using the generic data on atmospheric dispersion and medium-high population density in the EPA's CAP88-PC software. Doses were calculated to the maximally exposed individual (MEI) downwind of a smelter. For all radionuclides considered except ^{238}U, the most restrictive concentrations were based on doses to individual workers rather than to the downwind MEI.

Also evaluated were non-health-related impacts that industry may experience from residual contamination in recycled metals, such as those used in electronics and film. Upon investigation, it was found that most electronic components can withstand doses in excess 25 mrem. Thus, recycling the materials considered in this report at or below the contamination levels in Table S.1 would have little impact on the electronics industry.

TABLE S.1. Radiological Control Levels Based on an Individual Dose of 1 mrem/yr for Recycling and Reuse of DOE Metals or Equipment Containing Residual Radioactive Contamination[a]

Radionuclide	Bulk Contamination (pCi/g)				Surface Contamination (pCi/cm^2)
	Steel	Aluminum	Copper	Concrete	Tools & Equip[b]
^{3}H	5.7E+06	2.9E+07	8.8E+06	1.7E+08	2.6E+06
^{14}C	1.9E+05	8.3E+05	2.6E+05	5.6E+05	7.9E+04
^{36}Cl	---[c]	---	---	1.7E+04	---
^{41}Ca	---	---	---	8.5E+03	---
^{54}Mn	1.3E+01	3.6E+01	1.9E+02	5.9E+00	3.3E+03
^{55}Fe	1.5E+04	2.5E+04	2.1E+05	3.7E+03	1.5E+05
^{57}Co	9.0E+01	2.4E+02	1.4E+03	7.9E+01	1.8E+04
^{60}Co	4.4E+00	1.3E+01	6.7E+01	1.7E+00	1.1E+03
^{63}Ni	5.2E+05	2.6E+06	1.0E+06	5.5E+06	3.0E+05
^{65}Zn	1.7E+01	4.9E+01	2.6E+02	6.7E+00	3.3E+03
^{79}Se	4.3E+04	2.2E+05	6.7E+04	7.2E+05	2.0E+04
^{90}Sr	2.5E+03	1.3E+04	4.3E+03	3.3E+04	1.3E+03
^{93}Zr	3.8E+04	1.9E+05	1.9E+05	1.4E+05	6.7E+04
^{94}Nb	7.1E+00	2.0E+01	1.1E+02	3.5E+00	1.7E+03
^{99}Tc	1.3E+05	3.0E+05	4.3E+05	1.4E+05	1.2E+05
^{106}Ru	4.4E+01	1.3E+02	6.7E+02	2.4E+01	4.2E+03
^{110m}Ag	3.8E+00	1.1E+01	5.8E+01	1.7E+00	9.8E+02
^{125}Sb	2.4E+01	6.5E+01	3.6E+02	1.3E+01	5.6E+03
^{129}I	1.3E+03	1.2E+03	2.0E+03	5.3E+02	5.9E+02
^{134}Cs	6.3E+00	1.8E+01	9.6E+01	3.1E+00	9.7E+02
^{137}Cs	1.9E+01	5.3E+01	2.8E+02	9.7E+00	2.0E+03
^{144}Ce	7.2E+02	1.6E+03	1.2E+04	6.0E+02	6.6E+03
^{147}Pm	7.9E+04	4.0E+05	4.0E+05	2.5E+05	1.2E+05
^{151}Sm	1.0E+05	5.2E+05	5.2E+05	3.6E+05	2.4E+05
^{152}Eu	9.3E+00	2.6E+01	1.4E+02	3.9E+00	2.2E+03
^{154}Eu	8.8E+00	2.5E+01	1.3E+02	3.8E+00	2.0E+03
^{226}Ra	2.0E+02	1.0E+03	5.1E+02	1.3E+03	1.4E+02
^{228}Th	1.1E+01	5.3E+01	5.3E+01	3.4E+01	4.1E+01
^{229}Th	1.9E+00	9.6E+00	9.6E+00	6.1E+00	7.0E+00
^{230}Th	1.3E+01	6.3E+01	6.3E+01	4.0E+01	4.6E+01
^{232}Th	3.0E+00	1.5E+01	1.5E+01	9.5E+00	1.0E+01
^{232}U	5.0E+00	2.5E+01	2.5E+01	1.6E+01	2.1E+01
^{233}U	2.6E+01	1.3E+02	1.3E+02	8.0E+01	1.1E+02
^{234}U	2.6E+01	1.3E+02	1.3E+02	8.0E+01	1.1E+02
^{235}U	2.8E+01	1.4E+02	1.4E+02	8.6E+01	1.1E+02
^{238}U	2.8E+01	1.4E+02	1.4E+02	8.7E+01	1.1E+02
^{237}Np	6.4E+00	3.2E+01	3.2E+01	2.1E+01	1.7E+01
^{238}Pu	1.1E+01	5.5E+01	5.5E+01	3.5E+01	4.6E+01
^{239}Pu	1.0E+01	5.0E+01	5.0E+01	3.2E+01	4.1E+01

TABLE S.1. (Cont'd)

Radionuclide	Bulk Contamination (pCi/g)				Surface Contamination (pCi/cm^2)
	Steel	Aluminum	Copper	Concrete	Tools & Equip[b]
^{240}Pu	1.0E+01	5.0E+01	5.0E+01	3.2E+01	4.1E+01
^{241}Pu	5.8E+02	2.9E+03	2.9E+03	1.8E+03	2.4E+03
^{241}Am	6.0E+00	3.0E+01	3.0E+01	2.0E+01	1.6E+01

(a) Calculations were made using the EXTDF module from the GENII Software System (Napier et al. 1988).
(b) The values presented in this table are risk-based using a 1 mrem annual dose limit and assuming certain exposure conditions. The values for most radionuclides are greater than the surface contamination values that appear in the DOE *Radiological Control Manual* (DOE 1994). The DOE (1994) values were based on the values appearing in the U.S. NRC Regulatory Guide 1.86 (NRC 1974). These values are roughly related to instrumentation responses and an assumed dose rate of 10 mrem/yr, for selected categories of radionuclides.
(c) A dash (---) indicates a concrete activation product that was considered only for concrete-recycling scenarios.

Table S.2 summarizes the limiting concentrations in recycled materials based on a 0.2-mrad exposure to film stored for one month in a box constructed of either undiluted steel or concrete for each of the 42 radionuclides considered. This table shows the radiological control levels for bulk materials, in units of pCi/g, for steel and concrete, with and without an uncontaminated 0.5-cm lead lining. The control levels for film are more restrictive than those derived from doses to smelter workers or consumers for the photon-emitting radionuclides. This result is considered to be preliminary because of the conservative assumptions used in the analysis and because it is unlikely that storage areas for film would be constructed of undiluted, recycled steel or concrete. Further evaluation of the scenarios, assumptions, and data are needed.

TABLE S.2. Radiological Control Levels for Bulk Contamination Based on 0.2-mrad Exposure to Film Stored for One Month (pCi/g)[a]

Lead-Lined Radionuclide	Steel	Lead-Lined Steel	Concrete	Concrete
^{3}H	3.5E+09	---[b]	1.7E+09	---
^{14}C	2.1E+05	2.3E+12	1.2E+05	8.2E+12
^{36}Cl	NA[c]	NA	5.5E+03	4.6E+05
^{41}Ca	NA	NA	1.2E+03	---
^{54}Mn	3.6E+00	6.8E+00	2.7E+00	5.2E+00
^{55}Fe	3.5E+03	---	5.39E+02	---
^{57}Co	2.6E+01	6.5E+03	2.9E+01	5.4E+03
^{60}Co	1.2E+00	2.0E+00	8.3E-01	1.3E+00
^{63}Ni	2.8E+06	5.7E+13[d]	1.3E+06	5.7E+13[d]
^{65}Zn	4.8E+00	7.6E+00	3.2E+00	5.0E+00
^{79}Se	2.8E+05	6.9E+10[d]	1.6E+05	6.9E+10[d]
^{90}Sr	7.6E+03	1.5E+06	9.7E+03	3.2E+06
^{93}Zr	3.5E+06	2.5E+09[d]	1.5E+06	2.5E+09[d]
^{94}Nb	2.0E+00	4.3E+00	1.6E+00	3.4E+00
^{99}Tc	4.1E+04	6.7E+08	3.4E+04	1.8E+09
^{106}Ru	1.3E+01	3.6E+01	1.1E+01	2.9E+01
^{110m}Ag	1.1E+00	2.1E+00	8.0E-01	1.5E+00
^{125}Sb	6.7E+00	2.1E+01	5.7E+00	1.7E+01
^{129}I	8.6E+02	1.8E+08[d]	1.5E+02	1.8E+08[d]
^{134}Cs	1.8E+00	3.8E+00	1.4E+00	3.0E+00
^{137}Cs	5.3E+00	1.4E+01	4.5E+00	1.1E+01
^{144}Ce	2.2E+02	2.4E+06	2.2E+02	3.2E+06
^{147}Pm	8.2E+04	5.5E+09	5.9E+04	9.9E+09
^{151}Sm	5.0E+05	2.6E+13[d]	9.5E+04	2.6E+13[d]
^{152}Eu	2.6E+00	5.0E+00	1.8E+00	3.3E+00
^{154}Eu	2.5E+00	4.7E+00	1.8E+00	3.2E+00
^{226}Ra	7.4E+02	1.0E+06	8.6E+02	9.7E+05
^{228}Th	1.8E+03	2.8E+05	9.6E+02	3.0E+05
^{229}Th	4.9E+01	1.9E+04	4.4E+01	2.1E+04
^{230}Th	7.0E+03	4.8E+08	1.6E+03	5.0E+08
^{232}Th	8.9E+03	1.1E+05[d]	1.7E+03	1.1E+05[d]
^{232}U	6.1E+03	3.6E+08	1.2E+03	4.8E+08
^{233}U	9.1E+03	1.4E+08	3.2E+03	1.9E+08
^{234}U	7.7E+03	6.5E+08	1.4E+03	8.7E+08
^{235}U	2.9E+01	1.1E+04	3.3E+01	1.2E+04
^{238}U	1.0E+04	3.4E+05[d]	1.7E+03	3.4E+05[d]
^{237}Np	2.0E+02	4.0E+05	1.1E+02	4.3E+05
^{238}Pu	8.2E+03	1.7E+13[d]	1.3E+03	1.7E+13[d]
^{239}Pu	1.6E+04	5.4E+08	3.3E+03	7.3E+08

TABLE S.2. (Cont'd)

Lead-Lined Radionuclide	Steel	Lead-Lined Steel	Concrete	Concrete
^{240}Pu	8.6E+03	2.3E+11[d]	1.4E+03	2.3E+11[d]
^{241}Pu	1.1E+09	---	5.0E+08	---
^{241}Am	4.0E+02	3.4E+12[d]	1.6E+02	3.4E+12[d]

(a) Calculations were made assuming that the film was stored for one month in a rectangular container made from either steel or concrete, with or without lead shielding lining (0.5-cm thickness) the box. The radiological control levels were determined based on the 0.2 mrad (0.002 cGy) limit recommended by the NCRP (1989) for diagnostic x-ray film, and assumed 1 mrad = 1 mrem.

(b) For radionuclides having no gamma emissions, the lead lining reduced the dose to zero resulting in initial control levels that approached infinity. This is represented by (---) in the table.

(c) "NA" indicates that this concrete activation product was Not Applicable to this scenario and was considered only for concrete recycling scenarios.

(d) The specific activity of the radionuclide is provided. The calculated control level exceeds the specific activity of the radionuclide.

CONTENTS

FIGURES

TABLES

ACRONYMS AND ABBREVIATIONS

ALARA	as low as reasonably achievable
ANL	Argonne National Laboratory
CAP88-PC	Clean Air Act Assessment Package 1988 - Personal Computer code
CMOS	complementary metal oxide semiconductor
DCF	dose conversion factor
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EPA	U.S. Environmental Protection Agency
IAEA	International Atomic Energy Agency
IC	integrated circuit
LSI	large-scale integrated circuit
MEI	maximally exposed individual
MOS	metal oxide semiconductor
NCRP	National Council on Radiation Protection and Measurements
NESHAPS	National Emission Standards for Hazardous Air Pollutants
NMOS	n-channel metal oxide semiconductor
NRC	U.S. Nuclear Regulatory Commission
PC	printed circuit
PNL	Pacific Northwest Laboratory
RAM	random-access memory
ROM	read-only memory
RCL	radiological control level
RCRA	Resource Conservation and Recovery Act
SOS	silicon on sapphire
TREES	transient radiation effects on electronic systems
VLSI	very large-scale integrated circuit

1.0 INTRODUCTION

Maintaining, refurbishing, and decommissioning of U.S. Department of Energy (DOE) defense and research facilities results in the generation of materials and equipment that are radioactively contaminated or activated to varying degrees. A portion of this material and equipment has a high content of radionuclides or hazardous waste and is most suitably treated as radioactive or mixed waste. However, another portion has a low content of radionuclides and no hazardous materials and could be treated in different ways. Materials that contain only trivial quantities of radionuclides could be released from further regulatory control; that is, they could be released for recycling or reuse.

The DOE is currently determining the feasibility of developing radiological control criteria for the recycling or reuse of selected metals or equipment containing residual radioactive contamination resulting from DOE operations. The main objective of this report is to develop and apply a methodology for establishing radiological control criteria for recycling or reuse. The control levels calculated in this report are presented to demonstrate the methodology and do not necessarily represent DOE's criteria for application to material recycle. The scenarios and information developed by the International Atomic Energy Agency (IAEA) in Safety Series No. 111-P-1.1 (1992), Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities, were used as a point of departure for this study. This effort is a subtask of the Environmental Protection Support and Assistance Project within DOE's Office of Environmental Guidance, Air, Water, and Radiation Division (DOE/EH-232).

To determine if radioactively contaminated (or activated) materials can be released from regulatory controls, it is necessary to resolve two issues: whether future use of the materials is likely to cause radiation doses to members of the public in excess of dose criteria specified by DOE, and whether increased levels of radiation will cause non-health-related impacts on electronics and film. The final radiological control levels will be based on risk analyses performed using generic radiation exposure scenarios and pathways. The human-health-related analyses include evaluation of relevant

radionuclides, and the potential mechanisms of exposure to selected worker and consumer populations, as well as to the public downwind of a smelter facility. The analyses also include consideration of non-health-related impacts to the electronics and film industries. The criteria define the control levels for residual radionuclides that can be present in DOE-generated metals and equipment released for recycling or reuse. The results and data generated in this report, based on a generic assessment, are expected to be useful to DOE as it further explores the feasibility of developing final radiological control levels.

1.1 RELATED REGULATORY LIMITS

The regulatory limits considered in the development of the radiological control levels for recycling and reuse are summarized in Table 1.1. The values given represent applicable limits as specified in DOE Order 5400.5 (DOE 1990), U.S. Nuclear Regulatory Commission (NRC) 10 CFR 20 (NRC 1991), or U.S. Environmental Protection Agency (EPA) regulations (EPA 1989). Also included are the U.S. Department of Transportation's (DOT) limit for radioactive material requiring labeling, and the NRC limit for uranium and thorium that is received, possessed, used, transferred, or delivered without being considered source material. Appendix A contains a detailed description of each cited regulatory limit.

To derive the radiological control criteria for residual radioactivity that can be present in DOE-generated materials considered for recycling or reuse, alternative dose limits are considered. For process workers or individuals using consumer products made from recycled materials, dose limits of 1.0 mrem/yr and 10 mrem/yr are used in the analyses. For the public downwind from a smelter or processing facility, the dose limits used are 0.1 mrem/yr and 1.0 mrem/yr. Additionally, the discussion of the results includes the consideration of the DOT limit for radioactive material requiring labeling (i.e., of 2 nCi/g or higher).

1.2 ARRANGEMENT OF THIS REPORT

Background information related to recycling and reuse of various materials is presented in Section 2.0. Section 3.0 contains a description of

TABLE 1.1. Summary of Current Regulatory Limits

Description	Applicable Population	Limit	Reference
Dose limit	Public	100 mrem/yr	DOE: 5400.5 (1990) NRC: 10 CFR 20.1301 (1991)
Dose limit	Public-air pathway	10 mrem/yr	DOE: 5400.5 (1990) EPA: 40 CFR 61.92 (Subpart H) (1989)
Lifetime cancer risk	Public	10^{-4} lifetime cancer risk	EPA: 40 CFR 61 (Subpart H) (1989)
Dose constraint	Public - all pathways	10 mrem/yr	DOE: 5400.5 (1990)
Dose constraint	Public - stack monitoring	0.1 mrem/yr	DOE: 5400.1 (1988) EPA: 40 CFR 61.93 (Subpart H) (1989)
Dose constraint	Public - reporting requirement	100 person-rem/yr	DOE: 5400.1 (1988)
Dose constraint	Worker - restriction for unmonitored workers	100 mrem/yr	DOE: 5480.11 (1989)
Dose constraint	Uranium toxicity limit	10 mg/wk	NRC: 10 CFR 20.1201 (1991)
Concentration limits	Uranium and thorium	Source Material 0.05% by wt.	NRC: 10 CFR 20.2 10 CFR 20.1003 (1991)
Concentration limits	Labeling requirement	2 nCi/g	DOT: 49 CFR 173.401 (1988)

the generic radiation exposure scenarios and pathways used in the analyses. Section 4.0 contains a description of the methodology for performing the public dose assessment. Discussions of the potential non-health-related effects of residual radioactivity on electronics and on film are given in Sections 5.0 and 6.0, respectively. The control criteria for workers, the public, electronics, and film are summarized and discussed in Section 7.0. The detailed results, which are quite lengthy because of the number of scenarios, radionuclides, and exposure pathways considered in the analysis,

and because dose estimates were produced both for individuals and for the total population potentially exposed, are presented in Appendices E, F, G, and H. Section 8.0 contains a discussion of the control criteria.

2.0 OVERVIEW OF THE IAEA EXEMPTION PRINCIPLES FOR RECYCLING AND REUSE

The bulk of the total radioactive waste materials potentially recovered during decommissioning for recycling and reuse will be materials from DOE defense reactors, uranium enrichment facilities, plutonium reprocessing facilities, plutonium fabrication facilities, or various DOE research facilities. Perhaps the most detailed discussion of recycling and reuse, including the development of release criteria, was conducted by the IAEA. To provide background information and a point of departure, the following paragraphs describe the essential features associated with the development of control criteria for recycling and reuse, as defined by the IAEA.

2.1 IAEA PRINCIPLES OF EXEMPTION FROM REGULATORY CONTROL

Since the mid-1980s, the IAEA has been exploring the application of exemption principles to the release of various radioactive materials. As the basis for their work, the IAEA identified two basic radiation protection criteria for determining when a source or practice may be a candidate for exemption from regulatory control (IAEA 1988b):

- individual risks (or individual doses and collective doses)[(a)] must be sufficiently low as not to warrant regulatory concern
- radiation protection must be optimized (i.e., as low as reasonably achievable [ALARA]).

For the first basic criterion, i.e., keeping individual risk below the level of regulatory concern, limitations on individual risk are based on the radiation doses experienced by that group of individuals likely to have received the highest doses. The IAEA defines "critical group" as a collection of people who are representative of individuals receiving the highest levels of dose; however, having opted for a small annual dose limit, the IAEA does not attempt to define the maximally exposed individual (MEI) by selecting worst-case parameters or conditions. The "critical group" is defined as reasonably homogeneous over the exposure conditions and pathways considered.

(a) In this preliminary report, risks are assumed to be proportional to doses (IAEA 1988b; ICRP 1991).

To satisfy its second basic criterion, the IAEA requires that, of all of the available alternatives, exemption be declared only when it is the option that most clearly optimize radiation protection. The IAEA assumes that a collective dose commitment of less than about 1 man-sievert (100 person-rem) per year would have a detriment minimal enough to permit exception without further detailed examination of other options. The IAEA also cautions, however, that the formulation of an exemption should not allow circumventing otherwise applicable controls, for example, by deliberately diluting material.

For the purposes of developing radiological control levels for recycling or reuse, the IAEA concluded that "an individual dose, regardless of its origin, is likely to be regarded as trivial if it is of the order of some tens of μSv (several mrem) per year" (IAEA 1988a). The IAEA further stated that, to account for the possibility of exposure from several sources from exempt practices, "it may be reasonable for national authorities to apportion a fraction of that upper dose to each practice." Thus, the IAEA recommended the adoption of an individual dose limit to the critically exposed population group on the order of 1 mrem (10 μSv) in a year from each exempt practice. This report uses individual dose limits of 1.0 mrem/yr and 10 mrem/yr for process workers and individuals who use consumer products made from recycled materials; for members of the public downwind from a smelter, the report uses alternate limits of 0.1 mrem/yr and 1.0 mrem/yr.

2.2 RECYCLING AND REUSE AS PRACTICES

The IAEA defines the practice of recycling and reuse as "the set of activities starting from the release of the material out of the boundary within which regulatory control applies (i.e., the boundary of a nuclear site) and including all the operations, manipulations, and uses which lead to exposure of a critical group(s)" (IAEA 1992). A source is defined as "the radioactive material(s) to be recycled or reused, or as the nuclear facility(ies) releasing the material for recycling or reuse" (IAEA 1992).

With these definitions, several interpretations of the practice of recycling and reuse are possible. One option might be considering the "practice" to be all recycling or reuse of exempted materials. In this case, the "source" would be the total amount of radioactive materials being recycled

or reused on an annual basis from all nuclear facilities being renovated or decommissioned. This option would satisfy many basic needs, but it may introduce difficulties and complications in site-specific applications.

As an alternative to the first option, the IAEA has stated that "the exact scope of the defined practice and the definition of a source may depend on the features of these activities and on the particular exemption policy preferred by the regulatory agency" (IAEA 1992). For this option, the practice could be defined as the material handled for each site that produces material for recycling or reuse (i.e., an exemption for each nuclear facility to be decommissioned). Thus, under this option, the "practice" would be defined to cover only the material released from a given site; however, it should be ensured that the critical group and collective doses relative to that practice not be significantly affected by the contribution of materials released from other nuclear facilities. As a final option, because different materials (e.g., steel, concrete, or aluminum) are likely to be used in different ways and lead to the definition of different critical groups or populations, the IAEA concluded that it may be sensible to define each material as a different "source." For this option, the recycling and reuse of each separate material could be defined as a separate practice because the recycling and reuse of any particular material is likely to involve specific exposure pathways and critical groups/populations different from those for any other particular recycled or reused material.

The IAEA further cautioned that the total annual quantity of materials being exempted from all the nuclear facilities concerned should be taken into account. The agency stated that "given that certain classes of recycled materials, such as metal scrap, are traded in the international markets, consideration should be given to the contribution to exposure of the critical groups and the population from exempted materials originating from one country or destined for another" (IAEA 1992).

2.3 DECISION PROCESS FOR EVALUATING THE POTENTIAL FOR RECYCLING OR REUSE

The recent increase in recycling and reuse of materials and equipment is probably spurred most by economic opportunities, but society's growing desire to conserve raw materials and natural resources adds extra impetus to the drive. Minimizing waste also plays an important role, particularly for DOE-generated materials.

Materials recovered during decommissioning are candidates for various possible options for disposal, recycling, or reuse. The data and models contained in this report may be used by DOE (in coordination with other agencies) as input to analyses to select radiological control criteria for recycling that maintain radiation ALARA, while supporting environmental regulations. Figure 2.1 shows a draft decision-process that could be followed when considering recycling or reuse. The figure is a conceptual diagram showing how radiological control levels, once selected, might be applied in practice. The inclusion of a DOE-Only Recycle option is included as a potential logical step. However, the scope of this document is limited to the calculation of radiological control levels applicable to unrestricted release.

To determine if recycling is the "preferred" action or approach for management of material, DOE has identified two criteria: the action must be 1) environmentally acceptable and cost-effective and/or 2) environmentally preferred (resulting in minimal impact on the environment). This approach, which is one of several being considered, recognizes that some situations exist under which the direct costs may be higher for recycling than for burial but that certain environmental costs can actually be avoided by recycling (e.g., the recycling option reduces environmental insults associated with certain secondary impacts). That is, the short-term costs associated with the recycling activity can be balanced by the long-term benefits.[a]

(a) April 2, 1993, letter from Andrew Wallo (Director of the Air, Water, and Radiation Division of the U.S. Department of Energy Office of Environmental Guidance) to W.E. Kennedy, Jr. (Pacific Northwest Laboratory).

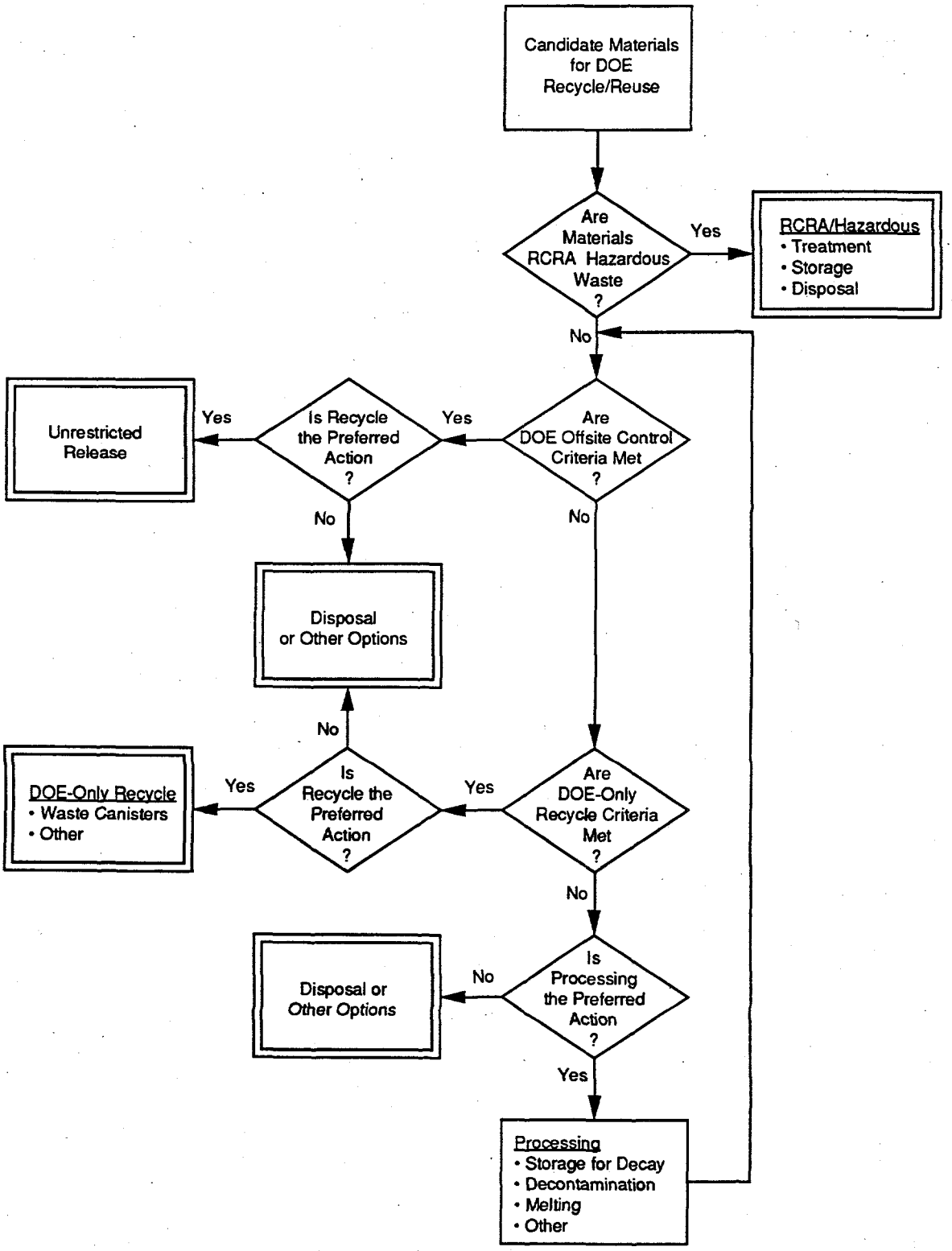


FIGURE 2.1. Decision Process for Evaluating the Potential for Recycling and Reuse

The processing steps in Figure 2.1 may include storage for decay, decontamination, size reduction, melting, or other operations designed to assist in reducing the concentration of radionuclides in or on the material; these steps may lead to a decision to recycle. However, processing should not include deliberate dilution for the sole purpose of meeting the control criteria. Technological improvements, including improvements in decontamination efficiency and material processing, could also have a future impact on decisions about whether to recycle and reuse.

3.0 DOSE ASSESSMENT METHODS

As discussed in Section 1.0, the main objective of this preliminary draft report is to develop and apply a methodology for establishing radiological control criteria for recycling or reuse of scrap materials generated by DOE operations. To determine if radioactively contaminated (or activated) materials can be released from regulatory controls, it is necessary to resolve the issue of whether future use of the materials is likely to cause radiation doses to members of the public in excess of dose criteria specified by the DOE. Generic radiation exposure scenarios may be used to model conceptually potential conditions, events, and processes that could result in radiation exposures to individuals or populations. These scenarios are a combination of radiation exposure pathways that contain specific conditions regarding concentrations of radionuclides in or on various media (such as surfaces, air, soil).

This section presents the methods used to calculate the radiological control levels based on individual doses following release of contaminated metals and concrete for recycling and of buildings, tools, and equipment for reuse. The discussion on calculational methods for public doses downwind of metal smelters, on the other hand, is given in Section 4.0. For more detailed discussions of the basic radiation exposure pathways and methods for calculating dose from inhalation, ingestion, and external exposures, see Appendices B, C, and D. The lengthy results of the dose calculations are presented in Appendices E, F, G, and H. The scenarios and models used to estimate radiation doses in this section are necessarily generic; however, based on a review of related literature, an attempt has been made to identify and evaluate the most important potential exposure pathways.

It should be noted that the generic methodology and scenarios described here (many originally presented in IAEA 1992) provide a basic framework for the numerical derivation of radiological control levels for recycling or reuse that will be adequately conservative for most situations. However, in determining control levels, or in performing an analytical assessment supporting the process of optimization, site-specific assessments may be required to best consider specific conditions beyond the generic conditions

included in this analysis. This may require the use of scenarios in addition to those discussed as part of this generic assessment or of alternative parameter values.

3.1 GENERAL ASSUMPTIONS

For the calculations that follow, it is assumed that 100 t of contaminated steel, aluminum, and concrete and 10 t of copper are recycled during a year. This assumption is intended to lead to the development of bulk contamination control levels. For the development of surface contamination control levels, individual tools or pieces of equipment for reuse are considered. The assumed concentration of each radionuclide considered in many cases in the initial calculations performed for this study is 1 μCi/g (3.7 x 10^4 Bq/g) for mass activity and 1 μCi/cm^2 (3.7 x 10^4 Bq/cm^2) for surface activity. For all dose assessments (individual, public, and film), the relative impact of each radionuclide was assessed separately.

3.2 REFERENCE RADIONUCLIDES

As listed in Table 3.1, 42 radionuclides were selected for consideration in the radiation exposure scenario/pathway analysis. These radionuclides were either reported to be "potentially present" in an analysis of Martin Marietta wastes during the development of Draft Release Limits for Contaminated Sludges and Soils by T. K. Chau et al. (1992), or they were known to be present in some DOE waste. Two radionuclides, ^{36}Cl and ^{41}Ca, are potentially present in activated concrete and are used only in scenarios dealing with concrete. The composite list was screened to eliminate the short half-lived radionuclides (<60 days) because these would likely decay in a waste stream before it arrives at an offsite treatment or disposal facility. A few long-lived isotopes were also excluded because of their anticipated low concentrations. Early daughter products in equilibrium with parent radionuclides have been assumed in all cases. For decay chains, the individual dose results have been reported for both the parent alone and for the parent-plus-daughter products in equilibrium with the parent.

For the calculations that follow, 40 reference radionuclides (plus two additional concrete activation products) have been selected. The

radionuclides considered and their physical half-lives are listed in Table 3.1. The 42 radionuclides could be grouped into categories according to their origins or physical characteristics:

- the origin of the radionuclides, whether natural uranium (^{238}U), uranium activation products (^{239}Pu, ^{241}Pu, and ^{241}Am), fission products (^{90}Sr, ^{99}Tc, and ^{137}Cs), or activation products (i.e., ^{36}Cl, ^{41}Ca, ^{54}Mn, ^{55}Fe, ^{60}Co, ^{63}Ni, ^{65}Zn, ^{94}Nb, ^{99}Tc, and ^{152}Eu)

- the half-life of the radionuclides, whether relatively short (^{65}Zn) or very long (i.e., ^{94}Nb or ^{239}Pu)

- the importance of the radionuclides in the context of bulk activation or surface contamination, that is, over the short term (^{55}Fe, ^{65}Zn, and ^{60}Co), long term (^{63}Ni, ^{137}Cs, and ^{152}Eu), or very long term (i.e., ^{238}U, ^{239}Pu, ^{94}Nb, and ^{99}Tc)

- mode of decay and internal dose conversion factors (DCF), whether alpha emitters with large DCF (^{228}Th, ^{229}Th, ^{230}Th, ^{232}Th, ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{238}U, ^{226}Ra, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am), beta/gamma emitters with large DCF (^{60}Co, ^{65}Zn, ^{94}Nb, ^{93}Zr, $^{110m}Ag/^{110}Ag$, ^{129}I, ^{134}Cs, $^{137}Cs/^{137m}Ba$, ^{144}Ce, ^{147}Pm, ^{151}Sm, ^{152}Eu, ^{154}Eu), non-photon emitters with moderate DCF (^{90}Sr, $^{106}Ru/^{106}Rh$, ^{241}Pu), beta/gamma emitters with low DCF (^{36}Cl, ^{54}Mn, ^{55}Fe, ^{57}Co, ^{99}Tc, ^{125}Sb), or non-photon emitters with moderate DCF (^{3}H, ^{14}C, ^{41}Ca, ^{63}Ni, ^{79}Se)

- the behavior of the radionuclides during recycling operations, that is, whether they are volatilized and escape or are concentrated or partitioned in slag or ingots (products).

For smelting of metals, it is probable that the majority of some radionuclides, such as ^{60}Co, remain in the ingot. However, a fraction of material will remain in the slag, and another portion will likely volatilize and be released with fumes and gases. The behavior of a specific radionuclide will depend on the chemistry of the radionuclide in question and the type of smelting process considered. Because the partitioning during smelting is not known for most radionuclides, the following dose calculations are based on the conservative assumption that, for each radionuclide, all of the activity is retained in each of the three phases of smelting: in the metal (steel, aluminum, or copper), in the slag, and released out of the stack. The slag is assumed to equal about 10% of initial mass of the steel, or about 10 t in this analysis. This triple accounting approach will overestimate the true doses; however, it will maximize the potential importance of the scenarios associated

TABLE 3.1. Radionuclides Considered in the Recycling and Reuse Analysis

Nuclide	Half-Life (yr)	Nuclide	Half-Life (yr)
^{3}H	12.3	^{144}Ce	0.78
^{14}C	5.7×10^{3}	^{147}Pm	2.62
^{36}Cl	3×10^{5}	^{151}Sm	87
^{41}Ca	1.3×10^{5}	^{152}Eu	13.6
^{54}Mn	0.86	^{154}Eu	8.8
^{55}Fe	2.7	^{226}Ra	1.6×10^{3}
^{57}Co	0.74	^{228}Th	1.91
^{60}Co	5.3	^{229}Th	7.34×10^{3}
^{63}Ni	99.9	^{230}Th	7.7×10^{4}
^{65}Zn	0.67	^{232}Th	1.4×10^{10}
^{79}Se	6.5×10^{4}	^{232}U	72
$^{90}Sr + Y$	28.5	^{233}U	1.5×10^{5}
^{93}Zr	1.5×10^{6}	^{234}U	2.47×10^{5}
^{94}Nb	2.0×10^{4}	^{235}U	7.1×10^{8}
^{99}Tc	2.13×10^{5}	^{238}U	4.51×10^{9}
^{106}Ru	1.01	^{237}Np	2.14×10^{6}
^{110m}Ag	0.68	^{238}Pu	87.6
^{125}Sb	2.8	^{239}Pu	2.4×10^{4}
^{129}I	1.6×10^{7}	^{240}Pu	6.57×10^{3}
^{134}Cs	2.06	^{241}Pu	14.4
^{137}Cs	30.1	^{241}Am	4.34×10^{2}

with each possibility and should serve as an adequate basis for developing radiological control levels. This methodology is consistent with that used in the IAEA document (1992).

3.3 RADIATION EXPOSURE PATHWAYS

The exposure of humans to radiation may occur in three main ways:

- as a result of exposure to external radiation from radioactive material
- from the inhalation of radioactive gases or of small particles of radioactive material
- from the ingestion of radioactive material.

The basic methods adopted in this report for assessing radiation doses via these exposure routes are described in Appendices B (external dose), C (internal dose), and D (dose calculation program), with special reference given to recycling and reuse. The following paragraphs describe some of the special features of the assessment methods used in this study.

3.3.1 External Radiation Exposure

A detailed description of the methods used for determining the external dose conversion factors is found in Appendix B. That appendix provides the details of the calculational procedures used for the various external exposure categories considered for the six recycling and reuse categories considered (Section 3.4). Appendix B also describes the source geometries, which usually consist of cylinder, half-cylinder, or disk source representations used to model either recycled materials or the use of consumer products or tools. Also included are the resulting dose conversion factors, in units of mrem/hr per μCi/g (Sv/hr per Bq/g) for bulk sources, and of mrem/hr per μCi/cm^2 (Sv/hr per Bq/cm^2) for surface sources. The information in Appendix B is largely consistent with the information provided by the IAEA in Safety Series 111-P-1.1 (1992).

3.3.2 Inhalation Exposure

Appendix C presents, in part, a discussion of the methods for estimating radiation doses from inhalation of airborne radioactive materials. That appendix presents the mathematical expression for estimating the committed dose by inhalation during the recycling exposure scenarios. For this calculation, the concentration of respirable airborne radioactive materials present at each work location is a key factor. The airborne particulate concentration depends on a variety of factors, including the physical condition of the material being handled, the quantity of the material present, and the building ventilation (IAEA 1992). Because it is difficult to predict generic particulate concentrations that may be present in air for specific recycling steps, certain "reference" air concentrations have been used. These air concentrations are listed in Appendix C and range from about 10^{-5} to 10^{-3} g/m^3. In determining these air concentrations, information from the literature and from previous studies was examined to consider the potential

impacts from recycling and reuse of radioactive metals (IAEA 1992; CEC 1988; NRC 1980; NRC 1984; O'Donnell et al. 1978). A listing of the dose conversion factors used in the assessment of dose via inhalation is also given in Appendix C, in units of rem/μCi inhaled, for the radionuclides of interest in this analysis. These factors were obtained from Internal Dose Conversion Factors for Calculation of Dose to the Public (DOE 1988b), based on Publications 30 and 48 of the International Commission on Radiological Protection (ICRP 1977-1982; ICRP 1986). All doses are estimated for adult reference man, as discussed in ICRP Publication 23 (ICRP 1975).

