APPENDIX D
COLLECTIVE DOSE ASSESSMENTS FOR CONCRETE, STEEL AND TRASH

1. Introduction

The NRC is considering regulatory requirements for the disposition of solid, potentially clearable materials that are under license by the NRC and its Agreement States. As part of its regulatory decision-making process, the NRC evaluates the advantages and disadvantages associated with a range of alternatives. This appendix assesses potential doses to workers and members of the public that could result from the implementation of the alternatives currently being evaluated. Potential collective doses are estimated for each alternative for concrete, ferrous metals[1], and trash. The information in this appendix is based on an evaluation of doses analyzed in part in NUREG-1640 (NRC 2003d), which is discussed in Appendix E.

This appendix summarizes the results of a draft report, available on NRC's webpage, entitled, "Collective Doses Associated with Clearance of Materials from NRC/Agreement State - Licensed Facilities," Rev. 2, December 31, 2003 (SC&A 2003). References in this summary pertaining to additional, detailed information correspond to their respective locations in the above referenced collective dose report (SC&A 2003). The objective of the report was to evaluate and compare the amounts and radionuclide characteristics of potentially clearable material (e.g., different types of metals, equipment, tools, concrete, and trash) and their associated radiation health impacts. For this purpose, assessments are made for the collective doses to radiation workers and members of the public that might result for each of the rulemaking alternatives.

The Draft GEIS defines five alternatives, two of which are subdivided into five options, as follows:

(1) No Action. This alternative is the baseline for comparison of alternatives and generally corresponds to material radioactivity concentration levels specified in Regulatory Guide 1.86 (USAEC 1974).

(2) Unrestricted Release. This alternative places no restrictions on what can be done with material that is released. Considerations for choosing a meaningful range of options for this alternative resulted in specifying 5 dose levels. The options include: zero above background (which was analyzed at 0.03 mrem/yr), 0.1 mrem/yr, 1 mrem/yr, and 10 mrem/yr. A realistic lower-bound dose limit of 0.03 mrem/yr was chosen because it is a small value at, or marginally above, detectable levels. The dose options used the

[1] In the context of this analysis, "ferrous metals" is used as an all inclusive term for all alloys whose major constituent is iron (Fe). Ferrous metals include such metals as carbon steel, stainless steel, forged steel, galvanized steel, cast iron, etc.

normalized doses in NUREG-1640[2] for unrestricted use to derive their respective radionuclide concentrations in specific materials.

(3) Disposal in EPA/State-Regulated Disposal Facility. This alternative places restrictions on the method of material dispositions. Specifically, material could only be disposed of in a RCRA Subtitle D landfill at or below the activity concentrations allowable for a defined dose option. The result is that a greater amount of activity could be released to landfills than the amount that would be released to general commerce under the Unrestricted Release Alternative. The options include: zero above background (which was analyzed at 0.03 mrem/yr), 0.1 mrem/yr, 1 mrem/yr, and 10 mrem/yr. The dose options used the normalized doses in NUREG-1640 for disposal in a RCRA Subtitle D landfill to derive their respective radionuclide concentrations in specific materials. The RCRA Subtitle D regulations encompass both municipal and industrial landfills. Construction and demolition (C&D) or other industrial waste landfills were included in the collective dose report (SC&A 2003). For further discussion of EPA/State-regulated landfills, see Appendix J.

(4) Disposal in a LLW Disposal Facility. This alternative is also referred to as the prohibition alternative, because any of the material considered would be disposed of only in an NRC-licensed low-level radioactive waste (LLW) disposal facility.

(5) Limited Dispositions. In this alternative, solid material would be released, but NRC would allow only certain authorized dispositions to limit the potential for public exposure. All materials to be released would undergo a radiation survey and the measured level of radiation would be compared against the criterion for release for limited dispositions. Solid materials with measured radiation levels below the established criterion would be released for pre-approved limited dispositions, while solid materials with radiation levels above the criterion would be sent to a LLW disposal facility. NRC regulations in 10 CFR Part 20 would be amended to add a regulation on limited dispositions. Any requests to release material other than to these limited end uses or at higher doses would require case-specific approval from NRC.

2. Design Objectives and Overall Approach

The overarching design objectives of this investigation are realism, clear and complete presentation, accuracy, consistency, and full disclosure of uncertainty in the derivation of the collective doses associated with each rulemaking alternative. In addition, the approach is required to be consistent and compatible with the methods used to derive individual normalized doses as provided in NUREG-1640 (NRC 2003d).

Consideration was given in the calculation of the collective doses to all categories of NRC/Agreement State licensees, a full range of exposure scenarios and/or population groups,

[2] "Radiological Assessments for Clearance of Materials from Nuclear Facilities - Main Report." NUREG-1640, Volume 1. Washington, DC: U.S. Nuclear Regulatory Commission, 2003 (NRC 2003d).

and a broad range of materials (ferrous metal (including steel), copper, aluminum, concrete, and trash). Since the number of categories and subcategories of licensees, types of materials, and exposure scenarios/population groups that can contribute to the collective doses is very large, the number of categories that are explicitly included was selected based on the following criterion:

- Capture enough of the categories and volume of material, and associated radionuclide inventory, and resultant collective doses associated with each rulemaking alternative for the preponderance of materials considered for disposal. If the exposure scenarios chosen are realistic, then these collective dose estimates can be used to provide very representative information on the dose impacts of the alternatives.

In the process of performing the analysis, secondary objectives included the following:

- Disclose which categories of NRC licensees and which materials are anticipated to be responsible for most of the collective doses.

- Create a mosaic of exposure categories that reveal the scenarios/population groups that are anticipated to experience the largest collective doses.

These objectives were achieved by using a combination of scoping/screening analyses and detailed calculations. The scoping calculations were generally deterministic and used to obtain a reasonable upper bound for the category selected. The detailed calculations could only be done when significant amounts of information were available, and over a realistic range of scenario specific situations. Then, performing random sampling over a very large number of potential exposure realizations, the collective dose can be estimated statistically (Monte Carlo method). This type of collective dose estimate has significant generic applicability because it is a valid representation of an average, or expected, value and its attendant range of uncertainty. As demonstrated through the scoping/screening calculations, steel, concrete, and trash were found to be the dominant sources of potentially clearable material in terms of volume of material, radionuclide inventory, and potential collective dose. In addition, the collective doses associated with end-use products made from recycled released steel were found to be responsible for the overwhelming majority of the collective doses.

The criteria for selecting the categories of recycled products to include in the collective dose assessment are described in Section 9.1 of the collective dose report (SC&A 2003). The criteria and methodology used to select categories of recycled products were specifically developed so as not to underestimate the collective dose. Also, for the collective dose assessment for the No Action and Unrestricted Release Alternatives it is assumed that the entire available inventory of solid material was used in the production of recycled products, and that none of the available inventory of solid material was disposed of in landfills, an assumption that maximizes the collective dose. Thus, although some specific types of products that could be made from recycled solid material may not be explicitly included in the collective dose assessment, the assumption that the entire available inventory of solid material is recycled accounts for the impact of the recycling on the collective dose (SC&A 2003).

All categories of licensees, types of materials, and exposure scenarios/population groups were addressed, but not to the same level of detail. As indicted above, a primary difference between the detailed analyses and the scoping/screening analyses results is that the detailed analyses are considered realistic estimates of the collective doses and sufficiently developed to be provided as a function of time, while the scoping/screening analyses are considered upper-end estimates of collective doses and cannot be represented as time dependent.

The analytical approach used in estimating collective doses from the use of products involved tracking and accounting for the inventories of radioactivity as materials moved from the point of release at licensees, to the incorporation of radioactivity in products or through the environment, and ultimately to dose receptors. In order to do this, ‘reference’ products (e.g., the generic category representing cars) were developed that are assumed to be representative of all products of a given type (e.g., the end product, namely the specific product being considered, such as “a car”). For example, a reference automobile was assumed that is representative of all types of automobiles (including pickup trucks, sport utility vehicles, etc.). The total collective dose remains the same, whether it is distributed over one car, two cars or 50 cars. This methodology is based on the assumption that, as the end-product concentration for a given amount of released radioactivity goes down, the number of individuals using that end-product goes up proportionally. In other words, the product of the concentration times the number of end-product users is always a constant. Analytically, when all the small contributions are added, it validates the methodology of calculating the collective dose from a single reference product. That single reference product is assumed to be representative of all products of that type.

For metals, the modeling is a cumulative total of all source terms and pathways having significance. For example, for power reactors, the source term is available for 50 years (or until the reactor is decommissioned) and it’s cumulative effects carried out for an additional 250 years. For everything else, because the available data was not sufficiently as refined, bounding, realistically, conservative estimates were made.

Since the detailed analysis employs probabilistic methods, uncertainty in quantities of material and the collective doses associated with the rulemaking alternatives are explicitly addressed by assigning uncertainty distributions to the input parameters, which are used throughout the calculations, and yield results that are presented as uncertainty distributions that disclose the mean, median, standard error of the mean, and the 5th and 95th percentile values for the results of the calculations.

For further elaboration on methodology, it is emphasized that the scoping/screening analyses do not employ probabilistic methods to assess uncertainties. Instead, a semi-quantitative analysis/discussion of the uncertainty in the analysis is provided which discloses the uncertainty in the quantities of material and collective doses in a less rigorous manner than those used in the more elaborate Monte Carlo analyses. In general, the scoping/screening analyses are designed to demonstrate that a given category of licensees, type of material, or exposed population group are not important contributors to the overall quantities of material or collective doses associated with each rulemaking alternative. As such, simplifying assumptions are used that provide a high level of assurance that the collective doses and quantities of materials are not underestimated.

As a final point, it is important to recognize that the concept of uncertainty, when addressing collective doses, is different from the concept of uncertainty when addressing individual doses, as was done in NUREG-1640. In NUREG-1640, the uncertainty analysis was concerned with estimating the uncertainty and mean values of the normalized doses to the individuals that comprise the critical groups. This report is concerned with the mean values and the uncertainty in the mean of real, but unknown, collective doses to population groups. It is not concerned with the variability of the doses to the individuals that comprise a given population group.

From a statistical perspective, NUREG-1640 (NRC 2003d) is concerned with the mean and standard deviation of the doses to individuals, while the collective dose report (SC&A 2003) is concerned with the mean and standard error of the mean of the collective doses to a given population group. This difference is important because individual variabilities within a population group "average out" when deriving the collective doses, resulting in uncertainties in the collective doses for a given population group that are relatively small as compared to the variabilities and uncertainties associated with the doses to the individuals that comprise the group.