The air concentrations used for various work locations in this study are consistent with those found in IAEA (1992). These values were compared with those of other studies and with generic conditions in the metal smelting and fabrication industry. The values used were found to be conservative estimates but were chosen to represent the conditions that could be encountered. The air concentrations were also compared with those used to set control criteria for mixed waste incineration as part of a separate study for DOE (Aaberg et al. 1995), and they were found to be relatively consistent. Therefore, they have been adopted in this preliminary assessment without further modification.

3.3.3 Ingestion Exposure

Also presented in Appendix C is a discussion of the methods used to estimate doses from ingestion of radioactive materials. For this study, ingestion is assumed to occur by one of three separate routes:

- ingestion of removable radioactive materials
- ingestion of corrosion from using frying pans or water pipes
- ingestion of food products contaminated by airborne plumes arising from a smelter.

Ingestion of removable radioactive contamination found on recycled metals or reused equipment can occur when the contamination is transferred from a surface to hands, foodstuffs, cigarettes, or other items that enter the mouth. A review of previous work on estimating the radiation doses associated with this pathway determined that very few quantitative data for this pathway are available. Quantitative data for ingestion of lead by children, however,

were found, and these were used in determining an assumed quantity for ingestion of removable radioactive contamination by adults. Admittedly, this approach is based on a different situation and may be too conservative for adults. However, an assumed quantity of 10 mg of contamination per hour of direct contact exposure is assumed in the calculations for ingestion of selected adult workers at a smelter. No consumers or individuals who handle the final products from such a smelter are assumed to be exposed through ingestion because the potential for removable surface contamination is markedly reduced in the final products.

Ingestion of contaminated metal corroded from frying pans during cooking is considered a separate ingestion pathway. A previous study of the potential impact of recycling considered cast iron pans with an assumed worst-case corrosion rate of 0.127 cm/yr (O'Donnell et al. 1978). An O'Donnel et al. review of the literature for the corrosion rates of different steels revealed a more reasonable value of 0.13 mm/yr. Despite the fact that stainless steel, copper, and aluminum pans have different corrosion rates, a value of 0.13 mm/yr is conservatively used for this study for all types of pans.

A potential source of public exposure from metal recycling is from the materials that may volatilize and be released through the stack during smelting. The methods used to estimate doses to individuals downwind from an operating smelter are described in Section 4.0 and Appendix G.

3.4 SPECIFIC INDIVIDUAL DOSE SCENARIOS AND ASSUMPTIONS

For this analysis, six separate categories of contaminated (or activated) materials and future conditions are considered:

- recycling of steel
- recycling of aluminum
- recycling of copper
- recycling of concrete (as aggregate)
- reuse of a contaminated room within a facility
- reuse of tools or equipment (with surface contamination).

These materials were selected because they have been identified as major potential elements to be considered for recycling and reuse. The major constituents of nuclear facilities, both in volume and weight, are concrete, steel, and other high-value metals. Steel and other metals have a high scrap value. For nuclear reactors, the dominant material is steel; for enrichment plants, aluminum can be the main constituent of recyclable materials, and it has an even higher scrap value than steel; for linear accelerators, copper is a likely recyclable material that also has high scrap value.

The six categories considered in this assessment are further subdivided into various exposure scenarios, describing the activities of specific individuals or groups of individuals. The range of scenarios evaluated has been chosen based on previous dose estimates for recycling and reuse (IAEA 1992; CEC 1988; O'Donnell et al. 1978; NRC 1980) to adequately represent those scenarios likely to be of generic importance and relevance to all DOE nuclear facilities. It should be noted that this document addresses a subset of the scenarios considered in a document by the IAEA (1992); that is, the scenarios representing the highest potential doses for each category of recycled material and radionuclide grouping. Details of the scenarios considered, the relevant assumptions, and the parameter values are discussed in the following sections.

3.4.1 Scenarios for Recycled Steel

The general steps for recycling of steel were divided by the IAEA (1992) into the sets of representative exposure scenarios given in Table 3.2. These scenarios were developed by considering the types of scenarios identified in previous studies of the impacts of recycling contaminated steel (CEC 1988; O'Donnell et al. 1978; NRC 1980). Based on the IAEA (1992) study, Table 3.2 presents, for each steel recycling step, a list of representative radiation exposure scenarios and a brief description of the associated events. The scenarios were assigned numbers and external exposure categories in the IAEA study to help keep track of the exposure conditions (i.e., source geometry, source radius, source thickness, source density, and exposure distance).

TABLE 3.2. Data for Individual and Collective Dose Estimates for Metal-Recycling Steps and Scenarios (IAEA 1992)

Recycling Step	Scenario Considered	External Exposure Category[a]	Description of Events
1. Scrap delivery	1.1 Loader 1	I	Load/unload trucks
	1.2 Truck driver	II	Transport scrap to smelter
2. Scrap processing	2.1 Processor	III	Shred, cut, smash, chop, bail, and band
3. Smelting			
	3.1 Worker 1	IV	Yard workers at smelter
	3.2 Loader 1[b]	V	Crane operators at smelter
	3.2 Loader 2	Va	Crane operators at smelter
	3.3 Operator 1[c]	VI	Furnace operators at smelter
	3.3 Operator 2	VII	Furnace operators at smelter
4. Industrial product/ by-product	4.1 Caster 1[d]	VIII	Casting large ingots at smelter
	4.1 Caster 2	VIIIa	Casting large ingots at smelter
	4.2 Caster 3	IX	Casting small objects at smelter
	4.3 Slag worker	X	Slag collection at smelter
	4.4 Loader 2	XI	Load, unload trucks
	4.5 Truck Driver 2	XII	Transport ingots to fabrication plant
5. Initial fabrication			
	5.1 Worker 2	XIII	Yard workers at fabrication plant
	5.2 Sheet worker	XIV	Sheet workers at fabrication plant
	5.3 Coil worker	XV	Coil workers at fabrication plant
6. Final fabrication			
	6.1 Sheet worker	XIV	Sheet workers at fabrication plant
	6.2 Coil worker	XV	Product workers at fabrication plant
7. Distribution			
	7.1 Loader 2	XI	Load, unload trucks
	7.2 Truck driver 2	XII	Transport products to market
	7.3 Sheet worker	XIV	Building assembly
	7.4 Worker 3	XVI	Warehouse workers
8. Consumer use			
	8.1 Parking lot	XVII	Slag use in asphalt parking lot
	8.2 Room	XVIII	Building (room) use
	8.3 Appliance	XIX	Home appliance use
	8.4 Automobile	XX	Automobile use
	8.5 Frying pan	XXI	Frying pan use
	8.6 Large equipment	XXV	Large machine use
9. Stack emissions	9.1 Downwind individual	-	Residential/garden

(a) Each category defines a specific geometry, source radius, source thickness, and density used in estimating external effective dose equivalent factors as described in Appendix B of IAEA (1992).
(b) Two loaders are defined as: Loader 1 for a 100 t smelter and Loader 2 for a 10-t smelter.
(c) Two furnace operators are defined as: Operator 1 for a 100 t smelter and Operator 2 for a 10-t smelter.
(d) Three casters are defined as: Caster 1 for a 100 t smelter, Caster 2 for a 10-t smelter, both casting ingots, and Caster 3, casting small objects.

In the IAEA study (1992), the first seven recycling steps and associated scenarios shown in Table 3.2 were intended to model metal recycling, from delivery of the scrap metal to distribution of the manufactured consumer products. These steps include the smelting operation and initial and final fabrication steps. The IAEA considered conditions at both large and small smelters by assuming two different furnace sizes: a large 100 t/yr smelter

and a small 10 t/yr smelter. Different source geometries and work conditions were evaluated for the different-sized furnaces through scenarios involving specific workers (i.e., loaders, operators, and casters). The IAEA scenarios also included consideration of the potential doses to workers who handle the slag materials generated during smelting.

Additional details concerning the IAEA radiation exposure scenarios considered for steel recycling are given in Table 3.3. The information presented includes the internal exposure pathways considered, the duration of exposure for both individuals and population groups, and the number of individuals assumed to be exposed for each scenario.

The three scenarios that are highlighted in Table 3.3 (i.e., 4.3 Slag Worker, 8.4 Automobile, and 8.6 Large Equipment) represent those that were found to be the most limiting for recycling of metals (IAEA 1992). These three scenarios are selected for consideration in this study.

For each scenario, exposure durations and the total number of individuals exposed are estimated based on the analysis procedures of previous studies (IAEA 1992; CEC 1988; O'Donnell et al. 1978; NRC 1980). Assumptions for the three exposure scenarios for recycling of steel are as follows:

- Slag Worker. This scenario models exposure conditions associated with handling the 10 t of slag produced from the 100 t of recycled steel. Ten workers at the smelter are assumed to spend 25 hr handling the slag, as shown in Table 3.3. The radionuclide content in the slag is assumed to be 10 times more concentrated than the recycled metal, or 10 μCi/g.

- Automobile Use. A recycled steel is assumed to be used in the body of automobiles. The automobiles are modeled assuming three thin cylinder sources, as described in Appendix B. Each cylinder is assumed to have a radius of 1.5 m and a thickness of 0.1 cm. The total steel used per automobile is estimated to be about 167 kg. The recycled steel (100 t) could be used to construct as many as 600 automobiles. For the maximum individual dose calculations, an individual (perhaps a taxi driver) is assumed to spend 2000 hr/yr in the automobile (IAEA 1992; CEC 1988). For collective dose estimates, an average occupancy of two individuals per automobile is assumed. These individuals are assumed to travel about 15,000 km/yr at a speed of about 50 km/hr, for a total annual exposure of 300 hr (as shown in Table 3.3).

TABLE 3.3. Data for Annual Individual and Collective Dose Estimates for Steel-Recycling Scenario[a]

Scenario/External Exposure Category	Internal Pathways[b]	Individual Exposure Duration (hr)	Collective Exposure Duration (hr)	Number of Exposed Individuals	Air Concentration (g/m^3)
1.1 Loader 1	Inh.	4	4	2	0.0005
1.2 Truck driver	-[c]	4	4	5	-
2.1 Processor	Inh. & Ing.	12	12	3	0.0001
3.1 Worker 1	Inh.	80	80	10	0.0001
3.2 Loader 1	Inh. & Ing.	4	4	5	0.001
3.2 Loader 2	Inh. & Ing.	20	20	5	0.001
3.3 Operator 1	Inh. & Ing.	5	5	3	0.001
3.3 Operator 2	Inh. & Ing.	50	50	3	0.001
4.1 Caster 1	Inh. & Ing.	2.5	2.5	2	0.001
4.1 Caster 2	Inh. & Ing.	25	25	2	0.001
4.2 Caster 3	Inh. & Ing.	50	50	2	0.001
4.3 Slag worker	Inh. & Ing.	25	25	10	0.001
4.4 Loader 2	-	2	2	2	-
4.5 Truck driver 2	-	5	5	5	-
5.1 Worker 2	-	40	40	10	-
5.2 Sheet worker	Inh. & Ing.	1	1	15	0.0001
5.3 Coil worker	Inh. & Ing.	1	1	1	0.0001
6.1 Sheet worker	-	1	1	20	-
6.2 Coil worker	-	80	80	5	-
7.1 Loader 2	-	20	20	2	-
7.2 Truck driver 2	-	8	8	5	-
7.3 Sheet worker	-	20	20	20	-
7.4 Worker 3	-	2000	2000	5	-
8.1 Parking lot	-	2000	2000	1	-
8.2 Room	-	1500	500	440	-
8.3 Appliance	-	1000	500	4300	-
8.4 Automobile	-	2000	300	1200	-
8.5 Frying pan	Ing.	180	60	3300	-
8.6 Large equipment	2000	1000	200	-	-

(a) Highlighted scenarios are the most limiting scenarios for this category.
(b) *Inh.* represents the inhalation pathway and *Ing.* represents the ingestion pathway.
(c) A dash indicates that the pathway is not considered for the scenario shown.

- Large Equipment Use. For this scenario, recycled steel is assumed to be used in the manufacture of a large piece of equipment such as a metal lathe. This equipment is assumed to weigh 0.5 t, with a total production of 200 items (equalling 100 t of recycled steel). The exposure conditions are intended to model industrial workers who may work near or operate the equipment. For the analysis, an individual is assumed to work near the equipment for 2000 hr/yr. For collective dose estimates, 200 people are assumed to work near the equipment for an average of 1000 hr/yr, each.

The IAEA (1992) considered scenarios that account for the potential doses resulting from use of consumer products made from contaminated steel or slag. The consumer scenarios considered by the IAEA include slag use in

asphalt (assumed to be used in constructing a parking lot); sheet steel use in constructing buildings, appliances, and automobiles; and consumer use of frying pans and large items of equipment. The estimated radiation doses for each category of consumer product are based on the assumption that all of the recycled material (slag or steel) is used for only one type of product. That is, cases that include multiple uses of the steel (i.e., for buildings, appliances, and/or cars) were not considered by the IAEA. For this study, two generic exposure scenarios are considered for the use of consumer products made from contaminated recycled steel and slag.

3.4.2 Scenarios for Recycled Aluminum

In general, the long-lived activation of aluminum in reactor components is negligible. However, surfaces may be contaminated through contact with the primary coolant at a reactor or through contact with other sources at research facilities. Thus, the same 40 reference radionuclides considered for steel are also considered for the aluminum recycling scenarios. The surface contamination is assumed to be uniformly spread throughout the melted material, with a final concentration of 1 μCi/g, consistent with the analysis method used for steel.

The IAEA (1992) modified the general recycling steps shown in Table 3.3 by establishing sets of representative exposure scenarios for aluminum recycling, as shown in Table 3.4. The general steps associated with aluminum recycling are quite similar to those previously described for steel recycling. The IAEA scenarios include conditions that represent loaders, truck drivers, processors, general workers, furnace operators, casters, and sheet workers. Also listed in Table 3.4 are the generic data used by the IAEA in analyzing aluminum recycling for issues of individual and collective exposure durations, number of people exposed, and internal exposure pathways. For consumer products, the IAEA assumed that recycled aluminum was used in manufacturing building materials (home siding), constructing of automobile components, and making frying pans. The IAEA data for the analysis of the doses resulting from use of these consumer products are listed in Table 3.4. Because aluminum is less dense than steel, the IAEA assumed that almost three times as many aluminum as steel automobiles and frying pans are produced per unit mass. The IAEA scenarios consider the recycling of 100 t of aluminum used for building

TABLE 3.4. Data for Annual Individual and Collective Dose Estimates for Aluminum-Recycling Scenario[a]

Scenario/External Exposure Category	Internal Pathways[b]	Individual Exposure Duration (hr)	Collective Exposure Duration (hr)	Number of Exposed Individuals	Air Concentration (g/m^3)
1.1 Loader	Inh.	4	4	2	0.0005
1.2 Truck driver	-[c]	4	4	5	-
2.1 Processor	Inh. & Ing.	12	12	3	0.0001
3.1 Worker	Inh.	80	80	10	0.0001
3.2 Operator	Inh. & Ing.	50	50	3	0.001
4.1 Caster	Inh. & Ing.	25	25	2	0.001
4.2 Worker 2	Inh. & Ing.	48	48	5	0.001
5.1 Sheet worker	Inh. & Ing.	1	1	15	0.0001
5.2 Worker 3	-	80	80	5	-
6.1 Siding	-	1500	1500	500	-
6.2 Automobile	-	2000	300	3400	-
6.3 Frying pan	Ing.	180	60	10000	-

(a) Highlighted scenarios represent the most limiting scenarios for this category.
(b) *Inh.* represents the inhalation pathway, and *Ing.* represents the ingestion pathway.
(c) A dash indicates that the pathway is not considered for the scenario shown.

materials and automobiles, and the recycling of 10 t of aluminum for frying pan construction. The ingestion rate from corrosion of the frying pan is assumed to be 0.003 g/hr of use.

For the IAEA calculations, those scenarios found to be the most limiting (i.e., those that produced the largest individual doses) have been highlighted in Table 3.4. These scenarios are used as the basis of this study to establish control levels for the recycling and reuse of aluminum from DOE sites.

3.4.3 Scenarios for Recycled Copper

The IAEA (1992) study did not directly consider recycling of copper. However, three scenarios for copper recycling have been developed for this study by analogy to the scenarios for steel and aluminum. The three scenarios include an industrial-worker scenario and two consumer-products scenarios. These scenarios, namely smelter operator, consumer use of a copper pan, and household use of copper water pipes, are described below. Table 3.5 lists the generic data used in analyzing copper recycling in terms of the individual and

TABLE 3.5. Data for Individual and Collective Dose Estimates for Copper-Recycling Scenario

Scenario/External Exposure Category	Internal Pathways	Individual Exposure Duration (hr)	Collective Exposure Duration (hr)	Number of Exposed Individuals	Air Concentration (g/m^3)
Operator	Inh. & Ing.[a]	50	50	3	0.001
Water pipes	Ing.[b]	6000	2000	1000	---[c]
Frying pan	Ing.	180	60	8000	---

(a) *Inh.* represents the inhalation pathway and *Ing.* represents the ingestion pathway.
(b) Ingestion of copper from pipes = 0.27 g/yr, based on scenario from O'Donnell et al. 1978 (pg. 108).
(c) A dash indicates that the pathway is not considered for the scenario shown.

collective exposure durations, the number of people exposed, and the internal exposure pathways considered. The following paragraphs discuss the assumptions for each scenario:

- Operator. The furnace-operator scenario is defined to model work conditions at a small (10 t) furnace. Three furnace operators are assumed to work 50 hr. External exposure for the operator is modelled using a 10-t full-cylinder furnace charge, with a length of 117 cm and a radius of 59 cm. The distance from the operator to the source is assumed to be 3 m.

- Frying Pan Use. The recycled copper is assumed to be used to make frying pans that are distributed to the general population. Frying pans are represented by a cylinder source of 15-cm radius and 0.2-cm thickness. Each pan is estimated to contain 1.2 kg of copper. The individual is assumed to be 60 cm from the source. A total of 10 t of copper is assumed to be used in making the pans, providing a total production of 8000 frying pans. The exposure hours assumed for this study are 180 hr/yr for the maximum individual and 60 hr/yr for members of the population group. The ingestion rate from corrosion of the pan is assumed to be 0.01 g/hr of use.

- Water Pipes. A 10-t batch of recycled copper is assumed to be made into copper tubing used in the plumbing of 250 new residences. The maximally exposed individual is assumed to be located in a room containing a double set of copper pipes 2 m long, shielded with 0.5-inch-thick sheetrock at a distance of 1 m for 6000 hr/yr. An average individual is assumed to be in such a room for 2000 hr/yr. A total of 1000 individuals are assumed to be exposed, based on four residents per house. The ingestion of a small amount of contaminated copper (270 mg/yr) is based on a concentration of 1 mg/L in water and ingestion at home of half the daily water intake of 1.5 L (40 CFR 143).

3.4.4 Scenarios for Recycled Concrete

Large quantities of activated or contaminated concrete will be encountered during decommissioning of DOE defense and research facilities. Because there is an economic incentive to avoid the costs of transport and disposal of radioactive concrete as radioactive waste, recycling of concrete as feedstock for further concrete manufacture has been considered by many European countries (IAEA 1992). Before such reuse could be authorized, it is clear that any building would have to pass an extensive radiation survey to assure compliance with existing national regulations.

For the IAEA (1992) study, the recycled concrete was assumed to be used to build a new structure in which individuals live or work for 6000 hr/yr. The initial concrete is assumed to be contaminated to an average level of 1 μCi/g. Although a very large dilution could occur during the manufacture of new concrete structures, for the development of DOE initial control levels, a 1:10 dilution is assumed (i.e., a contamination level of 0.1 μCi/g was assumed for this scenario). Detailed data describing the scenarios considered by the IAEA for concrete recycle are given in Table 3.6, with the limiting scenarios used in this study highlighted.

Because many of the manufacturing and construction steps are similar to those previously described for metal recycling (i.e., transport workers, loaders, general workers, processors) their descriptions are not repeated here. Again, the concrete activation products, ^{36}Cl and ^{41}Ca, are included in the list of radionuclides to account for the potential activation of the concrete. Descriptions of the source geometries and assumptions used for external exposure are found in Appendix B and descriptions of the models and assumptions for inhalation and ingestion are found in Appendix C.

3.4.5 Scenarios for Reuse of Concrete Buildings

Situations of both surface contamination and volume contamination for concrete building materials must be addressed. Section 3.4.4 provides scenarios for recycling of concrete with volume contamination. This section presents an evaluation of doses received from surface activity through exposure to one of two types of buildings: a) existing buildings that have

TABLE 3.6. Data for Individual and Collective Dose Estimates for Concrete-Recycling Scenario[a]

Scenario/External Exposure Category	Internal Pathways[b]	Individual Exposure Duration (hr)	Collective Exposure Duration (hr)	Number of Exposed Individuals	Air Concentration (g/m^3)
1.1 Loader	Inh.	4	4	2	0.0005
1.2 Truck driver	-[c]	4	4	5	-
2.1 Processor	Inh. & Ing.	12	12	3	0.0001
3.1 Worker1	Inh.	80	80	10	0.001
3.2 Loader	Inh. & Ing.	20	20	5	$1x10^{-5}$
3.3 Truck driver	-	5	5	5	-
4.1 Const. worker	Inh.	80	80	20	$1x10^{-5}$
5.1 Parking lot	-	2000	2000	1	-
5.2 New room	-	6000	2000	10	-

(a) Highlighted scenarios represent the most limiting scenarios for this category.
(b) *Inh.* represents the inhalation pathway, and *Ing.* represents the ingestion pathway.
(c) A dash indicates that the pathway is not considered for the scenario shown.

been released for reuse after contamination or b) new buildings made of concrete constructed of rubble from old, previously contaminated buildings. The scenario considered is intended to account for normal occupancy, as described in an evaluation of residual radioactive contamination conducted for the NRC by Kennedy and Strenge (1992). Detailed data describing this scenario for surface contamination in buildings are found in Table 3.7.

For the building-occupancy scenario for reused concrete buildings, individuals are assumed to work in a building after unrestricted release. Although the residence time could vary, a normal work year of 2000 hr/yr is assumed. For populations, the occupancy period is assumed to be 500 hr/yr, consistent with the previous scenarios for collective exposure in buildings constructed from recycled materials. The primary exposure pathways are by external exposure to penetrating sources of radiation and by inhalation and ingestion of radioactive materials. Because decontamination efforts performed prior to release focused on the removal of surface sources, the removable level of surface contamination is assumed to be 10% of the levels assumed for other scenarios in this report. That is, for this scenario, the air concentration is assumed to be 10^{-5} g/m^3 and the ingestion rate is assumed to be 1.0 mg/hr of exposure. External dose conversion factors for surface sources are described in Appendix B.

TABLE 3.7. Data for Individual and Collective Dose Estimates for Building and Tool/Equipment Reuse Scenario

Scenario/External Exposure Category	Internal Pathways[a]	Individual Exposure Duration (hr)	Collective Exposure Duration (hr)	Number of Exposed Individuals
Building Reuse				
1. Building Occupancy	Inh. & Ing.	2000	500	10
2. Building renovation	Inh. & Ing.	200	40	10
Tool/Equipment Reuse				
1.1 Hand tools	Inh. & Ing.	600	100	100
1.2 Small motors	Inh. & Ing.	600	100	100
1.3 Large pump	Inh. & Ing.	200	80	40
1.4 Large equipment	Inh. & Ing.	200	80	40

(a) *Inh.* represents the inhalation pathway, and *Ing.* represents the ingestion pathway.

3.4.6 Scenarios for Reuse of Equipment or Tools

During decommissioning, discrete pieces of contaminated equipment (including hand tools, pumps, small motors, furniture, and storage tanks) may be salvaged and released for unrestricted use if they meet radiological control criteria. For the IAEA (1992) study, it was assumed that the fixed contamination present on the surfaces of the tools or equipment was 10 times higher than the removable fraction as measured by swabbing. The radiation exposure scenarios that may be most limiting involve a hobbyist's use of hand tools that incorporate a small motor (e.g., an electric hand drill or saw) because of the potential presence of difficult-to-monitor contamination on the inner surfaces of the motor. A high duration of exposure (600 hr/yr) is assumed because of the relative proximity of power tools to workers involved in construction activities. The exposure pathways considered by the IAEA include exposure to external radiation, ingestion of contamination transferred from the surfaces of the tool to the hands and then to the mouth, and inhalation of localized airborne material from the hand tool.

In addition to reuse of small items, the IAEA determined that larger items could also be candidates for reuse. These items are likely to contain surface contamination; thus, the same exposure considerations as for hand tools apply, with modifications accounting for the size of the item. A summary of the exposure categories and conditions assumed in the IAEA analysis

is given in Table 3.7. The limiting scenarios (i.e., those that produce the highest radiation doses), as highlighted in Table 3.7, are used to establish the radiological control levels for this study.

3.5 RADIOLOGICAL CONTROL LEVELS

The radionuclide-specific doses to individual members of selected worker and consumer populations are reported in terms of effective dose equivalent (mrem/yr) per μCi/g of scrap material. Using the following equation, the radiological control levels (RCL_i for radionuclide i) in bulk materials are then calculated for the most restrictive, or most limiting, conditions for each radionuclide under each recycling/reuse category:

$$RCL_i = \frac{L}{kD_i} \qquad (3.1)$$

where RCL_i = radiological control level for radionuclide, *i*, pCi/g

k = proportionality factor, 1×10^{-6} μCi/pCi

L = regulatory dose limit; either 1.0 mrem/yr or 10 mrem/yr

D_i = dose to individual from 1 μCi/g of radionuclide, *i*, in feedstock, mrem/yr per μCi/g.

Similar radiological control levels were calculated for the scenarios applying to surface contamination, where the resulting units are pCi/cm^2. Not surprisingly, in all cases, the most restrictive conditions were found where the feedstock material is assumed to be composed of 100% DOE scrap material (i.e., no dilution) with a dose limit of 1.0 mrem/yr to the individual.

Details of the resulting doses to individuals in these selected populations and with these radiological control levels are given in Appendix E. Table E.1 summarizes the individual-based radiological control levels for each of the six recycling/reuse categories. Tables E.2 through E.7 present the limiting conditions (i.e., with respect to scenario, radiation exposure pathway, and resulting effective dose equivalent) that were used to calculate the control levels for the recycling of steel, aluminum, copper, and concrete scrap, and for the reuse of buildings and tools/equipment.

4.0 PUBLIC DOSES FROM SMELTERS

A potential source of radiation exposure to members of the general public from metal recycling is from the materials that may volatilize and be released through stacks during smelting. This section describes the modeling approach used to establish the radiological control levels when the estimated radiation doses to the public downwind of a smelter are used as the basis for the calculations. For purposes of this report, dose to a representative MEI is estimated. Future efforts, such as performing an analytical assessment supporting the process of optimization (IAEA 1988), will expand the analysis to consider populations surrounding selected smelters for the calculation of collective dose estimates.

4.1 DESCRIPTION OF CURRENT CONDITIONS AT SMELTERS

The values of parameters used in the dose calculations are based as much as possible on realistic data drawn from the steel industry. Smelting is a high-temperature operation used to extract metal from its impurities. The basic smelting processes are similar for all metals of interest. The main differences are the types of furnaces used, the proportions of scrap and metal used as feedstock, and the casting methods (O'Donnell et al. 1978).

For smelting, the metal is put into a furnace and melted. The pure metal settles to the bottom of the smelter, and the impurities are drawn off in the slag that is formed during the smelting process. The primary products of smelting are the metal ingots and the slag. Slag is generated as a by-product of the smelting process. Dust and fumes released during the smelting process may also be discharged from the plant.

The small amount of information currently available on the apportioning of the contaminating radionuclides among the different smelting products precludes providing an adequate description of the airborne concentrations that would be expected to be released from a smelter. The types and dispersion of particles and fumes discharged from the facility will be dictated by the feed materials being processed, the furnace type, the emission control systems, and the meteorological conditions.

4.2 PUBLIC DOSE CALCULATIONS

In Section 3.0 and Appendices B and C, the methods are presented for estimating the dose to workers and consumers from radioactive materials in recycled scrap metals and concrete. This section presents the methods for estimating the radiological control levels for the 42 radionuclides of interest when the public dose, rather than worker or consumer dose, is used as the underlying basis. For this report, the public doses were estimated using the CAP88-PC (Clean Air Act Assessment Package - 1988 for the Personal Computer [PC] Environment) software package version 1.0 (Parks 1992), developed by the EPA for estimation of dose and risk from radionuclide emissions to air. The CAP88-PC code was not used in the 1992 IAEA Safety Series No. 111-P-1.1. Instead, generic atmospheric dispersion and ground deposition models were used to calculate radiation doses for individuals exposed by inhalation, direct external exposure, and ingestion of food products pathways.

4.2.1 Methodology

As discussed in Section 3.2, it was conservatively assumed that all of the activity in the smelter feedstock was released via the stack. The radioactive effluents of the stack releases were used to estimate radiation doses to the MEI using the CAP88-PC code (Parks 1992).

The CAP88-PC software was developed by the EPA to perform dose and risk assessments for the purpose of demonstrating compliance with 40 CFR 61.93(a) of the National Emission Standards for Hazardous Air Pollutants (NESHAPS) rules. The CAP88-PC software package is a collection of computer programs, databases, and associated utility programs for estimating dose and risk from radionuclide emissions to air. This package contains modified versions of the codes AIRDOSE-EPA and DARTAB, as well as a database of dose and risk factors generated by RADRISK. The CAP88-PC calculations account for dispersion in air, deposition on ground surfaces, resuspension as dust, and subsequent incorporation of radioactive materials in the food chain.

The dose to an individual that would result from a release to the atmosphere from smelting or melting is proportional to the concentration of radioactive material in the ground-level air at that individual's location.

For an elevated release via a stack, the maximum ground-level concentration will occur at some distance downwind. This downwind distance is a function of the stack height, the temperature of the released plume, and the local atmospheric dispersion conditions. Most of the common generic models for estimating ground-level concentrations are based on the Gaussian Plume model (IAEA 1980; IAEA 1982b). In the CAP88-PC package, a modification of the Gaussian plume equation is used to estimate the average dispersion of an airborne emission from an elevated stack. The radionuclide concentrations in air and the rates of deposition on ground surfaces are also calculated. The output data from the atmospheric transport models are coupled with the terrestrial food chain models of the NRC Regulatory Guide 1.109 (NRC 1977) to obtain estimates of the radionuclide concentration in food.

The radiation exposure pathways considered in the analysis include inhalation of airborne material, external exposure to penetrating radiation, and ingestion of contaminated foods. The effective dose equivalent is calculated using the ICRP 26 weighting factors (ICRP 1977). The dose for the MEI is determined by identifying the location or sector-segment in the radial assessment grid where the risk is the highest. The MEI is assumed to reside at this location. The MEI committed effective dose equivalent is reported as mrem/yr for a 50-yr exposure.

The committed effective dose equivalent is expressed in the output for both a "selected individual" (mrem/yr) and a "collective population" (person-rem/yr). The selected-individual values are reported as the dose to the MEI. The MEI results are used in the subsequent calculations to determine the control criteria for the release of materials to smelter feedstock.

The collective population dose is estimated by summing the intake and exposure rates for all sector-segments and multiplying by the appropriate dose conversion factors. The collective population dose is reported in units of person-rem/yr. Calculations to estimate the collective or population doses were not attempted in this report because of the number of additional assumptions that must be made concerning the size of the population and the population distribution in each downwind direction.

4.2.2 Modeling Assumptions

For the public dose CAP88-PC calculations, it is assumed that the air emission source from a generic smelter is a single elevated stack, 30 m high, 2 m in diameter, with a forced air stack emission (momentum type), and a plume stack gas exit velocity of 5.4 m/s. These assumptions and values are roughly similar to those used by O'Donnell et al. (1978) and are similar to those used for mixed waste incineration as part of a separate study for DOE (Aaberg et al. 1995). For this analysis, the conservative assumption was made that all of the activity (100%) associated with the recycled steel was released to the air from the smelter. The average concentration of radionuclides in the feedstock corresponding to stack releases of 1 Ci/y are 1×10^{-4} μCi/g for 100 t/yr of smelter throughput and 1×10^{-3} μCi/g for 10 t/yr of throughput. The dose to an individual residing near the facility was then assessed. The radiation exposure pathways that were considered include inhalation of material in the plume, external exposure from material deposited on the ground, and ingestion of food products contaminated by material deposited from the plume.

For this analysis, a representative meteorological data set and its associated population distribution were chosen to determine the location of a representative maximally-exposed member of the public. The Argonne National Laboratory facility population files of the Chicago area, which are readily available in CAP88-PC, were selected to represent a medium/high-density metropolitan population. The meteorological data files for Chicago (ORD0452.WND) were used with the Argonne population files. Default values resident in the CAP88-PC code were used for the precipitation, temperature, and height of atmospheric lid parameters. The agricultural data requires the user to choose "urban" for the food source parameter. The state of Illinois defaults were used for the rest of the agricultural data.

Doses were calculated for concentric circles around the generic smelter in 16 compass directions to a distance of 80,000 m (43.5 mi) at 13 distance fields (i.e., centered on 250; 750; 1,500; 2,500; 3,500; 4,500; 7,500; 15,000; 25,000; 35,000; 45,000; 55,000; and 70,000 m). The location of the MEI was calculated to be 750 m south-southwest of the hypothetical smelter.

4.3 CALCULATION OF RADIOLOGICAL CONTROL LEVELS

Using the above methods, the CAP88-PC code output is in terms of committed effective dose equivalent (mrem/yr) to the MEI per Ci/yr released from the smelter. The radiological control levels for bulk scrap as they apply to a member of the public surrounding a smelter are calculated as follows:

$$RCL_i = \frac{kL}{D_{MEI,i}\,W} \qquad (4.1)$$

where RCL_i = the radiological control level for radionuclide i, pCi/g

k = proportionality factor, 1×10^{12} pCi/Ci

L = assumed regulatory dose limit, either 0.1 mrem/yr or 1.0 mrem/yr

$D_{MEI,i}$ = CAP88-PC dose to the MEI for 1 Ci/yr stack releases for radionuclide i, mrem/yr per Ci/yr

W = weight of scrap steel recycled annually to smelter, either 1.016×10^8 g/yr or 1.016×10^7 g/yr.

For this analysis, two different feedstock weights and two different regulatory dose limits were considered. Table 4.1 summarizes the various reductions of the right-hand side of Equation 4.1, using all combinations of throughput and dose limits. In all cases, the most limiting conditions resulted when the material is composed of 100% DOE scrap material (i.e., no dilution), with a 100-t/yr facility throughput and a dose limit of 0.1 mrem/yr to the MEI.

Details of doses to the MEI in the downwind public are given in Appendix G. Tables G.1 and G.2 summarize the initial public-based radiological control levels for 100 t/yr and 10 t/yr, respectively, using the alternative 0.1 mrem/yr and 1.0 mrem/yr dose limits. Tables G.3 through G.6 list the initial control levels for various dilutions of DOE scrap material in the smelter feedstock (i.e., 1%, 5%, 10%, 50%, and 100% DOE scrap material):

- Table G.3 is for 100 t/yr at the 0.1-mrem/yr dose limit.
  Table G.4 is for 100 t/yr at the 1.0-mrem/yr dose limit.

- Table G.5 is for 10 t/yr at the 0.1-mrem/yr dose limit.
  Table G.6 is for 10 t/yr at the 1.0-mrem/yr dose limit.

TABLE 4.1. Equations Used to Estimate the Radiological Control Criteria for the Public Exposure Scenario

Annual Feedstock Volume	Dose Limit	
	0.1 mrem/yr	1.0 mrem/yr
10 t/yr	$\frac{9.48 \times 10^{3}}{D_{MEI,i}}$	$\frac{9.48 \times 10^{4}}{D_{MEI,i}}$
100 t/yr	$\frac{9.48 \times 10^{2}}{D_{MEI,i}}$	$\frac{9.48 \times 10^{3}}{D_{MEI,i}}$

5.0 POTENTIAL EFFECTS ON ELECTRONICS

A potential concern regarding the unrestricted use of recycled metals containing residual radiation contamination is that they may be used in the manufacture of electronic components. The following paragraphs discuss of the current knowledge regarding radiation effects on electronics and applicable radiological control levels related to recycled metals.

5.1 TYPES OF RADIATION DAMAGE

Most electronic systems are composed of semiconductor devices, which, in turn, are made primarily of silicon (Messenger and Ash 1986). The crystalline structure of the silicon makes it useful as a semiconductor. When atoms combine to form a crystalline lattice, quantization of the electron energies is produced, creating electron-energy-momentum bands and gaps. The metal oxide semiconductor (MOS) action is based on the interaction between the valence band and the conduction band (Figure 5.1). The lowermost energy band of electron state-of-energy is defined as the valence band and is almost completely filled. Above the valence band is a nearly empty band of energy states, called the conduction band. Between the two bands lies the energy gap of 1.11 eV, where energy states are forbidden. Upon excitation, an electron will jump from the valence band, across the forbidden gap to the conduction band. The vacancy left by the electron in the valence band is called a "hole," which is assumed to have properties similar to an electron's, except that it has a positive charge.

One significant effect of ionizing radiation on electronic components is the increase in conductivity of the material through the formation of hole-electron pairs. Experimental data suggest that it takes 3.6 eV, about three times the 1.1-eV energy of the forbidden gap, to create a hole-electron pair in silicon. When 1 rad of radiation is absorbed by silicon (Si), about 4.2×10^{13} hole-electron pairs are formed per cm^3. Here, the dose unit of rad (Si) is the quantity of ionizing radiation that creates 4.2×10^{13} hole-electron pairs per cm^3 in silicon (Si), where 1 rad is equal to 100 erg/g

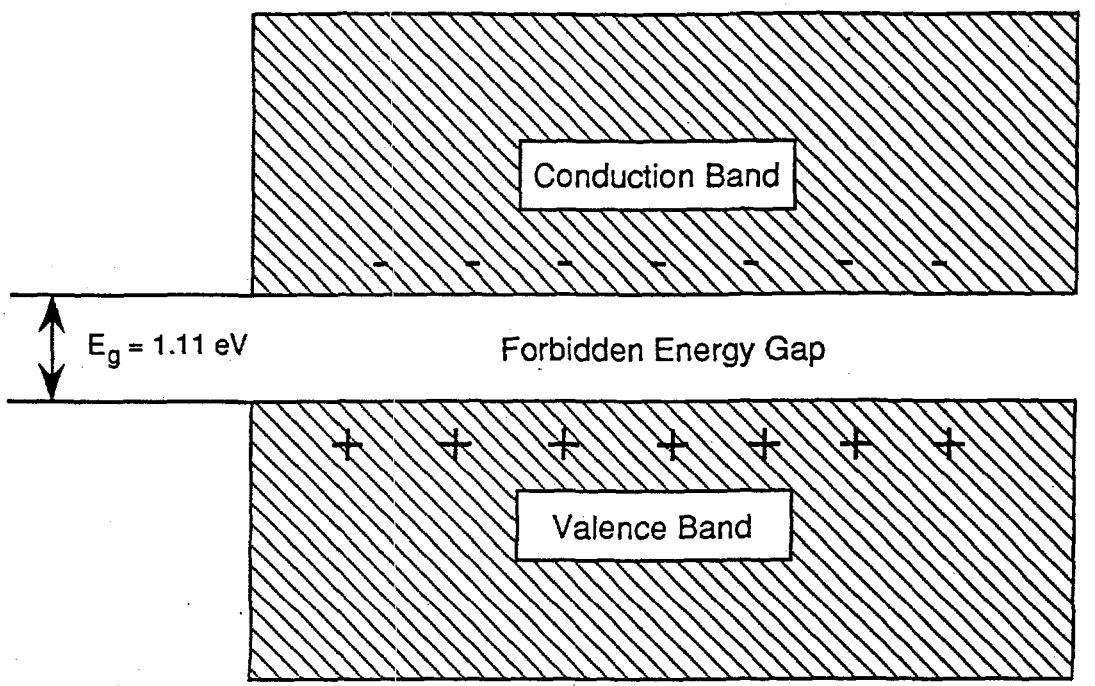


FIGURE 5.1. The Energy Band Structure of Silicon

(Messenger and Ash 1986). This number is calculated using the following equation, where h-e stands for hole-electron:

$$4.2 \times 10^{13} \text{ h-e pairs/cm}^3 = \frac{(100 \text{ ergs/g})\ (2.42 \text{ g/cm}^3)}{(1.6 \times 10^{-12} \text{ ergs/eV})\ (3.6 \text{ eV/h-e pair})} \qquad (5.1)$$

5.1.1 Effects from Various Types of Radiation

For transient radiation effects on electronic systems (TREES), the two main types of radiation considered are photons and neutrons (Messenger and Ash 1986). Incident gamma rays and x-rays cause radiation damage to semiconductor component devices through the effects of ionization. The principal ionization-induced changes in bulk material are conductivity increases through production of excess hole-electron pairs, trapped charges in insulators, production of electric and magnetic fields, and chemical effects. Electrons that cross the junction produce transient currents that will appear at the diode leads. Also, upon creation of the hole-electron pairs, any excess

energy is transferred to the lattice as thermal energy, which can cause further damage to the component device. Junction breakdown can be caused by migration of the holes from the collector to the emitter, thus causing a voltage drop. The level of damage is proportional to the amount of ionizing energy deposited therein; if severe enough, the damage may become permanent.

Neutrons, which are heavy uncharged particles, not only cause ionization of atoms within the semiconductor, but can cause displacement of atoms from their lattice sites within the crystal through direct collision with these atoms. The vacancies left by the displaced atoms are mobile (Passenheim 1988).

Protons may damage electronic components through ionization or displacement. The low-energy components (i.e., less than 1 MeV) of these charged particles have ranges sufficiently small that they can be stopped by a device's packaging material. The high-energy component has an interaction cross-section that decreases inversely with energy. The heavy mass of protons can result in neutron-like displacement damage.

Alpha particles can cause damage through ionization effects. Uranium and thorium, and their daughter nuclei, are part of the earth's crust and will be found in trace amounts in almost every material used to manufacture integrated circuit (IC) chips and their package. The ionization from alpha particles emanating from sources within a device package has been given as the principal causative factor for single-event upset errors in the IC chips. When a suitable protective layer is added around the chip, the alpha particles can be prevented from hitting the chip.

5.1.2 General Types of Damage to Electronic Components

Printed circuit (PC) boards are usually composed of a fiberglass board with electronic components connected together with thin metal strips called *lands*. Ionizing radiation will induce currents to flow on PC lands and be collected by every exposed surface (Passenheim 1988). These induced currents disrupt normal functioning of the devices.

Latchup is generally defined to have occurred when the semiconductor device is transformed to a state where it no longer responds to input signals. *Upset* is a general term used to indicate an action or state of exceeding a

tolerance level. *Data upset* is usually applied to shift registers and memories. For example, randon-access memory (RAM) stores information in flip-flop circuitry. A *single-event upset* can "burn in" the information into a given individual cell in the binary memory, partially transforming the device to read-only memory (ROM). Single-event upsets can be caused by very energetic heavy-ion components of cosmic rays or alpha particles.

In a high-density IC chip, single-event upsets are the main radiation-induced error that can occur. When capacitors are subjected to ionizing radiation, they experience a partial loss of charge. The TREES in resistors are small relative to those found with semiconductors and capacitors. The radiation-induced increase in conductivity acts to form internal leakage paths to shunt the resistor, thus lowering its resistance value.

With respect to MOSs, the n-channel MOSs (NMOSs) are the least radiation-resistant, with failures occurring in the 0.7 - 7.0 krad (Si) range for static and dynamic RAMs and microprocessors. (Here, a microprocessor is taken to include RAM and ROM memories, the central processing unit, and input/output [I/O] ports constructed on one large chip.) N-channel MOSs are used in most commercial hand-held calculators and minicomputers. They are also the mainstay of large-scale integrated circuits (LSI) and very large-scale (VLSI) integrated circuits in commercial microprocessors because they can be made for low-threshold voltage operation (i.e., can work off of a 5-volt supply), and they have greater electron mobility, making them faster. When exposed to ionizing radiation, the damage to the components causes shifts in voltage, changes in both channel resistance and hole-electron mobility, and increases in leakage currents. While the advantages of using NMOSs in microprocessors have led to large demands for these devices in the commercial market, little attention has been directed to the development of hardened (radiation-resistant) NMOS components for commercial or military use (Messenger and Ash 1986).

Methods commonly used to increase the hardness of electronic components include the following:

- making the gating material more radiation resistant
- using low-circuit impedances

- using clamping methods to prevent transient currents
- using circumvention methods to clamp memory devices or I/O devices
- making electronic systems redundant.

With the increasing complexity of electronic systems, their reliability is expected to decrease. One technique used to cope with this condition is to build in corresponding electronics in redundant configurations during manufacture. The redundant systems will aid in protecting against effects resulting from radiation damage.

5.2 TOTAL DOSE THRESHOLDS

As discussed above, electronic components can sustain damage from ionization effects caused by radiation. The total dose threshold and the dose rate range for various components are shown in Table 5.1. For comparison, natural background dose rates are about 0.100 rad/yr, or 1×10^{-5} rad/hr. All electronic components listed in Table 5.1 can withstand radiation doses higher than 25 mrem/yr.

5.3 CONTROL LIMITS FOR COMPONENTS IN ELECTRONICS

Most electronic components, even those that are not hardened to radiation effects, can withstand absorbed energy from radiation doses well above those that would be encountered by the radiological control limits that are proposed in this report. Therefore, the development of specialized radiological control levels for electronic components was determined to be unnecessary. The limiting radiological control levels discussed in Section 8.0, which are based on doses to the public and to film, should be well below the levels that would cause a significant level of unwanted TREES.

TABLE 5.1. Summary of Total Dose and Dose Rate Thresholds for Radiation Damage to Electronic Components[a]

Electronic Component Type	Dose Threshold Range (rad [Si])[b]	Dose Rate Range (rad/hr)
n-Channel MOS[c] (NMOS)	$5x10^2$ - $5x10^3$	$5x10^{-3}$ - $5x10^{-2}$
NMOS - hardened	$5x10^2$ - $5x10^3$	$5x10^1$ - $5x10^0$
Complementary MOS (CMOS)	$5x10^3$ - $5x10^4$	$5x10^2$ - $5x10^{-1}$
CMOS - hardened	$1x10^5$ - $1x10^6$	$1x10^0$ - $1x10^1$
CMOS on Sapphire (Al_2O_3) (CMOS/SOS)[d]	$5x10^3$ - $5x10^4$	$5x10^{-2}$ - $5x10^{-1}$
CMOS/SOS - hardened	$1x10^5$ - $1x10^6$	$1x10^0$ - $1x10^1$

(a) The values shown were calculated assuming a 10-year component lifetime with dose thresholds shown having a continuous 24-hr/d exposure. Information was summarized from Messenger and Ash (1986).
(b) The dose unit, rad (Si), is defined in this report as the deposition of 100 ergs of energy absorbed from ionizing radiation per gram of silicon (Si).
(c) MOS = metal oxide semiconductors.
(d) SOS = silicon on sapphire

6.0 POTENTIAL EFFECTS ON FILM

One potential concern related to the recycling of metals and concrete containing residual radioactive contamination is that these recycled materials may be used as material for making film-storage boxes. It is well known that film is sensitive to exposure to radiation. In fact, two major uses of film are in the fields of medical and industrial radiography. To attest to the sensitivity of film to radiation, the National Council on Radiation Protection and Measurements (NCRP) has commented that more shielding will be required to protect film stored near hospital x-ray rooms than that required to protect personnel (NCRP 1976).

The following paragraphs present potential radiation exposure scenarios for film after the release of contaminated metals or concrete for recycling. Also noted are factors to be considered when developing radiation control criteria for situations in which film will be exposed to such contaminated materials. The scenarios used are necessarily generic; however, an attempt has been made to identify and evaluate the most important potential exposure pathways.

6.1 TYPES OF RADIATION DAMAGE

Photographic emulsions are sensitive to many factors that can cause the reduction of silver halide, including heat, pressure, moisture, chemicals (particularly reducing agents), light, and radiation. Film is darkened when energy is given up to an orbital electron that combines with a silver ion to become metallic silver. For radiation, the film darkening is proportional to the amount of ionizing radiation absorbed by the film. The film darkening or optical density of the exposed film can be determined by photometric measurement and is defined by

$$S = \log \frac{I_0}{I_x} \qquad (6.1)$$

where S = photometric measurement

I_0 = light intensity transmitted through an unexposed film

I_x = light intensity transmitted through an exposed film.

The density of the film darkening is also dependent on the energy of the incident radiation for a given radiation dose. The absorption peak of most photographic emulsions is near 40 keV. The sensitivity of films at this energy is approximately 15 to 40 times greater than at 1 MeV, where the minimum sensitivity generally occurs (Kiefer and Maushart 1972). The higher the energy of the radiation, the lower the probability of an interaction with an orbital electron around an atom in the emulsion (Rogers 1981).

Different films are manufactured to have different sensitivities to radiation. This is accomplished primarily by varying the speed of the film: i.e., the faster the film, the greater the sensitivity it has to radiation exposure. As shown in Table 6.1, certain types of film may be sensitive to gamma exposure doses down in the 10^{-4} Roentgen (R) range, which equates approximately to 0.1 mrad if 1 R is assumed equal to roughly 1 rad.

Photographic emulsions have also been used in autoradiography to image alpha and beta particles emanating from a specimen that is in close contact with the emulsion. As early as 1896, Henri Becquerel found that uranium salts caused the blackening of emulsions of silver chloride and iodine. His work, together with the work of the Curies led to the recognition of radioactivity (Rogers 1981).

The films used in medical radiographic (x-ray) procedures may be sensitive to an exposure of less than 1 mR (NCRP 1976). If an undeveloped film was exposed to radiation before being used for patient studies, the decrease in film contrast may cause general film darkening or shadows, hampering the physician's ability to interpret the developed film. Thus, the diagnostic usefulness of the film would be decreased significantly.

TABLE 6.1. Ranges of Gamma Exposure Doses That Can Be Recorded by Various Types of Emulsions on Dosimeter Films[a]

Film Emulsion Type	Usable Range of Dose Measurement (R)
Printing emulsion	1×10^{2} to 1×10^{8}
Printing emulsion + fluorescence intensification	5×10^{1} to 5×10^{7}
Film	5×10^{-1} to 5×10^{4}
Film + fluorescence intensification	1×10^{-4} to 1×10^{3}

(a) Kiefer and Maushart 1972

6.2 TOTAL DOSE THRESHOLDS

To help prevent undesirable darkening or fogging of the films prior to use, the National Council on Radiation Protection and Measurements (NCRP) has recommended that radiographic film stored in darkrooms or film storage areas should not be exposed to more than 0.2 mrad (0.0002 cGy) of stray radiation prior to developing (NCRP 1976; 1989). For design specifications for film-storage areas, the NCRP recommends assuming a one-month storage time as an average if the exact time is unknown. Table 6 in Appendix C of NCRP Report No. 49 (NCRP 1976) shows lead or concrete shielding requirements that reduce the radiation from clinical x-ray units incident to the film to the 0.2 mrad limit. These data are provided for different storage periods and different distances from the radiation source.

6.3 CONTROL LEVELS BASED ON FILM

This section discusses the scenarios employed and how the results were used to develop radiological control levels for recycled materials.

6.3.1 Scenario Descriptions

There are a variety of exposure conditions and types of films that may be considered when assessing the potential exposure to film from recycled or reused materials. The following generic scenarios include the parameters that may control this type of exposure. The 1992 IAEA study did not include an analysis for the exposure of film.

The film storage scenario assumes that the film is stored in a rectangular container made from contaminated steel or along a wall made from contaminated concrete. A film storage vault similar to that used in medical x-ray departments was used as a guide for developing this scenario. To model the contaminated concrete, the film was assumed to be in a container with sides constructed of contaminated concrete.

The external dose factors were calculated using the GENII Software System (Napier et al. 1988) EXTDF module. The dimensions of the film storage box were taken to be 46 cm x 46 cm x 76 cm (approximately 18 in. x 18 in. x 30 in.), with the walls of the container modeled as rectangular slab shields. Both the steel and concrete containers (Figures 6.1a and 6.1b, respectively) were modeled with and without a 0.5-cm uncontaminated lead lining between the steel or concrete and the film. The recycled steel or concrete was conservatively assumed to represent 100% of the material used in constructing the film storage box. No correction was made for differences in film sensitivity for the different radiation energies of the radionuclides considered in this analysis.

For all scenarios, the film dose rate was estimated in the middle of the container, 23 cm (9 in.) from each side wall and 38 cm (15 in.) from the top and bottom of the container. This was done by assuming the film received four times the dose rate from any side, plus twice the dose rate received from the top. The thickness of the steel used for the container was taken to be 0.2 cm (0.08 in.). For the concrete container, only one side wall of the container was assumed to be recycled concrete. The dimensions of the container were similar to that for the steel container, except that the concrete wall was assumed to be 20 cm (about 8 in.) thick. The other walls were assumed to be made of unspecified, uncontaminated material.

6.3.2 Film-Based Radiological Control Levels

The dose rate factors, generated for four scenarios using the EXTDF program, are shown in Table 6.2. The scenarios included

- steel box (four sides, top, and bottom each at 0.2-cm thickness)
- same steel box with lead lining (0.5-cm thickness)

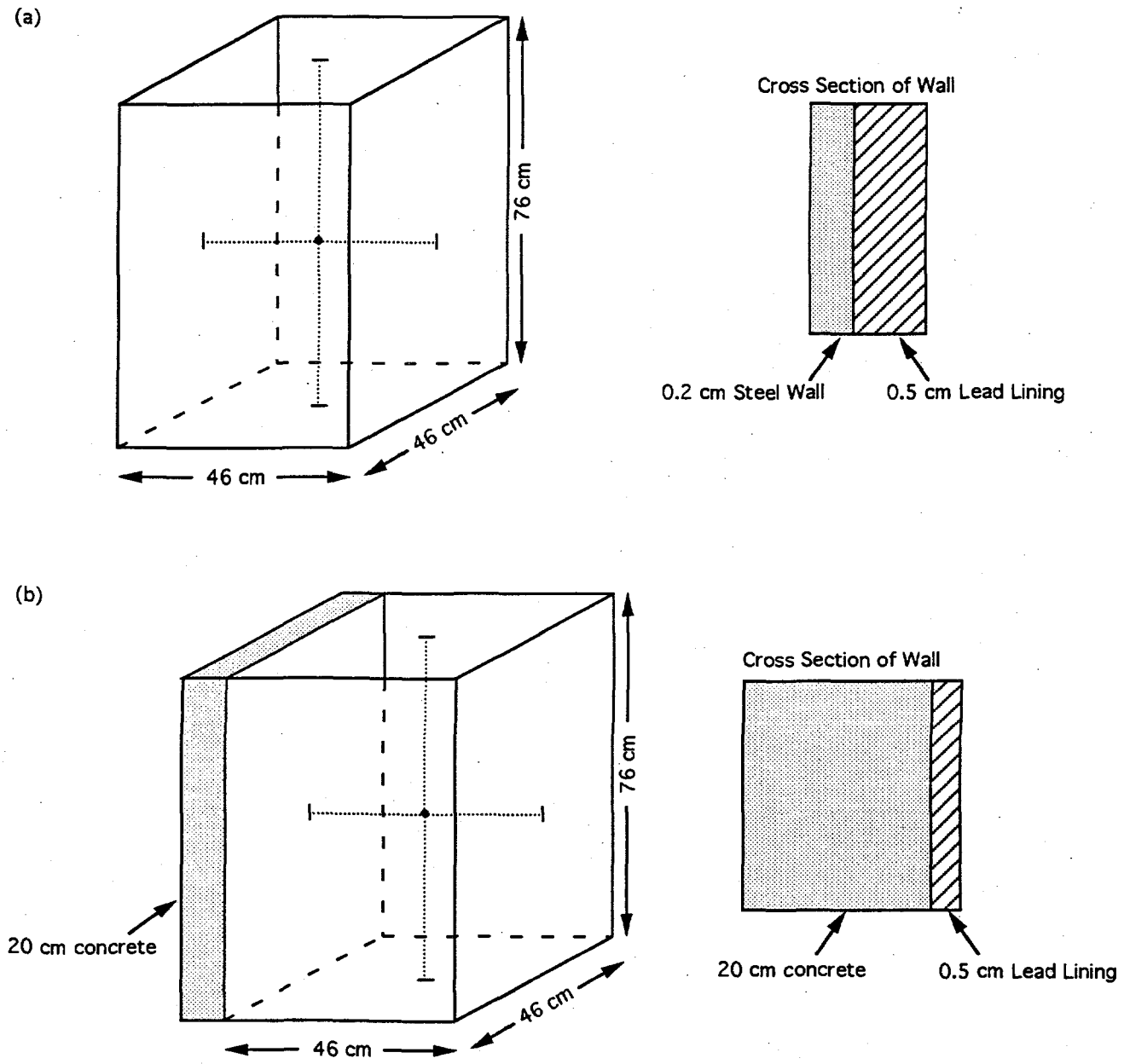


FIGURE 6.1. Diagrams of Steel and of Concrete Film Storage Boxes

- box with one side wall of concrete (one side at 20-cm thickness)
- same concrete box with a lead lining.

The radiological control levels for the steel and concrete used to construct the film storage box were calculated using the dose rate factors with an assumed one-month (730.5-hr) storage time and a dose limit of 0.2 mrad. The general equation used to generate these film-based radiological control levels is

$$RCL_i = \frac{k\ L}{DF_i\ T} \qquad (6.2)$$

where RCL_i = radiological control level for radionuclide i, pCi/g

L = assumed radiation limit, 0.2 mrad (= 0.2 mrem)

k = proportionality factor, 1×10^6 pCi/μCi

DF_i = calculated dose rate factor for radionuclide i, mrad/hr per μCi/g

T = time film is stored in box, 730.5 hr (= 1 month).

The resulting control levels for steel and concrete based on the film storage scenarios are given in Table 6.3. While these scenarios are not the most limiting that can be thought of for the use of recycled materials with film, they should provide approximations for a wide range of possible scenarios.

However, several arguments can be made that these scenarios are overly conservative. First, because is unlikely that the film-storage boxes would be made entirely of DOE recycled materials, assuming that the steel and concrete materials used were made of 100% DOE recycled materials would produce an overly conservative estimate. Second, no correction was made for differences in film sensitivity for the different radiation energies of the radionuclides considered in this analysis. These constraints should be kept in mind when the film-based control levels are compared to those based on individual doses or public doses.

TABLE 6.2. External Dose Rate Factors for the Film-Storage-Box Scenarios

Nuclide	External Dose Rate Factor (mrad/hr per μCi/g)			
	Steel	Lead-Lined Steel	Concrete	Lead-Lined Concrete
^{3}H	7.8E-08	-[a]	1.6E-07	-
^{14}C	1.3E-03	1.2E-10	2.3E-03	3.4E-11
^{36}Cl	NA[b]	NA	5.0E-02	6.0E-04
^{41}Ca	NA	NA	2.2E-01	-
^{54}Mn	7.7E+01	4.0E+01	1.0E+02	5.3E+01
^{55}Fe	7.9E-02	-	5.1E-01	-
^{57}Co	1.0E+01	4.3E-02	9.5E+00	5.1E-02
^{60}Co	2.2E+02	1.4E+02	3.3E+02	2.1E+02
^{63}Ni	9.6E-05	2.5E-19	2.2E-04	8.8E-20
^{65}Zn	5.8E+01	3.6E+01	8.7E+01	5.5E+01
^{79}Se	1.0E-03	6.4E-11	1.8E-03	1.8E-11
^{90}Sr	3.6E-02	1.8E-04	2.8E-02	8.6E-05
^{93}Zr	7.8E-05	9.7E-21	1.8E-04	3.4E-21
^{94}Nb	1.4E+02	6.4E+01	1.7E+02	8.2E+01
^{99}Tc	6.8E-03	4.1E-07	8.2E-03	1.6E-07
^{106}Ru	2.2E+01	7.7E+00	2.5E+01	9.5E+00
^{110m}Ag	2.6E+02	1.3E+02	3.4E+02	1.8E+02
^{125}Sb	4.1E+01	1.3E+01	4.8E+01	1.6E+01
^{129}I	3.2E-01	7.0E-11	1.9E+00	1.9E-11
^{134}Cs	1.5E+02	7.1E+01	1.9E+02	9.2E+01
^{137}Cs	5.2E+01	2.0E+01	6.2E+01	2.4E+01
^{144}Ce	1.3E+00	1.2E-04	1.2E+00	8.5E-05
^{147}Pm	3.3E-03	5.0E-08	4.7E-03	2.8E-08
^{151}Sm	5.5E-04	5.7E-16	2.9E-03	1.7E-16
^{152}Eu	1.1E+02	5.5E+01	1.5E+02	8.3E+01
^{154}Eu	1.1E+02	5.9E+01	1.5E+02	8.6E+01
^{226}Ra	3.7E-01	2.6E-04	3.2E-01	2.8E-04
^{228}Th	1.5E-01	9.8E-04	2.8E-01	9.2E-04
^{229}Th	5.6E+00	1.4E-02	6.3E+00	1.3E-02
^{230}Th	3.9E-02	7.3E-07	1.7E-01	5.4E-07
^{232}Th	3.1E-02	4.5E-07	1.6E-01	3.3E-07
^{232}U	4.5E-02	7.7E-07	2.3E-01	5.7E-07
^{233}U	3.0E-02	1.9E-06	8.6E-02	1.4E-06
^{234}U	3.6E-02	4.2E-07	2.0E-01	3.2E-07
^{235}U	9.5E+00	2.5E-02	8.4E+00	2.3E-02
^{238}U	2.7E-02	7.1E-12	1.6E-01	6.2E-12
^{237}Np	1.4E+00	6.9E-04	2.4E+00	6.4E-04
^{238}Pu	3.3E-02	2.3E-16	2.1E-01	2.3E-16

TABLE 6.2. (Cont'd)

Nuclide	External Dose Rate Factor (mrad/hr per μCi/g) Steel	Lead-Lined Steel	Concrete	Lead-Lined Concrete
^{239}Pu	1.7E-02	5.1E-07	8.3E-02	3.7E-07
^{240}Pu	3.2E-02	2.5E-16	2.0E-01	2.6E-16
^{241}Pu	2.6E-07	-	5.4E-07	-
^{241}Am	6.8E-01	1.3E-11	1.7E+00	1.2E-11

(a) For radionuclides with no gamma emissions, the lead lining reduced the dose to zero, resulting in initial control levels that approach infinity (shown as - in table).

(b) "NA" indicates a concrete activation product that is considered in the concrete recycling scenario and is Not Applicable to the steel recycling scenario.

TABLE 6.3. Radiological Control Levels Based on 0.2-mrad Exposure of Film Stored for One Month[a]

Radionuclide	Bulk Contamination (pCi/g)			
	Steel	Lead-Lined Steel	Concrete	Lead-Lined Concrete
^{3}H	3.5E+09	---[b]	1.7E+09	---
^{14}C	2.1E+05	2.3E+12	1.2E+05	8.2E+12
^{36}Cl	NA[c]	NA	5.5E+03	4.6E+05
^{41}Ca	NA	NA	1.2E+03	---
^{54}Mn	3.6E+00	6.8E+00	2.7E+00	5.2E+00
^{55}Fe	3.5E+03	---	5.39E+02	---
^{57}Co	2.6E+01	6.5E+03	2.9E+01	5.4E+03
^{60}Co	1.2E+00	2.0E+00	8.3E-01	1.3E+00
^{63}Ni	2.8E+06	5.7E+13[d]	1.3E+06	5.7E+13[d]
^{65}Zn	4.8E+00	7.6E+00	3.2E+00	5.0E+00
^{79}Se	2.8E+05	6.9E+10[d]	1.6E+05	6.9E+10[d]
^{90}Sr	7.6E+03	1.5E+06	9.7E+03	3.2E+06
^{93}Zr	3.5E+06	2.5E+09[d]	1.5E+06	2.5E+09[d]
^{94}Nb	2.0E+00	4.3E+00	1.6E+00	3.4E+00
^{99}Tc	4.1E+04	6.7E+08	3.4E+04	1.8E+09
^{106}Ru	1.3E+01	3.6E+01	1.1E+01	2.9E+01
^{110m}Ag	1.1E+00	2.1E+00	8.0E-01	1.5E+00
^{125}Sb	6.7E+00	2.1E+01	5.7E+00	1.7E+01
^{129}I	8.6E+02	1.8E+08[d]	1.5E+02	1.8E+08[d]
^{134}Cs	1.8E+00	3.8E+00	1.4E+00	3.0E+00
^{137}Cs	5.3E+00	1.4E+01	4.5E+00	1.1E+01
^{144}Ce	2.2E+02	2.4E+06	2.2E+02	3.2E+06
^{147}Pm	8.2E+04	5.5E+09	5.9E+04	9.9E+09
^{151}Sm	5.0E+05	2.6E+13[d]	9.5E+04	2.6E+13[d]
^{152}Eu	2.6E+00	5.0E+00	1.8E+00	3.3E+00
^{154}Eu	2.5E+00	4.7E+00	1.8E+00	3.2E+00
^{226}Ra	7.4E+02	1.0E+06	8.6E+02	9.7E+05
^{228}Th	1.8E+03	2.8E+05	9.6E+02	3.0E+05
^{229}Th	4.9E+01	1.9E+04	4.4E+01	2.1E+04
^{230}Th	7.0E+03	4.8E+08	1.6E+03	5.0E+08
^{232}Th	8.9E+03	1.1E+05[d]	1.7E+03	1.1E+05[d]
^{232}U	6.1E+03	3.6E+08	1.2E+03	4.8E+08
^{233}U	9.1E+03	1.4E+08	3.2E+03	1.9E+08
^{234}U	7.7E+03	6.5E+08	1.4E+03	8.7E+08
^{235}U	2.9E+01	1.1E+04	3.3E+01	1.2E+04
^{238}U	1.0E+04	3.4E+05[d]	1.7E+03	3.4E+05[d]

TABLE 6.3. (Cont'd)

Radionuclide	Bulk Contamination (pCi/g) Steel	Lead-Lined Steel	Concrete	Lead-Lined Concrete
^{237}Np	2.0E+02	4.0E+05	1.1E+02	4.3E+05
^{238}Pu	8.2E+03	1.7E+13[d]	1.3E+03	1.7E+13[d]
^{239}Pu	1.6E+04	5.4E+08	3.3E+03	7.3E+08
^{240}Pu	8.6E+03	2.3E+11[d]	1.4E+03	2.3E+11[d]
^{241}Pu	1.1E+09	---	5.0E+08	---
^{241}Am	4.0E+02	3.4E+12[d]	1.6E+02	3.4E+12[d]

(a) Calculations were made assuming that the film was stored for one month in a rectangular container made from either steel or concrete, with or without lead shielding lining (0.5-cm thickness) the box. The radiological control levels were determined based on the 0.2-mrad (0.002-cGy) limit recommended by the NCRP (1989) for diagnostic x-ray film, and assumed 1 mrad = 1 mrem.

(b) For radionuclides having no gamma emissions, the lead lining reduced the dose to zero resulting in initial control levels that approached infinity. This is represented by a long dash (---) in the table.

(c) "NA" indicates that this concrete activation product was Not Applicable to this scenario and was considered only for concrete recycling scenarios.

(d) The specific activity of the radionuclide is provided. The calculated control level exceeds the specific activity of the radionuclide.

7.0 RESULTS

Radiological control levels have been evaluated using a variety of exposure scenarios, parameters, and radionuclides. These radiological control levels are presented to demonstrate the methodology and do not necessarily represent the criteria the U.S. Department of Energy will apply to unrestricted release of the materials evaluated. The derived radiological control levels are represented by the concentrations of the various radionuclides in recycled and reused materials that would result in an individual dose rate of 1 mrem/yr or a public dose rate of 0.1 mrem/yr. Because the detailed results are quite lengthy, only the summary information will be presented in this section. The detailed results are presented in Appendices E and F for individual dose and Appendices G and H for public dose. The control levels associated with individual dose, public dose, and film exposure are summarized in Tables 7.1, 7.2, and 7.3, respectively. The radiological control levels summarized in Table 7.1 indicate the concrete bulk contamination levels. The surface contamination control levels are available in Appendix E, Table E.8.

The surface contamination radiological control levels of Table 7.1 are noted to be above the surface contamination limits of the DOE *Radiological Control Manual* (DOE 1994). The values listed in Table 7.1 are risk-based using 1 mrem/yr and the scenarios reported in Chapter 3. The DOE (1994) values are roughly related to instrumentation responses and an assumed dose rate of 10 mrem/yr for selected categories of radionuclides.

TABLE 7.1. Radiological Control Levels Based on an Individual Dose of 1 mrem/yr for Recycling and Reuse of DOE Metals or Equipment Containing Residual Radioactive Contamination

Radionuclide	Bulk Contamination (pCi/g)				Surface Contamination (pCi/cm²)
	Steel	Aluminum	Copper	Concrete	Tools & Equip.[b]
^{3}H	5.7E+06	2.9E+07	8.8E+06	1.7E+08	2.6E+06
^{14}C	1.9E+05	8.3E+05	2.6E+05	5.6E+05	7.9E+04
^{36}Cl	NA[a]	NA	NA	1.7E+04	5.1E+04
^{41}Ca	NA	NA	NA	8.5E+03	1.2E+05
^{54}Mn	1.3E+01	3.6E+01	1.9E+02	5.9E+00	3.3E+03
^{55}Fe	1.5E+04	2.5E+04	2.1E+05	3.7E+03	1.5E+05
^{57}Co	9.0E+01	2.4E+02	1.4E+03	7.9E+01	1.8E+04
^{60}Co	4.4E+00	1.3E+01	6.7E+01	1.7E+00	1.1E+03
^{63}Ni	5.2E+05	2.6E+06	1.0E+06	5.5E+06	3.0E+05
^{65}Zn	1.7E+01	4.9E+01	2.6E+02	6.7E+00	3.3E+03
^{79}Se	4.3E+04	2.2E+05	6.7E+04	7.2E+05	2.0E+04
^{90}Sr	2.5E+03	1.3E+04	4.3E+03	3.3E+04	1.3E+03
^{93}Zr	3.8E+04	1.9E+05	1.9E+05	1.4E+05	6.7E+04
^{94}Nb	7.1E+00	2.0E+01	1.1E+02	3.5E+00	1.7E+03
^{99}Tc	1.3E+05	3.0E+05	4.3E+05	1.4E+05	1.2E+05
^{106}Ru	4.4E+01	1.3E+02	6.7E+02	2.4E+01	4.2E+03
^{110m}Ag	3.8E+00	1.1E+01	5.8E+01	1.7E+00	9.8E+02
^{125}Sb	2.4E+01	6.5E+01	3.6E+02	1.3E+01	5.6E+03
^{129}I	1.3E+03	1.2E+03	2.0E+03	5.3E+02	5.9E+02
^{134}Cs	6.3E+00	1.8E+01	9.6E+01	3.1E+00	9.7E+02
^{137}Cs	1.9E+01	5.3E+01	2.8E+02	9.7E+00	2.0E+03
^{144}Ce	7.2E+02	1.6E+03	1.2E+04	6.0E+02	6.6E+03
^{147}Pm	7.9E+04	4.0E+05	4.0E+05	2.5E+05	1.2E+05
^{151}Sm	1.0E+05	5.2E+05	5.2E+05	3.6E+05	2.4E+05
^{152}Eu	9.3E+00	2.6E+01	1.4E+02	3.9E+00	2.2E+03
^{154}Eu	8.8E+00	2.5E+01	1.3E+02	3.8E+00	2.0E+03
^{226}Ra	2.0E+02	1.0E+03	5.1E+02	1.3E+03	1.4E+02
^{228}Th	1.1E+01	5.3E+01	5.3E+01	3.4E+01	4.1E+01
^{229}Th	1.9E+00	9.6E+00	9.6E+00	6.1E+00	7.0E+00
^{230}Th	1.3E+01	6.3E+01	6.3E+01	4.0E+01	4.6E+01
^{232}Th	3.0E+00	1.5E+01	1.5E+01	9.5E+00	1.0E+01
^{232}U	5.0E+00	2.5E+01	2.5E+01	1.6E+01	2.1E+01
^{233}U	2.6E+01	1.3E+02	1.3E+02	8.0E+01	1.1E+02
^{234}U	2.6E+01	1.3E+02	1.3E+02	8.0E+01	1.1E+02
^{235}U	2.8E+01	1.4E+02	1.4E+02	8.6E+01	1.1E+02
^{238}U	2.8E+01	1.4E+02	1.4E+02	8.7E+01	1.1E+02
^{237}Np	6.4E+00	3.2E+01	3.2E+01	2.1E+01	1.7E+01
^{238}Pu	1.1E+01	5.5E+01	5.5E+01	3.5E+01	4.6E+01
^{239}Pu	1.0E+01	5.0E+01	5.0E+01	3.2E+01	4.1E+01
^{240}Pu	1.0E+01	5.0E+01	5.0E+01	3.2E+01	4.1E+01
^{241}Pu	5.8E+02	2.9E+03	2.9E+03	1.8E+03	2.4E+03
^{241}Am	6.0E+00	3.0E+01	3.0E+01	2.0E+01	1.6E+01

(a) "NA" indicates a concrete activation product that was considered in the concrete recycling scenarios only and is Not Applicable to the other recycling scenarios.

(b) The values presented in this table are risk-based using 1 mrem/yr and assuming certain exposure conditions. The values for most radionuclides are greater than the surface contamination values that appear in the DOE *Radiological Control Manual* (DOE 1994). The DOE (1994) values were based on the values appearing in the U.S. NRC Regulatory Guide 1.86 (NRC 1974). These values are roughly related to instrumentation responses and an assumed dose rate of 10 mrem/yr. for selected categories of radionuclides.

TABLE 7.2. Public Dose-Based Radiological Control Levels for Metal Feedstock Volumes of 10 t/yr and 100 t/yr and Dose Limits of 0.1 mrem/y and 1.0 mrem/y

	10-t Feedstock		100-t Feedstock	
Nuclide	pCi/g at 0.1 mrem/yr	pCi/g at 1.0 mrem/yr	pCi/g at 0.1 mrem/yr	pCi/g at 1.0 mrem/yr
^{3}H	5.2E+08	5.2E+09	5.2E+07	5.2E+08
^{14}C	1.1E+08	1.1E+09	1.1E+07	1.1E+08
^{54}Mn	1.3E+05	1.3E+06	1.3E+04	1.3E+05
^{55}Fe	3.7E+07	3.7E+08	3.7E+06	3.7E+07
^{57}Co	8.2E+05	8.2E+06	8.2E+04	8.2E+05
^{60}Co	8.7E+03	8.7E+04	8.7E+02	8.7E+03
^{63}Ni	2.9E+07	2.9E+08	2.9E+06	2.9E+07
^{65}Zn	2.3E+05	2.3E+06	2.3E+04	2.3E+05
^{79}Se[(b)]	---	---	---	---
^{90}Sr	2.2E+05	2.2E+06	2.2E+04	2.2E+05
^{93}Zr	1.6E+06	1.6E+07	1.6E+05	1.6E+06
^{94}Nb	1.9E+03	1.9E+04	1.9E+02	1.9E+03
^{99}Tc	3.4E+06	3.4E+07	3.4E+05	3.4E+06
^{106}Ru	1.6E+05	1.6E+06	1.6E+04	1.6E+05
^{110m}Ag	4.9E+04	4.9E+05	4.9E+03	4.9E+04
^{125}Sb	8.1E+04	8.1E+05	8.1E+03	8.1E+04
^{129}I	9.6E+03	9.6E+04	9.6E+02	9.6E+03
^{134}Cs	3.0E+04	3.0E+05	3.0E+03	3.0E+04
^{137}Cs	9.8E+03	9.8E+04	9.8E+02	9.8E+03
^{144}Ce	1.9E+05	1.9E+06	1.9E+04	1.9E+05
^{147}Pm	1.9E+06	1.9E+07	1.9E+05	1.9E+06
^{151}Sm	2.5E+06	2.5E+07	2.5E+05	2.5E+06
^{152}Eu	8.3E+03	8.3E+04	8.3E+02	8.3E+03
^{154}Eu	1.1E+04	1.1E+05	1.1E+03	1.1E+04
^{226}Ra	8.0E+03	8.0E+04	8.0E+02	8.0E+03
^{228}Th	3.0E+02	3.0E+03	3.0E+01	3.0E+02
^{229}Th	1.1E+02	1.1E+03	1.1E+01	1.1E+02
^{230}Th	3.0E+02	3.0E+03	3.0E+01	3.0E+02
^{232}Th	2.1E+02	2.1E+03	2.1E+01	2.1E+02
$^{232}Th+D$[(a)]	1.1E+02	1.1E+03	1.1E+01	1.1E+02
^{232}U	1.6E+02	1.6E+03	1.6E+01	1.6E+02
^{233}U	5.7E+02	5.7E+03	5.7E+01	5.7E+02
^{234}U	5.7E+02	5.7E+03	5.7E+01	5.7E+02
^{235}U	6.0E+02	6.0E+03	6.0E+01	6.0E+02
^{238}U	6.4E+02	6.4E+03	6.4E+01	6.4E+02
$^{238}U+D$[(a)]	1.3E+02	1.3E+03	1.3E+01	1.3E+02
^{237}Np	1.5E+02	1.5E+03	1.5E+01	1.5E+02
^{238}Pu	2.4E+02	2.4E+03	2.4E+01	2.4E+02
^{239}Pu	2.2E+02	2.2E+03	2.2E+01	2.2E+02
^{240}Pu	2.2E+02	2.2E+03	2.2E+01	2.2E+02
^{241}Pu	1.5E+04	1.5E+05	1.5E+03	1.5E+04
^{241}Am	1.4E+02	1.4E+03	1.4E+01	1.4E+02

(a) *D* indicates that the daughter products of this decay chain were included in the calculation.
(b) No dose or risk factors were available in CAP88PC, which was used for the dose calculations.