The difference in these two concepts is equivalent to asking the question "what is the variability of the heights of the individuals that comprise the population of the U.S.," as opposed to asking the question "what is the uncertainty in the average height of the individuals of the U.S. population." In the case of the former, the variabilities are very large (the range of heights of adults likely span several feet). In the case of the latter, there is a real, but unknown average height of adults in the United States. Estimates of that value are based on measurements made on a sample of the total population, and the uncertainty in that value is probably less than a few inches.

3. Scrap Metal, Concrete, and Other Potentially Clearable Materials

The calculation of the amount and activity of potentially clearable material focused on commercial nuclear power plants, because they were determined to generate a major fraction of the total mass of potentially clearable materials from the decommissioning of all licensed facilities (96 percent of all ferrous metals and 99 percent of all concrete). The total mass of copper and aluminum from all licensed facilities combined is estimated to be less than 2 percent of the total mass of ferrous metals from nuclear power plants. Therefore, copper and aluminum were addressed by a deterministic scoping analysis.

Materials from nuclear power plants were characterized as clean, potentially clearable, or contaminated. Contaminated materials are expected to be disposed of as low-level radioactive waste (LLW), although some portion may be releasable if decontaminated. It is anticipated that some fraction of the potentially clearable materials would be disposed in a LLW facility under one or more of the regulatory alternatives being considered in this study. In addition to ferrous metals and concrete, the analysis also considered trash generated from operating nuclear power plants since some of this trash is generated within the restricted or impacted areas.

In addition to providing estimates for the expected masses of materials from boiling-water reactors (BWRs) and pressurized-water reactors (PWRs), the report also develops mass-to-surface area ratios. These ratios were needed to convert mass-based normalized effective dose equivalents (EDEs) to surficial dose factors. The analysis also determined contamination distributions needed to assess the impact of various regulatory alternatives on the mass of releasable materials and presents data on the fractional mix of radionuclides expected in the potentially clearable materials.

It was estimated that about 2 million metric tons of ferrous metals, 20 million metric tons of concrete, and about 200,000 metric tons of trash might fall within the scope of the proposed rule. About 45 percent of the radioactivity in these materials is from Co-60, with Cs-137 contributing another 16 percent.

4. Radionuclide Composition of Releasable Material Produced from Light-Water Reactors as a Function of Time

This information is presented, in part, in the form of curves that depict the volume of a given type of material on the y-axis and the radionuclide levels of the material on the x-axis. Then, using the normalized doses in NUREG-1640, or the explicitly defined clearance levels defined in Regulatory Guide 1.86 (USAEC 1974), the quantity of material and radionuclide composition of the potentially clearable material were calculated for the following four cases:

- Case A—Use NUREG-1640 <u>material-specific</u> limiting scenario normalized EDEs[3] for concrete and ferrous metals and trash limiting scenario normalized doses.

- Case B—Use NUREG-1640 (and trash) <u>material-independent</u> limiting scenario normalized EDEs.

- Case C—Use NUREG-1640 <u>Municipal Solid Waste (MSW) landfill (Subtitle D)</u> material-specific limiting scenario normalized EDEs for ferrous metals and concrete and limiting trash surrogate normalized doses.

- Case D—Use NUREG-1640 <u>industrial landfill</u> (Subtitle D) material-specific limiting scenario normalized EDEs for ferrous metals and concrete and limiting trash surrogate normalized doses.

This was accomplished by dividing the rulemaking alternative (in units of mrem/yr) by the normalized doses (in units of mrem/yr per pCi/g) to yield the release levels, (in units of pCi/g). Once the release level is determined for a given material and rulemaking alternative, curves were used to determine the quantity and radionuclide composition of the releasable material for a given alternative and case.

[3] NUREG-1640 provides normalized doses based either on the recommendations in ICRP Publication 26 (ICRP 1977) referred to as Effective Dose Equivalents (EDEs), or on the recommendations in ICRP Publication 60 (ICRP 1991) referred to as Effective Dose. This report uses the normalized EDEs.

Monte-Carlo calculations were employed to provide statistical representations for the mean value of the total collective dose and its range of uncertainty. The potential variability of the differing radiological source terms was included in these calculations, and the end results incorporate the uncertainty over all variable parameters considered, including this parameter.

The mean values of the total calculated activity in the releasable material are shown in Table D-1 for all four cases analyzed. As Table D-1 shows, the release of trash generates the largest amount of activity for three of the four cases analyzed. Only when the material-independent concentration limit (based on NUREG-1640) is used (i.e., Case B), is the activity generated dominated by the release of concrete. The activity in material released as a function of time is shown in the collective dose report (SC&A 2003).

Table D-1 Mean Value of Total Activity Released (Ci)

	Regulatory Options				
Case	**0.03 mrem/yr**	**0.1 mrem/yr**	**1 mrem/yr**	**10 mrem/yr**	**No Action**
		All Material			
Case A	0.848	3.36	41.14	538.29	4.33
Case B	0.177	0.449	2.23	6.11	
Case C	1.27	4.48	45.55	549.35	
Case D	0.927	3.57	41.99	541.09	
		Ferrous Metal			
Case A	0.092	0.395	2.86	12.74	1.76
Case B	0.008	0.048	0.745	4.33	
Case C	0.484	1.46	7.09	23.81	
Case D	0.138	0.550	3.54	15.55	
		Concrete			
Case A	0.168	0.401	1.49	1.73	0.243
Case B	0.168	0.401	1.49	1.73	
Case C	0.201	0.460	1.67	1.73	
Case D	0.201	0.460	1.67	1.73	
		Trash			
Case A	0.588	2.56	36.79	523.81	2.32
Case B	0.00001	0.00005	0.0015	0.043	
Case C	0.588	2.56	36.79	523.81	
Case D	0.588	2.56	36.79	523.81	

The mean values of the total calculated mass of the releasable material are shown in Table D-2 for all four cases analyzed. As Table D-2 shows, the release of concrete generates the largest mass of material for all of the cases analyzed. The mass of material released as a function of time is shown in SC&A 2003. For the dose options analyzed, there is very little difference in the mass of material released. This is also true for the other three cases analyzed. The reason for this is that there is a large mass of material at very low concentrations that would be released under any regulatory alternative, but as the regulatory alternatives become more liberal, the additional mass of material at the higher releasable concentrations is much less.

Table D-2 Mean Value of Total Mass of Material Released (million t)

Case	Regulatory Options				
	0.03 mrem/yr	**0.1 mrem/yr**	**1 mrem/yr**	**10 mrem/yr**	**No Action**
All Material					
Case A	16.0	19.1	21.9	22.3	18.3
Case B	15.5	18.4	21.4	22.1	
Case C	17.2	19.9	21.2	22.3	
Case D	16.7	19.5	22.0	22.3	
Ferrous Metal					
Case A	0.970	1.49	2.20	2.45	2.06
Case B	0.441	0.786	1.74	2.30	
Case C	1.57	2.00	2.38	2.48	
Case D	1.10	1.62	2.26	2.47	
Concrete					
Case A	15.0	17.6	19.6	19.8	16.2
Case B	15.0	17.6	19.6	19.8	
Case C	15.6	17.9	19.7	19.8	
Case D	15.6	17.9	19.7	19.8	
Trash					
Case A	0.014	0.021	0.041	0.066	0.020
Case B	0.0002	0.0004	0.002	0.006	
Case C	0.014	0.021	0.041	0.066	
Case D	0.014	0.021	0.041	0.066	

In SC&A 2003, Chapter 3 and Appendix A, detailed statistical data analysis was performed to determine the potential amounts of material considered releasable versus their associated levels of (measured) surface radioactivity. This type of distribution provides quantitative information on the shape of the probability distribution that characterizes the range of radioactivity levels likely to be found for the inventory of materials considered for release. The probability of finding contamination was mostly found to be very low and within the range of background radiation (approximately a null amount of radioactive contribution). Some small amount of material was found to increase beyond this, but within a very narrow range. This small increase, while possibly real, might also be partially caused by measurement uncertainty at these very low radiation levels. Beyond that range, there were very small, if any, amounts of material found until a much higher radioactivity level was reached. At this higher level, any materials found would require disposal at an NRC licensed LLW facility. Intuitively, such a probability distribution makes sense because anything that would be considered for release would, from materials process considerations, be expected to have no radioactivity or be at a very low level because of cleanup activities routinely performed as industry practice. Because decontamination is a destructive process, and cleanup efficiencies are usually high, very little, if any radioactivity would be expected to be found for these potentially clearable materials. Beyond this, other potentially radioactive materials would be expected to be at much higher levels, comparable to those requiring LLW disposal. Generally, although from a measurement aspect it is more difficult to demonstrate as well, materials considered for release having volumetric distributions of radioactivity would also be expected to have probability distributions similar to those found for surface radioactivity. Based on all of the above, the total amount of radioactivity from all released materials is expected to be small as compared to that contained in LLW. For example, the amount of radioactivity shipped to the three operating LLW facilities ranges from 1,400 to 420,000 Curies as annual averages. Regarding release, Table D-1 indicates that the estimated Curie inventory is expected to range from 0.2 to 549 Curies, over all regulatory options. Because the collective dose is proportional to the total radioactivity amount available for potential exposures, it also would be expected to be small.

The analysis included the possibility that some small amounts of ferrous scrap could be released after decontamination, based on operating experience with power reactors. For this to occur, the initial contamination levels would have to be relatively low and the decontamination factor high enough in order to meet release criteria. An item, such as a steam generator shell, might be considered worth decontaminating by licensees, given that such services are already available commercially. Generally a decontamination factor (DF) of around 10 is what is considered realistic, although some higher DFs can be achieved in limited specific cases. It is expected, from the aspect of implementation and cost concerns, that generally, based on process knowledge, only those contaminated materials that were within the DF range of 10 would be considered for decontamination. For such cases, it would be expected that any clean material of this type would be in the very low to none range of radioactivity concentration, and could be added to the clearance inventory.

5. Collective Doses Associated with Materials Other than those Modeled for Commercial Nuclear Power Plants

In addition to the release of ferrous metal, concrete, and trash from commercial nuclear power plants, the release of trash and other materials were analyzed for other types of NRC and/or Agreement State licensees. Also, the release of copper and aluminum from nuclear power plants was analyzed. Each of these analyses was performed via a screening (or bounding) calculation that compared the calculated mass and activity of material being analyzed to the mass and activity of ferrous metal, concrete, and trash from commercial nuclear power plants. The results of these scoping analyses are summarized below.