TABLE 7.3. Radiological Control Levels Based on a 0.2-mrad Exposure to Film Stored for One Month[a]

Radionuclide	Bulk Contamination (pCi/g) Lead-Lined Steel	Lead-Lined Steel	Concrete	Concrete
^{3}H	3.5E+09	-[b]	1.7E+09	-
* ^{14}C	2.1E+05	2.3E+12	**1.2E+05**[c]	8.2E+12
* ^{36}Cl	NA[d]	NA	**5.5E+03**	NA
* ^{41}Ca	NA	NA	1.2E+0	NA
* ^{54}Mn	**3.6E+00**	6.8E+00	**2.7E+00**	5.2E+00
* ^{55}Fe	**3.5E+03**	-	**5.4E+02**	-
* ^{57}Co	**2.6E+01**	6.5E+03	**2.9E+01**	5.4E+03
* ^{60}Co	**1.2E+00**	2.0E+00	**8.3E-01**	1.3E+00
* ^{63}Ni	2.8E+06	5.7E+13[e]	**1.3E+06**	5.7E+13[e]
* ^{65}Zn	**4.8E+00**	7.6E+00	**3.2E+00**	5.0E+00
^{79}Se	2.8E+05	6.9E+10[e]	**1.6E+05**	6.9E+10[e]
^{90}Sr	7.8E+03	1.5E+06	**9.7E+03**	3.2E+06
^{93}Zr	3.5E+06	2.5E+09[e]	1.5E+06	2.5E+09[e]
* ^{94}Nb	**2.0E+00**	4.3E+00	**1.6E+00**	3.4E+00
* ^{99}Tc	**4.1E+04**	6.7E+08	**3.4E+04**	1.8E+09
* ^{106}Ru	**1.3E+01**	3.6E+01	**1.1E+01**	2.9E+01
* ^{110m}Ag	**1.1E+00**	2.1E+00	**8.0E-01**	1.5E+00
* ^{125}Sb	**6.7E+00**	2.1E+01	**5.7E+00**	1.7E+01
* ^{129}I	**8.6E+02**	1.8E+08[e]	**1.5E+02**	1.8E+08[e]
* ^{134}Cs	**1.8E+00**	3.8E+00	**1.4E+00**	3.0E+00
* ^{137}Cs	**5.3E+00**	1.4E+01	**4.5E+00**	1.1E+01
* ^{144}Ce	**2.2E+02**	2.4E+06	**2.2E+02**	3.2E+06
* ^{147}Pm	8.2E+04	5.5E+09	**5.9E+04**	9.9E+09
* ^{151}Sm	5.0E+05	2.6E+13[e]	**9.5E+04**	2.6E+13[e]
* ^{152}Eu	**2.6E+00**	5.0E+00	**1.8E+00**	3.3E+00
* ^{154}Eu	**2.5E+00**	4.7E+00	**1.8E+00**	3.2E+00
^{226}Ra	7.4E+02	1.0E+06	8.6E+02	9.7E+05
^{228}Th	1.8E+03	2.8E+05	9.6E+02	3.0E+05
^{229}Th	4.9E+01	1.9E+04	4.4E+01	2.1E+04
^{230}Th	7.0E+03	3.8E+08	1.6E+03	5.0E+08
^{232}Th	8.9E+03	1.1E+05[e]	1.7E+03	1.1E+05[e]
^{232}U	6.1E+03	3.6E+08	1.2E+03	4.8E+08
^{233}U	9.1E+03	1.4E+08	3.2E+03	1.9E+08
^{234}U	7.7E+03	6.5E+08	1.4E+03	8.7E+08
* ^{235}U	2.9E+01	1.1E+04	**3.3E+01**	1.2E+04
^{238}U	1.0E+04	3.4E+05[e]	1.7E+03	3.4E+05[e]

TABLE 7.3. (Cont'd)

Radionuclide	Bulk Contamination (pCi/g) Lead-Lined Steel	Lead-Lined Steel	Concrete	Concrete
^{237}Np	2.0E+02	4.0E+05	1.1E+02	4.3E+05
^{238}Pu	8.2E+03	1.7E+13[e]	1.3E+03	1.7E+13[e]
^{239}Pu	1.6E+04	5.4E+08	3.3E+03	7.3E+08
^{240}Pu	8.6E+03	2.3E+11[e]	1.4E+03	2.3E+11[e]
^{241}Pu	1.1E+09	-	5.0E+08	-
^{241}Am	4.0E+02	3.4E+12[e]	1.6E+02	3.4E+12[e]

(a) Calculations assumed that the film was stored for one month in a rectangular container made from either steel or concrete, with or without lead-shielding (lining 0.5-cm thick) the box. The radiological control levels were based on the 0.2-mrad (0.002-cGy) limit recommended by the NCRP (1989) for diagnostic x-ray film and assumed 1 mrad = 1 mrem.

(b) For radionuclides with no gamma emissions, the lead lining reduced the dose to zero, resulting in control levels that approach infinity (shown as "-" in the table).

(c) The film control levels that are more restrictive than the control levels based on individual dose (Table 7.1) are shown in **BOLD** type with an * preceding the respective radionuclide.

(d) "NA" indicates that this concrete activation product was Not Applicable to this scenario and was considered only in the concrete recycling scenarios.

(e) The specific activity of the radionuclide is provided. The calculated control level exceeds the specific activity of the radionuclide.

8.0 DISCUSSION

Decommissioning of nuclear facilities could generate large quantities of radioactive and nonradioactive contaminated materials. All of these materials will require categorization as radioactive, nonradioactive, or exempt from regulatory control, followed by disposal, reuse, or recycling (IAEA 1990). In order to limit the volume of low-level wastes that are generated and to allow recycling or reuse of the materials, a defined set of radiological control limits is needed. The criteria upon which these control limits should be chosen are 1) the doses where the possibility of detrimental health effects occurring in any individual is extremely low and 2) the effects on industry should be minimized so that they are economically acceptable. In the application of these control limits it should be determined whether their use is 1) environmentally acceptable and cost-effective and/or 2) environmentally preferred. Various types of high-technology industries, such as those that manufacture integrated electronic circuits, background-radiation detectors, or film, seek high-quality starting materials. If recycled and reused metals are going to gain acceptance, all criteria should be satisfied. The control levels calculated in this report are presented to demonstrate the methodology and do not necessarily represent DOE's criteria for application to material recycle.

8.1 LINKING LIMITS FOR PUBLIC, WORKER, ELECTRONICS, AND FILM

In this study, radiological control levels were determined for three main groups or sets of circumstances:

- individual workers and consumers of final products manufactured from recycled materials
- the MEI in a population situated downwind of a generic smelter
- non-health-related doses to film

Control levels were not calculated for electronic components since the results of a literature review indicated that these components could withstand the radiation doses resulting from residual radioactivity based on any of the

above-mentioned three groups (see Section 5.0). In order to compare control levels having similar units, the surface contamination control levels for tools/equipment and buildings were not included in the following analysis. In the same vein, the results of the film scenarios dealing with the concrete film-storage boxes were excluded from the analysis because it is unlikely that the storage areas for film would be constructed of undiluted, recycled concrete. As mentioned before, the results of this analysis are considered preliminary because of the conservative assumptions used throughout.

The most limiting bulk contamination radiological control levels resulting from each of the three groups were combined and analyzed as a single group. A summary of the rankings of the derived radiological control levels for each radionuclide considered for recycling and reuse is presented in Table 8.1. For each radionuclide, each scenario's control level is ranked with the most restrictive control level (or lowest concentration) identified as RANK 1 in Table 8.1.

For 23 of the 42 radionuclides, including all but one of the alpha-emitters, the most restrictive control level is based on the steel-recycle individual dose. For one radionuclide (^{129}I), the concrete-recycle is limiting. In 17 of the 42 radionuclides considered, the most restrictive control level was associated with the scenario involving film exposure from recycled steel used in manufacturing a film-storage box. For the most part, these radionuclides were either the beta/gamma emitters or the no-photon emitters. One radionuclide, namely ^{238}U, resulted in the most restrictive control level being based on the dose to the MEI in a population surrounding a smelter. (Tables listing the limiting scenarios and control levels for all recycled materials considered are given in Appendix E.)

For further comparison, all RANK 1 control levels were sorted from lowest (most restrictive) to highest (least restrictive) (Table 8.2). As a result of this sorting, 17 of the radionuclides were found to have control levels less than or equal to 10 pCi/g, and 30 radionuclides were found to have control levels less than 2 nCi/g (see Figure 8.1). The 2 nCi/g level is the limit identified by the DOT for transportation of radioactive material on

TABLE 8.1. Rankings of Radiological Control Levels (pCi/g) for the Selected Radionuclides Considered for Recycling and Reuse[a]

Grouping	Nuclide	Rank 1	Rank 2	Rank 3	Rank 4	Rank 5	Rank 6	Rank 7
A[b]	^{229}Th	1.9E+00	6.1E+00	9.6E+00	9.6E+00	1.1E+01	4.9E+01	1.9E+04
		IND1[c]	IND4	IND2	IND3	PUBLC	FILM1	FILM2
A	^{232}Th	3.0E+00	9.5E+00	1.1E+01	1.5E+01	1.5E+01	8.9E+03	6.2E+08
		IND1	IND4	PUBLC	IND2	IND3	FILM1	FILM2
A	^{232}U	5.0E+00	1.6E+01	1.6E+01	2.5E+01	2.5E+01	6.1E+03	3.6E+08
		IND1	PUBLC	IND4	IND3	IND2	FILM1	FILM2
A	^{237}Np	6.4E+00	1.5E+01	2.1E+01	3.2E+01	3.2E+01	2.0E+02	4.0E+05
		IND1	PUBLC	IND4	IND2	IND3	FILM1	FILM2
A	^{239}Pu	1.0E+01	2.2E+01	3.2E+01	5.0E+01	5.0E+01	1.6E+04	5.4E+08
		IND1	PUBLC	IND4	IND3	IND2	FILM1	FILM2
A	^{240}Pu	1.0E+01	2.2E+01	3.2E+01	5.0E+01	5.0E+01	8.6E+03	1.1E+18
		IND1	PUBLC	IND4	IND2	IND3	FILM1	FILM2
A	^{241}Pu	5.8E+02	1.5E+03	1.8E+03	2.9E+03	2.9E+03	1.1E+09	
		IND1	PUBLC	IND4	IND3	IND2	FILM1	
A	^{241}Am	6.0E+00	1.4E+01	2.0E+01	3.0E+01	3.0E+01	4.0E+02	2.1E+13
		IND1	PUBLC	IND4	IND3	IND2	FILM1	FILM2
A	^{228}Th	1.1E+01	3.0E+01	3.4E+01	5.3E+01	5.3E+01	1.8E+03	2.8E+05
		IND1	PUBLC	IND4	IND3	IND2	FILM1	FILM2
A	^{230}Th	1.3E+01	3.0E+01	4.0E+01	6.3E+01	6.3E+01	7.0E+03	3.8E+08
		IND1	PUBLC	IND4	IND3	IND2	FILM1	FILM2
A	^{238}U	1.3E+01	2.8E+01	8.7E+01	1.4E+02	1.4E+02	1.0E+04	3.8E+13
		PUBLC	IND1	IND4	IND3	IND2	FILM1	FILM2
A	^{238}Pu	1.1E+01	2.4E+01	3.5E+01	5.5E+01	5.5E+01	8.2E+03	1.2E+18
		IND1	PUBLC	IND4	IND3	IND2	FILM1	FILM2
A	^{233}U	2.6E+01	5.7E+01	8.0E+01	1.3E+02	1.3E+02	9.1E+03	1.4E+08
		IND1	PUBLC	IND4	IND3	IND2	FILM1	FILM2
A	^{234}U	2.6E+01	5.7E+01	8.0E+01	1.3E+02	1.3E+02	7.7E+03	6.5E+08
		IND1	PUBLC	IND4	IND3	IND2	FILM1	FILM2
A	^{235}U	2.8E+01	2.9E+01	6.0E+01	8.6E+01	1.4E+02	1.4E+02	1.1E+04
		IND1	FILM1	PUBLC	IND4	IND3	IND2	FILM2
A	^{226}Ra	2.0E+02	5.1E+02	7.4E+02	8.0E+02	1.0E+03	1.3E+03	1.0E+06
		IND1	IND3	FILM1	PUBLC	IND2	IND4	FILM2
B	^{60}Co	1.2E+00	1.7E+00	2.0E+00	4.4E+00	1.3E+01	6.7E+01	8.7E+02
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
B	^{65}Zn	4.8E+00	6.7E+00	7.6E+00	1.7E+01	4.9E+01	2.6E+02	2.3E+04
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
B	^{94}Nb	2.0E+00	3.5E+00	4.3E+00	7.1E+00	2.0E+01	1.1E+02	1.9E+02
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
B	^{110m}Ag	1.1E+00	1.7E+00	2.1E+00	3.8E+00	1.1E+01	5.8E+01	4.9E+03
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
B	^{134}Cs	1.8E+00	3.1E+00	3.8E+00	6.3E+00	1.8E+01	9.6E+01	3.0E+03
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
B	^{137}Cs	5.3E+00	9.7E+00	1.4E+01	1.9E+01	5.3E+01	2.8E+02	9.8E+03
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
B	^{152}Eu	2.6E+00	3.9E+00	5.0E+00	9.3E+00	2.6E+01	1.4E+02	8.3E+02
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
B	^{154}Eu	2.5E+00	3.8E+00	4.7E+00	8.8E+00	2.5E+01	1.3E+02	1.1E+03
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
B	^{93}Zr	3.8E+04	1.4E+05	1.6E+05	1.9E+05	1.9E+05	3.5E+06	2.8E+22
		IND1	IND4	PUBLC	IND2	IND3	FILM1	FILM2
B	^{129}I	5.3E+02	8.6E+02	9.6E+02	1.2E+03	1.3E+03	2.0E+03	3.9E+12
		IND4	FILM1	PUBLC	IND2	IND1	IND3	FILM2

TABLE 8.1. (Cont'd)

Grouping	Nuclide	Rank 1	Rank 2	Rank 3	Rank 4	Rank 5	Rank 6	Rank 7
B	^{144}Ce	2.2E+02	6.0E+02	7.2E+02	1.6E+03	1.2E+04	1.9E+04	2.4E+06
		FILM1	IND4	IND1	IND2	IND3	PUBLC	FILM2
B	^{147}Pm	7.9E+04	8.2E+04	1.9E+05	2.5E+05	4.0E+05	4.0E+05	5.5E+09
		IND1	FILM1	PUBLC	IND4	IND2	IND3	FILM2
B	^{151}Sm	1.0E+05	2.5E+05	3.6E+05	5.0E+05	5.2E+05	5.2E+05	4.8E+17
		IND1	PUBLC	IND4	FILM1	IND3	IND2	FILM2
C	^{106}Ru	1.3E+01	2.4E+01	3.6E+01	4.4E+01	1.3E+02	6.7E+02	1.6E+04
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
C	^{90}Sr	2.5E+03	4.3E+03	7.6E+03	1.3E+04	2.3E+04	3.3E+04	1.5E+06
		IND1	IND3	FILM1	IND2	PUBLC	IND4	FILM2
D	^{54}Mn	3.6E+00	5.9E+00	6.8E+00	1.3E+01	3.6E+01	1.9E+02	1.3E+04
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
D	^{125}Sb	6.7E+00	1.3E+01	2.1E+01	2.4E+01	6.5E+01	3.6E+02	8.1E+03
		FILM1	IND4	FILM2	IND1	IND2	IND3	PUBLC
D	^{57}Co	2.6E+01	7.9E+01	9.0E+01	2.4E+02	1.4E+03	6.5E+03	8.2E+04
		FILM1	IND4	IND1	IND2	IND3	FILM2	PUBLC
D	^{36}Cl	3.4E+03	1.1E+04	1.7E+04	3.6E+04	1.8E+05	2.4E+05	
		FILM1	IND1	IND4	IND2	IND3	FILM2	
D	^{55}Fe	3.5E+03	3.7E+03	1.5E+04	2.5E+04	2.1E+05	3.7E+06	
		FILM1	IND4	IND1	IND2	IND3	PUBLC	
D	^{99}Tc	4.1E+04	1.3E+05	1.4E+05	3.0E+05	3.4E+05	4.3E+05	6.7E+08
		FILM1	IND1	IND4	IND2	PUBLC	IND3	FILM2
E	^{3}H	5.7E+06	8.8E+06	2.9E+07	5.2E+07	1.7E+08	3.5E+09	
		IND1	IND3	IND2	PUBLC	IND4	FILM1	
E	^{14}C	1.9E+05	2.1E+05	2.6E+05	5.6E+05	8.3E+05	1.1E+07	2.3E+12
		IND1	FILM1	IND3	IND4	IND2	PUBLC	FILM2
E	^{41}Ca	7.9E+03	8.5E+03	3.5E+04	5.7E+04	4.6E+05	1.5E+07	
		FILM1	IND4	IND1	IND2	IND3	PUBLC	
E	^{63}Ni	5.2E+05	1.0E+06	2.6E+06	2.8E+06	2.9E+06	5.5E+06	1.1E21
		IND1	IND3	IND2	FILM1	PUBLC	IND4	FILM2
E	^{79}Se	4.3E+04	6.7E+04	2.2E+05	2.8E+05	7.2E+05	4.3E+12	
		IND1	IND3	IND2	FILM1	IND4	FILM2	

(a) The value for the most restrictive types (i.e., having the lowest concentration) of controlling doses considered for a given radionuclide is shown as Rank 1.

(b) The nuclides were grouped by mode of decay and internal DCFs:
- A = alpha emitters with large DCF (^{228}Th, ^{229}Th, ^{230}Th, ^{232}Th, ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{238}U, ^{226}Ra, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am)
- B = beta/gamma emitters with large DCF (^{60}Co, ^{65}Zn, ^{94}Nb, ^{93}Zr, $^{110m}Ag/^{110}Ag$, ^{129}I, $^{134}Cs/^{137m}Ba$, ^{137}Cs, ^{144}Ce, ^{147}Pm, ^{151}Sm, ^{152}Eu, ^{154}Eu)
- C = non-photon emitters with moderate DCF (^{90}Sr, $^{106}Ru/^{106}Rh$, ^{241}Pu)
- D = beta/gamma emitters with low DCF (^{36}Cl, ^{54}Mn, ^{55}Fe, ^{57}Co, ^{99}Tc, ^{125}Sb)
- E = non-photon emitters with moderate DCF (^{3}H, ^{14}C, ^{41}Ca, ^{63}Ni, ^{79}Se).

(c) The dose type associated with the listed control level for a given radionuclide is given immediately below that value. The abbreviations used are as follows:
- IND1 = Most restrictive radiological control level for steel when based on bulk contamination contributing to a 1.0 mrem/yr dose to the individual.
- IND2 = Most restrictive radiological control level for aluminum when based on bulk contamination contributing to a 1.0-mrem/yr dose to the individual.
- IND3 = Most restrictive radiological control level for copper when based on bulk contamination contributing to a 1.0-mrem/yr dose to the individual.
- IND4 = Most restrictive radiological control level for concrete when based on bulk contamination contributing to a 1.0-mrem/yr dose to the individual.
- PUBLC = Recycling control level based on a 0.1-mrem/yr dose the MEI in a population surrounding a smelter that handles 100 t of recycled metal per year.
- FILM1 = Radiological control level based on 0.2-mrad exposure to film stored for one month in a rectangular box made of undiluted, recycled steel.
- FILM2 = Radiological control level based on 0.2-mrad exposure to film stored for one month in a rectangular box made of undiluted, recycled steel with a 0.5-cm thick lead lining.

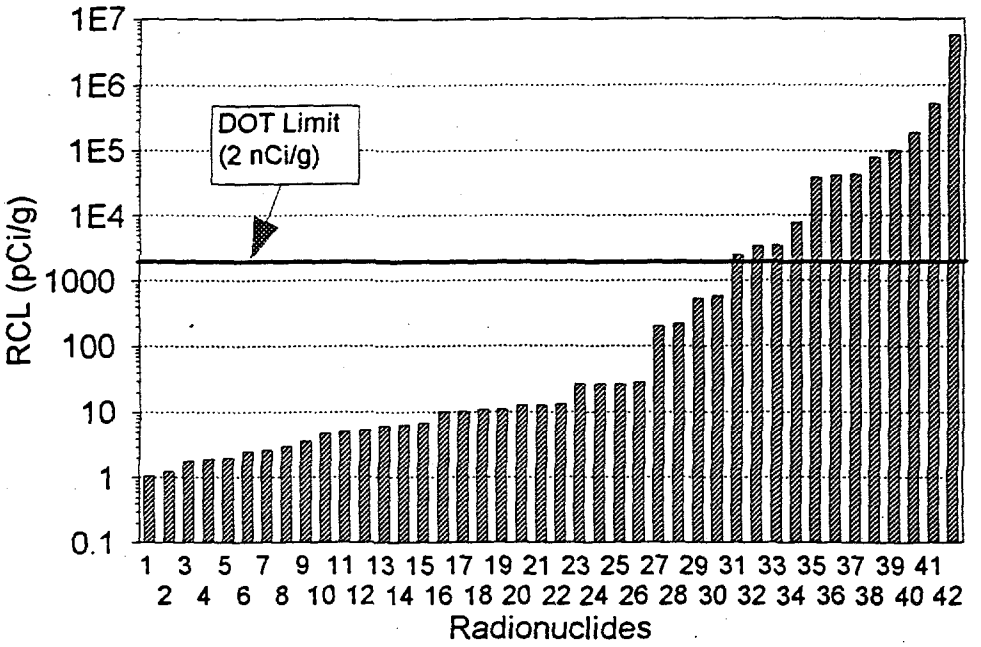


FIGURE 8.1. Radiological Control Levels for 42 Radionuclides Considered in *Recycling and Reuse* (*The numerical designations for* radionuclides applies to the overall rank; a key is given in Table 8.2.)

public roads (DOT 1985). The DOT transportation limit has a major impact on recycled materials because, in all cases, the material has to be transported to the recycler or smelter.

To determine if the derived control levels would be detectable by generally accepted survey techniques, the control levels were compared with the general detection limits for bulk material used at a DOE site (see footnote (d), Table 8.2). The results of such a comparison will, of course, vary from site to site since the detection capabilities themselves will change from site to site. As shown in Table 8.2, 12 of the 42 limiting radiological control levels were less than the associated lower limit of detection used in this comparison.

8.2 ISSUES AND QUESTIONS

As with any generic analysis, there are uncertainties in the results that are associated with the generic parameter values and assumptions chosen for use in the analysis. As a result of this study and the study performed by the IAEA (1992), the following areas were identified where additional data are needed:

- partitioning of radionuclides during smelting
- detailed descriptions of smelters DOE expects to use for the various recycled metals
- acceptable levels of the lower limit of detection of radionuclides in bulk material
- realistic scenarios for film and concrete
- information on the maximum permissible levels of contaminants in metals and concrete

In regard to the last bullet, Table 7.3 indicates control levels from lead-lined film container scenarios whose calculated control level exceeds the specific activity of the pure isotope. In these cases, the calculated control level was replaced with the specific activity of the isotope. The media (i.e., steel or concrete) identified with a control level equal to the specific activity of an isotope is no longer identifiable as that media, but is the pure isotopic element. In turn, there are maximum levels of a contaminant that will affect the physical and mechanical properties of a media. The maximum concentrations of contaminants permissible in each media considered needs to be evaluated and compared to the concentrations that result from the calculated control criteria.

In establishing radiological control levels for unrestricted release, decisions should be based on limiting conditions for doses to individuals, doses to the public, and the collective dose. In this report, the first two conditions have been addressed; but because of uncertainty about the appropriate generic population distribution and meteorology to use, collective doses have not been assessed. As additional data are collected, this assessment will need to be revised.

TABLE 8.2. Comparison of Most Restrictive Control Levels with Example Lower Limits of Detection for Selected Radionuclides

Nuclide Reference Number		Nuclide[a]	Most Restrictive Control Level[b] (pCi/g)	Dose Type Associated with Control Level[c]	LLD[d] (pCi/g)
1	*	^{110m}Ag	1.1	FILM1	10
2	*	^{60}Co	1.2	FILM1	10
3	*	^{134}Cs	1.8	FILM1	10
4	*	^{229}Th	1.9	IND1	2
5	*	^{94}Nb	2.0	FILM1	10
6	*	^{154}Eu	2.5	FILM1	20
7	*	^{152}Eu	2.6	FILM1	10
8		^{232}Th	3.0	IND1	2
9	*	^{54}Mn	3.6	FILM1	30
10	*	^{65}Zn	4.8	FILM1	10
11		^{232}U	5.0	IND1	2
12	*	^{137}Cs	5.3	FILM1	10
13		^{241}Am	6.0	IND1	2
14		^{237}Np	6.4	IND1	2
15	*	^{125}Sb	6.7	FILM1	10
16		^{239}Pu	10	IND1	2
17		^{240}Pu	10	IND1	2
18		^{228}Th	11	IND1	2
19		^{238}Pu	11	IND1	2
20		^{106}Ru	13	FILM1	12
21		^{238}U	13	PUBLC	2
22		^{230}Th	13	IND1	2
23		^{233}U	26	IND1	2
24		^{234}U	26	IND1	2
25	*	^{57}Co	26	FILM1	30
26		^{235}U	28	IND1	2
27		^{226}Ra	200	IND1	2
28		^{144}Ce	220	FILM1	10
29		^{129}I	150	IND4	25
30		^{241}Pu	530	IND1	3
31		^{90}Sr	2,500	IND1	10
32		^{36}Cl	3,430	FILM1	30
33		^{55}Fe	3,450	FILM1	30
34		^{41}Ca	7,880	FILM1	30
35		^{93}Zr	38,000	IND1	10
36		^{99}Tc	40,500	FILM1	30

TABLE 8.2. (Cont'd)

Nuclide Reference Number	Nuclide[a]	Most Restrictive Control Level[b] (pCi/g)	Dose Type Associated with Control Level[c]	LLD[d] (pCi/g)
37	^{79}Se	43,000	IND1	10
38	^{147}Pm	79,000	IND1	30
39	^{151}Sm	100,000	IND1	10
40	^{14}C	190,000	IND1	30
41	^{63}Ni	525,000	IND1	30
42	^{3}H	5,700,000	IND1	30

(a) Those radionuclides for which the associated control level is less than the example LLD are shown in **bold** with an asterisk (*) preceding the nuclide.

(b) For the seven types of controlling doses considered for a given radionuclide, the value for the most restrictive (i.e., the lowest concentration) is shown.

(c) The abbreviations used for the dose types are as follows:

- IND1 = Steel control level based on bulk contamination contributing to a 1.0-mrem/yr dose to the individual.
- IND2 = Aluminum control level based on bulk contamination contributing to a 1.0-mrem/yr dose to the individual.
- IND3 = Copper control level based on bulk contamination contributing to a 1.0-mrem/yr dose to the individual.
- IND4 = Concrete control level based on bulk contamination contributing to a 1.0-mrem/yr dose to the individual.
- IND5 = Reuse control level based on surface contamination on tools and equipment contributing to a 1.0-mrem/yr dose to the individual.
- PUBLC = Recycling control level based on a 0.1-mrem/yr dose to the MEI in a population surrounding a smelter that handles 100 t of recycled metal per year.
- FILM1 = Recycling and reuse control level based on a 0.2-mrad exposure to film stored for one month in a rectangular box made of steel.
- FILM2 = Recycling and reuse control level based on a 0.2-mrad exposure to film stored for one month in a rectangular box made of steel with a 0.5-cm-thick lead lining.

(d) Information concerning example lower limit of detection (LLD) values obtained from Westinghouse Hanford Company, Richland, WA.

- For an alpha emitter not listed, the LLD is 2 pCi/g.
- For any pure beta emitter not listed having a maximum beta energy of less than 100 keV, the LLD is 30 pCi/g.
- For any pure beta emitter not listed having a maximum beta energy equal to or greater than 100 keV, the LLD is 10 pCi/g.
- For any beta-gamma emitter not listed, the LLD is 10 pCi/g.
- Gamma emitter detection limits are acceptable, assuming single non-Compton interferant isotopes. Samples with more than one isotope will likely have higher effective LLDs due to high-energy interference. For example, when ^{137}Cs is found in the presence of ^{60}Co, the LLD for ^{137}Cs will be adversely affected.

9.0 REFERENCES

Aaberg, R.L., D.A. Baker, K. Rhoads, M.F. Jarvis, W.E. Kennedy, Jr. 1995. Radiation Dose Assessment Methodology and Preliminary Dose Estimates to Support U.S. Department of Energy Radiation Control Criteria for Regulated Treatment and Disposal of Hazardous Wastes and Materials. PNL-9405. Richland, WA: PNL.

Chau, T.K., R. Shuman, R.D. Baird, and E.A. Jennrich. 1992. Draft Release Limits for Contaminated Sludges and Solids. RAE-9005/7-2. Prepared by Rodgers and Associates Engineering for Martin Marietta Energy Systems K-25 Plant, Oak Ridge, TN: K-25.

Commission of The European Communities (CEC). 1988. Radiological Protection Criteria for the Recycling of Materials from the Dismantling of Nuclear Installations. Radiation Protection No. 43. Luxembourg: CEC.

International Atomic Energy Agency (IAEA). 1980. Atmospheric Dispersion in Nuclear Power Plant Siting - A Safety Guide. Safety Series No. 50-SG-S3. Vienna: IAEA.

International Atomic Energy Agency (IAEA). 1982a. Basic Safety Standards for Radiation Protection. 1982 Edition, Jointly Sponsored by IAEA, ILO, NEA(OECD), WHO. Vienna: IAEA.

International Atomic Energy Agency (IAEA). 1982b. Generic Models and Parameters for Assessing the Environmental Transfer of Radionuclides from Routine Releases. Safety Series No. 57. Vienna: IAEA.

International Atomic Energy Agency (IAEA). 1988a. Factors Relevant to the Recycling or Reuse of Components Arising From the Decommissioning and Refurbishment of Nuclear Facilities. Technical Reports Series No. 293. Vienna: IAEA.

International Atomic Energy Agency (IAEA). 1988b. Principles for the Exemption of Radiation Sources and Practices from Regulatory Control. Safety Series No. 89. Vienna: IAEA.

International Atomic Energy Agency (IAEA). 1990. The Regulatory Process for the Decommissioning of Nuclear Facilities. Safety Series No. 105. Vienna: IAEA.

International Atomic Energy Agency (IAEA). 1992. Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities. Safety Series No. 111-P-1.1. Vienna: IAEA.

International Commission on Radiological Protection (ICRP). 1975. Report of the Task Group on Reference Man. ICRP Publication 23. Oxford: Pergamon.

International Commission on Radiological Protection (ICRP). 1977. Recommendations of the ICRP. ICRP Publication 26. Oxford: Pergamon.

International Commission on Radiological Protection (ICRP). 1977-82. Limits for Intakes of Radionuclides by Workers. ICRP Publication 30, vols 2-8. Oxford: Pergamon Press.

International Commission on Radiological Protection (ICRP). 1986. The Metabolism of Plutonium and Related Elements. ICRP Publication 48. Oxford: Pergamon Press.

International Commission on Radiological Protection (ICRP). 1991. 1990 Recommendations of the International Commission on Radiological Protection. ICRP Publication 60; vol. 21, No. 1-3. Oxford: Pergamon Press.

Kennedy, W.E., Jr., and D.L. Strenge. 1992. Residual Radioactive Contamination from Decommissioning: Technical Basis for Translating Contamination Levels to Annual Dose. NUREG/CR-5512. Washington, DC: NRC.

Kiefer, H., and R. Maushart. 1972. Radiation Protection Measurement. Oxford: Pergamon Press.

Messenger, G.C., and M.S. Ash. 1986. The Effects of Radiation on Electronic Systems. New York: Van Nostrand Reinhold Co.

Napier, B.A., D.L. Strenge, R.A. Peloquin, and J.V. Ramsdell. 1988. GENII - The Hanford Environmental Radiation Dosimetry Software System. PNL-6584, Volumes 1-3. Richland, WA: PNL.

National Council on Radiation Protection and Measurements (NCRP). 1976. Structural Shielding Design and Evaluation for Medical Use of X Rays and Gamma Rays of Energies Up to 10 MeV. NCRP Report No. 49. Washington, DC: NCRP.

National Council on Radiation Protection and Measurements (NCRP). 1989. Medical X-Ray, Electron Beam and Gamma-Ray Protection For Energies Up to 50 MeV. (Equipment Design, Performance and Use). NCRP Report No. 102. Washington, DC: NCRP.

O'Donnell, F.R., S.J. Cotter, D.C. Kocher, E.L. Etnier, and A.P. Watson. 1978. Potential Radiation Dose to Man From Recycle of Metals Reclaimed From a Decommissioned Nuclear Power Plant. NUREG/CR-0134, ORNL/NUREG/TM-215. Prepared by Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission. Washington, DC: NRC.

Parks, B.S. 1992. User's Guide for CAP88-PC, Version 1.0. 402-B-92-001. Prepared by the U.S. Environmental Protection Agency (EPA) for the U.S. Department of Energy. Las Vegas, NV: EPA.

Passenheim, B.C. 1988. How to Do Radiation Tests. A "Do It Yourself" Manual. San Diego, CA: Ingenuity Ink, 5838 Camber Drive.

Rogers, A.W. 1981. Practical Autoradiography. Amersham Review 20. Arlington Heights, IL: Amersham Corporation.

U.S. Department of Energy (DOE) 1988a. "General Environmental Protection Program Requirements." Order 5400.1. Washington, DC: DOE.

U.S. Department of Energy (DOE). 1988b. Internal Dose Conversion Factors for Calculation of Dose to the Public. DOE/EH-0071. Washington, DC: DOE.

U.S. Department of Energy (DOE). 1988c. "Radiation Protection for Occupational Work." Order 5480.11. Washington, DC: DOE.

U.S. Department of Energy (DOE). 1990. "Radiation Protection of the Public and the Environment." Order 5400.5. Washington, DC: DOE.

U.S. Department of Energy (DOE). 1994. Radiological Control Manual. Washington, DC: DOE.

U.S. Department of Transportation (DOT). 1985. "Radioactive Materials -- Definitions." 49 CFR 173.403, Subpart 1. Washington, DC: DOT.

U.S. Environmental Protection Agency (EPA). 1989. National Emission Standards for Emissions of Radionuclides Other than Radon From Department of Energy Facilities. 40 CFR 61, Subpart H. Washington, DC: EPA.

U.S. Environmental Protection Agency (EPA). 1991. National Secondary Drinking Water Regulations. 40 CFR 143. Washington, DC: EPA.

U.S. Nuclear Regulatory Commission (NRC). 1974. Termination of Operating Licenses for Nuclear Reactors. Regulatory Guide 1.86. Washington, DC: NRC.

U.S. Nuclear Regulatory Commission (NRC). 1977. 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. Regulatory Guide 1.109, Rev 1*. Washington, DC: NRC.

U.S. Nuclear Regulatory Commission (NRC). 1980. Draft Environmental Impact Statement Concerning Proposed Rulemaking: Exemption from Licensing Requirements for Smelted Alloys Containing Residual Technetium-99 and Low-Enriched Uranium. NUREG-0519. Washington, DC: NRC.

U.S. Nuclear Regulatory Commission (NRC). 1984. Final Environmental Impact Statement Concerning Proposed Rulemaking: Exemption from Licensing Requirements for Smelted Alloys Containing Residual Technetium-99 and Low-Enriched Uranium. NUREG-0518. Washington, DC: NRC.

U.S. Nuclear Regulatory Commission (NRC). 1991. Standards for Protection Against Radiation. 10 CFR 20. Washington, DC: NRC.

APPENDICES

APPENDIX A

Description of Cited Regulatory Limits

APPENDIX A: DESCRIPTION OF CITED REGULATORY LIMITS

A.1 OCCUPATIONAL DOSE LIMITS

- DOE 5480.11 Radiation Protection Standards for Internal and External Exposure for Occupational Workers.

The exposure of an occupational worker to radiation resulting from routine DOE activities shall not cause the limiting values for assessed dose specified herein to be exceeded:

The limiting value of annual effective dose equivalent from both internal and external sources received in any year by an occupational worker is 5 rem (0.05 Sv).

The limiting value of annual dose equivalent received in any year by an occupational worker, for individual organs and tissues is 15 rem (0.15 Sv) to the lens of the eye or 50 rem (0.50 Sv) to any other organ, tissue (including skin), or extremity of the body.

Occupational workers shall be monitored, as appropriate, to demonstrate compliance with the radiation protection standards and to estimate the dose equivalents received from external and internal sources of radiation. Workplaces shall be routinely monitored, as appropriate, for identification and control of potential exposure sources.

Personnel dosimetry programs shall be adequate to demonstrate compliance with the radiation protection standards. Personnel dosimeters shall be routinely calibrated and maintained and shall meet the requirements of the DOE Laboratory Accreditation Program for Personnel Dosimetry as specified in DOE 5480.15. Personnel dosimetry shall be provided to radiation workers who have the potential to exceed in a year any one of the following from external sources:

- 100 mrem (0.001 Sv) annual effective dose equivalent to the whole body
- 5 rem (0.05 Sv) annual dose equivalent to the skin
- 5 rem (0.05 Sv) annual dose equivalent to any extremity
- 1.5 rem (0.015) annual dose equivalent to the lens of the eye.

- 10 CFR 20.1201: Occupational Dose Limits for Adults.

The licensee shall control the occupational dose to individual adults, except for planned special exposures under § 20.1206, to the following dose limits.

(1) An annual limit, which is the more limiting of

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue, other than the lens of the eye, being equal to 50 rems (0.5 Sv).

In addition to the annual dose limits, the licensee shall limit the soluble uranium intake by an individual to 10 mg/week in consideration of chemical toxicity (see footnote 3 of appendix B to §§ 20.1001-20.2401).

Footnote 3 - For soluble mixtures of U-238, U-234, and U-235 in air, chemical toxicity may be the limiting factor. If the percent by weight (enrichment) of U-235 is not greater than five, the concentration value for a 40-hour work week is 0.2 milligrams uranium per cubic meter of air average. For any enrichment, the product of the average concentration and time of exposure during a 40-hour work week shall not exceed 8E-3 (SA) μCi-hr/ml, where SA is the specific activity of the uranium inhaled. The specific activity for natural uranium is 6.77E-7 curies per gram U.

- 52 Federal Register, No. 17: Presidential Guidelines for Occupational Exposure

The ICRP recommends that the effective (i.e., weighted) dose equivalent incurred in any year be limited to 5 rems.

A.2 PUBLIC DOSE LIMITS

- DOE 5400.5: Public dose means the dose received by member(s) of the public from exposure to radiation and to radioactive material released by a DOE facility or operation, whether the exposure is within a DOE site boundary or off-site. It does not include dose received from occupational exposures, doses received from naturally occurring "background" radiation, doses received as a patient from medical practices, or doses received from consumer products.