The development of a reference PWR nuclear power plant revealed that the masses of potentially clearable copper and aluminum were very small. (The best data for the analysis were from PWR ferrous metals inventory; BWRs were included via scaling factors.) The analysis also showed that the incremental radioactivity in releasable copper as compared to ferrous metals released under Case A was less than 7 percent for the 1 mrem/yr dose option. Similarly, the incremental radioactivity in aluminum released as compared to ferrous metals released under Case A was less than 0.5 percent for the 1 mrem/yr dose option. The total collective dose is directly proportional to the curie content of the released material. Because the addition of the copper's total radioactivity content is a small percent of the ferrous metal's (7 percent increment), it only has a small contributing effect to the total collective dose. The incremental increases for copper and aluminum are within the uncertainty bound of the analysis, and thus were not explicitly included in the Monte Carlo collective dose calculations but were included in the scoping analyses.

It was estimated that the total mass of potentially clearable trash generated by academic, industrial, government, and other unidentified sources would be about 13.4 times the mass generated by nuclear utilities. However, the activity contained in the much larger volume of the potentially clearable trash from these other licensees was calculated to be 40 percent of the activity in clearable trash from nuclear utility licensees. This increase in releasable activity was not included in the Monte Carlo collective dose calculations but was included in the scoping analyses and derivation of collective doses.

Other nuclear fuel cycle facilities that were analyzed include fuel fabrication facilities, uranium hexafluoride production facilities, and independent spent fuel storage installations (ISFSI). The bounding analysis for fuel fabrication facilities shows that exclusion of these facilities understates the total quantity of radioactivity in released ferrous metals by 3 percent for the 10 mrem/yr dose option, 0.2 percent for the 1 mrem/yr dose option, and has no effect on the remaining three dose options. Also, during the recycling of ferrous metals, uranium and physio-chemically-similar elements, such as zirconium and tungsten, especially with respect to their high refractory properties, partition to the slag and do not contribute to the collective dose in steel end-use products, such as automobiles and structural steel in buildings. For concrete, the impact of excluding fuel fabrication facilities on the quantity of radioactivity in released concrete varies from 6 percent for the zero above background dose option to 0.6 percent for the 10 mrem/yr dose option. The exclusion does not affect the no action results for concrete. Similar results were found from the bounding analyses of the other nuclear fuel cycle facilities.

Non-nuclear fuel cycle facilities that were analyzed include non-power reactors and medical centers. For non-power reactors, it was found that the radioactivity levels in released steel and concrete from all such reactors are very low — a maximum of 0.9 mCi for steel and 12 mCi for concrete for the 10 mrem/yr steel dose option. These quantities of radioactvity would not affect the collective dose results calculated elsewhere in the collective dose report (SC&A 2003) to any measurable degree.

The results of the screening analysis for medical centers revealed that the potential volume of material produced each year by all medical facilities in the U.S. that could be impacted by an NRC rulemaking addressing release could be as high as 38,000 tons/yr for the prohibition alternative (because such licensees could find it more feasible to disposition all materials as prohibited), and the radionuclide inventory in this material is about 1.3 mCi of tritium and 17 mCi of C-14. An upper-bound estimate of the potential collective doses associated with the release of this material showed that the collective dose from potentially released tritium in trash from impacted areas of all medical facilities in the U.S. would be <1 person-rem/yr, and the collective dose from the ingestion of released C-14 would be about 3.3 person-rem/yr. The actual collective doses are expected to be a very small fraction of these upper-bound values. For comparison to the calculated collective dose due to nuclear power reactor licensee operation and decommissioning, the values in Table 5.31 of the collective dose report (SC&A 2003) should be compared to the Case A collective doses reported in Table D-3.

6. Screening of Scenarios

There is a large number of potential exposure scenarios that could be used to evaluate the collective dose to the general public from the release of ferrous metal scrap and concrete rubble from NRC- or Agreement State-licensed facilities. A series of screening analyses was performed to eliminate scenarios that would not make a significant contribution to the total collective dose. These scenarios were selected from the radiological assessments of individuals exposed to materials released from nuclear facilities as presented in NUREG-1640 and in an earlier EPA study (Anigstein et al. 2001). Also included were five additional scenarios characterizing the population exposures to finished steel products.

The aim of the analyses was to calculate the collective doses from the release of 1 kt of released material with a total specific activity of 1 Bq/g, comprising the mix of radionuclides found in releasable material. These normalized collective doses were then ranked in decreasing order and the cumulative collective doses were tabulated. The results of the screening analyses show that, in the case of ferrous metal scrap, only five scenarios, all depicting population exposures to finished steel products, would make significant contributions to the total collective dose. In the case of concrete rubble, only the road use scenario plays a significant role in the collective dose analysis.

These screening analyses address most of the exposure scenarios described in NUREG-1640. Not included in the screening analysis are the scenarios characterizing the consumption of water from wells and surface runoff infiltrated by leachate from landfills or storage piles. These scenarios, which do not readily lend themselves to the deterministic screening analyses applied to the other scenarios, are addressed by the main collective dose analysis described in Section 10. Another class of scenarios omitted from the screening analyses is the population exposures from passing trucks carrying released materials. These scenarios are addressed by the main analysis in Section 10.

7. Collective Radiation Exposures to Workers Implementing the Rulemaking Alternatives at Light-Water Reactors

Two major activities need to be accomplished by radiation workers at licensed facilities when releasing material: (1) the performance of surveys; and (2) the decontamination of material to meet acceptable limits if the licensees elect to perform decontamination activities in support of release (some licensees may deem it not cost-effective to perform any decontamination activities for the purpose of material release).

The analysis of these major activities reveals that the collective exposures from surveying the entire inventory of releasable ferrous metals, trash, and concrete material for release following the Multi-Agency Radiation Survey and Site Manual (MARSSIM) (NRC 1997a) approach would be about 290 person-rem for the entire population of PWRs and BWRs. The overwhelming majority of this collective dose is due to surveys for trash because of the large surface area to volume ratio of trash, as compared to concrete and steel.

The collective exposure estimates are highly dependent on two variables, the exposure rate and unit survey effort. If the material is surveyed in place, where exposures could come from a variety of structures and sources other than the released material, then exposure rates could be higher by at least a factor of 10. If the potentially clearable material is removed from other sources of radioactivity, and the release levels are lower than Regulatory Guide 1.86 levels, then the exposure rates will likely be less than 0.005 mrem/hr above background. Conversations with representatives of the nuclear power industry reveal that, during the license termination process, the potentially clearable material would be segregated from the more contaminated areas and placed in low background areas so that the material could be surveyed with the lowest possible limits of detection. Accordingly, a radiation field of 0.005 mrem/hr above natural background (as used in this analysis) is consistent with this strategy and consistent with the types of radiation fields that may be expected from the material that is being surveyed.

In theory, the level of effort required to perform surveys would be expected to increase as the release level is reduced. However, investigations currently being performed by the NRC reveal that, with the exception of the zero above background dose option, the level of effort to perform surveys (for the radionuclide mix of concern at light water reactors) is expected to be approximately the same for all alternatives; i.e., about 3 minutes per square meter surveyed using conventional pancake probe survey techniques.

The collective exposures to workers performing decontamination in support of release is anticipated to be higher than the exposures from surveys, because decontamination activities are anticipated to be performed in more highly contaminated areas and require a greater level of effort per ton of material undergoing release. Estimates of collective dose associated with decontamination activities in support of release during operations and license termination is about 300 person-rem to decontaminate steel in the population of PWRs or BWRs. The estimated exposure rates and labor hours used in this estimate depend on assumptions regarding how decontamination is performed.

The results are unaffected by the rulemaking alternative, because it is assumed that the level of effort required to decontaminate the material to the release objective is the same for all rulemaking alternatives. This may not be the case for the very low dose options, such as the 0.1 mrem/yr alternative and the zero above background dose option. If it is determined that the level of effort required to achieve these criteria is twice as high, then the collective dose would double. In addition, if it is deemed plausible to decontaminate material that is up to 500,000 dpm/100 cm^2 (i.e., 100 times the limit set by Regulatory Guide 1.86 for Co-60), it is plausible that the radiation field in the vicinity of the decontamination operations could increase by a factor of 10 to 0.5 mrem/hr, thereby potentially increasing the collective dose by a factor of 10.

8. Radiation Doses to Workers Due to the Disposal of Releasable Materials at Licensed Low-Level Radioactive Waste Disposal Facilities

A scoping calculation of the collective doses to radiation workers at licensed LLW disposal facilities due to the licensed disposal of all potentially clearable material was performed. The approach used took advantage of the large amount of data that has been compiled on actual collective doses to radiation workers at LLW facilities, along with data characterizing the quantities of radioactive materials disposed. These data were used to derive empirically determined normalized collective doses to this population group expressed in terms of person-rem per curie disposed. This value was then multiplied by the total radionuclide inventory contained in potentially clearable material to derive the collective dose. In performing these calculations, C-14 and H-3 were not included in the analyses, since these radionuclides are large contributors to the number of curies disposed at LLW facilities, but do not contribute to the collective doses to workers. The results reveal that the collective dose to this population group from the licensed disposal of all potentially clearable material (as may be the case under a prohibition alternative) is less than 1 person-rem.

Because of the incrementally small quantities of radionuclides disposed at licensed facilities associated with release, as compared to the quantity of radionuclides currently being disposed, and the fact that the current doses to the nearby populations are a small fraction of the regulatory standards for offsite exposures set forth in 10 CFR 61 (i.e., 25 mrem/yr (0.25 mSv/yr)), a separate analysis of offsite exposures in the vicinity of LLW disposal facilities associated with release was not considered necessary and is not provided.

9. Collective Doses to Members of the Public from Recycling and Reuse of Released Material

Following release, residual activities in the released material will travel a complex route until final disposal in a landfill or until the radionuclides decay to stable forms. Along the way, some individuals will receive some degree of exposure to the radionuclides in the released material. The exposure scenarios and levels of exposure will vary by radionuclide and the type of released material (i.e., steel, copper, aluminum, concrete, and trash). This section divides the exposed population into population groups within which the individual members are anticipated to experience similar exposure from the released material. In this way, the collective doses to each population group could be estimated as the product of the average dose rate (i.e., rem/hr) to the members of each population group times the collective number of person hours of exposure in the population group.

The potential exposure scenarios vary depending on the type of material released, i.e., metal, concrete, or trash. For metal, the material is assumed to be either recycled or disposed of, as shown on Figure D-1. If it is recycled, then the potential exposure scenarios involve: (1) transportation of the material, (2) use of the products produced from the material, (3) the population surrounding the mill due to air emissions from the mill, (4) use of the slag (as road beds, and people traveling on those roads), and (5) disposal of the dust produced at the mill, as shown on the right side of Figure D-1. If the metal is disposed of in a landfill, it is assumed that it would be transported directly to the landfill, and the only exposures would be during transport and due to leaching of the material from the landfill, as shown on the left side of Figure D-1. As demonstrated by the scoping analysis, doses to workers along the released material's path (i.e., truck drivers, mill workers, landfill workers, road builders, etc.) do not contribute significantly to the collective dose, and are therefore not modeled.

Similar to Figure D-1 for metal, Figures D-2 and D-3 show the scenarios that are included in the collective dose model for the release of concrete and trash, respectively. Note that the section numbers indicated in Figures D-1 through D-3 refer to sections of the collective dose report (SC&A 2003).

Collective dose models were developed, and are described in detail in the report, for each of the exposure scenarios depicted in Figures D-1 through D-3, as follows:

- Exposures to Finished Steel Products
  - Beds
  - Automobiles
  - Office Buildings
  - Office Furniture
  - Home Appliances

- Exposures Due to Disposal of Released Material in a Landfill
  - Municipal Landfills
  - Industrial Landfills

- Exposures to Released Material Used in Road Construction
  - Concrete
  - Slag from Steel Mills

- Exposures to Airborne Releases
  - Steel Mill
  - Incinerator

- Exposures During Transportation
  - To Steel Mill
  - To Landfill
  - To Incinerator
  - To Road Construction Site

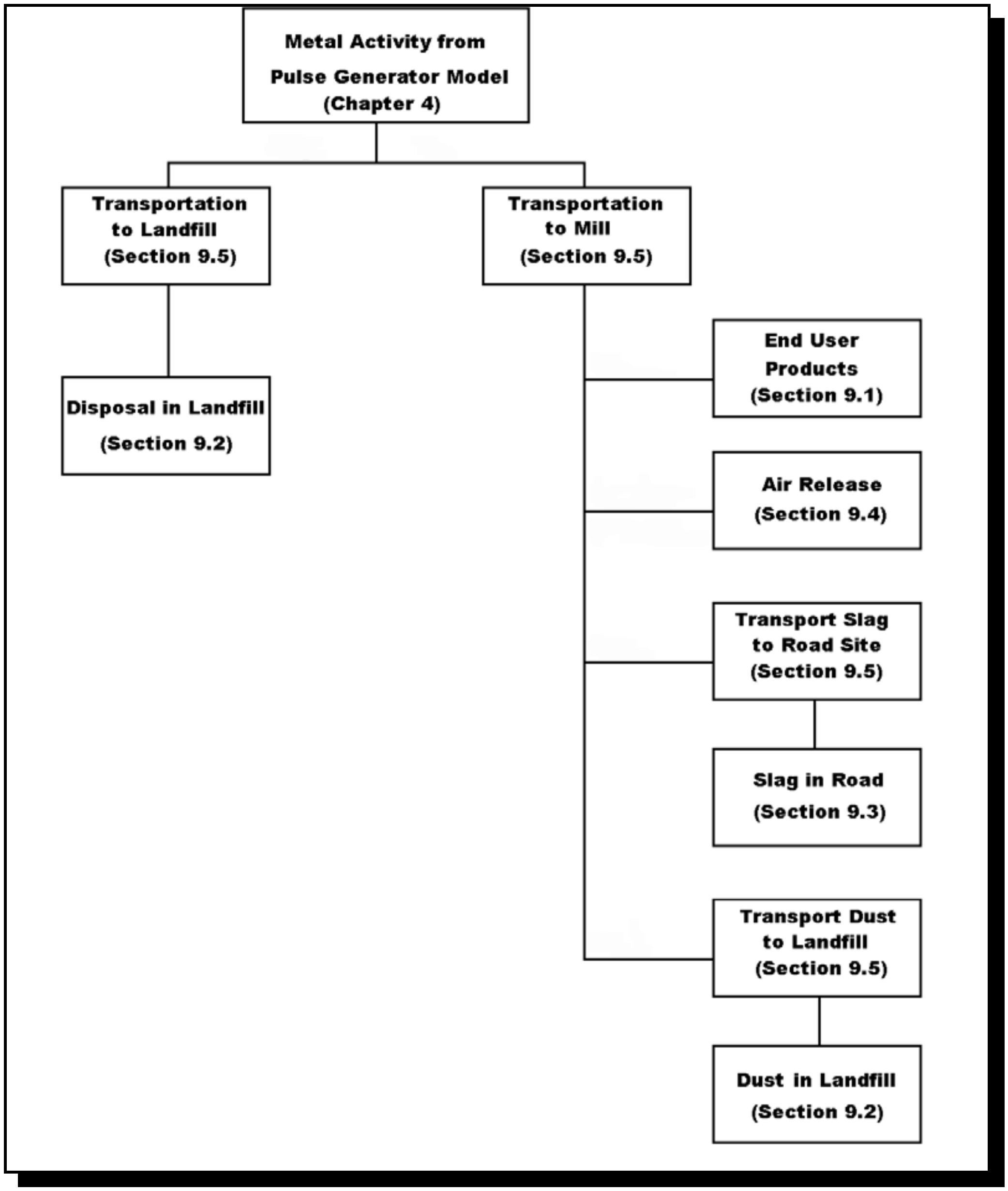

Figure D-1 Integrated Released Metal Collective Dose Model

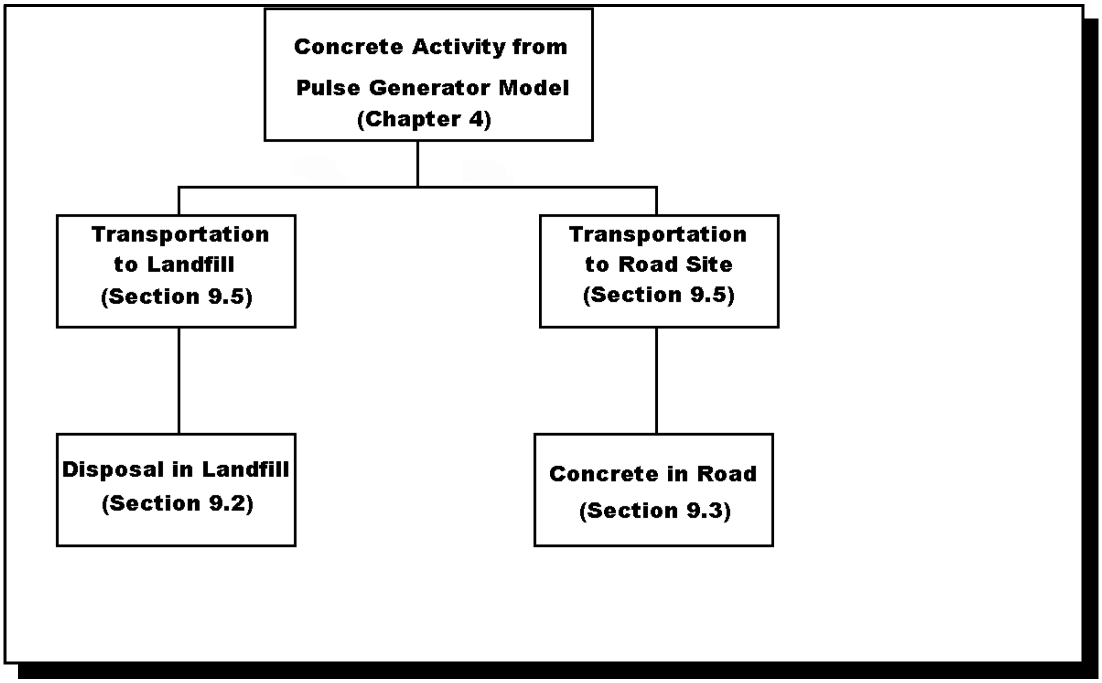

Figure D-2 Integrated Released Concrete Collective Dose Model

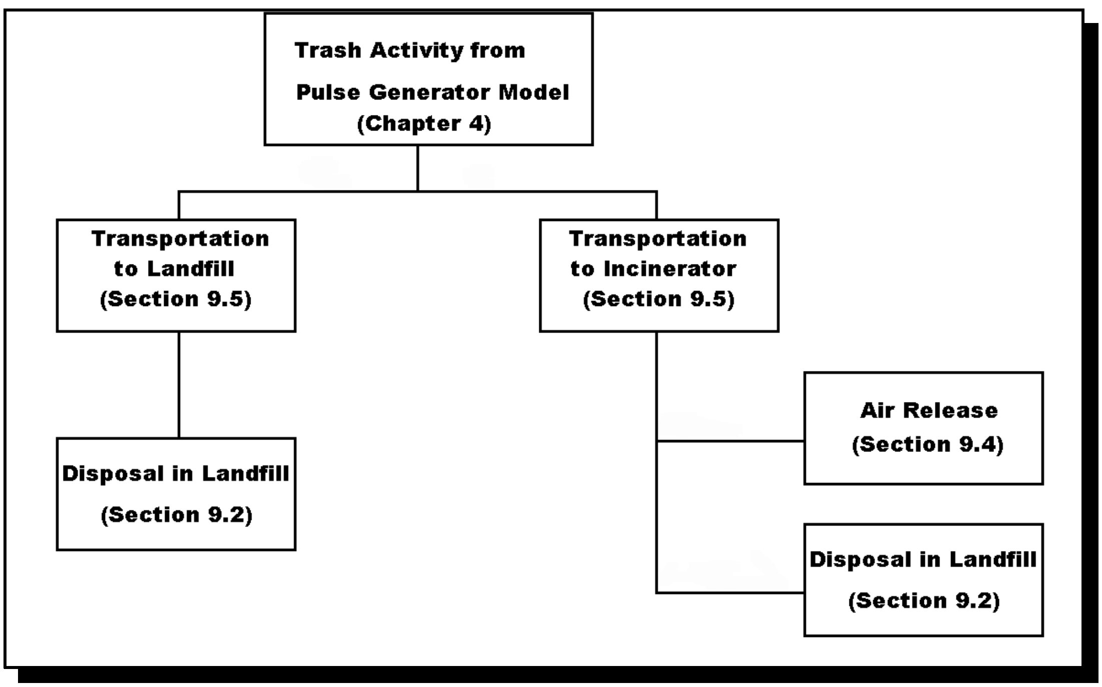

Figure D-3 Integrated Released Trash Collective Dose Model

9.1 External Collective Doses

For discussion purposes, the general form of the equation used to calculate the collective dose due to external exposure from radioactivity in released materials is presented:

$$D_{n,i,ext} = E_i \; O_i \; DF_{n,i} \; A_{n,i} \; \frac{1 - e^{-\lambda_n}}{\lambda_n} \; e^{-\lambda_n t} \qquad \text{D-1}$$

where:

$D_{n, i, ext}$	=	collective dose from radionuclide n and product i (person-rem)
E_i	=	average exposure duration to product i (hr/yr)
O_i	=	average occupancy/usage of product i (people)
$DF_{n, i}$	=	external collective dose factor for radionuclide n and product i (rem/hr per Ci)
$A_{n, i}$	=	activity of radionuclide n that is present in product i (Ci)
	=	integrating factor over a one year period for radionuclide n
λ_n	=	radiological decay constant for radionuclide n (yr^{-1})
t	=	time after release when exposure occurs (yrs)

To calculate the total collective dose, the terms in Equation D-1 would need to be evaluated for each specific case and summed over all cases for each of the times, products, radionuclides, and licensees.