The primary public dose limits include consideration of all exposure modes from all DOE activities (including remedial actions). DOE must also comply with legally applicable requirements (e.g., 40 CFR Parts 61, 191, and 192 and 10 CFR Parts 60 and 72).

Except as provided by II.1a(4), the exposure of members of the public to radiation sources as a consequence of all routine DOE activities shall

not cause, in a year, an effective dose equivalent greater than 100 mrem (1 mSv).

In addition, DOE operators are required to report DOE-related effective dose equivalent contributions of 10 mrem (0.10 mSv) or more in a year.

(II.1a(4)). Exceptions: Unusual circumstances could affect a DOE activity in such a manner that the potential public dose could exceed an effective dose equivalent of 100 mrem (1 mSv) in a year.

- 10 CFR 20.1301: Dose limits for Individual Members of the Public

  Each licensee shall conduct operations so that

  (1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contribution from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003, and

  (2) The dose in any unrestricted area from external sources does not exceed 0.002 rem (0.02 mSv) in any one hour.

A.3 LIMITS ON AIRBORNE EMISSIONS

- DOE 5400.1: Airborne radiation and radioactive materials discharged from DOE facilities shall comply with the requirements of 40 CFR Part 61, "National Emission Standards for Hazardous Air Pollutants."

- DOE 5400.5: Airborne emissions only, all DOE sources of radionuclides. To the extent required by the Clean Air Act, the exposure of members of the public to radioactive materials released to the atmosphere as a consequence of routine DOE activities shall not cause members of the public to receive, in a year, an effective dose equivalent greater than 10 mrem (0.1 mSv). Exposures to, and releases of, radon-220, radon-222 and their respective decay products are subject to DOE limits (IV.4b, IV.6).

  (IV.4b). Generic guidelines for concentrations of airborne radon decay products shall apply to existing occupied or habitable structures on private property that are intended for release without restriction; structures that will be demolished or buried are excluded. The applicable guideline is 40 CFR Part 192: In any occupied or habitable building, the objective of remedial action shall be, and a reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 WL. In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL. Remedial actions by DOE are not required to comply with this guideline when there is reasonable

assurance that residual radioactive material is not the source of the radon concentration.

(IV.6b(2)). Controls shall be designed such that Rn-222 concentrations in the atmosphere above facility surfaces or openings in addition to background levels will not exceed

(a) 100 pCi/L at any given point

(b) An annual average concentration of 30 pCi/L over the facility site

(c) An annual average concentration of 3 pCi/L at or above any location outside the facility site

Flux rates from the storage of radon-producing wastes shall not exceed 20 $pCi/m^2 \cdot sec$, as required by 40 CFR Part 61.

- 40 CFR 61.92 (Subpart H): Emissions of radionuclides to the ambient air from Department of Energy facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem/yr.

- 40 CFR 61.93 (Subpart H): Radionuclide emission rates from point sources (stacks or vents) shall be measured in accordance with the following requirements or other procedures for which EPA has granted prior approval.

  Radionuclide emission measurements in conformance with the requirements shall be made at all release points that have the potential to discharge radionuclides into the air in quantities that could cause an effective dose equivalent in excess of 1% of the standard (i.e., 0.1 mrem/yr).

- 40 CFR 61 (see page 51656, 54 Federal Register, No. 240, 12-15-89)

  The Agency recognizes that consideration of maximum individual risk (MIR) (the maximum estimated risk of contracting cancer following a lifetime of exposure to the emitted pollutant) must take into account the strengths and weaknesses of this measure of risk. It is estimated based on the assumption of continuous exposure for 24 hr/day for 70 yr. As such, it does not necessarily reflect the true risk, but displays a conservative risk level, which is an upperbound that is unlikely to be exceeded. The Administrator believes that an MIR of approximately 1 in 10 thousand (1×10^{-4}) should ordinarily be the upper end of the range of acceptability.

A.4 DOT LIMITS

- 49 CFR 173.403: "Radioactive material" means any material having a specific activity greater than 0.002 microcuries per gram (2 nCi/gram). Each package of radioactive material, unless excepted by 173.421, 173.422, 173.424, 173.425(b) or 173.427, shall be labeled as provided in Subpart E of Part 172 of this subchapter ("RADIOACTIVE MATERIAL").

A.5 LIMITS ON SOURCE MATERIALS

- 10 CFR 20.2: "Source Material" means: (i) Uranium or thorium, or any combination thereof, in any physical or chemical form; or (ii) ores which contain by weight one-twentieth of one percent (0.05%) or more of (a) uranium, (b) thorium or (c) any combination thereof. Source material does not include special nuclear material.

A.6 REFERENCES

49 CFR 173.401. 1988. "Shippers--General Requirements for Shipments and Packages; Subpart 1--Radioactive Materials." U.S. Department of Transportation. Washington, DC: DOT.

50 FR 41898. October 16, 1985. 49 CFR 173.403, Subpart 1, "Radioactive Materials--Definitions." Department of Transportation. Washington, DC: DOT.

52 FR 17. 1987. "Presidential Guidelines for Occupational Exposure." Federal Register.

54 FR 51695. 1990. 40 CFR 61, Subpart H, "National Emission Standards for Radionuclide Emissions from Department of Energy Facilities." Federal Register. U.S. Environmental Protection Agency. Washington, DC: EPA.

56 FR 23360-23474. May 21, 1991. "Standards for Protection Against Radiation." 10 CFR 20, Federal Register. Washington, DC: NRC.

U.S. Department of Energy (DOE). 1988. "General Environmental Protection Program Requirements." Order 5400.1. Washington, DC: DOE.

U.S. Department of Energy (DOE). 1990. "Radiation Protection of the Public and the Environment." Order 5400.5. Washington, DC: DOE.

U.S. Department of Energy (DOE). 1988. "Radiation Protection for Occupational Work." Order 5480.11. Washington, DC: DOE.

APPENDIX B

Methods for External Dose Calculations

APPENDIX B: METHODS FOR EXTERNAL DOSE CALCULATIONS

The generic assessment methodology for estimating individual and collective effective dose equivalents from external exposures to radionuclides is described in the following paragraphs. This methodology is used in the recycling and reuse scenarios defined in Section 3.0 of this report to estimate the potential radiation doses resulting from recycling contaminated metals or concrete and reuse of contaminated buildings or tools and equipment.

B.1 ESTIMATION OF EXTERNAL DOSE RATES

The general equation for estimating the effective dose equivalent received by the individuals defined in the recycling exposure scenarios for radionuclide i for volume and surface contamination are shown in Equations B.1 and B.2, respectively. The source is considered to be volume contamination (μCi/g) for all scenarios except the use of buildings, as well as, tools and equipment, which were modeled as surface contamination (μCi/cm^2).

$$H_{EXT,i,s} = t\ C_{i,w}\ DF_{EXT,i,s}\ W\ P \qquad (B.1)$$

$$H_{EXT,i,s} = t\ C_{i,s}\ DF_{EXT,i,s}\ W\ P \qquad (B.2)$$

where $H_{EXT,i,s}$ = the effective dose equivalent from one year's external exposure to radionuclide i (or to photons of energy i) in source category s, mrem/yr

t = the duration of exposure for the individual, hr/yr

$C_{i,w}$ = the initial concentration of radionuclide i (or of photons of energy i) in the exempt material being recycled, μCi/g (assumed to be 1.0 in the analysis, except 10 for the slag worker scenario)

$C_{i,s}$ = the initial concentration of radionuclide i present as surface contamination, μCi/cm^2 (assumed to be 1.0 in the analysis)

$DF_{EXT,i,s}$ = the effective dose equivalent from external exposures to radionuclide i (or to photons of energy i) in source category s, mrem/hr per $\mu Ci/g$ or mrem/hr per $\mu Ci/cm^2$, depending on the scenario (from the tables in this appendix)

W = the quantity of exempt material handled by workers divided by the quantity of material handled by workers (taken to be 1.0 for this evaluation)

P = the number of people exposed (P = 1 for maximum individual dose calculations).

B.2 SCENARIO DOSE CONVERSION FACTORS (DCF)

The recycling scenarios used in this study are typical of those expected in the recycling of many metals. Each scenario revolves around an object (source) that contains some or all of the contaminated metal being considered. These sources (including their representations and dimensions) and the distances from the sources to an exposed person are combined into what are called external exposure categories for each material considered. These categories and source descriptions are listed in Table B.1 for all recycling and reuse scenarios. The table describes the exposure conditions and modeling assumptions for both recycling and reuse scenarios.

The sources considered in this study are generally represented by a self-absorbing, homogeneous, cylindrical volume with the dose point on the axis of the cylinder--or by a surface-contaminated source. External absorbers and shields were ignored except under the following circumstances: the furnace-operator scenarios for aluminum and copper, which could present the potential for exposure to molten metals contained in the furnaces; the large-pump scenario when it involves a shielded cylinder; and the copper-pipe scenario when it includes the presence of sheetrock. This procedure tends to maximize the estimated dose equivalents.

Table B.2 describes the source modeled for each scenario, lists the source representation used, and gives its dimensions (radius, length, and density). In many situations, a source can be represented better by a half cylinder than by a full cylinder. For these situations, a full cylinder was

TABLE B.1. Source Descriptions for External Dose Factors by Scenario

No.	Scenario Description	Source Description	Source[a] Representation	Density (g/cm^3)	Length[b] (cm)	Radius[c] (cm)	Distance[d] (cm)	Shields[e] (cm)		Source[f] Material
	Steel									
1	Slag worker	100-Mg slag pile	0.5 (half) cylinder	2.70	455	228	1500	0	0	Magnetite concrete
2	Automobile	3 sheets	3 full cylinders	7.86	0.1	150	50	0	0	Iron
3	Large equipment	1-Mg casting(.5)	0.5 (full) cylinder	7.86	1	201	100	0	0	Iron
	Aluminum									
4	Operator	25-Mg charge	1 full cylinder	2.70	228	114	2.5	30.5	267	Aluminum
5	Automobile	3 sheets	3 full cylinders	2.70	0.1	150	50	0	0	Aluminum
6	Frying pan	1 fry pan	1 full cylinder	2.70	1	15	60	0	0	Aluminum
	Concrete									
7	Concrete worker	100-Mg pile	0.5 (half) cylinder	2.30	480	240	1000	0	0	Concrete
8a	New room	4 walls	2 (4 half) cylinders	2.30	15	308	300	0	0	Concrete
8b	New room	Floor	1 full cylinder	2.30	15	305	100	0	0	Concrete
	Building Reuse									
9a	Building	4 walls	2 (4 half) disks	-	300	308	0	0	0	Concrete
9b	Renovation	Floor	1 full (disk)	-	100	305	0	0	0	Concrete
9c	Renovation	Ceiling	1 full (disk)	-	150	305	0	0	0	Concrete
	Tools/Equipment									
10a	Hand tools	1 small item	0.5 (half) disk	-	60	14	0	0	0	Iron
10b	Hand tools		0.5 (half) disk	-	68	14	0	0	0	Iron
11	Large pump	1 small item	5 cylinder (side)	7.86	0.1[g]	20[h]	0.5	59.5	0	Iron
	Copper									
12	Operator	10-Mg charge	1 full cylinder	8.92	117	59	2.5	30.5	267	Nickel
13	Frying pan	1 fry pan	1 full cylinder	8.92	0.2	15	60	0	0	Nickel
14	Water pipes	Water pipes	2 cylinders (side)	8.92	0.5[g]	200[h]	1.	100	0	Nickel

(a) The number preceding the description is a multiplier of the dose factor to account for partial or multiple sources.
(b) Parameter in GENII EXTDF: T1.
(c) Parameter in GENII EXTDF: SLTH.
(d) Parameter in GENII EXTDF: T2.
(e) Parameters in GENII EXTDF: T3, T4.
These distances represent additional shields. For Scenarios 4 and 12, the distances T2 and T3 represent steel furnace wall and firebrick, and T4 is distance to the receptor (shield is air). For Scenario 11, T2 represents a steel shield, and T3 is the distance through air.
(f) Source material is taken as the closest match with shield material specifications available in EXTDF. Magnetite concrete is the most similar to slag. Nickel is the closest to copper.
(g) For this geometry (cylinder side), dimension T1 represents the cylinder diameter.
(h) For this geometry (cylinder side), dimension SLTH represents the cylinder length.

TABLE B.2. Nuclide-Specific Effective Dose Equivalent Rate Factors for Recycling Scenarios (mrem/hr per μCi/g) and for Reuse Scenarios (mrem/hr per μCi/cm^2)

			External Dose Factor for Each Scenario (mrem/hr per μCi/g)						
			1	2	3	4	5	6	7
Nuclide	Atomic No.	Halflife Days	Steel Slag Worker	Steel Auto	Steel Large Equipment	Aluminum Operator	Aluminum Auto	Aluminum Fry Pan	Concrete Worker
^{3}H	1	4.49E+03	3.1E-12	3.2E-08	3.0E-08	0.0E+00	9.0E-09	5.3E-10	4.6E-11
^{14}C	6	2.09E+06	6.9E-06	7.6E-04	4.6E-04	5.5E-16	4.1E-04	2.0E-05	1.5E-05
^{36}Cl	17	1.10E+08	3.8E-04	4.6E-02	2.5E-02	7.6E-08	1.4E-02	9.4E-04	8.2E-04
^{41}Ca	20	3.67E+07	3.3E-06	1.4E-02	1.4E-02	0.0E+00	8.8E-03	5.2E-04	6.2E-05
^{54}MNn	25	3.13E+02	1.1E+00	3.9E+01	3.6E+01	1.8E-02	1.4E+01	1.3E+00	2.9E+00
^{55}Fe	26	9.86E+02	7.6E-06	3.3E-02	3.1E-02	0.0E+00	2.0E-02	1.2E-03	1.4E-04
^{57}Co	27	2.71E+02	7.1E-02	5.6E+00	3.2E+00	8.3E-06	2.1E+00	1.8E-01	1.8E-01
^{60}Co	27	1.92E+03	4.1E+00	1.1E+02	1.1E+02	2.2E-01	4.0E+01	3.7E+00	1.0E+01
^{63}Ni	28	3.65E+04	2.9E-07	4.5E-05	4.1E-05	4.6E-31	2.5E-05	1.0E-06	6.8E-07
^{65}Zn	30	2.44E+02	1.1E+00	2.9E+01	2.8E+01	5.7E-02	1.0E+01	9.5E-01	2.6E+00
^{79}Se	34	2.37E+07	5.1E-06	5.7E-04	3.6E-04	9.2E-17	3.1E-04	1.5E-05	1.1E-05
^{90}Sr	38	1.04E+04	1.7E-04	2.1E-02	1.1E-02	8.6E-09	7.3E-03	4.5E-04	3.8E-04
^{93}Zr	40	5.58E+08	2.0E-07	3.6E-05	3.3E-05	9.0E-36	1.9E-05	7.9E-07	4.9E-07
^{94}Nb	41	7.41E+06	1.9E+00	7.1E+01	6.4E+01	2.3E-02	2.5E+01	2.3E+00	4.9E+00
^{99}Tc	43	7.77E+07	3.1E-05	3.9E-03	2.1E-03	7.7E-12	1.6E-03	9.2E-05	6.8E-05
^{106}Rr	44	3.68E+02	2.8E-01	1.1E+01	9.7E+00	5.0E-03	3.8E+00	3.5E-01	6.9E-01
^{110m}Ag	47	2.50E+02	4.0E+00	1.3E+02	1.2E+02	1.5E-01	4.6E+01	4.2E+00	1.0E+01
^{125}Sb	51	1.01E+03	5.2E-01	2.1E+01	1.8E+01	2.5E-03	7.7E+00	6.8E-01	1.3E+00
^{129}I	53	5.73E+09	7.7E-03	1.9E-01	1.7E-01	1.1E-16	4.0E-01	1.4E-02	2.1E-02
^{134}Cs	55	7.53E+02	2.2E+00	7.9E+01	7.2E+01	3.3E-02	2.8E+01	2.5E+00	5.5E+00
^{137}Cs	55	1.10E+04	6.8E-01	2.7E+01	2.4E+01	4.1E-03	9.5E+00	8.6E-01	1.7E+00
^{144}Ce	58	2.84E+02	9.7E-03	7.0E-01	3.8E-01	4.8E-09	3.1E-01	2.4E-02	2.5E-02
^{144}Pm	61	9.58E+02	1.7E-05	1.9E-03	1.1E-03	1.7E-12	9.2E-04	4.9E-05	3.8E-05
^{151}Sm	62	3.29E+04	3.0E-06	2.5E-04	2.4E-04	1.4E-26	2.5E-04	9.7E-06	8.9E-06
^{152}Eu	63	4.96E+03	1.8E+00	5.4E+01	5.0E+01	9.3E-02	1.9E+01	1.8E+00	4.5E+00
^{154}Eu	63	3.21E+03	1.8E+00	5.7E+01	5.2E+01	7.3E-02	2.0E+01	1.8E+00	4.5E+00
^{226}Ra	88	5.84E+05	2.6E-03	2.0E-01	1.1E-01	3.0E-08	7.7E-02	6.6E-03	6.7E-03
^{228}Th	90	6.98E+02	9.6E-04	7.9E-02	4.5E-02	3.2E-08	3.4E-02	2.7E-03	2.5E-03
^{229}Th	90	2.68E+06	3.9E-02	3.1E+00	1.5E+00	4.6E-07	1.3E+00	1.0E-01	1.0E-01
^{230}Th	90	2.81E+07	1.3E-04	1.9E-02	1.3E-02	3.1E-11	1.1E-02	7.1E-04	3.5E-04
^{232}Th	90	5.13E+12	6.7E-05	1.4E-02	1.1E-02	1.9E-11	8.3E-03	5.2E-04	2.0E-04
^{232}U	92	2.36E+04	9.2E-05	2.0E-02	1.6E-02	3.2E-11	1.2E-02	7.4E-04	2.8E-04
^{233}U	92	5.58E+07	1.3E-04	1.5E-02	1.0E-02	8.1E-11	6.7E-03	4.9E-04	3.6E-04
^{234}U	92	8.29E+07	5.3E-05	1.6E-02	1.3E-02	1.8E-11	9.2E-03	5.7E-04	1.8E-04
^{235}U	92	2.57E+11	6.8E-02	5.1E+00	2.9E+00	8.0E-07	1.9E+00	1.6E-01	1.7E-01
^{238}U	92	1.63E+12	2.1E-05	1.2E-02	1.0E-02	3.3E-23	7.1E-03	4.3E-04	9.1E-05
^{237}Np	93	7.81E+08	9.2E-03	7.5E-01	3.6E-01	2.3E-08	3.8E-01	2.7E-02	2.5E-02
^{238}Pu	94	3.20E+04	1.2E-05	1.4E-02	1.3E-02	4.5E-31	8.7E-03	5.1E-04	7.8E-05
^{239}Pu	94	8.81E+06	3.6E-05	7.8E-03	6.3E-03	2.1E-11	4.2E-03	2.7E-04	1.1E-04
^{240}Pu	94	2.40E+06	1.2E-05	1.3E-02	1.2E-02	5.0E-31	8.3E-03	4.9E-04	7.8E-05
^{241}Pu	94	5.26E+03	1.0E-11	1.1E-07	1.0E-07	0.0E+00	3.0E-08	1.8E-09	1.5E-10
^{241}Am	95	1.58E+05	6.7E-03	4.6E-01	1.5E-01	6.5E-23	3.1E-01	2.0E-02	1.6E-02

TABLE B.2. (Cont'd)

Nuclide	Atomic No.	Halflife Days	External Dose Factor for Each Scenario						
			8 Concrete New Room mrem/hr /μCi/g	9 Reuse Building Renovations mrem/hr /μCi/cm²	10 Reuse Hand Tools mrem/hr /μCi/cm²	11 Reuse Large Pump mrem/hr /μCi/g	12 Copper Operator mrem/hr /μCi/g	13 Copper Water Pipe mrem/hr /μCi/g	14 Copper Fry Pan mrem/hr /μCi/g
^{3}H	1	4.49E+03	1.4E-07	8.6E-08	1.6E-09	3.9E-22	0.0E+00	8.3E-10	2.0E-31
^{14}C	6	2.09E+06	3.0E-03	1.9E-03	2.3E-05	1.9E-07	1.0E-15	1.4E-05	5.2E-09
^{36}Cl	17	1.10E+08	1.0E-01	4.8E-02	4.8E-04	2.4E-05	7.2E-08	9.3E-04	1.7E-06
^{41}Ca	20	3.67E+07	2.0E-01	1.2E-01	2.1E-03	1.7E-16	0.0E+00	3.4E-04	8.4E-26
^{43}Mn	25	3.13E+02	2.8E+02	5.4E+01	4.9E-01	2.7E-02	7.2E-03	8.6E-01	3.3E-03
^{55}Fe	26	9.86E+02	4.5E-01	2.7E-01	4.9E-03	4.0E-16	0.0E+00	7.8E-04	1.9E-25
^{57}Co	27	2.71E+02	2.1E+01	8.7E+00	8.4E-02	3.4E-03	3.6E-06	1.2E-01	2.5E-04
^{60}Co	27	1.92E+03	9.6E+02	1.6E+02	1.4E+00	7.8E-02	7.7E-02	2.5E+00	1.0E-02
^{63}Ni	28	3.65E+04	2.4E-04	1.5E-04	2.2E-06	1.8E-09	1.5E-30	1.1E-06	4.7E-11
^{65}Zn	30	2.44E+02	2.5E+02	4.1E+01	3.6E-01	2.0E-02	1.9E-02	6.4E-01	2.6E-03
^{79}Se	34	2.37E+07	2.3E-03	1.4E-03	1.8E-05	1.3E-07	2.1E-16	1.1E-05	3.5E-09
^{90}Sr	38	1.04E+04	5.0E-02	2.7E-02	2.8E-04	1.0E-05	8.9E-09	4.2E-04	6.5E-07
^{93}R	40	5.58E+08	1.9E-04	1.2E-04	1.8E-06	1.1E-09	4.4E-35	8.5E-07	2.7E-11
^{94}ZNb	41	7.41E+06	4.8E+02	9.7E+01	8.7E-01	4.8E-02	9.3E-03	1.6E+00	5.8E-03
^{99}Tc	43	7.77E+07	1.2E-02	6.9E-03	8.3E-05	1.5E-06	1.1E-11	7.5E-05	6.6E-08
^{106}Rr	44	3.68E+02	6.8E+01	1.5E+01	1.3E-01	7.6E-03	2.0E-03	2.5E-01	8.9E-04
^{110m}Ag	47	2.50E+02	9.8E+02	1.8E+02	1.6E+00	8.9E-02	5.5E-02	2.9E+00	1.1E-02
^{125}Sb	51	1.01E+03	1.3E+02	3.0E+01	2.7E-01	1.4E-02	1.1E-03	4.6E-01	1.6E-03
^{129}I	53	5.73E+09	3.1E+00	2.0E+00	2.0E-02	2.0E-05	2.5E-16	3.4E-03	5.7E-07
^{134}Ca	55	7.53E+02	5.4E+02	1.1E+02	9.7E-01	5.3E-02	1.3E-02	1.7E+00	6.5E-03
^{137}Ca	55	1.10E+04	1.7E+02	3.7E+01	3.3E-01	1.8E-02	1.8E-03	5.9E-01	2.1E-03
^{144}Ce	58	2.84E+02	2.8E+00	1.2E+00	1.1E-02	4.1E-04	3.8E-09	1.4E-02	2.9E-05
^{147}Pm	61	9.58E+02	6.6E-03	3.9E-03	4.7E-05	6.5E-07	1.7E-12	3.6E-05	2.7E-08
^{151}Sm	62	3.29E+04	3.8E-03	1.9E-03	2.9E-05	6.4E-09	3.8E-26	5.7E-06	1.2E-10
^{152}Eu	63	4.96E+03	4.3E+02	7.6E+01	6.8E-01	3.7E-02	3.3E-02	1.2E+00	4.6E-03
^{154}Eu	63	3.21E+03	4.4E+02	7.9E+01	7.0E-01	3.9E-02	2.6E-02	1.2E+00	4.8E-03
^{226}Ra	88	5.84E+05	7.4E-01	3.1E-01	2.9E-03	1.2E-04	1.6E-08	4.2E-03	8.8E-06
^{228}Th	90	6.98E+02	4.2E-01	2.0E-01	2.6E-03	4.1E-05	2.2E-08	1.6E-03	2.8E-06
^{229}Th	90	2.68E+06	1.3E+01	5.6E+00	5.7E-02	1.8E-03	3.2E-07	6.1E-02	1.1E-04
^{230}Th	90	2.81E+07	1.7E-01	9.7E-02	1.6E-03	5.2E-06	2.4E-11	3.9E-04	2.4E-07
^{232}Th	90	5.13E+12	1.5E-01	8.8E-02	1.5E-03	2.3E-06	1.5E-11	3.1E-04	1.3E-07
^{232}U	92	2.36E+04	2.2E-01	1.3E-01	2.2E-03	3.5E-06	2.5E-11	4.5E-04	2.1E-07
^{233}U	92	5.58E+07	9.9E-02	5.2E-02	8.1E-04	6.3E-06	6.3E-11	3.2E-04	4.6E-07
^{234}U	92	8.29E+07	1.8E-01	1.1E-01	1.9E-03	1.9E-06	1.4E-11	3.6E-04	1.1E-07
^{235}U	92	2.57E+11	1.9E+01	7.8E+00	7.3E-02	3.1E-03	5.5E-07	1.1E-01	2.4E-04
^{238}U	92	1.63E+12	1.5E-01	8.7E-02	1.6E-03	7.2E-07	5.5E-23	2.7E-04	1.6E-08
^{237}Np	93	7.81E+08	3.8E+00	2.0E+00	2.3E-02	3.8E-04	1.6E-08	1.4E-02	1.9E-05
^{238}Pu	94	3.20E+04	1.9E-01	1.1E-01	2.0E-03	2.2E-07	9.5E-31	3.3E-04	4.9E-09
^{239}Pu	94	8.81E+06	8.0E-02	4.6E-02	8.0E-04	1.7E-06	1.6E-11	1.8E-04	1.2E-07
^{240}Pu	94	2.40E+06	1.8E-01	1.1E-01	1.9E-03	2.4E-07	1.1E-30	3.1E-04	5.4E-09
^{241}Pu	94	5.26E+03	4.8E-07	2.9E-07	5.2E-09	1.3E-21	0.0E+00	2.8E-09	6.8E-31
^{241}Am	95	1.58E+05	2.5E+00	1.6E+00	1.8E-02	1.7E-04	1.1E-22	6.5E-03	3.7E-06

defined such that the area of its flat surface was twice that actually needed. The effective dose equivalent was then calculated using the full cylinder and dividing the resulting dose by two. For one category of consumer products, namely the reuse of large pumps, the source was modeled by a shielded cylindrical source from the side, rather than end-on. For some of the consumer product categories, such as the automobile, an individual was assumed to be exposed to two or more identical, symmetrically arranged sources. Figure B.1 shows the source geometry used for cylinder and half-cylinder sources in this study with definition of the dose receptor point; of the cylinder length, L; and of the cylinder radius, r.

The calculations were performed using the EXTDF module for the GENII Software System (Napier et al. 1988). The calculations are analogous to those included in the IAEA study (IAEA 1992) where the CONDOS code was used (O'Donnell et al. 1981).

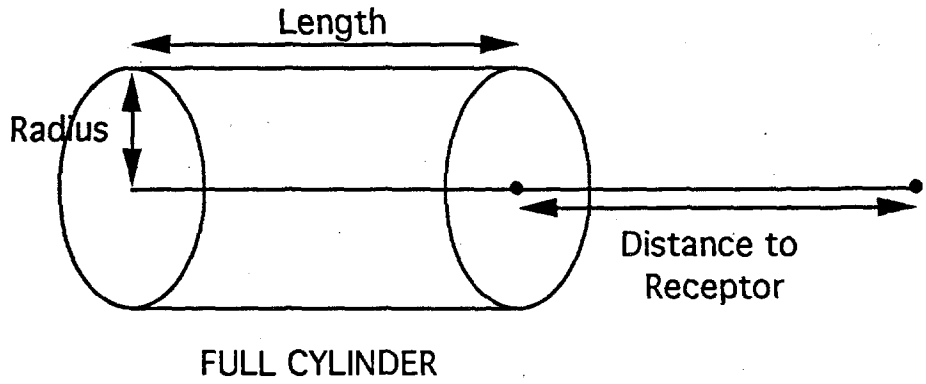
Table B.2 lists the effective dose equivalent rates obtained for the scenarios and the radionuclides considered in this study. The results can be applied to other radionuclide concentrations, different from 1.0 μCi/g, and to different exposure times. External dose conversion factors are available for the 251 radionuclides currently supported in the GENII Software System (Napier et al. 1988). Dose factors for other radionuclides may be estimated by using surrogate radionuclides with similar gamma energy distributions.

B.3 REFERENCES

International Atomic Energy Agency (IAEA). 1992. Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities. Safety Series No. 111-P-1.1. Vienna: IAEA.

O'Donnell, F. R., D. C. Kocher, O. W. Burk, and F. H. Clark. 1981. CONDOS-II - A Tool for Estimating Radiation Doses from Radionuclide-Containing Consumer Projects. NUREG/CR-2068; ORNL/NUREG/TM-454. Washington, D.C.: NRC.

Napier, B.A., R.A. Peloquin, D.L. Strenge, and J.V. Ramsdell. 1988. GENII - The Hanford Environmental Radiation Dosimetry Software System. PNL-6584 Vol 1-3. Prepared by Pacific Northwest Laboratory for the U.S. Department of Energy. Richland, WA: PNL.



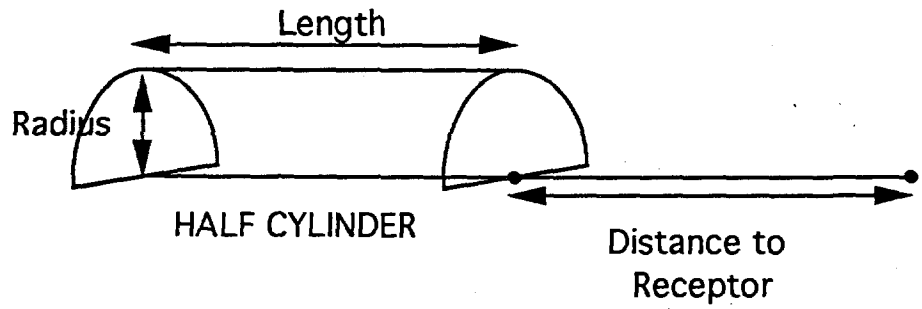


FIGURE B.1. Cylinder and Half-Cylinder Source Geometries

APPENDIX C

Methods for Internal Dose Calculations

APPENDIX C: INTERNAL DOSE CALCULATIONS

The following paragraphs describe the generic assessment methodology for estimating the individual and collective doses resulting from inhalation or ingestion of radioactive materials. These methods are used in the recycling scenarios defined in Section 3.0 of this report to estimate the potential radiation doses resulting from recycling of contaminated materials. This methodology is intended to serve as an example of the types of exposure conditions and pathway analyses that may have to be considered when attempting to establish exempt quantities for recycling on a national basis. The analysis described includes a variety of exposure pathways and scenarios in an attempt to provide a detailed assessment and produce generic criteria.

C.1 INHALATION OF AIRBORNE RADIOACTIVE MATERIALS

The general equations for estimating the committed dose by inhalation received by the individuals defined in the recycling exposure scenarios for radionuclide i for volume and surface contamination, respectively, are as follows:

$$H_{INH,i} = V\ t\ DF_{INH,i}\ W\ (C_d\ C_{w,i}) \qquad (C.1)$$

$$H_{INH,i} = V\ t\ DF_{INH,i}\ W\ (C_{s,i}\ RF\ TF_{INH}) \qquad (C.2)$$

where $H_{INH,i}$ = the committed effective dose equivalent from one year's intake of radionuclide i by inhalation, rem

V = the breathing rate of the worker, m^3/h (taken to be 1.2 in this analysis [ICRP 1975])

t = the duration of exposure for the individual, hr/yr

$DF_{INH,i}$ = the committed effective dose from inhalation of 1 μCi of radionuclide i, rem/μCi

W = fraction of the material handled by the workers that is exempt material being recycled (taken to be 1.0 in this analysis)

C_d = concentration of respirable dust in air, g/m^3 (see Table C.1)

$C_{w,i}$ = the undiluted concentration of radionuclide i in the dust, assumed to be equal to that of the exempt material being recycled, $\mu Ci/g$ (taken to be 1 $\mu Ci/g$ in this analysis)

$C_{s,i}$ = the concentration of radionuclide i present as surface contamination, $\mu Ci/cm^2$ (taken to be 10^4 $\mu Ci/m^2$, or 1 $\mu Ci/cm^2$; in this analysis)

RF = the resuspension factor for surface activity, m^{-1} (taken to be $10^{-6}/m$ in this analysis)

TF_{INH} = the transfer factor for inhalation of surface activity, the fraction of surface contamination available for resuspension, m^{-1} (taken to be $10^{-6}/m$ in this analysis).

The committed effective dose equivalent factors used in this study were taken from Publication 30 of the ICRP (ICRP 1977-1982). These factors are listed in Table C.1 for the radionuclides of interest in the example generic analysis.

The concentration of respirable dust in the air will vary depending upon a variety of factors, including the physical condition of the material being handled, the quantity of the material present, and the building's ventilation. Thus, it is difficult to predict the concentrations that may be present during any recycling step. However, so that a complete analysis may be performed, air concentrations have been assumed for those recycling steps where the potential for inhalation is most likely. These air concentrations are listed in Table C.2 for the general types of work conditions defined in Section 3.0. In determining these air concentrations, information from the literature on airborne dust loadings was examined, and previous studies that considered the potential impacts from the recycling of radioactive metals were reviewed (O'Donnell et al. 1978; NRC 1980; CEC 1985). In general, the air concentrations reported in the literature vary between about 10^{-3} g/m^3 and 10^{-5} g/m^3.

TABLE C.1. Internal Dose Factors for Radionuclides Included in Analysis

Nuclide	CEDE per Unit Intake[a] Inhalation (rem/μCi)	Ingestion (rem/μCi)	Nuclide	CEDE Per Unit Intake[a] Inhalation (rem/μCi)	Ingestion (rem/μCi)
^{3}H	6.3×10^{-5}	6.3×10^{-5}	^{144}Ce	3.5×10^{-1}	2.0×10^{-2}
^{14}C	2.4×10^{-5}	2.1×10^{-3}	^{147}Pm	3.4×10^{-2}	9.5×10^{-4}
^{36}Cl	2.0×10^{-2}	3.0×10^{-3}	^{151}Sm	2.9×10^{-2}	3.4×10^{-4}
^{41}Ca	1.3×10^{-3}	1.2×10^{-3}	^{152}Eu	2.2×10^{-1}	6.0×10^{-3}
^{54}Mn	6.4×10^{-3}	2.7×10^{-3}	^{154}Eu	2.6×10^{-1}	9.1×10^{-3}
^{55}Fe	1.2×10^{-3}	5.8×10^{-4}	^{226}Ra	$7.9 \times 10^{+0}$	$1.1 \times 10^{+0}$
^{57}Co	7.5×10^{-3}	6.6×10^{-4}	^{228}Th	$3.1 \times 10^{+2}$	3.8×10^{-1}
^{60}Co	1.5×10^{-1}	1.0×10^{-2}	^{229}Th	$1.7 \times 10^{+3}$	$3.5 \times 10^{+0}$
^{63}Ni	1.9×10^{-3}	5.4×10^{-4}	^{230}Th	$2.6 \times 10^{+2}$	5.3×10^{-1}
^{65}Zn	1.8×10^{-2}	1.4×10^{-2}	^{232}Th	$1.1 \times 10^{+3}$	$2.8 \times 10^{+0}$
^{79}Se	8.9×10^{-3}	8.3×10^{-3}	^{232}U	$6.7 \times 10^{+2}$	6.8×10^{-0}
^{90}Sr + Y	2.3×10^{-1}	1.3×10^{-1}	^{233}U	$1.3 \times 10^{+2}$	2.5×10^{-1}
^{93}Zr	7.4×10^{-2}	1.6×10^{-3}	^{234}U	$1.3 \times 10^{+2}$	2.6×10^{-1}
^{94}Nb	3.3×10^{-1}	5.1×10^{-3}	^{235}U	$1.2 \times 10^{+2}$	2.5×10^{-1}
^{99}Tc	7.5×10^{-3}	1.3×10^{-3}	^{238}U	$1.2 \times 10^{+2}$	2.3×10^{-1}
^{106}Ru	4.4×10^{-1}	2.1×10^{-2}	^{237}Np	$4.9 \times 10^{+2}$	$3.9 \times 10^{+0}$
^{110m}Ag	5.3×10^{-2}	1.1×10^{-2}	^{238}Pu	$3.0 \times 10^{+2}$	5.4×10^{-2}
^{125}Sb	9.8×10^{-3}	2.6×10^{-3}	^{239}Pu	$3.3 \times 10^{+2}$	5.8×10^{-2}
^{129}I	1.8×10^{-1}	2.8×10^{-1}	^{240}Pu	$3.3 \times 10^{+2}$	5.8×10^{-2}
^{134}Cs	4.7×10^{-2}	7.4×10^{-2}	^{241}Pu	$5.7 \times 10^{+0}$	9.2×10^{-4}
^{137}Cs	3.2×10^{-2}	5.0×10^{-2}	^{241}Am	$5.2 \times 10^{+2}$	$4.5 \times 10^{+0}$

(a) Data for the dose conversion factors (committed effective dose equivalent [CEDE] per unit intake) were obtained from DOE/EH-0071 (1988). For all radionuclides, the dose conversion factor for the least soluble form of the element is shown.

TABLE C.2. Air Concentrations and Exposure Durations for Estimating Inhalation Doses for Recycling and Reuse Scenarios

Scenario Considered [a]	Exposure Duration (h)	Air Concentration (g/m^3)
Metal and Concrete:		
1.1 Loader 1	4	$5x10^{-4}$
2.1 Processor	12	10^{-4}
3.1 Worker 1	80	10^{-4}
3.2 Loader 1[b]	4	10^{-3}
3.2 Loader 2[b]	20	10^{-3}
3.3 Operator 1[b]	5	10^{-3}
3.3 Operator 2[b]	50	10^{-3}
4.1 Caster 1[b]	2.5	10^{-3}
4.1 Caster 2[b]	25	10^{-3}
4.2 Caster 3	50	10^{-3}
4.3 Slag worker	25	10^{-3}
5.2 Sheet worker	1	10^{-4}
5.3 Coil worker	80	10^{-4}
Building Reuse	2000	surface[c]
Building Renovation	---	10^{-3}
Reuse of Tools:		
1.1 Hand tools	600	surface
1.2 Small motors	600	surface
1.3 Large Pump	200	surface
1.4 Large equipment	200	surface

(a) The scenarios are described in Section 3.0.
(b) For loaders, operators, and casters, *1* indicates work conditions at a 100-t steel smelter, and *2* indicates work conditions at a 10-t steel smelter.
(c) "Surface" indicates that a resuspension factor was used to derive an air concentration.