One of the keys to evaluating the terms in Equation D-1 is to identify or calculate the appropriate values for the various parameters. The EPA's Exposure Factors Handbook (EPA 1987) was used as the primary source of most of the exposure duration data. The occupancy factors came from various sources, depending on which type of product was being modeled. For example, the occupancy factor for automobiles was taken from the 1995 Edition of "Nationwide Personal Transportation Survey." (DOT 1997)

Perhaps the most important parameter is the product-specific dose factor ($DF_{n, i}$). In short, single reference products were developed for each type of end product (including their mass and geometry) and the MCNP computer program was used to calculate dose factors for each reference product. See Sections 9.1.2 through 9.1.7 of the collective dose report (SC&A 2003) for the specifics of how the dose factors were developed for each end-use product.

9.2 Collective Doses Due to Food

The general form of the equation used to calculate the collective dose due to the consumption of food and water containing residual radioactivity from released material is:

$$D_{n,i,ing} = DF_n \; A_{n,i} \; e^{-\lambda_n t} \qquad \text{D-2}$$

where:

$D_{n, i, ing}$	=	collective dose from radionuclide n and product i (person-rem),
DF_n	=	ingestion dose conversion factor for radionuclide n (rem/Ci),

$A_{n,i}$ = amount of radionuclide n from released material that is present in product i (Ci),
λ_n = radiological decay constant for radionuclide n, and
t = time after release when exposure occurs (yrs).

As with external dose, to calculate the total collective dose, the terms in Equation D-2 would need to be evaluated and summed over all cases, for each of the foods, radionuclides, and licensees.

The ingestion dose conversion factors used in this analysis were taken from EPA's Federal Guidance Report No. 11. The key to solving Equation D-2 is the calculation of the amount of activity that is present in each food type ($A_{n,i}$). The specific methodologies used to estimate the activity in food are presented in detail in the report. The total activity was apportioned to the various potential pathways. For example, based on national averages, a certain percentage of ground water was apportioned for use as potable water, and then a certain percentage of that potable water was apportioned as drinking water. In that fashion, the activity consumed via the drinking water pathway was calculated.

10. Collective Doses Associated with Materials from Commercial Nuclear Power Plants

The results of the calculation of the activity and mass of the released material are presented above in Section 4, while the deterministic results of the collective dose scenario screening calculations are presented above in Section 5. This section focuses on the results of the probabilistic calculation of the collective doses using Monte Carlo methods for materials from commercial nuclear power plants.

The detailed probabilistic analysis was performed for five cases – the four material release cases, plus a Case C2 in which the trash is assumed to be incinerated, with its ash being disposed of in a landfill. Table D-3 presents the collective dose from all released materials for all regulatory options.

For Cases A and B, the collective doses are dominated by the external exposure of the population to released ferrous metal. Because Co-60 (with a 5-year half-life) is the primary radionuclide that partitions to the metal from the mix being used in this analysis, the exposure rapidly decreases when there is no new material being released. The other primary radionuclide from the mix (Cs-137) partitions primarily to the dust at the steel mill, and, for this analysis, the dust is assumed to be disposed of in a landfill, which is very unlikely to result in exposures to a large segment of the population (see the following discussion for Cases C and D).

For Cases C and D, all of the released material (ferrous metal, concrete, and trash) is assumed to be disposed of in a municipal or industrial landfill. The exposure of industrial workers (e.g., scrap truck drivers, landfill operators) is a primary contributor to the collective dose for these cases. The only potential pathway for cases C and D for exposure of other members of the general population is via the leaching/ground-water pathway. Since it frequently takes ground water a very long time to travel to surface water or to a well, many of the shorter-lived radionuclides will have significantly decayed before they have an opportunity to expose the population. Also, even those radionuclides with long half-lives that travel relatively fast through the unsaturated zone (e.g., I-129 and C-14) take more than the 250 years to reach the accessible environment, which is the maximum assessment period of the analysis (SC&A 2003). The contribution of the longer-lived radionuclides beyond 250 years is a very small fraction of the total collective dose associated with all radionuclides.

Case C2 used the same activity release distribution as calculated for Case C, but assumed that all trash would first be sent to an incinerator, with the resulting ash and any air pollution control device media being sent to a landfill (i.e., any activity that is not released via the stack of the incinerator was assumed to go to a landfill). Comparing the Case C2 to Case C collective doses in Table D-3 shows that the portion of the released activity that escapes the incinerator's air pollution control device results in exposures that are about three orders of magnitude higher than if the released activity is sent directly to a landfill.

As with the activity and mass results, additional details regarding the calculated collective doses resulting from material being released are provided in the collective dose report (SC&A 2003), including a breakdown that provides the mean, standard deviation, median, 5th percentile and 95th percentile values annually for each material type, regulatory option, case analyzed, and year of the analysis.

Table D-3 Mean Collective Dose Results (person-rem)

Case ID	Regulatory Options				
	0.03 mrem/yr	0.1 mrem/yr	1 mrem/yr	10 mrem/yr	No Action
All Material					
Case A	2.07E+02	8.88E+02	6.40E+03	2.84E+04	3.93E+03
Case B	2.16E+01	1.13E+02	1.68E+03	9.68E+03	3.93E+03
Case C	1.20E-01	3.46E-01	1.70E+00	6.43E+00	3.78E-01
Case C2	1.62E+01	7.01E+01	1.01E+03	1.44E+04	6.35E+01
Case D	5.29E-02	1.69E-01	1.01E+00	4.82E+00	3.78E-01
Ferrous Metal					
Case A	2.05E+02	8.81E+02	6.38E+03	2.84E+04	3.92E+03
Case B	1.88E+01	1.07E+02	1.66E+03	9.65E+03	3.92E+03
Case C	9.42E-02	2.84E-01	1.38E+00	4.63E+00	3.43E-01
Case C2	9.42E-02	2.84E-01	1.38E+00	4.63E+00	3.43E-01
Case D	2.70E-02	1.07E-01	6.86E-01	3.02E+00	3.43E-01
Concrete					
Case A	2.76E+00	6.54E+00	2.39E+01	2.78E+01	3.91E+00
Case B	2.76E+00	6.54E+00	2.39E+01	2.78E+01	3.91E+00
Case C	2.41E-02	5.53E-02	2.00E-01	2.09E-01	2.92E-02
Case C2	2.41E-02	5.53E-02	2.00E-01	2.09E-01	2.92E-02
Case D	2.41E-02	5.53E-02	2.00E-01	2.09E-01	2.92E-02

Table D-3 Mean Collective Dose Results (person-rem) (continued)

Case ID	Regulatory Options				
	0.03 mrem/yr	0.1 mrem/yr	1 mrem/yr	10 mrem/yr	No Action
Trash					
Case A	1.77E-03	6.90E-03	1.24E-01	1.58E+00	6.60E-03
Case B	2.71E-08	1.49E-07	4.75E-06	1.19E-04	6.60E-03
Case C	1.77E-03	6.90E-03	1.24E-01	1.58E+00	6.60E-03
Case C2	1.60E+01	6.98E+01	1.01E+03	1.44E+04	6.31E+01
Case D	1.77E-03	6.90E-03	1.24E-01	1.58E+00	6.60E-03

11. Collective Doses for IAEA RS-G-1.7 Clearance Levels

This section addresses collective doses for the IAEA clearance levels, which were published in Radiation Safety Guide No. RS-G-1.7 (IAEA 2004). As was described earlier, all collective doses were initially derived using release levels from NUREG-1640 and its supplement on trash (NRC 2003c, 2005c). The IAEA clearance levels are based on a deterministic analysis, while NUREG-1640 release levels were developed using a Monte Carlo method. As is noted earlier, the Draft GEIS focuses on impacts associated with materials that are expected to be released from nuclear power plants. As a result, radionuclide distributions and their relative fractions (mix) reflect those commonly found at both pressurized and boiling water reactors. Because the amounts of materials expected to be released from nuclear power plants are dominant, the Draft GEIS assumes that impacts associated with such releases are bounding and envelope those of other types of licensees.

The derivation of collective doses based on IAEA clearance levels relies on the application of adjustment factors to scale the collective doses derived from NUREG-1640. The adjustment factors take into consideration differences between the release levels of NUREG-1640 and clearance levels of the IAEA safety guide. The adjustment factors were developed for each type of material considered in the Draft GEIS (metals, concrete, and trash). Two other methods were considered but rejected as the chosen method provides reasonable estimates of adjusted collective doses given the differences in the modeling approaches between NUREG-1640 and the IAEA safety guide and associated uncertainties in model assumptions and parameters. One method considered the use of a qualitative analysis, where the discussions would address only the direction and magnitude of changes in collective doses by comparing the release and clearance levels of the most commonly found radionuclides (e.g., C-14, Co-60, Sr-90, or Cs-137). This method was not used because it does not offer the means to quantify differences in collective doses. Another method considered a series of duplicate Monte Carlo analyses of all collective doses by applying the IAEA clearance levels without using any information from NUREG-1640. This method was rejected because of the necessity to develop probability distribution functions for the assumptions and parameters used in the IAEA deterministic analysis.