C.2 SECONDARY INGESTION OF RADIOACTIVE MATERIALS

Ingestion of removable radioactive contamination found on recycled metals can occur when the contamination is transferred from a surface to hands, foodstuffs, cigarettes, or other items that enter the mouth. These pathways of potential ingestion are referred to as "secondary" ingestion pathways. Previous dose evaluations of the ingestion of surface contamination have been directed toward chronic occupational exposure situations (Dunster 1962; Healy 1971; Gibson and Wrixon 1979; Kennedy et al. 1981). Additional

studies have been conducted in estimating the quantities of lead contamination that could be ingested by different age groups (Sayre et al. 1974; Lepow et al. 1975; Walter et al. 1980; Gallacher et al. 1984). Table C.3 summarizes the referenced surface contamination ingestion data.

A review of previous work on secondary ingestion pathways indicates that no quantitative data for radioactive materials are available. Because of this lack of data, previous dose estimates for secondary ingestion have relied on assumed ingestion rates. Quantitative data for ingestion of lead by children indicate that they may ingest 11 to 50 mg of lead from hand surfaces with a frequency of up to 10 times per day (Lepow et al. 1975; Walter et al. 1980; Gallacher et al. 1984; LaGoy 1987). The total quantity ingested per day by children may range upward from about 0.1 g. For this analysis, adults are assumed to ingest less removable contamination than children. To estimate the potential radiation doses resulting from secondary ingestion, adult workers are assumed to ingest 10 mg of contamination per hour of exposure. Only those workers who are in direct contact with the recycled metals are assumed to be exposed by this pathway. No consumers or individuals who handle the final product are considered to be exposed through secondary ingestion.

The general equations for estimating the committed dose by ingestion received by the individuals defined in the recycling exposure scenarios for radionuclide *i* are as follows:

$$H_{ING,i} = t\ DF_{ING,i}\ W\ I\ C_{ING,i} \qquad (C.3)$$

$$H_{ING,i} = t\ DF_{ING,i}\ W\ I2\ TF_{ING}\ C_{s,i} \qquad (C.4)$$

where $H_{ING,i}$ = the committed effective dose equivalent from one year's intake of radionuclide i by ingestion, rem

t = the duration of exposure for the individual, hr/yr

$DF_{ING,i}$ = the committed effective dose equivalent from ingestion of 1 μCi of radionuclide i, rem/μCi

W = the fraction of material handled by workers that is exempt material being recycled (taken to be 1.0 in this analysis)

I = the rate of secondary ingestion of removable surface contamination, g/hr (taken to be 0.01 g/hr in this analysis)

$C_{ING.i}$ = the concentration of radionuclide i in the removable dust, μCi/g (taken to be 1.0 in this analysis, except 10.0 for the slag worker scenario)

$I2$ = the rate of secondary ingestion of removable surface contamination, m^2/hr (taken to be 10^{-4} m^2/hr in this analysis)

TF_{ING} = the transfer factor for ingestion of surface activity (taken to be 0.01 in this analysis)

$C_{s.i}$ = the concentration of radionuclide *i* present as surface contamination, $\mu Ci/m^2$ (taken to be 10^4 $\mu Ci/m^2$ or 1 $\mu Ci/cm^2$ in this analysis).

TABLE C.3. Referenced Surface Contamination Ingestion Data

Author and Reference	Reported Ingestion Rate or Other Value	Comments
Dunster 1962	10^{-3} m^2/d	Chronic ingestion of MPC_w values of ^{226}Ra, ^{90}Sr, and ^{210}Pb to derive permissible levels of skin contamination.
Gibson and Wrixon 1979	10^{-3} m^2/day	Assuming chronic ingestion. No data available to improve Dunster's model (MPC_w analysis).
Healy 1971	10^{-4} m^2/hr	Chronic ingestion during 8 h for workers, 24 h for members of the public. These are arbitrary assumptions in an effort to account for presumed higher intake by children, i.e., 2.4×10^{-3} m^2/d.
Kennedy et al. 1981	10^{-4} m^2/hr	Chronic ingestion of removable surface contamination on transportation containers. Dose estimates for both workers and members of the public (2000 hr/yr exposure).
Sayre et al. 1974	0.018 m^2 of dust on hands (children)	Dirt-hand-mouth ingestion route for intake of lead dust by children. Measurement of the amount of lead dust on children's hands compared to the level present on surfaces in a house.
Lepow et al. 1975	0.1 g/d (children)	"Hands-in-mouth" exposure route is the principal cause of excessive lead ingestion. Mean measured weight on children's hands was 11 mg. Assuming a mouthing frequency of 10 times/d for small children yields an estimate of 0.1 g of dirt ingested/d.
Walter et al. 1980	Age-dependent	Secondary risk factors for lead ingestion were found to be age-dependent. Household dustiness is a factor for ages 2 yr and under; soil lead is a factor for ages 2 to 7 yr.
Gallacher et al. 1984	10^{-3} to 3×10^{-3} m^2 (children)	Data comparing environmental and hand contamination of lead on children's hands was equivalent to 20 to 50 mg. This level was estimated to equal about 1×10^{-3} to 3×10^{-3} m^2 at the level present in outdoor areas.

C.3 REFERENCES

Commission of the European Communities (CEC), Nuclear Science and Technology. 1985. Methodology for Evaluating Radiological Consequences of the Management of Very Low-Level Solid Waste Arising from Decommissioning of Nuclear Power Plants. CEC Report EUR 10058 EN. Luxembourg: CEC.

Dunster, H. J. 1962. "Maximum Permissible Levels of Skin Contamination, United Kingdom Atomic Energy Authority." *AHSB(RP)*:R28.

Gallacher, J. E. J., P. C. Elwood, K. M. Phillips, B. E. Davies, and D. T. Jones. 1984. "Relation Between Pica and Blood Lead in Areas of Differing Lead Exposure." *Archives of Disease in Childhood* 59:40-44.

Gibson, J. A. B., and A. D. Wrixon. 1979. "Methods for the Calculation of Derived Working Limits for Surface Contamination by Low-Toxicity Radionuclides." *Health Physics* 36(3): 311-321.

Healy J. W. 1971. Surface Contamination: Decision Levels. LA-4558-MA, Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

International Commission on Radiological Protection (ICRP). 1975. Report of the Task Group on Reference Man. ICRP Publication 23. Oxford: Pergamon Press.

International Commission on Radiological Protection (ICRP). 1977-82. Limits for Intakes of Radionuclides by Workers. ICRP Publication 30, vols 208. Oxford: Pergamon Press.

Kennedy, W. E., Jr., E. C. Watson, D. W. Murphy, B. J. Harrer, R. Harty, and J. M. Aldrich. 1981. A Review of Removable Surface Contamination on Radioactive Materials Transportation Containers. NUREG/CR-1858. Washington, D.C.: NRC.

LaGoy, P. K. 1987. "Estimated Soil Ingestion Rates for Use in Risk Assessment." *Risk Analysis* 7(3):355-359.

Lepow, M. L., L. Bruckman, M. Gillette, S. Varkowitx, , R. Robino, and J. Kapish. 1975. "Investigations into Sources of Lead in the Environment of Urban Children." *Environmental Research* 10:414-426.

O'Donnell, F. R., S. J. Cotter, D. C. Kocher, E. L. Etnier, and A. P. Watson. 1978. Potential Radiation Dose to Man from Recycle of Metals Reclaimed from a Decommissioned Nuclear Power Plant. NUREG/CR-0134, ORNL/NUREG/TM-215. Washington, DC: NRC.

Sayre, J. W., E. Charney, J. Vostal, and I. B. Pless. 1974. "House and Hand Dust as a Potential Source of Childhood Lead Exposure." *American Journal of Diseases of Children* 127:167-170.

U.S. Department of Energy (DOE). Internal Dose Conversion Factors for Calculation of Dose to the Public. DOE/EH-0071. Washington, D.C.: DOE.

U.S. Nuclear Regulatory Commission (NRC). 1980. Draft Environmental Impact Statement Concerning Proposed Rulemaking: Exemption from Licensing Requirements for Smelted Alloys Containing Residual Technetium-99 and Low-Enriched Uranium. NUREG-0518. Washington, D.C.: NRC.

Walter, S. D., A. J. Yankel, and I. H. Von Lindern. 1980. "Age-Specific Risk Factors for Lead Absorption in Children." *Archives of Environmental Health* 53(1):53-58.

APPENDIX D

Dose Calculation Program

APPENDIX D: DOSE CALCULATION PROGRAM

The screening calculations are performed using a BASIC program named RECYCL.BAS. Three excerpts from the self-documenting program are listed in this appendix:

- the program title page
- the scenario subroutine that lists the parameters used
- the dose calculation subroutine that specifies equations used.

RECYCL.BAS Recycle Project Worker & Consumer Doses No. W24459

Program estimates doses from various WORKER & CONSUMER recycling scenarios.

Steel:
- 1 Slag Worker
- 2 Automobile
- 3 Large Equipment

Aluminum:
- 4 Operator
- 5 Automobile
- 6 Frying Pan

Concrete:
- 7 Worker 1
- 8 New Room

Building Reuse:
- 9 Building Renovation

Tools and Equipment Reuse
- 10 Hand Tools
- 11 Large Pump

Copper Recycle
- 12 Operator
- 13 Water Pipes
- 14 Frying Pan

Film Storage Box
- 15 Steel
- 16 Lead-lined Steel
- 17 Concrete
- 18 Lead-lined Concrete

Program generates various Tables for the Report: PNL-xxxx
Radiation Dose Assessments to Support Evaluations of Radiation Control Criteria for Recycle or Reuse of Materials and Equipment. RL Hill, et al.

Author: D. A. Baker, PNL Date 21 Feb 93 start date

Contact: D. A. Baker, PNL, (509) 375 3809

```
'---------------- Generic Scenario Parameters ----------------------------

  FResus = .000001                          ' Resuspension Factor, 1/m

  FCM = 1                                   ' Fraction Contaminated Material, "W"

  RInhal = 1.2                              ' Inhalation rate, m3/h
  Finhal = .01                              ' Inhalation transfer factor

  RInges = .01                              ' Primary Ingestion rate, g/h
  RInges2 = 1                               ' Secondary Ingestion rate, cm2/h
  FInges2 = .01                             ' Secondary Ingestion transfer factor

'--------------------------- Scenario Selection ---------------------------

  SELECT CASE SN

'--------------------------- Steel

    CASE 1                                  ' Slag Worker

      CBulk = 10                            ' Mass Concentration, uCi/g
      CSurf = 0                             ' Surface Concentration, uCi/cm2
      ConcAir = .001                        ' Air Concentration of dust, g/m3
      TExtern(1) = 25                       ' External exposure time, h
      TInhal = 25                           ' Inhalation exposure time, h
      TInges = 25                           ' Ingestion exposure time, h
      TPop = 25                             ' Population exposure time, h
      Pop = 5                               ' Population, persons

    CASE 2                                  ' Automobile

      CBulk = 1                             ' Mass Concentration, uCi/g
      CSurf = 0                             ' Surface Concentration, uCi/cm2
      ConcAir = 0                           ' Air Concentration of dust, g/m3
      TExtern(2) = 2000                     ' External exposure time, h
      TInhal = 0                            ' Inhalation exposure time, h
      TInges = 0                            ' Ingestion exposure time, h
      TPop = 300                            ' Population exposure time, h
      Pop = 1200                            ' Population, persons

    CASE 3                                  ' Large Equipment

      CBulk = 1                             ' Mass Concentration, uCi/g
      CSurf = 0                             ' Surface Concentration, uCi/cm2
      ConcAir = 0                           ' Air Concentration of dust, g/m3
      TExtern(3) = 2000                     ' External exposure time, h
      TInhal = 0                            ' Inhalation exposure time, h
      TInges = 0                            ' Ingestion exposure time, h
      TPop = 1000                           ' Population exposure time, h
      Pop = 200                             ' Population, persons


'--------------------------- Aluminum

    CASE 4                                  ' Operator

      CBulk = 1                             ' Mass Concentration, uCi/g
      CSurf = 0                             ' Surface Concentration, uCi/cm2
      ConcAir = .001                        ' Air Concentration of dust, g/m3
      TExtern(4) = 50                       ' External exposure time, h
      TInhal = 50                           ' Inhalation exposure time, h
      TInges = 50                           ' Ingestion exposure time, h
      TPop = 50                             ' Population exposure time, h
      Pop = 3                               ' Population, persons
```

```
CASE 5                                  ' Automobile

  CBulk = 1                             ' Mass Concentration, uCi/g
  CSurf = 0                             ' Surface Concentration, uCi/cm2
  ConcAir = 0                           ' Air Concentration of dust, g/m3
  TExtern(5) = 2000                     ' External exposure time, h
  TInhal = 0                            ' Inhalation exposure time, h
  TInges = 0                            ' Ingestion exposure time, h
  TPop = 300                            ' Population exposure time, h
  Pop = 3400                            ' Population, persons

CASE 6                                  ' Frying Pan

  CBulk = 1                             ' Mass Concentration, uCi/g
  CSurf = 0                             ' Surface Concentration, uCi/cm2
  ConcAir = 0                           ' Air Concentration of dust, g/m3
  TExtern(6) = 180                      ' External exposure time, h
  TInhal = 0                            ' Inhalation exposure time, h
  TInges = 180                          ' Ingestion exposure time, h
  TPop = 60                             ' Population exposure time, h
  Pop = 10000                           ' Population, persons

------------------------ Concrete

CASE 7                                  ' Worker 1

  CBulk = 1                             ' Mass Concentration, uCi/g
  CSurf = 0                             ' Surface Concentration, uCi/cm2
  ConcAir = .001                        ' Air Concentration of dust, g/m3
  TExtern(7) = 80                       ' External exposure time, h
  TInhal = 80                           ' Inhalation exposure time, h
  TInges = 0                            ' Ingestion exposure time, h
  TPop = 80                             ' Population exposure time, h
  Pop = 10                              ' Population, persons
```

```
CASE 8                                  ' New Room

  CBulk = .1                            ' Mass Concentration, uCi/g
  CSurf = 0                             ' Surface Concentration, uCi/cm2
  ConcAir = 0                           ' Air Concentration of dust, g/m3
  TExtern(8) = 6000                     ' External exposure time, h
  TInhal = 0                            ' Inhalation exposure time, h
  TInges = 0                            ' Ingestion exposure time, h
  TPop = 2000                           ' Population exposure time, h
  Pop = 10                              ' Population, persons

------------------------ Building Reuse

CASE 9                                  ' Building Renovation (Surface)

  CBulk = 0                             ' Mass Concentration, uCi/g
  CSurf = 1                             ' Surface Concentration, uCi/cm2
  ConcAir = .001                        ' Air Concentration of dust, g/m3
  Finhal = .001                         ' Inhalation transfer factor
  FInges2 = .001                        ' Secondary ingestion factor
  TExtern(9) = 200                      ' External exposure time, h
  TInhal = 200                          ' Inhalation exposure time, h
  TInges = 200                          ' Ingestion exposure time, h
  TPop = 40                             ' Population exposure time, h
  Pop = 10                              ' Population, persons

------------------------ Tools and Equipment Reuse

CASE 10                                 ' Hand Tools (Surface)

  CBulk = 0                             ' Mass Concentration, uCi/g
  CSurf = 1                             ' Surface Concentration, uCi/cm2
  ConcAir = 0                           ' Air Concentration of dust, g/m3
  TExtern(10) = 600                     ' External exposure time, h
  TInhal = 600                          ' Inhalation exposure time, h
  TInges = 600                          ' Ingestion exposure time, h
  TPop = 100                            ' Population exposure time, h
  Pop = 100                             ' Population, persons

CASE 11                                 ' Large Pump

  CBulk = 0                             ' Mass Concentration, uCi/g
  CSurf = 1                             ' Surface Concentration, uCi/cm2
  ConcAir = 0                           ' Air Concentration of dust, g/m3
  TExtern(11) = 200                     ' External exposure time, h
  TInhal = 200                          ' Inhalation exposure time, h
  TInges = 200                          ' Ingestion exposure time, h
  TPop = 80                             ' Population exposure time, h
  Pop = 40                              ' Population, persons
```

```
'---------------------- Copper Recycle

    CASE 12                                    ' Operator

      CBulk = 1                                ' Mass Concentration, uCi/g
      CSurf = 0                                ' Surface Concentration, uCi/cm2
      ConcAir = .001                           ' Air Concentration of dust, g/m3
      TExtern(12) = 50                         ' External exposure time, h
      TInhal = 50                              ' Inhalation exposure time, h
      TInges = 50                              ' Ingestion exposure time, h
      TPop = 50                                ' Population exposure time, h
      Pop = 3                                  ' Population, persons

    CASE 13                                    ' Water Pipes

      CBulk = 1                                ' Mass Concentration, uCi/g
      CSurf = 0                                ' Surface Concentration, uCi/cm2
      ConcAir = 0                              ' Air Concentration of dust, g/m3
      TExtern(13) = 6000                       ' External exposure time, h
      TInhal = 0                               ' Inhalation exposure time, h
      TInges = 6000                            ' Ingestion exposure time, h
      RInges = 4.5E-08                         ' Ingestion rate, g/h
                                               '    [value used to make ingestion of 0.27 mg ( 4.5E-8 g/h x 6000 h)]
      TPop = 2000                              ' Population exposure time, h
      Pop = 1000                               ' Population, persons

    CASE 14                                    ' Frying Pan

      CBulk = 1                                ' Mass Concentration, uCi/g
      CSurf = 0                                ' Surface Concentration, uCi/cm2
      ConcAir = 0                              ' Air Concentration of dust, g/m3
      TExtern(14) = 180                        ' External exposure time, h
      TInhal = 0                               ' Inhalation exposure time, h
      TInges = 180                             ' Ingestion exposure time, h
      TPop = 60                                ' Population exposure time, h
      Pop = 8000                               ' Population, persons

'---------------------- Film Storage Box

    CASE 15                                    ' Film Box - Steel

      CBulk = 1                                ' Mass Concentration, uCi/g
      CSurf = 0                                ' Surface Concentration, uCi/cm2
      ConcAir = 0                              ' Air Concentration of dust, g/m3
      TExtern(15) = 730.5                      ' External exposure time, h
      TInhal = 0                               ' Inhalation exposure time, h
      TInges = 0                               ' Ingestion exposure time, h
      TPop = 0                                 ' Population exposure time, h
      Pop = 0                                  ' Population, persons
```

```
CASE 16                                  ' Film Box - Lead-Lined Steel

  CBulk = 1                              ' Mass Concentration, uCi/g
  CSurf = 0                              ' Surface Concentration, uCi/cm2
  ConcAir = 0                            ' Air Concentration of dust, g/m3
  TExtern(16) = 730.5                    ' External exposure time, h
  TInhal = 0                             ' Inhalation exposure time, h
  TInges = 0                             ' Ingestion exposure time, h
  TPop = 0                               ' Population exposure time, h
  Pop = 0                                ' Population, persons

CASE 17                                  ' Film Box - Concrete

  CBulk = 1                              ' Mass Concentration, uCi/g
  CSurf = 0                              ' Surface Concentration, uCi/cm2
  ConcAir = 0                            ' Air Concentration of dust, g/m3
  TExtern(17) = 730.5                    ' External exposure time, h
  TInhal = 0                             ' Inhalation exposure time, h
  TInges = 0                             ' Ingestion exposure time, h
  TPop = 0                               ' Population exposure time, h
  Pop = 0                                ' Population, persons

CASE 18                                  ' Film Box - Lead-lined Concrete

  CBulk = 1                              ' Mass Concentration, uCi/g
  CSurf = 0                              ' Surface Concentration, uCi/cm2
  ConcAir = 0                            ' Air Concentration of dust, g/m3
  TExtern(18) = 730.5                    ' External exposure time, h
  TInhal = 0                             ' Inhalation exposure time, h
  TInges = 0                             ' Ingestion exposure time, h
  TPop = 0                               ' Population exposure time, h
  Pop = 0                                ' Population, persons
```

```
'-------------------------------------------------------------------------------

'------------ Doses based on 1 uCi/g of material:  Dose per Conc

'--------------- Inhalation

  Cinhal = ConcAir * CBulk + CSurf * FResus * Finhal * CM2perM2            ' Nuclide air concentration, uCi/m3
           g/m3    uCi/g  uCi/cm2   1/m       -       cm2/m2

  DOSE(NUM, SN, 1) = Cinhal * TInhal * RInhal * DFInhal(NUM) * FCM         ' mrem Inhalation
                     uCi/m3     h      m3/h       mrem/uCi      -

  DOSEPOP(NUM, SN, 1) = Cinhal * TPop * Pop * RInhal * DFInhal(NUM) * FCM  ' MAN-mrem Inhalation
                        uCi/m3    h     man    m3/h      mrem/uCi     -

'--------------- Ingestion

  REInges = RInges * CBulk + RInges2 * FInges2 * CSurf                     ' Nuclide ingestion rate, uCi/h
            g/h     uCi/g   cm2/h       -       uCi/cm2

  DOSE(NUM, SN, 2) = REInges * TInges * DFInges(NUM) * FCM                 ' mrem Ingestion
                      uCi/h      h        mrem/uCi      -


  DOSEPOP(NUM, SN, 2) = REInges * TPop * Pop * DFInges(NUM) * FCM          ' MAN-mrem Ingestion
                        uCi/h     h     man     mrem/uCi      -

'---------------  External

  IF SN = 9 OR SN = 10 OR SN = 11 THEN                                     ' Determine surface or mass concentration
    UNIT$(SN) = "(pCi/cm2)"
    CExt = CSurf                                                           ' Effective External concentration, uCi/m2
           uCi/cm2
  ELSE
    UNIT$(SN) = " (pCi/g) "
    CExt = CBulk                                                           ' Effective External concentration, uCi/g
           uCi/g
  END IF

  DOSE(NUM, SN, 3) = CExt * TExtern(SN) * DFEXT(NUM, SN) * FCM             ' mrem External
                     uCi/g        h      mrem/h per uCi/g    -
                  or uCi/cm2           or    per uCi/cm2

  DOSEPOP(NUM, SN, 3) = CExt * TPop * Pop * DFEXT(NUM, SN) * FCM           ' MAN-mrem External
                        uCi/g   h     man  mrem/h per uCi/g   -
                     or uCi/cm2             or    per uCi/cm2

'-------------------------------------------------------------------------------
```

```
'=============================================================================
'     Generation of Table files for limiting concentrations for each nuclide
'     for the film storage box scenarios.
'
'     FileName$       Name of output file                              INPUT
'     Nucs            Number of nuclides for table                     INPUT
'     S1              Start scenario number in material group          INPUT
'     S2              Last scenario number in material group           INPUT
'     Title$          Table title                                      INPUT
'=============================================================================

  DEFINT A-Z:   DEFSNG C-D

'-----------------------------------------------------------------------------

  PRINT :  PRINT "Building "; Title$:

  DScreen = .2                          ' Screening Dose, mrad/y

  F1$ = "   0.0E+00;;    Z\ero  "
  F2$ = "   0.0E+00;;      ∞    "       ' Non\e  "
  F3$ = "   0.0E+00"
  N$ = "         "

  FileNum = FREEFILE
  OPEN FileName$ FOR OUTPUT AS #FileNum

  PRINT #FileNum, "Table "; Title$; TAB(80); "RECYCLE Project   "; TIMESTAMP$
  PRINT #FileNum,
  PRINT #FileNum,
  PRINT #FileNum, "                                  Bulk Contamination                "
  PRINT #FileNum, "                  ------------------------ pCi/g --------------------"
  PRINT #FileNum,
  PRINT #FileNum, "                                  LEAD-LINED                LEAD-LINED"
  PRINT #FileNum, "Nuclide          STEEL             STEEL       CONCRETE      CONCRETE"
  PRINT #FileNum,

  FOR N = 1 TO Nucs

    PRINT ".";
    HY = INSTR(NUC$(N), "-")
    EL$ = LEFT$(NUC$(N), HY - 1)
    NUM$ = MID$(NUC$(N), HY + 1)

    LSET N$ = NUM$ + EL$
    PRINT #FileNum, "   " + N$;

    REDIM SumDOSE!(1 TO 5)
    REDIM Climit!(1 TO 5)

    FOR S = S1 TO S2
  SumDOSE!(S - S1 + 1) = SumDOSE!(S - S1 + 1) + DOSE(N, S, 1) + DOSE(N, S, 2) + DOSE(N, S, 3) 'Total dose-all 3 paths
    NEXT

    FOR S = 1 TO 4
      IF SumDOSE!(S) > 0 THEN                                  ' Calculate Limiting concentration
        Climit(S) = DScreen / SumDOSE!(S) * pCiperuCi          ' pCi/g
      ELSE
        Climit(S) = 0
      END IF
    NEXT

    LINE$ = "  " + FORMAT$(Climit(1), F2$) + "     " + FORMAT$(Climit(2), F2$)
    LINE$ = LINE$ + "     " + FORMAT$(Climit(3), F2$) + "     " + FORMAT$(Climit(4), F2$)
```

```
      PRINT #FileNum, LINE$

      IF N MOD 3 = 0 THEN PRINT #FileNum,

    NEXT

    PRINT #FileNum, CRFF$; :    PRINT
    CLOSE #FileNum

END SUB  '▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒▒ TableS2
```

APPENDIX E

Control Criteria Based on Individual Dose

APPENDIX E: CONTROL CRITERIA BASED ON INDIVIDUAL DOSE

The dose to individual members of the public is one of the three main factors used in establishing the radiological control levels (the other two being the dose to the public downwind of a smelter and the non-health-related impact on electronics and film). In the context of recycle and reuse of scrap metals and concrete, the individual dose is estimated for worker populations at metal smelters and concrete manufacturing facilities, and for consumers who may use items that are made from recycled metals or items after unrestricted release.

Table E.1 contains a summary of the nuclide concentrations that represent the radiological control levels. These control levels were based on a 1 mrem/yr dose to an individual for the most restrictive scenario for each recycled material type or reused material considered. Tables E.2 through E.8 present the limiting conditions for each category considered for recycle and reuse for each of the 40 radionuclides (plus two concrete activation products).

TABLE E.1. Summary of Radiological Control Levels for Each Material Considered for Recycle or Reuse as Based on an Individual Dose of 1 mrem/y

Radionuclide	Bulk Contamination (pCi/g)				Surface Contamination (pCi/cm^2)
	Steel	Aluminum	Copper	Concrete	Tools & Equip[b]
^{3}H	5.7E+06	2.9E+07	8.8E+06	1.7E+08	2.6E+06
^{14}C	1.9E+05	8.3E+05	2.6E+05	5.6E+05	7.9E+04
^{36}Cl	---[c]	---	---	1.7E+04	---
^{41}Ca	---	---	---	8.5E+03	---
^{54}Mn	1.3E+01	3.6E+01	1.9E+02	5.9E+00	3.3E+03
^{55}Fe	1.5E+04	2.5E+04	2.1E+05	3.7E+03	1.5E+05
^{57}Co	9.0E+01	2.4E+02	1.4E+03	7.9E+01	1.8E+04
^{60}Co	4.4E+00	1.3E+01	6.7E+01	1.7E+00	1.1E+03
^{63}Ni	5.2E+05	2.6E+06	1.0E+06	5.5E+06	3.0E+05
^{65}Zn	1.7E+01	4.9E+01	2.6E+02	6.7E+00	3.3E+03
^{79}Se	4.3E+04	2.2E+05	6.7E+04	7.2E+05	2.0E+04
^{90}Sr	2.5E+03	1.3E+04	4.3E+03	3.3E+04	1.3E+03
^{93}Zr	3.8E+04	1.9E+05	1.9E+05	1.4E+05	6.7E+04
^{94}Nb	7.1E+00	2.0E+01	1.1E+02	3.5E+00	1.7E+03
^{99}Tc	1.3E+05	3.0E+05	4.3E+05	1.4E+05	1.2E+05
^{106}Ru	4.4E+01	1.3E+02	6.7E+02	2.4E+01	4.2E+03
^{110m}Ag	3.8E+00	1.1E+01	5.8E+01	1.7E+00	9.8E+02
^{125}Sb	2.4E+01	6.5E+01	3.6E+02	1.3E+01	5.6E+03
^{129}I	1.3E+03	1.2E+03	2.0E+03	5.3E+02	5.9E+02
^{134}Cs	6.3E+00	1.8E+01	9.6E+01	3.1E+00	9.7E+02
^{137}Cs	1.9E+01	5.3E+01	2.8E+02	9.7E+00	2.0E+03
^{144}Ce	7.2E+02	1.6E+03	1.2E+04	6.0E+02	6.6E+03
^{147}Pm	7.9E+04	4.0E+05	4.0E+05	2.5E+05	1.2E+05
^{151}Sm	1.0E+05	5.2E+05	5.2E+05	3.6E+05	2.4E+05
^{152}Eu	9.3E+00	2.6E+01	1.4E+02	3.9E+00	2.2E+03
^{154}Eu	8.8E+00	2.5E+01	1.3E+02	3.8E+00	2.0E+03
^{226}Ra	2.0E+02	1.0E+03	5.1E+02	1.3E+03	1.4E+02
^{228}Th	1.1E+01	5.3E+01	5.3E+01	3.4E+01	4.1E+01
^{229}Th	1.9E+00	9.6E+00	9.6E+00	6.1E+00	7.0E+00
^{230}Th	1.3E+01	6.3E+01	6.3E+01	4.0E+01	4.6E+01
^{232}Th	3.0E+00	1.5E+01	1.5E+01	9.5E+00	1.0E+01
^{232}U	5.0E+00	2.5E+01	2.5E+01	1.6E+01	2.1E+01
^{233}U	2.6E+01	1.3E+02	1.3E+02	8.0E+01	1.1E+02
^{234}U	2.6E+01	1.3E+02	1.3E+02	8.0E+01	1.1E+02
^{235}U	2.8E+01	1.4E+02	1.4E+02	8.6E+01	1.1E+02
^{238}U	2.8E+01	1.4E+02	1.4E+02	8.7E+01	1.1E+02
^{237}Np	6.4E+00	3.2E+01	3.2E+01	2.1E+01	1.7E+01
^{238}Pu	1.1E+01	5.5E+01	5.5E+01	3.5E+01	4.6E+01
^{239}Pu	1.0E+01	5.0E+01	5.0E+01	3.2E+01	4.1E+01

TABLE E.1. (Cont'd)

Radionuclide	Bulk Contamination (pCi/g)				Surface Contamination (pCi/cm^2)
	Steel	Aluminum	Copper	Concrete	Tools & Equip[b]
^{240}Pu	1.0E+01	5.0E+01	5.0E+01	3.2E+01	4.1E+01
^{241}Pu	5.8E+02	2.9E+03	2.9E+03	1.8E+03	2.4E+03
^{241}Am	6.0E+00	3.0E+01	3.0E+01	2.0E+01	1.6E+01

(a) Calculations were made using the EXTDF module from the GENII Software System (Napier et al. 1988).

(b) The values presented in this table are risk-based using 1 mrem/yr and assuming certain exposure conditions. The values for most radionuclides are greater than the surface contamination values that appear in the DOE *Radiological Control Manual* (DOE 1994). The DOE (1994) values were based on the values appearing in the U.S. NRC Regulatory Guide 1.86 (1974). These values are roughly related to instrumentation responses and an assumed dose rate of 10 mrem/yr, for selected categories of radionuclides.

(c) A dash (---) indicates a concrete activation product that was considered only for concrete-recycling scenarios.

TABLE E.2. Limiting Scenarios and Corresponding Control Levels for the Recycled Steel Category

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/g)
H-3	Steel -- Slag Worker	Ingestion	1.8E-01	5.7E+06
C-14	Steel -- Slag Worker	Ingestion	5.3E+00	1.9E+05
CL-36	Steel -- Automobile	External	9.1E+01	1.1E+04
CA-41	Steel -- Automobile	External	2.9E+01	3.5E+04
MN-54	Steel -- Automobile	External	7.9E+04	1.3E+01
FE-55	Steel -- Automobile	External	6.6E+01	1.5E+04
CO-57	Steel -- Automobile	External	1.1E+04	9.0E+01
CO-60	Steel -- Automobile	External	2.3E+05	4.4E+00
NI-63	Steel -- Slag Worker	Ingestion	1.9E+00	5.2E+05
ZN-65	Steel -- Automobile	External	5.8E+04	1.7E+01
SE-79	Steel -- Slag Worker	Ingestion	2.3E+01	4.3E+04
SR-90	Steel -- Slag Worker	Ingestion	3.9E+02	2.5E+03
ZR-93	Steel -- Slag Worker	Inhalation	2.6E+01	3.8E+04
NB-94	Steel -- Automobile	External	1.4E+05	7.1E+00
TC-99	Steel -- Automobile	External	7.9E+00	1.3E+05
RU-106	Steel -- Automobile	External	2.3E+04	4.4E+01
AG-110M	Steel -- Automobile	External	2.6E+05	3.8E+00
SB-125	Steel -- Automobile	External	4.2E+04	2.4E+01
I-129	Steel -- Slag Worker	Ingestion	7.6E+02	1.3E+03
CS-134	Steel -- Automobile	External	1.6E+05	6.3E+00
CS-137	Steel -- Automobile	External	5.4E+04	1.9E+01
CE-144	Steel -- Automobile	External	1.4E+03	7.2E+02
PM-147	Steel -- Slag Worker	Inhalation	1.3E+01	7.9E+04
SM-151	Steel -- Slag Worker	Inhalation	9.6E+00	1.0E+05
EU-152	Steel -- Automobile	External	1.1E+05	9.3E+00
EU-154	Steel -- Automobile	External	1.1E+05	8.8E+00
RA-226	Steel -- Slag Worker	Ingestion	5.1E+03	2.0E+02
TH-228	Steel -- Slag Worker	Inhalation	9.4E+04	1.1E+01
TH-229	Steel -- Slag Worker	Inhalation	5.2E+05	1.9E+00
TH-230	Steel -- Slag Worker	Inhalation	7.9E+04	1.3E+01
TH-232	Steel -- Slag Worker	Inhalation	3.4E+05	3.0E+00
U-232	Steel -- Slag Worker	Inhalation	2.0E+05	5.0E+00
U-233	Steel -- Slag Worker	Inhalation	3.9E+04	2.6E+01
U-234	Steel -- Slag Worker	Inhalation	3.9E+04	2.6E+01
U-235	Steel -- Slag Worker	Inhalation	3.6E+04	2.8E+01
U-238	Steel -- Slag Worker	Inhalation	3.6E+04	2.8E+01
NP-237	Steel -- Slag Worker	Inhalation	1.6E+05	6.4E+00
PU-238	Steel -- Slag Worker	Inhalation	9.0E+04	1.1E+01
PU-239	Steel -- Slag Worker	Inhalation	9.9E+04	1.0E+01
PU-240	Steel -- Slag Worker	Inhalation	9.9E+04	1.0E+01
PU-241	Steel -- Slag Worker	Inhalation	1.7E+03	5.8E+02
AM-241	Steel -- Slag Worker	Inhalation	1.7E+05	6.0E+00
SR-90&D	Steel -- Automobile	External	1.6E+03	6.3E+02
RA-226&D	Steel -- Automobile	External	1.7E+05	6.0E+00
TH-232&D	Steel -- Slag Worker	Inhalation	4.4E+05	2.3E+00
NP-237&D	Steel -- Slag Worker	Inhalation	1.6E+05	6.4E+00
U-238&D	Steel -- Automobile	External	1.9E+05	5.4E+00

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TABLE E.3. Limiting Scenarios and Corresponding Control Levels for the Recycled Aluminum Category

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/g)
H-3	Aluminum -- Operator	Ingestion	3.5E-02	2.9E+07
C-14	Aluminum -- Frying Pan	Ingestion	1.2E+00	8.3E+05
CL-36	Aluminum -- Automobile	External	2.8E+01	3.6E+04
CA-41	Aluminum -- Automobile	External	1.8E+01	5.7E+04
MN-54	Aluminum -- Automobile	External	2.8E+04	3.6E+01
FE-55	Aluminum -- Automobile	External	4.1E+01	2.5E+04
CO-57	Aluminum -- Automobile	External	4.2E+03	2.4E+02
CO-60	Aluminum -- Automobile	External	7.9E+04	1.3E+01
NI-63	Aluminum -- Operator	Ingestion	3.8E-01	2.6E+06
ZN-65	Aluminum -- Automobile	External	2.1E+04	4.9E+01
SE-79	Aluminum -- Frying Pan	Ingestion	4.5E+00	2.2E+05
SR-90	Aluminum -- Operator	Ingestion	7.9E+01	1.3E+04
ZR-93	Aluminum -- Operator	Inhalation	5.2E+00	1.9E+05
NB-94	Aluminum -- Automobile	External	5.0E+04	2.0E+01
TC-99	Aluminum -- Automobile	External	3.3E+00	3.0E+05
RU-106	Aluminum -- Automobile	External	7.5E+03	1.3E+02
AG-110M	Aluminum -- Automobile	External	9.2E+04	1.1E+01
SB-125	Aluminum -- Automobile	External	1.5E+04	6.5E+01
I-129	Aluminum -- Automobile	External	8.0E+02	1.2E+03
CS-134	Aluminum -- Automobile	External	5.6E+04	1.8E+01
CS-137	Aluminum -- Automobile	External	1.9E+04	5.3E+01
CE-144	Aluminum -- Automobile	External	6.2E+02	1.6E+03
PM-147	Aluminum -- Operator	Inhalation	2.5E+00	4.0E+05
SM-151	Aluminum -- Operator	Inhalation	1.9E+00	5.2E+05
EU-152	Aluminum -- Automobile	External	3.9E+04	2.6E+01
EU-154	Aluminum -- Automobile	External	4.0E+04	2.5E+01
RA-226	Aluminum -- Operator	Ingestion	1.0E+03	1.0E+03
TH-228	Aluminum -- Operator	Inhalation	1.9E+04	5.3E+01
TH-229	Aluminum -- Operator	Inhalation	1.0E+05	9.6E+00
TH-230	Aluminum -- Operator	Inhalation	1.6E+04	6.3E+01
TH-232	Aluminum -- Operator	Inhalation	6.7E+04	1.5E+01
U-232	Aluminum -- Operator	Inhalation	4.0E+04	2.5E+01
U-233	Aluminum -- Operator	Inhalation	7.8E+03	1.3E+02
U-234	Aluminum -- Operator	Inhalation	7.8E+03	1.3E+02
U-235	Aluminum -- Operator	Inhalation	7.2E+03	1.4E+02
U-238	Aluminum -- Operator	Inhalation	7.2E+03	1.4E+02
NP-237	Aluminum -- Operator	Inhalation	3.1E+04	3.2E+01
PU-238	Aluminum -- Operator	Inhalation	1.8E+04	5.5E+01
PU-239	Aluminum -- Operator	Inhalation	2.0E+04	5.0E+01
PU-240	Aluminum -- Operator	Inhalation	2.0E+04	5.0E+01
PU-241	Aluminum -- Operator	Inhalation	3.4E+02	2.9E+03
AM-241	Aluminum -- Operator	Inhalation	3.3E+04	3.0E+01
SR-90&D	Aluminum -- Automobile	External	3.6E+02	2.8E+03
RA-226&D	Aluminum -- Automobile	External	5.8E+04	1.7E+01
TH-232&D	Aluminum -- Operator	Inhalation	8.7E+04	1.1E+01
NP-237&D	Aluminum -- Operator	Inhalation	3.1E+04	3.2E+01
U-238&D	Aluminum -- Automobile	External	6.6E+04	1.5E+01

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TABLE E.4. Limiting Scenarios and Corresponding Control Levels for the Recycled Copper Category

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/g)
H-3	Copper -- Frying Pan	Ingestion	1.1E-01	8.8E+06
C-14	Copper -- Frying Pan	Ingestion	3.8E+00	2.6E+05
CL-36	Copper -- Water Pipes	External	5.6E+00	1.8E+05
CA-41	Copper -- Frying Pan	Ingestion	2.2E+00	4.6E+05
MN-54	Copper -- Water Pipes	External	5.2E+03	1.9E+02
FE-55	Copper -- Water Pipes	External	4.7E+00	2.1E+05
CO-57	Copper -- Water Pipes	External	7.0E+02	1.4E+03
CO-60	Copper -- Water Pipes	External	1.5E+04	6.7E+01
NI-63	Copper -- Frying Pan	Ingestion	9.7E-01	1.0E+06
ZN-65	Copper -- Water Pipes	External	3.9E+03	2.6E+02
SE-79	Copper -- Frying Pan	Ingestion	1.5E+01	6.7E+04
SR-90	Copper -- Frying Pan	Ingestion	2.3E+02	4.3E+03
ZR-93	Copper -- Operator	Inhalation	5.2E+00	1.9E+05
NB-94	Copper -- Water Pipes	External	9.3E+03	1.1E+02
TC-99	Copper -- Frying Pan	Ingestion	2.3E+00	4.3E+05
RU-106	Copper -- Water Pipes	External	1.5E+03	6.7E+02
AG-110M	Copper -- Water Pipes	External	1.7E+04	5.8E+01
SB-125	Copper -- Water Pipes	External	2.8E+03	3.6E+02
I-129	Copper -- Frying Pan	Ingestion	5.0E+02	2.0E+03
CS-134	Copper -- Water Pipes	External	1.0E+04	9.6E+01
CS-137	Copper -- Water Pipes	External	3.5E+03	2.8E+02
CE-144	Copper -- Water Pipes	External	8.4E+01	1.2E+04
PM-147	Copper -- Operator	Inhalation	2.5E+00	4.0E+05
SM-151	Copper -- Operator	Inhalation	1.9E+00	5.2E+05
EU-152	Copper -- Water Pipes	External	7.1E+03	1.4E+02
EU-154	Copper -- Water Pipes	External	7.4E+03	1.3E+02
RA-226	Copper -- Frying Pan	Ingestion	2.0E+03	5.1E+02
TH-228	Copper -- Operator	Inhalation	1.9E+04	5.3E+01
TH-229	Copper -- Operator	Inhalation	1.0E+05	9.6E+00
TH-230	Copper -- Operator	Inhalation	1.6E+04	6.3E+01
TH-232	Copper -- Operator	Inhalation	6.7E+04	1.5E+01
U-232	Copper -- Operator	Inhalation	4.0E+04	2.5E+01
U-233	Copper -- Operator	Inhalation	7.8E+03	1.3E+02
U-234	Copper -- Operator	Inhalation	7.8E+03	1.3E+02
U-235	Copper -- Operator	Inhalation	7.2E+03	1.4E+02
U-238	Copper -- Operator	Inhalation	7.2E+03	1.4E+02
NP-237	Copper -- Operator	Inhalation	3.1E+04	3.2E+01
PU-238	Copper -- Operator	Inhalation	1.8E+04	5.5E+01
PU-239	Copper -- Operator	Inhalation	2.0E+04	5.0E+01

TABLE E.4. (Cont'd)

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/g)
PU-240	Copper -- Operator	Inhalation	2.0E+04	5.0E+01
Pu-241	Copper -- Operator	Inhalation	3.4E+02	2.9E+03
AM-241	Copper -- Operator	Inhalation	3.3E+04	3.0E+01
SR-90&D	Copper -- Frying Pan	Ingestion	2.5E+02	4.0E+03
RA-226&D	Copper -- Frying Pan	Ingestion	1.4E+04	7.1E+01
TH-232&D	Copper -- Operator	Inhalation	8.7E+04	1.1E+01
NP-237&D	Copper -- Operator	Inhalation	3.1E+04	3.2E+01
U-238&D	Copper -- Water Pipes	External	1.2E+04	8.2E+01

TABLE E.5. Limiting Scenarios and Corresponding Control Levels for the Recycled Concrete Category

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/g)
H-3	Concrete -- Worker 1	Ingestion	6.0E-03	1.7E+08
C-14	Concrete -- New Room	External	1.8E+00	5.6E+05
CL-36	Concrete -- New Room	External	6.0E+01	1.7E+04
CA-41	Concrete -- New Room	External	1.2E+02	8.5E+03
MN-54	Concrete -- New Room	External	1.7E+05	5.9E+00
FE-55	Concrete -- New Room	External	2.7E+02	3.7E+03
CO-57	Concrete -- New Room	External	1.3E+04	7.9E+01
CO-60	Concrete -- New Room	External	5.8E+05	1.7E+00
NI-63	Concrete -- Worker 1	Ingestion	1.8E-01	5.5E+06
ZN-65	Concrete -- New Room	External	1.5E+05	6.7E+00
SE-79	Concrete -- New Room	External	1.4E+00	7.2E+05
SR-90	Concrete -- New Room	External	3.0E+01	3.3E+04
ZR-93	Concrete -- Worker 1	Inhalation	7.1E+00	1.4E+05
NB-94	Concrete -- New Room	External	2.9E+05	3.5E+00
TC-99	Concrete -- New Room	External	7.0E+00	1.4E+05
RU-106	Concrete -- New Room	External	4.1E+04	2.4E+01
AG-110M	Concrete -- New Room	External	5.9E+05	1.7E+00
SB-125	Concrete -- New Room	External	8.0E+04	1.3E+01
I-129	Concrete -- New Room	External	1.9E+03	5.3E+02
CS-134	Concrete -- New Room	External	3.3E+05	3.1E+00
CS-137	Concrete -- New Room	External	1.0E+05	9.7E+00
CE-144	Concrete -- New Room	External	1.7E+03	6.0E+02
PM-147	Concrete -- New Room	External	4.0E+00	2.5E+05
SM-151	Concrete -- Worker 1	Inhalation	2.8E+00	3.6E+05
EU-152	Concrete -- New Room	External	2.6E+05	3.9E+00
EU-154	Concrete -- New Room	External	2.6E+05	3.8E+00
RA-226	Concrete -- Worker 1	Ingestion	7.6E+02	1.3E+03
TH-228	Concrete -- Worker 1	Inhalation	3.0E+04	3.4E+01
TH-229	Concrete -- Worker 1	Inhalation	1.6E+05	6.1E+00
TH-230	Concrete -- Worker 1	Inhalation	2.5E+04	4.0E+01
TH-232	Concrete -- Worker 1	Inhalation	1.1E+05	9.5E+00
U-232	Concrete -- Worker 1	Inhalation	6.4E+04	1.6E+01
U-233	Concrete -- Worker 1	Inhalation	1.2E+04	8.0E+01
U-234	Concrete -- Worker 1	Inhalation	1.2E+04	8.0E+01
U-235	Concrete -- New Room	External	1.2E+04	8.6E+01
U-238	Concrete -- Worker 1	Inhalation	1.2E+04	8.7E+01
NP-237	Concrete -- Worker 1	Inhalation	4.7E+04	2.1E+01
PU-238	Concrete -- Worker 1	Inhalation	2.9E+04	3.5E+01
PU-239	Concrete -- Worker 1	Inhalation	3.2E+04	3.2E+01

TABLE E.5. (Cont'd)

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/g)
PU-240	Concrete -- Worker 1	Inhalation	3.2E+04	3.2E+01
PU-241	Concrete -- Worker 1	Inhalation	5.5E+02	1.8E+03
AM-241	Concrete -- Worker 1	Inhalation	5.0E+04	2.0E+01
SR-90&D	Concrete -- New Room	External	1.1E+03	8.8E+02
RA-226&D	Concrete -- New Room	External	4.1E+05	2.5E+00
TH-232&D	Concrete -- New Room	External	6.0E+05	1.7E+00
NP-237&D	Concrete -- Worker 1	Inhalation	4.7E+04	2.1E+01
U-238&D	Concrete -- New Room	External	4.1E+05	2.4E+00

TABLE E.6. Limiting Scenarios and Corresponding Control Levels for Both Reuse Scenarios, Tool/Equipment Reuse and Building Reuse

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/cm^2)
H-3	Tools & Equip. -- Hand Tools	External	3.8E-01	2.6E+06
C-14	Tools & Equip. -- Hand Tools	External	1.3E+01	7.9E+04
CL-36	Tools & Equip. -- Hand Tools	External	2.0E+01	5.1E+04
CA-41	Building Reuse -- Renovation	Ingestion	2.3E+01	4.3E+04
MN-54	Building Reuse -- Renovation	External	1.1E+04	9.2E+01
FE-55	Building Reuse -- Renovation	Ingestion	5.3E+01	1.9E+04
CO-57	Building Reuse -- Renovation	External	1.7E+03	5.7E+02
CO-60	Building Reuse -- Renovation	External	3.1E+04	3.2E+01
NI-63	Tools & Equip. -- Hand Tools	External	3.4E+00	3.0E+05
ZN-65	Building Reuse -- Renovation	External	8.1E+03	1.2E+02
SE-79	Tools & Equip. -- Hand Tools	External	5.0E+01	2.0E+04
SR-90	Tools & Equip. -- Hand Tools	External	8.0E+02	1.3E+03
ZR-93	Tools & Equip. -- Hand Tools	External	1.5E+01	6.7E+04
NB-94	Building Reuse -- Renovation	External	1.9E+04	5.1E+01
TC-99	Tools & Equip. -- Hand Tools	External	8.4E+00	1.2E+05
RU-106	Building Reuse -- Renovation	Inhalation	2.9E+03	3.4E+02
AG-110M	Building Reuse -- Renovation	External	3.6E+04	2.8E+01
SB-125	Building Reuse -- Renovation	External	6.1E+03	1.7E+02
I-129	Tools & Equip. -- Hand Tools	External	1.7E+03	5.9E+02
CS-134	Building Reuse -- Renovation	External	2.2E+04	4.6E+01
CS-137	Building Reuse -- Renovation	External	7.5E+03	1.3E+02
CE-144	Building Reuse -- Renovation	Inhalation	2.5E+02	4.0E+03
PM-147	Tools & Equip. -- Hand Tools	External	8.2E+00	1.2E+05
SM-151	Tools & Equip. -- Hand Tools	External	4.1E+00	2.4E+05
EU-152	Building Reuse -- Renovation	External	1.5E+04	6.6E+01
EU-154	Building Reuse -- Renovation	External	1.6E+04	6.4E+01
RA-226	Tools & Equip. -- Hand Tools	External	7.2E+03	1.4E+02
TH-228	Tools & Equip. -- Hand Tools	External	2.5E+04	4.1E+01
TH-229	Tools & Equip. -- Hand Tools	External	1.4E+05	7.0E+00
TH-230	Tools & Equip. -- Hand Tools	External	2.2E+04	4.6E+01
TH-232	Tools & Equip. -- Hand Tools	External	9.6E+04	1.0E+01
U-232	Tools & Equip. -- Hand Tools	External	4.9E+04	2.1E+01
U-233	Tools & Equip. -- Hand Tools	External	9.5E+03	1.1E+02
U-234	Tools & Equip. -- Hand Tools	External	9.5E+03	1.1E+02
U-235	Tools & Equip. -- Hand Tools	External	8.8E+03	1.1E+02
U-238	Tools & Equip. -- Hand Tools	External	8.8E+03	1.1E+02
NP-237	Tools & Equip. -- Hand Tools	External	5.9E+04	1.7E+01
PU-238	Tools & Equip. -- Hand Tools	External	2.2E+04	4.6E+01
PU-239	Tools & Equip. -- Hand Tools	External	2.4E+04	4.1E+01

TABLE E.6. (Cont'd)

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/cm^2)
PU-240	Tools & Equip. -- Hand Tools	External	2.4E+04	4.1E+01
PU-241	Tools & Equip. -- Hand Tools	External	4.2E+02	2.4E+03
AM-241	Tools & Equip. -- Hand Tools	External	6.4E+04	1.6E+01
SR-90&D	Tools & Equip. -- Hand Tools	External	8.6E+02	1.2E+03
RA-226&D	Tools & Equip. -- Hand Tools	External	5.0E+04	2.0E+01
TH-232&D	Tools & Equip. -- Hand Tools	External	1.3E+05	7.6E+00
NP-237&D	Tools & Equip. -- Hand Tools	External	5.9E+04	1.7E+01
U-238&D	Building Reuse -- Renovation	Inhalation	2.6E+04	3.8E+01

TABLE E.7. Limiting Scenarios and Corresponding Control Levels for the Tool/Equipment Reuse Category

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/cm^2)
H-3	Tools & Equip. -- Hand Tools	Ingestion	3.8E-01	2.6E+06
C-14	Tools & Equip. -- Hand Tools	Ingestion	1.3E+01	7.9E+04
CL-36	Tools & Equip. -- Hand Tools	Ingestion	2.0E+01	5.1E+04
CA-41	Tools & Equip. -- Hand Tools	Ingestion	8.6E+00	1.2E+05
MN-54	Tools & Equip. -- Hand Tools	External	3.1E+02	3.3E+03
FE-55	Tools & Equip. -- Hand Tools	Ingestion	6.5E+00	1.5E+05
CO-57	Tools & Equip. -- Hand Tools	External	5.5E+01	1.8E+04
CO-60	Tools & Equip. -- Hand Tools	External	9.0E+02	1.1E+03
NI-63	Tools & Equip. -- Hand Tools	Ingestion	3.4E+00	3.0E+05
ZN-65	Tools & Equip. -- Hand Tools	External	3.0E+02	3.3E+03
SE-79	Tools & Equip. -- Hand Tools	Ingestion	5.0E+01	2.0E+04
SR-90	Tools & Equip. -- Hand Tools	Ingestion	8.0E+02	1.3E+03
ZR-93	Tools & Equip. -- Hand Tools	Inhalation	1.5E+01	6.7E+04
NB-94	Tools & Equip. -- Hand Tools	External	5.8E+02	1.7E+03
TC-99	Tools & Equip. -- Hand Tools	Ingestion	8.4E+00	1.2E+05
RU-106	Tools & Equip. -- Hand Tools	Inhalation	2.4E+02	4.2E+03
AG-110M	Tools & Equip. -- Hand Tools	External	1.0E+03	9.8E+02
SB-125	Tools & Equip. -- Hand Tools	External	1.8E+02	5.6E+03
I-129	Tools & Equip. -- Hand Tools	Ingestion	1.7E+03	5.9E+02
CS-134	Tools & Equip. -- Hand Tools	External	1.0E+03	9.7E+02
CS-137	Tools & Equip. -- Hand Tools	External	5.0E+02	2.0E+03
CE-144	Tools & Equip. -- Hand Tools	Inhalation	1.5E+02	6.6E+03
PM-147	Tools & Equip. -- Hand Tools	Inhalation	8.2E+00	1.2E+05
SM-151	Tools & Equip. -- Hand Tools	Inhalation	4.1E+00	2.4E+05
EU-152	Tools & Equip. -- Hand Tools	External	4.6E+02	2.2E+03
EU-154	Tools & Equip. -- Hand Tools	External	4.9E+02	2.0E+03
RA-226	Tools & Equip. -- Hand Tools	Ingestion	7.2E+03	1.4E+02
TH-228	Tools & Equip. -- Hand Tools	Inhalation	2.5E+04	4.1E+01
TH-229	Tools & Equip. -- Hand Tools	Inhalation	1.4E+05	7.0E+00
TH-230	Tools & Equip. -- Hand Tools	Inhalation	2.2E+04	4.6E+01
TH-232	Tools & Equip. -- Hand Tools	Inhalation	9.6E+04	1.0E+01
U-232	Tools & Equip. -- Hand Tools	Inhalation	4.9E+04	2.1E+01
U-233	Tools & Equip. -- Hand Tools	Inhalation	9.5E+03	1.1E+02
U-234	Tools & Equip. -- Hand Tools	Inhalation	9.5E+03	1.1E+02
U-235	Tools & Equip. -- Hand Tools	Inhalation	8.8E+03	1.1E+02
U-238	Tools & Equip. -- Hand Tools	Inhalation	8.8E+03	1.1E+02
NP-237	Tools & Equip. -- Hand Tools	Inhalation	5.9E+04	1.7E+01
PU-238	Tools & Equip. -- Hand Tools	Inhalation	2.2E+04	4.6E+01
PU-239	Tools & Equip. -- Hand Tools	Inhalation	2.4E+04	4.1E+01

TABLE E.7. (Cont'd)

Nuclide	Limiting Scenario	Limiting Exposure Pathway	*Total Dose* for Limiting Scenario (mrem)	*Corresponding* Radiological Control Level (pCi/cm^2)
PU-240	Tools & Equip. -- Hand Tools	Inhalation	2.4E+04	4.1E+01
PU-241	Tools & Equip. -- Hand Tools	Inhalation	4.2E+02	2.4E+03
AM-241	Tools & Equip. -- Hand Tools	Inhalation	6.4E+04	1.6E+01
SR-90&D	Tools & Equip. -- Hand Tools	Ingestion	8.6E+02	1.2E+03
RA-226&D	Tools & Equip. -- Hand Tools	Ingestion	5.0E+04	2.0E+01
TH-232&D	Tools & Equip. -- Hand Tools	Inhalation	1.3E+05	7.6E+00
NP-237&D	Tools & Equip. -- Hand Tools	Inhalation	5.9E+04	1.7E+01
U-238&D	Tools & Equip. -- Hand Tools	Inhalation	9.6E+03	1.0E+02

TABLE E.8. Limiting Scenarios and Corresponding Control Levels for the Building Reuse Category

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/cm^2)
H-3	Building Reuse -- Renovation	Ingestion	1.3E-02	7.8E+07
C-14	Building Reuse -- Renovation	Ingestion	7.9E-01	1.3E+06
CL-36	Building Reuse -- Renovation	Ingestion	1.0E+01	9.7E+04
CA-41	Building Reuse -- Renovation	Ingestion	2.3E+01	4.3E+04
MN-54	Building Reuse -- Renovation	External	1.1E+04	9.2E+01
FE-55	Building Reuse -- Renovation	Ingestion	5.3E+01	1.9E+04
CO-57	Building Reuse -- Renovation	External	1.7E+03	5.7E+02
CO-60	Building Reuse -- Renovation	External	3.1E+04	3.2E+01
NI-63	Building Reuse -- Renovation	Ingestion	1.4E-01	7.1E+06
ZN-65	Building Reuse -- Renovation	External	8.1E+03	1.2E+02
SE-79	Building Reuse -- Renovation	Ingestion	2.0E+00	5.1E+05
SR-90	Building Reuse -- Renovation	Ingestion	3.2E+01	3.1E+04
ZR-93	Building Reuse -- Renovation	Inhalation	5.2E-01	1.9E+06
NB-94	Building Reuse -- Renovation	External	1.9E+04	5.1E+01
TC-99	Building Reuse -- Renovation	Ingestion	1.7E+00	6.0E+05
RU-106	Building Reuse -- Renovation	Inhalation	2.9E+03	3.4E+02
AG-110M	Building Reuse -- Renovation	External	3.6E+04	2.8E+01
SB-125	Building Reuse -- Renovation	External	6.1E+03	1.7E+02
I-129	Building Reuse -- Renovation	Ingestion	4.6E+02	2.2E+03
CS-134	Building Reuse -- Renovation	External	2.2E+04	4.6E+01
CS-137	Building Reuse -- Renovation	External	7.5E+03	1.3E+02
CE-144	Building Reuse -- Renovation	Inhalation	2.5E+02	4.0E+03
PM-147	Building Reuse -- Renovation	Inhalation	1.1E+00	9.5E+05
SM-151	Building Reuse -- Renovation	Inhalation	5.3E-01	1.9E+06
EU-152	Building Reuse -- Renovation	External	1.5E+04	6.6E+01
EU-154	Building Reuse -- Renovation	External	1.6E+04	6.4E+01
RA-226	Building Reuse -- Renovation	Ingestion	3.0E+02	3.3E+03
TH-228	Building Reuse -- Renovation	Inhalation	8.6E+02	1.2E+03
TH-229	Building Reuse -- Renovation	Inhalation	5.9E+03	1.7E+02
TH-230	Building Reuse -- Renovation	Inhalation	7.5E+02	1.3E+03
TH-232	Building Reuse -- Renovation	Inhalation	3.2E+03	3.1E+02
U-232	Building Reuse -- Renovation	Inhalation	1.6E+03	6.1E+02
U-233	Building Reuse -- Renovation	Inhalation	3.3E+02	3.1E+03
U-234	Building Reuse -- Renovation	Inhalation	3.4E+02	3.0E+03
U-235	Building Reuse -- Renovation	Inhalation	1.8E+03	5.4E+02
U-238	Building Reuse -- Renovation	Inhalation	3.1E+02	3.2E+03
NP-237	Building Reuse -- Renovation	Inhalation	2.3E+03	4.3E+02
PU-238	Building Reuse -- Renovation	Inhalation	7.5E+02	1.3E+03
PU-239	Building Reuse -- Renovation	Inhalation	8.1E+02	1.2E+03
PU-240	Building Reuse -- Renovation	Inhalation	8.2E+02	1.2E+03

TABLE E.8. (Cont'd)

Nuclide	Limiting Scenario	Limiting Exposure Pathway	Total Dose for Limiting Scenario (mrem)	Corresponding Radiological Control Level (pCi/cm^2)
PU-241	Building Reuse -- Renovation	Inhalation	1.4E+01	7.2E+04
AM-241	Building Reuse -- Renovation	Inhalation	2.5E+03	4.1E+02
SR-90&D	Building Reuse -- Renovation	Ingestion	1.5E+02	6.7E+03
RA-226&D	Building Reuse -- Renovation	Ingestion	2.5E+04	4.1E+01
TH-232&D	Building Reuse -- Renovation	Inhalation	3.6E+04	2.8E+01
NP-237&D	Building Reuse -- Renovation	Inhalation	5.3E+03	1.9E+02
U-238&D	Building Reuse -- Renovation	Inhalation	2.6E+04	3.8E+01

APPENDIX F

Individual Doses from Unit Radionuclide Concentrations

APPENDIX F: INDIVIDUAL DOSES RESULTING FROM UNIT RADIONUCLIDE CONCENTRATIONS

Table F.1 presents the dose for each radionuclide that results when 1 μCi/g of bulk contamination or 1 μCi/cm^2 of surface contamination is assumed for each recycled material type or each reused material considered. As discussed in Section 3.5, these doses are used to determine the radiological control levels presented in Appendix E.

TABLE F.1. Individual Dose Corresponding to Unit Concentrations for Each Recycled Material Type or Each Reused Material Considered

	Limiting Dose (mrem/yr)					
	Bulk Contamination for Recycling at 1 μCi/g				Surface Contamination for Reuse at 1 μCi/cm²	
Nuclide	Steel	Aluminum	Copper	Concrete	Tools & Equip.	Building
^{3}H	1.9E-01	2.3E-01	2.3E-01	6.7E-03	7.6E-01	2.7E-01
^{14}C	5.8E+00	7.6E+00	7.6E+00	1.8E+00	2.5E+01	8.8E+00
^{36}Cl	---	---	---	6.0E+01	---	2.7E+01
^{41}Ca	---	---	---	1.2E+02	---	2.8E+01
^{54}Mn	7.9E+04	2.8E+04	5.2E+03	1.7E+05	3.2E+02	1.1E+04
^{55}Fe	6.6E+01	4.1E+01	4.7E+00	2.7E+02	1.0E+01	5.6E+01
^{57}Co	1.1E+04	4.2E+03	7.0E+02	1.3E+04	5.9E+01	1.7E+03
^{60}Co	2.3E+05	7.9E+04	1.5E+04	5.8E+05	9.6E+02	3.1E+04
^{63}Ni	2.1E+00	1.9E+00	1.9E+00	2.0E-01	6.6E+00	2.7E+00
^{65}Zn	5.8E+04	2.1E+04	3.9E+03	1.5E+05	3.9E+02	8.2E+03
^{79}Se	2.6E+01	3.0E+01	3.0E+01	1.4E+00	1.0E+02	3.6E+01
^{90}Sr	4.3E+02	4.7E+02	4.7E+02	3.0E+01	1.6E+03	5.9E+02
^{93}Zr	2.7E+01	6.5E+00	6.5E+00	7.8E+00	2.5E+01	2.6E+01
^{94}Nb	1.4E+05	5.0E+04	9.3E+03	2.9E+05	6.1E+02	2.0E+04
^{99}Tc	7.9E+00	4.7E+00	4.7E+00	7.0E+00	1.6E+01	8.6E+00
^{106}Ru	2.3E+04	7.5E+03	1.5E+03	4.1E+04	3.6E+02	3.1E+03
^{110m}Ag	2.6E+05	9.2E+04	1.7E+04	5.9E+05	1.1E+03	3.6E+04
^{125}Sb	4.2E+04	1.5E+04	2.8E+03	8.0E+04	2.0E+02	6.1E+03
^{129}I	8.3E+02	1.0E+03	1.0E+03	1.9E+03	3.4E+03	1.6E+03
^{134}Cs	1.6E+05	5.6E+04	1.0E+04	3.3E+05	1.5E+03	2.2E+04
^{137}Cs	5.4E+04	1.9E+04	3.5E+03	1.0E+05	8.0E+02	7.6E+03
^{144}Ce	1.4E+03	6.2E+02	8.4E+01	1.7E+03	2.7E+02	4.2E+02
^{147}Pm	1.3E+01	3.4E+00	3.4E+00	4.0E+00	1.4E+01	1.4E+01
^{151}Sm	9.7E+00	2.3E+00	2.3E+00	3.1E+00	6.2E+00	9.4E+00
^{152}Eu	1.1E+05	3.9E+04	7.1E+03	2.6E+05	4.9E+02	1.5E+04
^{154}Eu	1.1E+05	4.0E+04	7.4E+03	2.6E+05	5.5E+02	1.6E+04
^{226}Ra	5.4E+03	4.0E+03	4.0E+03	8.3E+02	1.4E+04	6.5E+03
^{228}Th	9.5E+04	2.1E+04	2.1E+04	3.3E+04	2.7E+04	8.3E+04
^{229}Th	5.2E+05	1.2E+05	1.2E+05	1.8E+05	1.6E+05	4.6E+05
^{230}Th	8.0E+04	1.8E+04	1.8E+04	2.7E+04	2.5E+04	7.1E+04
^{232}Th	3.4E+05	7.5E+04	7.5E+04	1.2E+05	1.1E+05	3.0E+05
^{232}U	2.0E+05	4.4E+04	4.4E+04	7.1E+04	4.9E+04	1.8E+05
^{233}U	3.9E+04	8.6E+03	8.6E+03	1.4E+04	9.7E+03	3.4E+04
^{234}U	3.9E+04	8.6E+03	8.6E+03	1.4E+04	9.7E+03	3.4E+04
^{235}U	3.6E+04	7.9E+03	7.9E+03	1.3E+04	9.0E+03	3.3E+04
^{238}U	3.6E+04	7.9E+03	7.9E+03	1.3E+04	8.9E+03	3.2E+04
^{237}Np	1.6E+05	3.6E+04	3.6E+04	5.2E+04	8.2E+04	1.5E+05
^{238}Pu	9.1E+04	2.0E+04	2.0E+04	3.2E+04	2.2E+04	7.9E+04
^{239}Pu	1.0E+05	2.2E+04	2.2E+04	3.5E+04	2.4E+04	8.7E+04
^{240}Pu	1.0E+05	2.2E+04	2.2E+04	3.5E+04	2.4E+04	8.7E+04
^{241}Pu	1.7E+03	3.8E+02	3.8E+02	6.0E+02	4.2E+02	1.5E+03
^{241}Am	1.7E+05	3.9E+04	3.9E+04	5.5E+04	9.1E+04	1.6E+05

TABLE F.1. (Cont'd)

Nuclide	Limiting Dose (mrem/yr)					
	Bulk Contamination for Recycling at 1 μCi/g				Surface Contamination for Reuse at 1 μCi/cm²	
	Steel	Aluminum	Copper	Concrete	Tools & Equip.	Building
$^{90}Sr+D^{(a)}$	1.6E+03	3.6E+02	2.5E+02	1.1E+03	8.6E+02	1.5E+02
$^{226}Ra+D^{(b)}$	1.7E+05	5.8E+04	1.4E+04	4.1E+05	5.0E+04	2.5E+04
$^{232}Th+D^{(c)}$	4.4E+05	8.7E+04	8.7E+04	6.0E+05	1.3E+05	3.6E+04
$^{237}Np+D^{(d)}$	1.6E+05	3.1E+04	3.1E+04	4.7E+04	5.9E+04	5.3E+03
$^{238}U+D^{(e)}$	1.9E+05	6.6E+04	1.2E+04	4.1E+05	9.6E+03	2.6E+04

(a) $^{90}Sr+D$ represents ^{90}Sr in equilibrium with its daughter product, ^{90}Y.
(b) $^{226}Ra+D$ represents ^{226}Ra in equilibrium with its daughter products, ^{222}Rn (with short-lived ^{218}Po, ^{218}At, ^{214}Pb, ^{214}Bi, ^{214}Po, and ^{210}Tl), ^{210}Pb, ^{210}Bi, and ^{210}Po.
(c) $^{232}Th+D$ represents ^{232}Th in equilibrium with its daughter products, ^{228}Ra, ^{228}Ac, ^{228}Th, ^{224}Ra (which includes short-lived ^{220}Rn and ^{216}Po), ^{212}Pb, ^{212}Bi, ^{212}Po, ^{208}Tl, and ^{208}Pb.
(d) $^{237}Np+D$ represents ^{237}Np in equilibrium with its daughter product, ^{233}Pa.
(e) $^{238}U+D$ represents ^{238}U in equilibrium with its daughter products, ^{234}Th, ^{234m}Pa, and ^{234}Pa.

APPENDIX G

Control Criteria Based On Public Doses

APPENDIX G: CONTROL CRITERIA BASED ON PUBLIC DOSE

The doses to the MEI were calculated using the CAP88-PC code, assuming a 1 Ci/yr emission from a smelter located in a medium-high population area (such as Chicago). The data are presented in Tables G.1 through G.5 for two assumed dose limits: 0.1 mrem/yr and 1.0 mrem/yr. For radionuclides that have decay chains, results of calculations are shown both with (+) and without consideration of the daughters. As discussed in Section 3.2, it is conservatively assumed that the radionuclide feedstock activity is equal to the stack emission activity. The specific activities of individual radionuclides in the feedstock were calculated assuming the smelter handled either 100 t/yr or 10 t/yr (where 1.016E+03 kg is assumed to equal 1 t).

TABLE G.1. Public-Based Radiological Control Levels for Various Radionuclides, Assuming a Metal Feedstock Volume of 100 t/yr[a]

Nuclide[b]	MEI mrem/yr	Ci/yr at 0.1 mrem/yr	Ci/yr at 1 mrem/yr	pCi/g[c] at 0.1 mrem/yr	pCi/g[c] at 1 mrem/yr
^{3}H	1.90E-05	5.26E+03	5.26E+04	5.18E+07	5.18E+08
^{14}C	8.77E-05	1.14E+03	1.14E+04	1.12E+07	1.12E+08
^{36}Cl[d]	---	---	---	---	---
^{41}Ca	6.57E-05	1.52E+03	1.52E+04	1.50E+07	1.50E+08
^{54}Mn	7.55E-02	1.32E+00	1.32E+01	1.30E+04	1.30E+05
^{55}Fe	2.63E-04	3.80E+02	3.80E+03	3.74E+06	3.74E+07
^{57}Co	1.20E-02	8.33E+00	8.33E+01	8.20E+04	8.20E+05
^{60}Co	1.13E+00	8.85E-02	8.85E-01	8.71E+02	8.71E+03
^{63}Ni	3.36E-04	2.98E+02	2.98E+03	2.93E+06	2.93E+07
^{65}Zn	4.24E-02	2.36E+00	2.36E+01	2.32E+04	2.32E+05
^{79}Se[e]	---	---	---	---	---
^{90}Sr	4.38E-02	2.28E+00	2.28E+01		2.25E+05
$^{90}Sr+Y$	4.51E-02	2.22E+00	2.22E+01	2.18E+04	2.18E+05
^{93}Zr	6.27E-03	1.59E+01	1.59E+02	1.57E+05	1.57E+06
^{94}Nb	5.09E+00	1.96E-02	1.96E-01	1.93E+02	1.93E+03
^{99}Tc	2.89E-03	3.46E+01	3.46E+02	3.41E+05	3.41E+06
^{106}Ru	6.29E-02	1.59E+00	1.59E+01	1.56E+04	1.56E+05
^{110m}Ag	2.01E-01	4.98E-01	4.98E+00	4.90E+03	4.90E+04
^{125}Sb	1.21E-01	8.26E-01	8.26E+00	8.13E+03	8.13E+04
^{129}I	1.03E+00	9.71E-02	9.71E-01	9.56E+02	9.56E+03
^{134}Cs	3.34E-01	2.99E-01	2.99E+00	2.95E+03	2.95E+04
^{137}Cs	7.15E-03	1.40E+01	1.40E+02	1.38E+05	1.38E+06
$^{137}Cs+$	1.01E+00	9.90E-02	9.90E-01	9.75E+02	9.75E+03
^{144}Ce	5.10E-02	1.96E+00	1.96E+01	1.93E+04	1.93E+05
^{147}Pm	5.10E-03	1.96E+01	1.96E+02	1.93E+05	1.93E+06
^{151}Sm	3.95E-03	2.53E+01	2.53E+02	2.49E+05	2.49E+06
^{152}Eu	1.18E+00	8.47E-02	8.47E-01	8.34E+02	8.34E+03
^{154}Eu	9.35E-01	1.07E-01	1.07E+00	1.05E+03	1.05E+04
^{226}Ra	1.23E+00	8.13E-02	8.13E-01	8.00E+02	8.00E+03
^{228}Th	3.30E+01	3.03E-03	3.03E-02	2.98E+01	2.98E+02
^{229}Th	9.18E+01	1.09E-03	1.09E-02	1.07E+01	1.07E+02
^{230}Th	3.27E+01	3.06E-03	3.06E-02	3.01E+01	3.01E+02
^{232}Th	4.71E+01	2.12E-03	2.12E-02	2.09E+01	2.09E+02
$^{232}Th+$	8.84E+01	1.13E-03	1.13E-02	1.11E+01	1.11E+02
^{232}U	6.24E+01	1.60E-03	1.60E-02	1.58E+01	1.58E+02
^{233}U	1.74E+01	5.75E-03	5.75E-02	5.66E+01	5.66E+02
^{234}U	1.72E+01	5.81E-03	5.81E-02	5.72E+01	5.72E+02
^{235}U	1.64E+01	6.10E-03	6.10E-02	6.00E+01	6.00E+02
^{238}U	1.53E+01	6.54E-03	6.54E-02	6.43E+01	6.43E+02
$^{238}U+$	7.85E+01	1.27E-03	1.27E-02	1.25E+01	1.25E+02

TABLE G.1. (Cont'd)

Nuclide[b]	MEI mrem/yr	Ci/yr at 0.1 mrem/yr	Ci/yr at 1 mrem/yr	pCi/g[c] at 0.1 mrem/yr	pCi/g[c] at 1 mrem/yr
^{237}Np	6.39E+01	1.56E-03	1.56E-02	1.54E+01	1.54E+02
^{238}Pu	4.13E+01	2.42E-03	2.42E-02	2.38E+01	2.38E+02
^{239}Pu	4.44E+01	2.25E-03	2.25E-02	2.22E+01	2.22E+02
^{240}Pu	4.44E+01	2.25E-03	2.25E-02	2.22E+01	2.22E+02
^{241}Pu	6.77E-01	1.48E-01	1.48E+00	1.45E+03	1.45E+04
^{241}Am	7.06E+01	1.42E-03	1.42E-02	1.39E+01	1.39E+02

(a) The doses to the maximally exposed individual (MEI) were calculated using CAP88PC, assuming a 1 Ci/yr emission from a smelter located in a medium-high population area (such as Chicago). The radionuclide feedstock activity, 100% from DOE scrap, is conservatively assumed to be equal to the stack emission activity. See Section 3.2 for further discussion.
(b) For radionuclides having decay chains, results of calculations are shown both with (+) and without consideration of the daughters.
(c) The specific activities of individual radionuclides in the feedstock were calculated assuming the smelter handled 100 t/yr (where 1.016E+03 kg is assumed to equal 1 t).
(d) The radionuclide ^{36}Cl is not available in the CAP88-PC code.
(e) Although this is listed as a valid radionuclide in the CAP88-PC code, no dose or risk factors are available.