The development of the adjustment factors is based on a procedure described in MARSSIM (NUREG-1575, Sect. 4.3.4, p.4-9 (NRC 2001)). The procedure involves deriving a gross activity concentration, C_{ga}, taking into account the release or clearance level of each radionuclide assumed to be present in each type of material and their relative fractions (SC&A 2003). The expression is:

$$C_{ga} = 1 / [(f_1 /CL_1) + (f_2 / CL_2) + ... + (f_n /CL_n)]$$

C_{ga} = gross activity concentration of the radionuclide mix, given the relative fraction of each radionuclide and IAEA clearance or NUREG-1640 release level, pCi/g

CL_i = clearance or release level, pCi/g, for radionuclide i, for a given dose, and relative fraction f_n

f_n = radionuclide fraction, unitless

The calculation is performed twice, once for NUREG-1640 release levels, and again for the clearance levels of IAEA RS-G-1.7. The resulting gross activity concentrations are compared and ratioed to derive an adjustment factor, as follows:

$$AF = C_{ga,\ IAEA} / C_{ga,\ NUREG\text{-}1640}$$

As derived using NUREG-1640, the collective doses are best estimates, based on a Monte Carlo analysis, and, consequently, the results reflect uncertainties associated with the statistical distributions of model parameters, including radionuclide mix. For example, it is known that the distribution of radionuclides and their relative mix would vary among nuclear power plants. At some power plants, Cs-137 may be more prevalent than Co-60, for instance. At other plants, the opposite may be true given different operational histories. A limited sensitivity analysis was conducted to assess differences in adjustment factors when changes are made to the radionuclide mix. Of the 17 radionuclides making up the mix used in the collective dose analysis, three comprise nearly 80% of the activity; they are Fe-55, Co-60, and Cs-137. New adjustment factors were derived for the entire distribution of 17 radionuclides by interchanging the relative fractions of these three radionuclides. These changes mimic alternate conditions where the relative mix may be different.

Table D-4 presents the resulting changes in gross activity concentrations and corresponding adjustment factors. A review indicates that the resulting adjustment factors vary from 1.2 to 2.3 over all four cases, with an average of 1.9 ± 0.5. For the three alternate cases, the adjustment factors vary from 1.2 to 2.2, with an average of 1.7 ± 0.5. The results indicate that changes up to a factor of three in the relative mix (0.46/0.15) yield lower adjustment factors than the base case. Lower adjustment factors imply that collective doses normalized to RS-G-1.7 clearance levels would be in closer agreement to those derived using the release levels of NUREG-1640.

In light of the above discussions, the uncertainties associated with collective doses derived using NUREG-1640 release levels are not translatable with the adjustment factors developed herein. As a result, collective doses representing IAEA clearance levels are assumed to be best single estimates, given the lack of information on the derivation of the IAEA clearance levels.

Table D-4 Adjustment Factors Versus Changes in Relative Mix of Three Radionuclides

Parameters	Radionuclide Mix			
	Base Case	**Case 1**	**Case 2**	**Case 3**
Fe-55	0.18	0.18	0.46	0.15
Co-60	0.46	0.15	0.18	0.46
Cs-137	0.15	0.46	0.15	0.18
RS-G-1.7 gross activity concentration (pCi/g)	3.8	3.8	6.3	3.7
NUREG-1640 gross activity concentration (pCi/g)	1.7	3.1	3.5	1.7
Adjustment factor (RS-G-1.7 / NUREG-1640)	2.3	1.2	1.8	2.2
Ratio of adjustment factors to the base case	1	0.52	0.78	0.96

Table D-5 summarizes the results and presents the adjustment factors for each material (concrete, steel, and trash) and a material independent case. A review of the results reveals some differences in adjustment factors, with NUREG-1640 and RS-G-1.7 being alternatively more limiting depending on the type of material. For concrete, the results indicate that RS-G-1.7 yields a more liberal gross activity concentration (3.8 pCi/g), while NUREG-1640 is more permissive for steel (9.1 pCi/g) and trash (4,070 pCi/g). For the material independent case, NUREG-1640 is limiting at 1.7 pCi/g. The adjustment factors for concrete and the material independent case are identical. This is because the material independent case is based on the most limiting clearance levels, which are those assigned to concrete. It should be noted that the results for trash are not directly comparable as RS-G-1.7 does not address materials that fit the definition of trash generated by nuclear power plants. Accordingly, the adjustment factor for trash is arbitrarily assigned to unity. In addition, the adjustment factor for concrete is rounded off since the method used and the data do not provide this level of precision.

Table D-5 Derived and Applied IAEA RS-G-1.7 and NUREG-1640 Adjustment Factors

Case	Concrete	Ferrous Metals	Trash	Material Independent
RS-G-1.7 gross activity concentration (pCi/g)	3.8	3.8	3.8	3.8
NUREG-1640 gross activity concentration (pCi/g)	1.7	9.1	4,070	1.7
Derived adjustment factors (RS-G-1.7 / NUREG-1640)	2.3	0.4	0.001	2.3
Limiting case	NUREG-1640	RS-G-1.7	RS-G-1.7	NUREG-1640
Applied adjustment factors	2	0.4	1	2

The adjustment factors are used to scale the collective doses generated with the release levels of NUREG-1640. The results imply that RS-G-1.7 collective doses for concrete are expected to be higher than that of NUREG-1640 release levels by a factor of about two. An adjustment factor of two is assigned to the case illustrating releases of material based on the most restrictive release criteria (i.e., material independent case). However, collective doses for ferrous metals, based on

RS-G-1.7 clearance levels, are expected to be lower by a factor of about 0.4. For trash, collective doses are assumed to be identical since an adjustment factor of one was arbitrarily assigned to trash.

12. Collective Doses Associated with the Reuse of Equipment

Collective doses associated with the reuse of equipment were evaluated for two categories of equipment, large and small. The approach used in estimating collective doses relies on a scoping analysis as practices associated with the reuse of equipment are known to be highly variable. For example, it is known that different types of equipment and tools are used in radiologically controlled areas and later taken out. The types of equipment that could be potentially released from nuclear facilities for reuse in an environment free of radiological controls ranges from small items, such as hand tools, to very large ones, such as mechanized equipment and industrial vehicles. The following are examples of potentially reusable equipment, tools, and miscellaneous items:

- small hand tools (wrenches, screw drivers, etc.) and power tools (drills, saws, etc.)
- electrical equipment, such as control panels, motors, pumps, and generators
- office furniture (desks, chairs, filing cabinets, etc.) and office equipment (copiers, computers, printers, fax machines, etc.)
- construction equipment, such as scaffolding, noise or dust-control barriers, wheelbarrows, etc.
- mechanized equipment, such as trucks, backhoes, bulldozers, and other vehicles
- materials and supplies for use in their original forms, but taken out as excess, such as piping, tubing, electrical wiring, floor covering, ductwork, sheet metal, pipe hangers, light fixtures, wall board, and sheet glass.

It should be noted that these examples are assumed to characterize as well the profile of equipment, tools, and miscellaneous items that may be released by other types of licensees.

It is recognized that the release of equipment is an extremely dynamic process involving different types of facilities and activities, such as routine operations, research and development, major and minor power plant outages, refurbishment, decommissioning, etc. In addition, this process is taking place simultaneously at thousands of facilities across the nation and conducted every hour of the day and every day of the week. As a result, it is not possible to define what types of objects and how many are routinely used in radiologically controlled areas, and what fraction is surveyed and released for reuse *versus* those that are discarded as LLW.

As is noted earlier, the GEIS focuses on impacts associated with materials that are expected to be released from nuclear power plants. As a result, radionuclide distributions and their relative fractions reflect those commonly found at both pressurized and boiling water reactors. Because the amounts of equipment and tools being routinely released from nuclear power plants are dominant, the GEIS assumes that impacts associated with such releases are bounding and envelope those of other types of licensees.

In practice, equipment and tools are surveyed before being taken out of radiologically controlled areas. The survey consists of conducting a scan with a portable radiation survey meter (e.g., gas flow proportional detector) and taking wipes to assess the presence of removable surface activity. The presence of radioactivity on wipes is evaluated using separate instrumentation (e.g., bench top beta or alpha particle counter). Some survey methods involve the introduction of the item into an instrument (e.g., gamma tool monitor) that measures radioactivity *in toto* from all external and internal surfaces. Depending on the results of the survey, the items are either cleaned to meet release criteria, not taken out of the controlled area and set aside for later use in the same work area, or simply discarded as LLW. Together, these ALARA practices are expected to mitigate the presence of residual radioactivity on released items and should result in residual levels that are well below release criteria.

Dose factors and their corresponding release levels, for both large and medium-sized equipment, were taken from the NUREG-1640 Supplement on Reuse (NRC 2005d). In addition, the analysis includes the clearance criteria approved by the International Atomic Energy Agency (IAEA) in "Radiation Safety Guide RS-G-1.7" (IAEA 2004). Since the IAEA criteria present only volumetric clearance levels, surficial clearance levels were derived by applying a mass-to-surface ratio (from NUREG-1640) to the IAEA clearance levels (NRC 2003d). This aspect is discussed later. The case addressing the No Action Alternative (status quo) assumes that current practices result in an annual dose of about 1 mrem. This is because the release level of 6,980 dpm/100 cm^2, derived for Co-60 and small equipment, is not much different than the 5,000 dpm/100 cm^2 limit defining current practices, i.e., both yield an annual dose of about 1 mrem (see discussion in Appendix B discussing current practices). However, for large equipment with a limit of 600 dpm/100 cm^2, the difference is attributed to the scenario developed in the Supplement to NUREG-1640 (NRC 2005d). This scenario considers a driver in a cab of a vehicle with all internal surfaces having residual radioactivity levels at the release criteria for Co-60.

Given that no specific information is available on the type of equipment that might be released for reuse and the frequency of their reuse, the analysis applies broad assumptions in estimating collective doses. The dose factors developed for the file cabinet are assumed to be representative of medium-sized equipment, such as bookcases, lockers, tool cabinets, outer cases of welding machines, work benches, and other objects with similar geometries, dimensions, and surface areas. In addition, the dose factors derived for the file cabinet are assigned to hand tools and other similar small objects. This approach is deemed to be generically appropriate because it relies on the full range of exposure pathways (i.e., external, inhalation, and incidental ingestion) that workers would encounter while using smaller equipment, hand tools, and small items (NRC 2005d). Another justification for this approach involves the difficulty in defining a representative set of items that would be typical of hand tools routinely subjected to release. Defining a representative set of hand tools would imply a degree of specificity that would be difficult to justify since alternate cases could be made with equally powerful technical arguments.

Another complication revolves around defining the inventory of residual radioactivity associated with released equipment and materials. The inventory is directly proportional to the surface area (cm^2) of each item and release level (dpm per 100 cm^2 or pCi/cm^2) for a given dose level. The product of the two would yield the inventory (dpm or pCi). However, this is not a simple problem since first one would need to identify all types of equipment or items that might be released, and define the surface area of each item. The surface area is also difficult to determine. For example, should the area include all surfaces (external and internal) or just external? Are there areas that are inaccessible to the survey method but are suspected of having some residual levels of activity? How would one define the total surface of inaccessible areas? Should equipment or items characterized by complex physical configurations be released using only volumetric release levels? What fraction of equipment and items used by the nuclear industry could be considered complex in configuration? Given the lack of specific information in addressing these questions and associated uncertainties, even if they were answerable, the scoping analysis applies the release level as a surrogate for the inventory of residual radioactivity. For example, if one were to assume that 3,000 dpm/100 cm^2 were associated with an annual dose of 1 mrem, then it follows that 300 dpm/100 cm^2 would result in an annual dose of 0.1 mrem, other aspects being equal. This ratioing is applied to each dose level and its corresponding release level; thereby, leading to a comparison of collective doses relative to the 1 mrem dose option and its release level.

In converting the volumetric IAEA criteria to surficial release levels, a mass-to-surface ratio is defined using the information presented in Appendix A to NUREG-1640 (NRC 2003d). In simple terms, the mass-to-surface ratio ($gram/cm^2$) is defined as the total mass (grams) of an object divided by its total surface area (cm^2), or equivalently, the product of the density and its effective thickness, i.e., $gram/cm^3$ x cm. A review of the data presented in NUREG-1640 indicates that mass-to-surface ratios vary significantly, depending on the type of equipment or item. For equipment with complex configuration, one would expect that there may be portions of the equipment with different features, each with its own mass-to-surface ratio. For example, a tank might have four mass-to-surface ratios, one for the wall making up the body of the tank, another for access openings and connection flanges, another for its support skid, and a fourth one that is an overall average for all features of the tank. Table D-6 illustrates the variability of mass-to-surface ratios for some ferrous metal components found at typical power plants.

A review indicates that ratios vary from about 2 to nearly 80 g/cm^2, with an overall average of about 5 gm/cm^2 as being representative of both types of plants. It is recognized that there may be circumstances when mass-to-surface ratios will be different. For example, some types of ventilation ductwork may be built with a thinner gauge of sheet metal or a valve body may be made of cast iron with thicker walls, all resulting in different ratios than that listed here. However, such case-specific differences are not expected to yield a much different average as the associated amounts of metal for such extreme cases are expected to make up a small fraction of the total inventory of released ferrous metals. Accordingly, the value of 5 gm/cm^2 is assumed to be a representative estimate for the purpose of bounding collective doses.

Table D-6 Mass-to-Surface Ratios for Some Steel Components Found at Power Plants

Item/Component	Minimum (gm/cm^2)	Maximum (gm/cm^2)	Average (gm/cm^2)	Notes
Tanks				
BWR	2.2	54.3	9.8	
PWR	2.6	52.9	11.3	
Piping				
BWR	5	56.6	13	
PWR	2.5	27.6	6	
Rebars	1.9	11.3	5.4	Over 11 sizes
HVAC Ductwork				
BWR	--	--	1.3	Average
PWR	--	--	1.1	" "
Valves				
BWR	10.3	79.9	38	External surfaces only
PWR	10.3	63.2	29.5	" " "
Structural steel	3.8	21.9	10	Total surface area
Pipe hangers	1.8	19.9	7.5	Weighted average
Heat Exchangers				
BWR	--	--	2.5	Aggregate average
PWR	--	--	3.1	" "
Overall average				
BWR	--	--	4.5	Aggregate average
PWR	--	--	5.1	" "

In light of the lack of specific information, average collective doses were estimated using the following general expression:

$$D_{ave} = L\,(C_{RU}/C_{RL})\,W\,K\,N_e\,IF_i\,D_i\,R^{-1} \qquad \text{Eq. 1}$$

where:

D_{ave} = average collective dose, person-rem, with each nuclide at the release level

L = dose limit, mrem/year, corresponding to release level, dpm/100 cm^2, summed over all radionuclides I

C_{RU} = reuse release level, dpm/100 cm^2, for dose limit, mrem/yr, summed over all radionuclides I

C_{RL} = reference release level, dpm/100 cm^2, for reference dose limit of 1 mrem/yr, summed over all radionuclides I

W = effective workforce assumed to reuse equipment, persons

K = conversion factor, 10^{-3} rem per mrem

N_e = number of equipment or items assumed to be release and reused

IF_i = integrating factor for radionuclide I, integrated over useful life of equipment, yr

$IF_i = [1 - \exp(-(\lambda t_a)] / \lambda$, where;

t_a = time of integration for assessment period, year

λ_i = decay constant for radionuclide, I

D_i = decay factor from time of release to beginning of assessment period, as defined below

$D_i = \exp(-(\lambda t_s)$, where;

t_s = elapsed time from release to beginning of assessment period, year

R = activity profile adjustment factor, max-to-mean surface residual radioactivity levels, unitless

The number of work-hours that workers are assumed to handle reused items is based on 2,080 hours per year adjusted for the fact that about 25% of the time is spent on administrative and support functions that do not require the use of any equipment. In addition, the analysis assumed that some equipment that are used are not a product of release, meaning that such equipment is of other origins and were never introduced in any radiologically controlled areas. This fraction was assumed to be 25% for both large and small equipment, lacking specific data. Given that such equipment is expected to have a productive life cycle, a useful life of 14 years was assigned for large equipment and three years for small equipment. The duration of the useful life of equipment is driven by operational conditions and economic considerations, taking into account replacement costs and cost of repairs, amortization, and cost of money. These factors are expected to be different among facilities. For large equipment, the useful life is assumed to be twice that of the typical amortization schedule of seven years for capital expenditures. This assumes that once amortized, the equipment is used for another seven years before being declared worn out and discarded. For small equipment, such as hand tools, the useful life (3 years) assumes that once worn out, these items are discarded given that replacement costs are usually less than repair costs.

The determination of the number of workers using equipment and items after release is complicated by the lack of information characterizing practices at various facilities. For example, the type of equipment and tools used during routine operation is expected to be different than that used during maintenance or plant outages. Similar differences would be expected between refurbishment and decommissioning. In all cases, the size of the work force would vary as well. Lacking specific information, the approach considers occupationally exposed workers as a surrogate for the population that might be using released equipment and tools. The worker population that uses equipment and tools in radiologically controlled areas can be considered to be the same population that uses released equipment. This assumption is valid because once equipment and tools are released and workers are out of a radiologically controlled area, it does not matter whether the use of the equipment is associated with the same worker or any other worker. The analysis is insensitive to the origin of the worker, and considers only that a "worker" uses equipment and tools that have been released. Accordingly, the use of radiation workers is deemed to be a surrogate in determining the size of the workforce as there is some information on the number of workers employed by the nuclear industry.

The number of workers is based on the NRC REIRS database (NRC,2003e). A review of the database indicates that about 108,000 workers at reactors and 11,800 at materials sites are badged and report exposures under the provisions of 10 CFR Part 20.2206. The number of workers from Agreement State (AS) licensees was estimated by ratioing the number of licenses between the AS and NRC data. Based on NUREG-1350 (NRC 2002c), the ratio was estimated to be 3.3 (16,253/4,922), thereby giving a total of material 38,900-badged workers. The total size of the workforce of both NRC and AS licensees is estimated to be 150,000 workers (108,000 + 38,900, rounded off). A further evaluation of the database indicates that for about 50% of the workforce, all exposures or doses are reported as “non-measurable.” This information indicates that a portion of the workforce, although required to be badged, perform duties in radiologically control areas that may not require “hands-on” activities. Such types of workers may include supervisors, security, engineers, inspectors, janitors, etc. Accordingly, some of that workforce might not perform “hands-on” functions in radiologically control areas and it is then unlikely that they would be using released equipment. Nevertheless, it is assumed that 25% of the work force is using large equipment, and 75% of the workforce is using small equipment, including hand tools and small items. The resulting assumptions and estimate of the size of the work force using released equipment are summarized in Table D-7.

Table D-7 Major Assumptions Used in Deriving Collective Doses Due to Reuse

Parameter	Large Equipment	Small Equipment	Notes
Annual work-hours	2,080	2,080	Assumes a full work year
Admin./Support functions	0.25	0.25	Time away from released equipment
Equipment distribution	0.25	0.25	Fraction of time using equipment of other origins
Annual work-hours per worker	1,040	1,040	Contact time with released equipment
Size of work force	150,000	150,000	Potential number of workers using equipment
Incidental workers	0.25	0.75	Fraction using released equipment
Effective work force	19,000	56,000	Aggregate number of workers using released equipment
Equipment useful life (years)	14	3	Time over which dose is integrated
Elapsed time from release to start of exposure (years)	0	0	Assumes no radioactive decay before use
Number of pieces of equipment in use per person	1	1	Dose multiplier - see text for details

Collective doses are based on the presence of 17 radionuclides commonly found at nuclear power plants, as beta, gamma, and alpha emitters (Table 3.21, SC&A 2003). No credit is taken for radioactive decay from the time of release to the beginning of the assessment period. Three sets of radionuclide distribution-weighted gross activity release levels were derived for both, large and small equipment, and one using IAEA Radiation Safety Guide RS-G-1.7 for both small and large equipment.

Release levels were derived for both small and large equipment using a procedure described in MARSSIM (NUREG-1575, Section 4.3.4, page 4-9 (NRC 2001a)). The procedure involves deriving a gross activity concentration, C_{ru}, taking into account the release or clearance level of each radionuclide assumed to be present and its relative fraction (SC&A 2003). The expression is:

$$C_{RU} = \sum C_i \qquad \text{Eq. 2}$$

where

C_{RU} = release level, dpm/100 cm^2, for dose limit, mrem/yr, summed over all radionuclides I

C_i = weighted concentration for the given mix and dose option, where;

$C_i = 1 / [(F_1/L_1) + (F_2/L_2) + ... (F_n/L_n)]$

F_n = relative fraction of radionuclide I, unitless

L_n = limit for radionuclide I, and given dose option

Collective doses were adjusted to represent average surface activity profiles of equipment being released. The adjustment applies a correction factor, max-to-mean surface residual radioactivity levels to release criteria. A single average factor was used in the calculation as opposed to applying a factor for each dose option. This approach was used because there is not enough information to develop a more definitive activity profile at each dose option.