TABLE G.2. Public-Based Radiological Control Levels for Various Radionuclides, Assuming a Metal Feedstock Volume of 10 t/yr[a]

Nuclide[b]	MEI mrem/yr	Ci/yr at 0.1 mrem/yr	Ci/yr at 1 mrem/yr	pCi/g[c] at 0.1 mrem/yr	pCi/g[c] at 1 mrem/yr
^{3}H	1.90E-05	5.26E+03	5.26E+04	5.18E+08	5.18E+09
^{14}C	8.77E-05	1.14E+03	1.14E+04	1.12E+08	1.12E+09
^{36}Cl[d]	---	---	---	---	---
^{41}Ca	6.57E-05	1.52E+03	1.52E+04	1.50E+08	1.50E+09
^{54}Mn	7.55E-02	1.32E+00	1.32E+01	1.30E+05	1.30E+06
^{55}Fe	2.63E-04	3.80E+02	3.80E+03	3.74E+07	3.74E+08
^{57}Co	1.20E-02	8.33E+00	8.33E+01	8.20E+05	8.20E+06
^{60}Co	1.13E+00	8.85E-02	8.85E-01	8.71E+03	8.71E+04
^{63}Ni	3.36E-04	2.98E+02	2.98E+03	2.93E+07	2.93E+08
^{65}Zn	4.24E-02	2.36E+00	2.36E+01	2.32E+05	2.32E+06
^{79}Se[e]	---	---	---	---	---
^{90}Sr	4.38E-02	2.28E+00	2.28E+01	2.25E+05	2.25E+06
$^{90}Sr+Y$	4.51E-02	2.22E+00	2.22E+01	2.18E+05	2.18E+06
^{93}Zr	6.27E-03	1.59E+01	1.59E+02	1.57E+06	1.57E+07
^{94}Nb	5.09E+00	1.96E-02	1.96E-01	1.93E+03	1.93E+04
^{99}Tc	2.89E-03	3.46E+01	3.46E+02	3.41E+06	3.41E+07
^{106}Ru	6.29E-02	1.59E+00	1.59E+01	1.56E+05	1.56E+06
^{110m}Ag	2.01E-01	4.98E-01	4.98E+00	4.90E+04	4.90E+05
^{125}Sb	1.21E-01	8.26E-01	8.26E+00	8.13E+04	8.13E+05
^{129}I	1.03E+00	9.71E-02	9.71E-01	9.56E+03	9.56E+04
^{134}Cs	3.34E-01	2.99E-01	2.99E+00	2.95E+04	2.95E+05
^{137}Cs	7.15E-03	1.40E+01	1.40E+02	1.38E+06	1.38E+07
$^{137}Cs+$	1.01E+00	9.90E-02	9.90E-01	9.75E+03	9.75E+04
^{144}Ce	5.10E-02	1.96E+00	1.96E+01	1.93E+05	1.93E+06
^{147}Pm	5.10E-03	1.96E+01	1.96E+02	1.93E+06	1.93E+07
^{151}Sm	3.95E-03	2.53E+01	2.53E+02	2.49E+06	2.49E+07
^{152}Eu	1.18E+00	8.47E-02	8.47E-01	8.34E+03	8.34E+04
^{154}Eu	9.35E-01	1.07E-01	1.07E+00	1.05E+04	1.05E+05
^{226}Ra	1.23E+00	8.13E-02	8.13E-01	8.00E+03	8.00E+04
^{228}Th	3.30E+01	3.03E-03	3.03E-02	2.98E+02	2.98E+03
^{229}Th	9.18E+01	1.09E-03	1.09E-02	1.07E+02	1.07E+03
^{230}Th	3.27E+01	3.06E-03	3.06E-02	3.01E+02	3.01E+03
^{232}Th	4.71E+01	2.12E-03	2.12E-02	2.09E+02	2.09E+03
$^{232}Th+$	8.84E+01	1.13E-03	1.13E-02	1.11E+02	1.11E+03
^{232}U	6.24E+01	1.60E-03	1.60E-02	1.58E+02	1.58E+03
^{233}U	1.74E+01	5.75E-03	5.75E-02	5.66E+02	5.66E+03
^{234}U	1.72E+01	5.81E-03	5.81E-02	5.72E+02	5.72E+03
^{235}U	1.64E+01	6.10E-03	6.10E-02	6.00E+02	6.00E+03
^{238}U	1.53E+01	6.54E-03	6.54E-02	6.43E+02	6.43E+03
$^{238}U+$	7.85E+01	1.27E-03	1.27E-02	1.25E+02	1.25E+03
^{237}Np	6.39E+01	1.56E-03	1.56E-02	1.54E+02	1.54E+03

TABLE G.2. (Cont'd)

Nuclide[b]	MEI mrem/yr	Ci/yr at 0.1 mrem/yr	Ci/yr at 1 mrem/yr	pCi/g[c] at 0.1 mrem/yr	pCi/g[c] at 1 mrem/yr
^{238}Pu	4.13E+01	2.42E-03	2.42E-02	2.38E+02	2.38E+03
^{239}Pu	4.44E+01	2.25E-03	2.25E-02	2.22E+02	2.22E+03
^{240}Pu	4.44E+01	2.25E-03	2.25E-02	2.22E+02	2.22E+03
^{241}Pu	6.77E-01	1.48E-01	1.48E+00	1.45E+04	1.45E+05
^{241}Am	7.06E+01	1.42E-03	1.42E-02	1.39E+02	1.39E+03

(a) The doses to the maximally exposed individual (MEI) were calculated using CAP88PC, assuming a 1 Ci/yr emission from a smelter located in a medium-high population area (such as Chicago). The radionuclide feedstock, 100% from DOE scrap, is conservatively assumed to be equal to the stack emission. See Section 3.2 for further discussion.
(b) For radionuclides having decay chains, results of calculations are shown both with (+) and without consideration of the daughters.
(c) The specific activities of individual radionuclides in the feedstock were calculated assuming the smelter handled 10 t/yr (where 1.016E+03 kg is assumed to equal 1 t).
(d) The radionuclide ^{36}Cl is not available in the CAP88-PC code.
(e) Although this is listed as a valid radionuclide in the CAP88-PC code, no dose or risk factors are available.

TABLE G.3. Radiological Control Levels (pCi/g) for Various Percentages of DOE Scrap in the Metal Feedstock Volume of 100 t/yr, Assuming a Public Dose of 0.1 mrem/yr[a]

			Percentage of DOE Scrap				
Nuclide[b]	MEI mrem/yr	Ci/yr	100%	50%	10%	5%	1%
^{3}H	1.90E-05	5.26E+04	5.18E+07	1.04E+08	5.18E+08	1.04E+09	5.18E+09
^{14}C	8.77E-05	1.14E+04	1.12E+07	2.24E+07	1.12E+08	2.24E+08	1.12E+09
^{36}Cl[c]	---	---	---	---	---	---	---
^{41}Ca	6.57E-05	1.52E+04	1.50E+07	3.00E+07	1.50E+08	3.00E+08	1.50E+09
^{54}Mn	7.55E-02	1.32E+01	1.30E+04	2.61E+04	1.30E+05	2.61E+05	1.30E+06
^{55}Fe	2.63E-04	3.80E+03	3.74E+06	7.48E+06	3.74E+07	7.48E+07	3.74E+08
^{57}Co	1.20E-02	8.33E+01	8.20E+04	1.64E+05	8.20E+05	1.64E+06	8.20E+06
^{60}Co	1.13E+00	8.85E-01	8.71E+02	1.74E+03	8.71E+03	1.74E+04	8.71E+04
^{63}Ni	3.36E-04	2.98E+03	2.93E+06	5.86E+06	2.93E+07	5.86E+07	2.93E+08
^{65}Zn	4.24E-02	2.36E+01	2.32E+04	4.64E+04	2.32E+05	4.64E+05	2.32E+06
^{79}Se[d]	---	---	---	---	---	---	---
^{90}Sr	4.38E-02	2.28E+01	2.25E+04	4.49E+04	2.25E+05	4.49E+05	2.25E+06
$^{90}Sr+Y$	4.51E-02	2.22E+01	2.18E+04	4.36E+04	2.18E+05	4.36E+05	2.18E+06
^{93}Zr	6.27E-03	1.59E+02	1.57E+05	3.14E+05	1.57E+06	3.14E+06	1.57E+07
^{94}Nb	5.09E+00	1.96E-01	1.93E+02	3.87E+02	1.93E+03	3.87E+03	1.93E+04
^{99}Tc	2.89E-03	3.46E+02	3.41E+05	6.81E+05	3.41E+06	6.81E+06	3.41E+07
^{106}Ru	6.29E-02	1.59E+01	1.56E+04	3.13E+04	1.56E+05	3.13E+05	1.56E+06
^{110m}Ag	2.01E-01	4.98E+00	4.90E+03	9.79E+03	4.90E+04	9.79E+04	4.90E+05
^{125}Sb	1.21E-01	8.26E+00	8.13E+03	1.63E+04	8.13E+04	1.63E+05	8.13E+05
^{129}I	1.03E+00	9.71E-01	9.56E+02	1.91E+03	9.56E+03	1.91E+04	9.56E+04
^{134}Cs	3.34E-01	2.99E+00	2.95E+03	5.89E+03	2.95E+04	5.89E+04	2.95E+05
^{137}Cs	7.15E-03	1.40E+02	1.38E+05	2.75E+05	1.38E+06	2.75E+06	1.38E+07
$^{137}Cs+$	1.01E+00	9.90E-01	9.75E+02	1.95E+03	9.75E+03	1.95E+04	9.75E+04
^{144}Ce	5.10E-02	1.96E+01	1.93E+04	3.86E+04	1.93E+05	3.86E+05	1.93E+06
^{147}Pm	5.10E-03	1.96E+02	1.93E+05	3.86E+05	1.93E+06	3.86E+06	1.93E+07
^{151}Sm	3.95E-03	2.53E+02	2.49E+05	4.98E+05	2.49E+06	4.98E+06	2.49E+07
^{152}Eu	1.18E+00	8.47E-01	8.34E+02	1.67E+03	8.34E+03	1.67E+04	8.34E+04
^{154}Eu	9.35E-01	1.07E+00	1.05E+03	2.11E+03	1.05E+04	2.11E+04	1.05E+05
^{226}Ra	1.23E+00	8.13E-01	8.00E+02	1.60E+03	8.00E+03	1.60E+04	8.00E+04
^{228}Th	3.30E+01	3.03E-02	2.98E+01	5.97E+01	2.98E+02	5.97E+02	2.98E+03
^{229}Th	9.18E+01	1.09E-02	1.07E+01	2.14E+01	1.07E+02	2.14E+02	1.07E+03
^{230}Th	3.27E+01	3.06E-02	3.01E+01	6.02E+01	3.01E+02	6.02E+02	3.01E+03
^{232}Th	4.71E+01	2.12E-02	2.09E+01	4.18E+01	2.09E+02	4.18E+02	2.09E+03
$^{232}Th+$	8.84E+01	1.13E-02	1.11E+01	2.23E+01	1.11E+02	2.23E+02	1.11E+03
^{232}U	6.24E+01	1.60E-02	1.58E+01	3.15E+01	1.58E+02	3.15E+02	1.58E+03
^{233}U	1.74E+01	5.75E-02	5.66E+01	1.13E+02	5.66E+02	1.13E+03	5.66E+03
^{234}U	1.72E+01	5.81E-02	5.72E+01	1.14E+02	5.72E+02	1.14E+03	5.72E+03
^{235}U	1.64E+01	6.10E-02	6.00E+01	1.20E+02	6.00E+02	1.20E+03	6.00E+03
^{238}U	1.53E+01	6.54E-02	6.43E+01	1.29E+02	6.43E+02	1.29E+03	6.43E+03
$^{238}U+$	7.85E+01	1.27E-02	1.25E+01	2.51E+01	1.25E+02	2.51E+02	1.25E+03

TABLE G.3. (Cont'd)

Nuclide[b]	MEI mrem/yr	Ci/yr	Percentage of DOE Scrap				
			100%	50%	10%	5%	1%
^{237}Np	6.39E+01	1.56E-02	1.54E+01	3.08E+01	1.54E+02	3.08E+02	1.54E+03
^{238}Pu	4.13E+01	2.42E-02	2.38E+01	4.77E+01	2.38E+02	4.77E+02	2.38E+03
^{239}Pu	4.44E+01	2.25E-02	2.22E+01	4.43E+01	2.22E+02	4.43E+02	2.22E+03
^{240}Pu	4.44E+01	2.25E-02	2.22E+01	4.43E+01	2.22E+02	4.43E+02	2.22E+03
^{241}Pu	6.77E-01	1.48E+00	1.45E+03	2.91E+03	1.45E+04	2.91E+04	1.45E+05
^{241}Am	7.06E+01	1.42E-02	1.39E+01	2.79E+01	1.39E+02	2.79E+02	1.39E+03

(a) The doses to the maximally exposed individual (MEI) were calculated using CAP88PC, assuming a 1 Ci/yr emission from a smelter located in a medium-high population area (such as Chicago). The radionuclide feedstock activity is conservatively assumed to be equal to the stack emission activity. See Section 3.2 for further discussion. The specific activities of individual radionuclides in the feedstock were calculated assuming the smelter handled 100 t/yr (where 1.016E+03 kg is assumed to equal 1 t).
(b) For radionuclides having decay chains, results of calculations are shown both with (+) and without consideration of the daughters.
(c) The radionuclide ^{36}Cl is not available in the CAP88-PC code.
(d) Although this is listed as a valid radionuclide in the CAP88-PC code, no dose or risk factors are available.

TABLE G.4. Radiological Control Levels (pCi/g) for Various Percentages of DOE Scrap in the Metal Feedstock Volume of 100 t/yr, Assuming a Public Dose of 1.0 mrem/yr[a]

Nuclide[b]	MEI mrem/yr	Ci/yr	Percentage of DOE Scrap				
			100%	50%	10%	5%	1%
^{3}H	1.90E-05	5.26E+04	5.18E+08	1.04E+09	5.18E+09	1.04E+10	5.18E+10
^{14}C	8.77E-05	1.14E+04	1.12E+08	2.24E+08	1.12E+09	2.24E+09	1.12E+10
^{36}Cl[c]	---	---	---	---	---	---	---
^{41}Ca	6.57E-05	1.52E+04	1.50E+08	3.00E+08	1.50E+09	3.00E+09	1.50E+10
^{54}Mn	7.55E-02	1.32E+01	1.30E+05	2.61E+05	1.30E+06	2.61E+06	1.30E+07
^{55}Fe	2.63E-04	3.80E+03	3.74E+07	7.48E+07	3.74E+08	7.48E+08	3.74E+09
^{57}Co	1.20E-02	8.33E+01	8.20E+05	1.64E+06	8.20E+06	1.64E+07	8.20E+07
^{60}Co	1.13E+00	8.85E-01	8.71E+03	1.74E+04	8.71E+04	1.74E+05	8.71E+05
^{63}Ni	3.36E-04	2.98E+03	2.93E+07	5.86E+07	2.93E+08	5.86E+08	2.93E+09
^{65}Zn	4.24E-02	2.36E+01	2.32E+05	4.64E+05	2.32E+06	4.64E+06	2.32E+07
^{79}Se[d]	---	---	---	---	---	---	---
^{90}Sr	4.38E-02	2.28E+01	2.25E+05	4.49E+05	2.25E+06	4.49E+06	2.25E+07
$^{90}Sr+Y$	4.51E-02	2.22E+01	2.18E+05	4.36E+05	2.18E+06	4.36E+06	2.18E+07
^{93}Zr	6.27E-03	1.59E+02	1.57E+06	3.14E+06	1.57E+07	3.14E+07	1.57E+08
^{94}Nb	5.09E+00	1.96E-01	1.93E+03	3.87E+03	1.93E+04	3.87E+04	1.93E+05
^{99}Tc	2.89E-03	3.46E+02	3.41E+06	6.81E+06	3.41E+07	6.81E+07	3.41E+08
^{106}Ru	6.29E-02	1.59E+01	1.56E+05	3.13E+05	1.56E+06	3.13E+06	1.56E+07
^{110m}Ag	2.01E-01	4.98E+00	4.90E+04	9.79E+04	4.90E+05	9.79E+05	4.90E+06
^{125}Sb	1.21E-01	8.26E+00	8.13E+04	1.63E+05	8.13E+05	1.63E+06	8.13E+06
^{129}I	1.03E+00	9.71E-01	9.56E+03	1.91E+04	9.56E+04	1.91E+05	9.56E+05
^{134}Cs	3.34E-01	2.99E+00	2.95E+04	5.89E+04	2.95E+05	5.89E+05	2.95E+06
^{137}Cs	7.15E-03	1.40E+02	1.38E+06	2.75E+06	1.38E+07	2.75E+07	1.38E+08
$^{137}Cs+$	1.01E+00	9.90E-01	9.75E+03	1.95E+04	9.75E+04	1.95E+05	9.75E+05
^{144}Ce	5.10E-02	1.96E+01	1.93E+05	3.86E+05	1.93E+06	3.86E+06	1.93E+07
^{147}Pm	5.10E-03	1.96E+02	1.93E+06	3.86E+06	1.93E+07	3.86E+07	1.93E+08
^{151}Sm	3.95E-03	2.53E+02	2.49E+06	4.98E+06	2.49E+07	4.98E+07	2.49E+08
^{152}Eu	1.18E+00	8.47E-01	8.34E+03	1.67E+04	8.34E+04	1.67E+05	8.34E+05
^{154}Eu	9.35E-01	1.07E+00	1.05E+04	2.11E+04	1.05E+05	2.11E+05	1.05E+06
^{226}Ra	1.23E+00	8.13E-01	8.00E+03	1.60E+04	8.00E+04	1.60E+05	8.00E+05
^{228}Th	3.30E+01	3.03E-02	2.98E+02	5.97E+02	2.98E+03	5.97E+03	2.98E+04
^{229}Th	9.18E+01	1.09E-02	1.07E+02	2.14E+02	1.07E+03	2.14E+03	1.07E+04
^{230}Th	3.27E+01	3.06E-02	3.01E+02	6.02E+02	3.01E+03	6.02E+03	3.01E+04
^{232}Th	4.71E+01	2.12E-02	2.09E+02	4.18E+02	2.09E+03	4.18E+03	2.09E+04
$^{232}Th+$	8.84E+01	1.13E-02	1.11E+02	2.23E+02	1.11E+03	2.23E+03	1.11E+04
^{232}U	6.24E+01	1.60E-02	1.58E+02	3.15E+02	1.58E+03	3.15E+03	1.58E+04
^{233}U	1.74E+01	5.75E-02	5.66E+02	1.13E+03	5.66E+03	1.13E+04	5.66E+04
^{234}U	1.72E+01	5.81E-02	5.72E+02	1.14E+03	5.72E+03	1.14E+04	5.72E+04
^{235}U	1.64E+01	6.10E-02	6.00E+02	1.20E+03	6.00E+03	1.20E+04	6.00E+04
^{238}U	1.53E+01	6.54E-02	6.43E+02	1.29E+03	6.43E+03	1.29E+04	6.43E+04
$^{238}U+$	7.85E+01	1.27E-02	1.25E+02	2.51E+02	1.25E+03	2.51E+03	1.25E+04

TABLE G.4. (Cont'd)

Nuclide[b]	MEI mrem/yr	Ci/yr	Percentage of DOE Scrap				
			100%	50%	10%	5%	1%
^{237}Np	6.39E+01	1.56E-02	1.54E+02	3.08E+02	1.54E+03	3.08E+03	1.54E+04
^{238}Pu	4.13E+01	2.42E-02	2.38E+02	4.77E+02	2.38E+03	4.77E+03	2.38E+04
^{239}Pu	4.44E+01	2.25E-02	2.22E+02	4.43E+02	2.22E+03	4.43E+03	2.22E+04
^{240}Pu	4.44E+01	2.25E-02	2.22E+02	4.43E+02	2.22E+03	4.43E+03	2.22E+04
^{241}Pu	6.77E-01	1.48E+00	1.45E+04	2.91E+04	1.45E+05	2.91E+05	1.45E+06
^{241}Am	7.06E+01	1.42E-02	1.39E+02	2.79E+03	1.39E+03	2.79E+03	1.39E+04

(a) The doses to the maximally exposed individual (MEI) were calculated using CAP88PC, assuming a 1 Ci/yr emission from a smelter located in a medium-high population area (such as Chicago). As a conservative assumption, the radionuclide feedstock activity is assumed to be equal to the stack emission activity. See Section 3.2 for further discussion. The specific activities of individual radionuclides in the feedstock were calculated assuming the smelter handled 100 t/yr (where 1.016E+03 kg is assumed to equal 1 t).
(b) For radionuclides having decay chains, results of calculations are shown both with (+) and without consideration of the daughters.
(c) The radionuclide ^{36}Cl is not available in the CAP88-PC code.
(d) Although this is listed as a valid radionuclide in the CAP88-PC code, no dose or risk factors are available.

TABLE G.5. Radiological Control Levels (pCi/g) for Various Percentages of DOE Scrap in the Metal Feedstock Volume of 10 t/yr, Assuming a Public Dose of 0.1 mrem/yr[a]

			Percentage of DOE Scrap				
Nuclide[b]	MEI mrem/yr	Ci/yr	100%	50%	10%	5%	1%
^{3}H	1.90E-05	5.26E+04	5.18E+08	1.04E+09	5.18E+09	1.04E+10	5.18E+10
^{14}C	8.77E-05	1.14E+04	1.12E+08	2.24E+08	1.12E+09	2.24E+09	1.12E+10
^{36}Cl[c]	---	---	---	---	---	---	---
^{41}Ca	6.57E-05	1.52E+04	1.50E+08	3.00E+08	1.50E+09	3.00E+09	1.50E+10
^{54}Mn	7.55E-02	1.32E+01	1.30E+05	2.61E+05	1.30E+06	2.61E+06	1.30E+07
^{55}Fe	2.63E-04	3.80E+03	3.74E+07	7.48E+07	3.74E+08	7.48E+08	3.74E+09
^{57}Co	1.20E-02	8.33E+01	8.20E+05	1.64E+06	8.20E+06	1.64E+07	8.20E+07
^{60}Co	1.13E+00	8.85E-01	8.71E+03	1.74E+04	8.71E+04	1.74E+05	8.71E+05
^{63}Ni	3.36E-04	2.98E+03	2.93E+07	5.86E+07	2.93E+08	5.86E+08	2.93E+09
^{65}Zn	4.24E-02	2.36E+01	2.32E+05	4.64E+05	2.32E+06	4.64E+06	2.32E+07
^{79}Se[d]	---	---	---	---	---	---	---
^{90}Sr	4.38E-02	2.28E+01	2.25E+05	4.49E+05	2.25E+06	4.49E+06	2.25E+07
$^{90}Sr+Y$	4.51E-02	2.22E+01	2.18E+05	4.36E+05	2.18E+06	4.36E+06	2.18E+07
^{93}Zr	6.27E-03	1.59E+02	1.57E+06	3.14E+06	1.57E+07	3.14E+07	1.57E+08
^{94}Nb	5.09E+00	1.96E-01	1.93E+03	3.87E+03	1.93E+04	3.87E+04	1.93E+05
^{99}Tc	2.89E-03	3.46E+02	3.41E+06	6.81E+06	3.41E+07	6.81E+07	3.41E+08
^{106}Ru	6.29E-02	1.59E+01	1.56E+05	3.13E+05	1.56E+06	3.13E+06	1.56E+07
^{110m}Ag	2.01E-01	4.98E+00	4.90E+04	9.79E+04	4.90E+05	9.79E+05	4.90E+06
^{125}Sb	1.21E-01	8.26E+00	8.13E+04	1.63E+05	8.13E+05	1.63E+06	8.13E+06
^{129}I	1.03E+00	9.71E-01	9.56E+03	1.91E+04	9.56E+04	1.91E+05	9.56E+05
^{134}Cs	3.34E-01	2.99E+00	2.95E+04	5.89E+04	2.95E+05	5.89E+05	2.95E+06
^{137}Cs	7.15E-03	1.40E+02	1.38E+06	2.75E+06	1.38E+07	2.75E+07	1.38E+08
$^{137}Cs+$	1.01E+00	9.90E-01	9.75E+03	1.95E+04	9.75E+04	1.95E+05	9.75E+05
^{144}Ce	5.10E-02	1.96E+01	1.93E+05	3.86E+05	1.93E+06	3.86E+06	1.93E+07
^{147}Pm	5.10E-03	1.96E+02	1.93E+06	3.86E+06	1.93E+07	3.86E+07	1.93E+08
^{151}Sm	3.95E-03	2.53E+02	2.49E+06	4.98E+06	2.49E+07	4.98E+07	2.49E+08
^{152}Eu	1.18E+00	8.47E-01	8.34E+03	1.67E+04	8.34E+04	1.67E+05	8.34E+05
^{154}Eu	9.35E-01	1.07E+00	1.05E+04	2.11E+04	1.05E+05	2.11E+05	1.05E+06
^{226}Ra	1.23E+00	8.13E-01	8.00E+03	1.60E+04	8.00E+04	1.60E+05	8.00E+05
^{228}Th	3.30E+01	3.03E-02	2.98E+02	5.97E+02	2.98E+03	5.97E+03	2.98E+04
^{229}Th	9.18E+01	1.09E-02	1.07E+02	2.14E+02	1.07E+03	2.14E+03	1.07E+04
^{230}Th	3.27E+01	3.06E-02	3.01E+02	6.02E+02	3.01E+03	6.02E+03	3.01E+04
^{232}Th	4.71E+01	2.12E-02	2.09E+02	4.18E+02	2.09E+03	4.18E+03	2.09E+04
$^{232}Th+$	8.84E+01	1.13E-02	1.11E+02	2.23E+02	1.11E+03	2.23E+03	1.11E+04
^{232}U	6.24E+01	1.60E-02	1.58E+02	3.15E+02	1.58E+03	3.15E+03	1.58E+04
^{233}U	1.74E+01	5.75E-02	5.66E+02	1.13E+03	5.66E+03	1.13E+04	5.66E+04
^{234}U	1.72E+01	5.81E-02	5.72E+02	1.14E+03	5.72E+03	1.14E+04	5.72E+04
^{235}U	1.64E+01	6.10E-02	6.00E+02	1.20E+03	6.00E+03	1.20E+04	6.00E+04
^{238}U	1.53E+01	6.54E-02	6.43E+02	1.29E+03	6.43E+03	1.29E+04	6.43E+04
$^{238}U+$	7.85E+01	1.27E-02	1.25E+02	2.51E+02	1.25E+03	2.51E+03	1.25E+04

TABLE G.5. (Cont'd)

Nuclide[b]	MEI mrem/yr	Ci/yr	Percentage of DOE Scrap				
			100%	50%	10%	5%	1%
^{237}Np	6.39E+01	1.56E-02	1.54E+02	3.08E+02	1.54E+03	3.08E+03	1.54E+04
^{238}Pu	4.13E+01	2.42E-02	2.38E+02	4.77E+02	2.38E+03	4.77E+03	2.38E+04
^{239}Pu	4.44E+01	2.25E-02	2.22E+02	4.43E+02	2.22E+03	4.43E+03	2.22E+04
^{240}Pu	4.44E+01	2.25E-02	2.22E+02	4.43E+02	2.22E+03	4.43E+03	2.22E+04
^{241}Pu	6.77E-01	1.48E+00	1.45E+04	2.91E+04	1.45E+05	2.91E+05	1.45E+06
^{241}Am	7.06E+01	1.42E-02	1.39E+02	2.79E+03	1.39E+03	2.79E+03	1.39E+04

(a) The doses to the maximally exposed individual (MEI) were calculated using CAP88PC, assuming a 1 Ci/yr emission from a smelter located in a medium-high population area (such as Chicago). The radionuclide feedstock activity is conservatively assumed to be equal to the stack emission activity. See Section 3.2 for further discussion. The specific activities of individual radionuclides in the feedstock were calculated assuming the smelter handled 10 t/yr (where 1.016E+03 kg is assumed to equal 1 t).
(b) For radionuclides having decay chains, results of calculations are shown both with (+) and without consideration of the daughters.
(c) The radionuclide ^{36}Cl is not available in the CAP88-PC code.
(d) Although this is listed as a valid radionuclide in the CAP88-PC code, no dose or risk factors are available.

TABLE G.6. Radiological Control Levels (pCi/g) for Various Percentages of DOE Scrap in the Metal Feedstock Volume of 10 t/yr, Assuming a Public Dose of 1.0 mrem/yr[a]

Nuclide[b]	MEI mrem/yr	Ci/yr	Percentage of DOE Scrap				
			100%	50%	10%	5%	1%
^{3}H	1.90E-05	5.26E+04	5.18E+09	1.04E+10	5.18E+10	1.04E+11	5.18E+11
^{14}C	8.77E-05	1.14E+04	1.12E+09	2.24E+09	1.12E+10	2.24E+10	1.12E+11
^{36}Cl[c]	---	---	---	---	---	---	---
^{41}Ca	6.57E-05	1.52E+04	1.50E+09	3.00E+09	1.50E+10	3.00E+10	1.50E+11
^{54}Mn	7.55E-02	1.32E+01	1.30E+06	2.61E+06	1.30E+07	2.61E+07	1.30E+08
^{55}Fe	2.63E-04	3.80E+03	3.74E+08	7.48E+08	3.74E+09	7.48E+09	3.74E+10
^{57}Co	1.20E-02	8.33E+01	8.20E+06	1.64E+07	8.20E+07	1.64E+08	8.20E+08
^{60}Co	1.13E+00	8.85E-01	8.71E+04	1.74E+05	8.71E+05	1.74E+06	8.71E+06
^{63}Ni	3.36E-04	2.98E+03	2.93E+08	5.86E+08	2.93E+09	5.86E+09	2.93E+10
^{65}Zn	4.24E-02	2.36E+01	2.32E+06	4.64E+06	2.32E+07	4.64E+07	2.32E+08
^{79}Sr[d]	---	---	---	---	---	---	---
^{90}SR	4.38E-02	2.28E+01	2.25E+06	4.49E+06	2.25E+07	4.49E+07	2.25E+08
$^{90}SR+Y$	4.51E-02	2.22E+01	2.18E+06	4.36E+06	2.18E+07	4.36E+07	2.18E+08
^{93}Zr	6.27E-03	1.59E+02	1.57E+07	3.14E+07	1.57E+08	3.14E+08	1.57E+09
^{94}Nb	5.09E+00	1.96E-01	1.93E+04	3.87E+04	1.93E+05	3.87E+05	1.93E+06
^{99}Tc	2.89E-03	3.46E+02	3.41E+07	6.81E+07	3.41E+08	6.81E+08	3.41E+09
^{106}Ru	6.29E-02	1.59E+01	1.56E+06	3.13E+06	1.56E+07	3.13E+07	1.56E+08
^{110m}A	2.01E-01	4.98E+00	4.90E+05	9.79E+05	4.90E+06	9.79E+06	4.90E+07
^{125}Sb	1.21E-01	8.26E+00	8.13E+05	1.63E+06	8.13E+06	1.63E+07	8.13E+07
^{129}I	1.03E+00	9.71E-01	9.56E+04	1.91E+05	9.56E+05	1.91E+06	9.56E+06
^{134}Cs	3.34E-01	2.99E+00	2.95E+05	5.89E+05	2.95E+06	5.89E+06	2.95E+07
^{137}Cs	7.15E-03	1.40E+02	1.38E+07	2.75E+07	1.38E+08	2.75E+08	1.38E+09
^{137}Cs	1.01E+00	9.90E-01	9.75E+04	1.95E+05	9.75E+05	1.95E+06	9.75E+06
^{144}Ce	5.10E-02	1.96E+01	1.93E+06	3.86E+06	1.93E+07	3.86E+07	1.93E+08
^{147}Pm	5.10E-03	1.96E+02	1.93E+07	3.86E+07	1.93E+08	3.86E+08	1.93E+09
^{151}Sm	3.95E-03	2.53E+02	2.49E+07	4.98E+07	2.49E+08	4.98E+08	2.49E+09
^{152}Eu	1.18E+00	8.47E-01	8.34E+04	1.67E+05	8.34E+05	1.67E+06	8.34E+06
^{154}Eu	9.35E-01	1.07E+00	1.05E+05	2.11E+05	1.05E+06	2.11E+06	1.05E+07
^{226}Ra	1.23E+00	8.13E-01	8.00E+04	1.60E+05	8.00E+05	1.60E+06	8.00E+06
^{228}Th	3.30E+01	3.03E-02	2.98E+03	5.97E+03	2.98E+04	5.97E+04	2.98E+05
^{229}Th	9.18E+01	1.09E-02	1.07E+03	2.14E+03	1.07E+04	2.14E+04	1.07E+05
^{230}Th	3.27E+01	3.06E-02	3.01E+03	6.02E+03	3.01E+04	6.02E+04	3.01E+05
^{232}Th	4.71E+01	2.12E-02	2.09E+03	4.18E+03	2.09E+04	4.18E+04	2.09E+05
^{232}Th	8.84E+01	1.13E-02	1.11E+03	2.23E+03	1.11E+04	2.23E+04	1.11E+05
^{232}U	6.24E+01	1.60E-02	1.58E+03	3.15E+03	1.58E+04	3.15E+04	1.58E+05
^{233}U	1.74E+01	5.75E-02	5.66E+03	1.13E+04	5.66E+04	1.13E+05	5.66E+05
^{234}U	1.72E+01	5.81E-02	5.72E+03	1.14E+04	5.72E+04	1.14E+05	5.72E+05
^{235}U	1.64E+01	6.10E-02	6.00E+03	1.20E+04	6.00E+04	1.20E+05	6.00E+05
^{238}U	1.53E+01	6.54E-02	6.43E+03	1.29E+04	6.43E+04	1.29E+05	6.43E+05
$^{238}U+$	7.85E+01	1.27E-02	1.25E+03	2.51E+03	1.25E+04	2.51E+04	1.25E+05
^{237}Np	6.39E+01	1.56E-02	1.54E+03	3.08E+03	1.54E+04	3.08E+04	1.54E+05

TABLE G.6. (Cont'd)

Nuclide[b]	MEI mrem/yr	Ci/yr	Percentage of DOE Scrap				
			100%	50%	10%	5%	1%
^{238}Pu	4.13E+01	2.42E-02	2.38E+03	4.77E+03	2.38E+04	4.77E+04	2.38E+05
^{239}Pu	4.44E+01	2.25E-02	2.22E+03	4.43E+03	2.22E+04	4.43E+04	2.22E+05
^{240}Pu	4.44E+01	2.25E-02	2.22E+03	4.43E+03	2.22E+04	4.43E+04	2.22E+05
^{241}Pu	6.77E-01	1.48E+00	1.45E+05	2.91E+05	1.45E+06	2.91E+06	1.45E+07
^{241}Am	7.06E+01	1.42E-02	1.39E+03	2.79E+03	1.39E+04	2.79E+04	1.39E+05

(a) The doses to the maximally exposed individual (MEI) were calculated using CAP88PC, assuming a 1 Ci/yr emission from a smelter located in a medium-high population area (such as Chicago). The radionuclide feedstock activity is conservatively assumed to be equal to the stack emission activity. See Section 3.2 for further discussion. The specific activities of individual radionuclides in the feedstock were calculated assuming the smelter handled 10 t/yr (where 1.016E+03 kg is assumed to equal 1 t).
(b) For radionuclides having decay chains, results of calculations are shown both with (+) and without consideration of the daughters.
(c) The radionuclide ^{36}Cl is not available in the CAP88-PC code.
(d) Although this is listed as a valid radionuclide in the CAP88-PC code, no dose or risk factors are available.

APPENDIX H

Public Doses from Unit Radionuclide Concentrations

APPENDIX H: PUBLIC DOSES RESULTING FROM UNIT RADIONUCLIDE CONCENTRATIONS

Table H.1 gives the doses that would result from each radionuclide from 1 pCi/g bulk contamination for each public dose scenario. The data were generated using CAP88-PC.

TABLE H.1. Public Dose Corresponding to Unit Concentrations (pCi/g) for Recycled Metals[a]

	Limiting		Limiting Dose	
	pCi/g per mrem/yr	pCi/g per mrem/yr	mrem/yr per pCi/g	mrem/yr per pCi/g
Nuclide	10 t	100 t	10 t	100 t
^{3}H	5.2E+09	5.2E+08	1.9E-10	1.9E-09
^{14}C	1.1E+09	1.1E+08	8.9E-10	8.9E-09
^{36}Cl[b]	---	---	---	---
^{41}Ca	1.5E+09	1.5E+08	6.7E-10	6.7E-09
^{54}Mn	1.3E+06	1.3E+05	7.7E-07	7.7E-06
^{55}Fe	3.7E+08	3.7E+07	2.7E-09	2.7E-08
^{57}Co	8.2E+06	8.2E+05	1.2E-07	1.2E-06
^{60}Co	8.7E+04	8.7E+03	1.2E-05	1.2E-04
^{63}Ni	2.9E+08	2.9E+07	3.4E-09	3.4E-08
^{65}Zn	2.3E+06	2.3E+05	4.3E-07	4.3E-06
^{79}Se[c]	---	---	---	---
^{90}Sr	2.3E+06	2.3E+05	4.5E-07	4.5E-06
$^{90}Sr+Y$	2.2E+06	2.2E+05	4.6E-07	4.6E-06
^{93}Zr	1.6E+07	1.6E+06	6.4E-08	6.4E-07
^{94}Nb	1.9E+04	1.9E+03	5.2E-05	5.2E-04
^{99}Tc	3.4E+07	3.4E+06	2.9E-08	2.9E-07
^{106}Ru	1.6E+06	1.6E+05	6.4E-07	6.4E-06
^{110m}Ag	4.9E+05	4.9E+04	2.0E-06	2.0E-05
^{125}Sb	8.1E+05	8.1E+04	1.2E-06	1.2E-05
^{129}I	9.6E+04	9.6E+03	1.1E-05	1.1E-04
^{134}Cs	3.0E+05	3.0E+04	3.4E-06	3.4E-05
^{137}Cs	1.4E+07	1.4E+06	7.3E-08	7.3E-07
$^{137}Cs+$	9.8E+04	9.8E+03	1.0E-05	1.0E-04
^{144}Ce	1.9E+06	1.9E+05	5.2E-07	5.2E-06
^{147}Pm	1.9E+07	1.9E+06	5.2E-08	5.2E-07
^{151}Sm	2.5E+07	2.5E+06	4.0E-08	4.0E-07
^{152}Eu	8.3E+04	8.3E+03	1.2E-05	1.2E-04
^{154}Eu	1.1E+05	1.1E+04	9.5E-06	9.5E-05
^{226}Ra	8.0E+04	8.0E+03	1.3E-05	1.3E-04
^{228}Th	3.0E+03	3.0E+02	3.4E-04	3.4E-03
^{229}Th	1.1E+03	1.1E+02	9.3E-04	9.3E-03
^{230}Th	3.0E+03	3.0E+02	3.3E-04	3.3E-03
^{232}Th	2.1E+03	2.1E+02	4.8E-04	4.8E-03
$^{232}Th+$	1.1E+03	1.1E+02	9.0E-04	9.0E-03
^{232}U	1.6E+03	1.6E+02	6.3E-04	6.3E-03
^{233}U	5.7E+03	5.7E+02	1.8E-04	1.8E-03

TABLE H.1. (Cont'd)

	Limiting		Limiting Dose	
	pCi/g per mrem/yr	pCi/g per mrem/yr	mrem/yr per pCi/g	mrem/yr per pCi/g
Nuclide	10 t	100 t	10 t	100 t
^{234}U	5.7E+03	5.7E+02	1.8E-04	1.8E-03
^{235}U	6.0E+03	6.0E+02	1.7E-04	1.7E-03
^{238}U	6.4E+03	6.4E+02	1.6E-04	1.6E-03
$^{238}U+$	1.3E+03	1.3E+02	8.0E-04	8.0E-03
^{237}Np	1.5E+03	1.5E+02	6.5E-04	6.5E-03
^{238}Pu	2.4E+03	2.4E+02	4.2E-04	4.2E-03
^{239}Pu	2.2E+03	2.2E+02	4.5E-04	4.5E-03
^{240}Pu	2.2E+03	2.2E+02	4.5E-04	4.5E-03
^{241}Pu	1.5E+05	1.5E+04	6.9E-06	6.9E-05
^{241}Am	1.4E+03	1.4E+02	7.2E-04	7.2E-03

(a) The doses to the maximally exposed individual (MEI) were calculated using the CAP88-PC code, with generic assumptions of a 1 Ci/y emission from a smelter located in a medium-high population area.
(b) The radionuclide ^{36}Cl is not available in the CAP88-PC code.
(c) Although this radionuclide was listed as a valid choice in the CAP88-PC code, no dose or risk factors associated with this radionuclide were available in the code.