The factors are as follows:

Dose Options	Max-to-mean factor
10 mrem/yr	21.6
1.0 mrem/yr	7.1
No action, 1 mrem/yr	5.4
0.1 mrem/yr	3.3
0.03 mrem/yr	2.6
Average factor	8

These observations indicate that residual radioactivity profiles on equipment and tools can vary depending on licensee practices. This analysis assumes that residual radioactivity profiles are characterized by a continuous spectrum, bounded by a range defined by non-detectable levels on the low side and release levels on the high side. Within this spectrum, the average is assumed to be the best single estimate of residual radioactivity levels. It should be noted that other max-to-mean factors might in fact be observed in isolated instances, depending on the situation of a specific licensee, and it may be difficult to narrow these ranges and provide a more robust estimate of the variability and best estimate of the average. Nevertheless, it is not plausible to assume that equipment and tools would be characterized only by higher activity profiles and would be released routinely by every licensee. Accordingly, this adjustment yields average collective doses for the exposed population of workers and not doses to the average member of the critical group.

The number of pieces of equipment being reused by each worker after release could be different between small and large equipment. The collective dose is assumed to be directly proportional to the total number of items being used at any one time. In considering large equipment, such as vehicles and mechanized equipment, it is assumed that a worker can only operate one piece of equipment at a time. Accordingly, the dose multiplier is assumed to be one for such a case. For small equipment, such as hand or small power tools, etc., it is conceivable that a worker could use a number of items or at least be surrounded by several such small tools while working. Accordingly, the dose multiplier could be greater than one in such instance. As noted earlier, the dose factors derived for the file cabinet are assigned to hand tools and other similar small objects. The large surface area of the file cabinet is assumed to represent a collection of small tools. For example, the surface area of a typical screw driver was estimated to be about 120 cm^2. Other tools could be physically larger than a screw driver, such as power tools, shovel, etc. Accordingly, the large surface area of the filing cabinet (about 3 m^2) makes up for the presence of numerous smaller tools being used or located in the immediate vicinity of a worker. For example, if the average total surface area of an average hand tool were 1,000 cm^2, this would correspond to the exposure associated with approximately 30 tools, based on the surface area of the file cabinet (derived as: $3\ m^2 \times 10^4\ cm^2/m^2 \div 1{,}000\ cm^2$) with other factors being equal. This approach is deemed to be adequately conservative as it retains the full range of exposure pathways (i.e., external, inhalation, and incidental ingestion) that workers would encounter while using hand tools and other small items. Accordingly, the dose multiplier for hand tools and small items is assumed to be one as well.

The results are presented in Tables D-8 and D-9. A review of the results indicates that for large equipment, collective doses vary from less than 1 to about 150 person-rems. For small equipment collective doses range from less than 1 to about 160 person-rems. At a release dose of 1 mrem/year, collective doses are about 15 person-rems for large equipment and 16 person-rems for small equipment.

The reason for the small differences in doses is the influence of competing factors. The competing factors include the useful life of equipment, 14 years for large equipment and 3 years small equipment; number of workers assumed to use released equipment, 19,000 for large equipment and 56,000 for small equipment; and assumed average residual surface radioactivity profiles, 140 dpm/100 cm^2 for large equipment and 1,600 dpm/100 cm^2 for small equipment. A shorter useful life tends to result in lower collective doses, a higher number of worker yields higher collective doses, and higher residual radioactivity profiles result in increased collective

Table D-8 Collective Doses Associated with the Reuse of Large Equipment From Nuclear Power Reactors

Reg. Dose Option	Assumed Criteria Profile (dpm/100 cm^2)		Collective Doses (person-rem)
	Mean	Max	Mean
IAEA RS-G-1.7	530	4,200	56
10 mrem/yr	1,400	11,000	150
1 mrem/yr	140	1,100	15
No action/Status quo	630	5,000	66
0.1 mrem/yr	14	110	1.5
0.03 mrem/yr	4.2	34	0.4

Note: Mean and max profile based on SC&A 2003, Table 4.4, p.4-8 and 4-9.
Collective doses are expressed only as averages since collective doses reflect impacts to the expected population of workers and not to the average member of the critical group.
For the "no action" case, the regulatory dose is assumed to be the same as the 1 mrem/yr option.
The IAEA volumetric criteria were converted to surficial limits using a mass-to-surface ratio of 5 g/cm^2.

Table D-9 Collective Doses Associated with the Reuse of Small Equipment From Nuclear Power Reactors

Reg. Dose Option	Assumed Criteria Profile (dpm/100 cm^2)		Collective Doses (person-rem)
	Mean	Max	Mean
IAEA RS-G-1.7	530	4,200	5
10 mrem/yr	16,000	130,000	160
1 mrem/yr	1,600	13,000	16
No action/Status quo	630	5,000	6
0.1 mrem/yr	160	1,300	1.6
0.03 mrem/yr	49	390	0.5

Note: Mean and max profile based on SC&A 2003, Table 4.4, p.4-8 and 4-9.
Collective doses are expressed only as averages since collective doses reflect impacts to the expected population of workers and not to the average member of the critical group.
For the "no action" case, the regulatory dose is assumed to be the same as the 1 mrem/yr option.
The IAEA volumetric criteria were converted to surficial limits using a mass-to-surface ratio of 5 g/cm^2.

doses. Another reason for differences is that collective doses, based on IAEA and NUREG-1640 clearance and release levels, reflect different analytical approaches. The IAEA clearance levels are based on a deterministic analysis, while NUREG-1640 release levels were developed using a Monte Carlo method.

As was noted earlier, a simplified approach was used in assessing collective doses and several assumptions were made without the benefit of supporting information. As a result, the collective dose estimates incorporate some uncertainties. The uncertainties are associated with the characterization of practices involving the release and reuse of equipment; the types of equipment that may be released *versus* those may be discarded as radioactive waste; physical features of released equipment and small items; how equipment are used after having being released; variation in the distribution of radionuclides and relative mix and their combined effect on the total inventory of residual radioactivity; and size of the workforce postulated to use equipment that have been released. In all cases, values were assigned to parameters using engineering judgement without the benefit of supporting data from licensees. Finally, the assessment focuses on the nuclear power industry because of the larger workforce and greater amounts of equipment being released and reused. However, it is recognized that the reuse of released equipment and materials is occurring in other industrial sectors, but the amounts of materials subject to release and associated workforce are expected to be smaller than that of the nuclear power industry. Accordingly, the collective doses estimated in this analysis are assumed to be bounding, even though there may be isolated differences in some instances, such as radionuclide distribution, type of equipment, and size of the workforce, among others.

13. Collective Doses for Trash Incineration Workers

As with prior assessments, a scoping analysis was performed to assess collective doses to incinerator workers processing trash released from nuclear power reactors. The amounts of trash and levels of residual radioactivity reflect estimates associated with each regulatory dose option described earlier. Collective doses were estimated using the following general expression:

$$D_{iw} = E_{iw} C_i L T W K$$

where:

D_{iw}	=	collective dose due to incineration, person-rem
E_{iw}	=	dose rate during handling, mrem/yr per pCi/g, summed over all nuclides
C_i	=	trash gross activity concentration, pCi/g
L	=	trash labor productivity rate, person-hours per ton of trash
T	=	tonnage of trash, metric tons
W	=	work hours per year, 2,000 hours
K	=	conversion factor, 10^{-3} rem per mrem

The total trash tonnage and concentrations are based on data presented in Table 4.4 of the SC&A report (SC&A 2003). The productivity factor to process trash is based on a labor rate of 0.5 person-hour per ton, assuming an incinerator with an average design capacity of 500 tons per day (NRC 1984). The levels of effort to process trash were estimated to range from 7,000 to 33,000 person-hours. The dose factor is estimated to be 2.44×10^{-4} mrem/year per pCi/g, assuming the combined presence of 17 radionuclides, as beta, gamma, and alpha emitters (SC&A 2003, NRC 2005c). The presence of these radionuclides reflects a specific mix based on nuclear utility data. Four radionuclides make up most (about 83 percent) of the activity assumed to be present in trash; they are, in decreasing order, Co-60, Fe-55, Cs-137, and Cs-134. The dose factor assumes

2,000 hours per work year. The dose factor includes various functions, such as handling of the trash, loading the incinerator with trash, and during routine servicing or maintenance. The gross activity concentrations of the trash define two conditions, mean and maximum. The conditions reflect a distribution of trash concentrations truncated at the upper end by activity levels defined for each regulatory dose option. Activity levels above these are assumed to be out of the realm of possible release since higher concentrations would warrant classifying trash as low-level radioactive waste. The mean concentration is assumed to represent a trash concentration within the distribution defined at its lower bound by essentially non-detectable levels and the regulatory dose option for upper activity levels. The maximum concentration represents the upper end of the distribution of activity levels, as defined for each regulatory dose option.

The results are presented in Table D-10. The results indicate that collective doses are low, expected to be less than 0.03 person-rem for the 10 mrem/year dose option. All other collective doses are lower by orders of magnitude. Collective doses are expected to vary because of several factors. As described earlier, the amounts of releaseed trash are expected to vary, both among power plants and as a function of time. Similarly, the levels of residual radioactivity and the associated mix of radionuclides will vary as well. For example, the spectrum of radionuclides associated with a major plant outage is different than that found during routine operations. Moreover, the handling and processing rates of trash at incinerators are anticipated to differ, thereby yielding working conditions that might differ from that assumed here in deriving dose factors. Finally, this analysis assumes that all trash generated by power plants would be incinerated, while this is not expected in practice since not all landfills use incineration as a precursor to disposal. For example, only landfills servicing large metropolitan centers are expected to use incineration. For rural areas, trash is typically buried as there may not be enough of a trash volume to warrant the use of incineration. It is expected that these variations would negate one another, thereby leading to conditions where concentrations might be higher, but are associated with smaller quantities of trash. Accordingly, it is expected that the collective doses estimated in this analysis represent central estimates, while recognizing that at times doses may be slightly lower or higher depending on specific conditions.

Table D-10 Collective Doses of Incinerator Workers Processing Trash from Nuclear Power Reactors

Reg. Dose Option	Trash Tonnage (metric tons)	Mean and Maximum Trash Gross Concentration (pCi/g)		Person hours	Collective Doses (person-rem)
		Mean	Max		Mean
10 mrem/year	66,000	7,825	41,604	33,000	3.2 E-02
1 mrem/year	41,000	898	4,160	20,500	2.3 E-03
No action	20,000	114	382	10,000	1.4 E-04
0.1 mrem/year	21,000	121	416	10,500	1.6 E-04
0.03 mrem/year	14,000	43	125	7,000	3.7 E-05

Note: Trash tonnage and gross concentrations taken from SC&A 2003, Table 4.4.
Labor hours based on 0.5 person-hour per ton and an average incinerator design capacity of 500 tons per day. The trash processing rate is taken from NUREG/CR-3585, Table C-4.

[this page intentionally left blank]