



Battelle

Pacific Northwest Laboratories
Battelle Boulevard
P.O. Box 999
Richland, Washington 99352
Telephone (509)

375-2626

June 3, 1994

Mr. Steven L. Baggett, Section Chief
Mail Stop 8-F-5
Sealed Source Safety Section
Division of Industrial and Medical Nuclear Safety
U.S. Nuclear Regulatory Commission
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852

Dear Mr. Baggett,

JOB CODE L2536: TASK 7, FINAL REPORT

Enclosed is the final technical letter report for Task 7 of the project "Review of Improper Transfer/Disposal Scenarios for Generally Licensed Devices Study." Task 7 is entitled "Final Review of the 1987 Report by Oak Ridge Associated Universities, 'Improper Transfer/Disposal Scenarios for Generally Licensed Devices.'" A separate copy has been sent to Mr. Huang by overnight express.

In attempting to perform a probabilistic risk analysis as requested in your letter of March 30, 1994, for a given scenario (namely, ^{137}Cs gamma gauges), we found that there were significant gaps in the data and that the numbers of devices in service include many with activities greater than those included in the ORAU Report's scope of work (that is, limited to 20 mCi or less). We also found that incorporating realistic scenarios (e.g., different fractions-taken-in and different time-and-proximity factors for each individual in multiple-person incidents) was more difficult than the remaining funds would allow us to complete. For these reasons, we have included in Appendix E the dose equivalent values for "reference" 20 mCi sources due to ingestion, inhalation, and external exposure for "reference worst-plausible-case" scenarios of fractions-taken-in of 10^{-4} and time-and-proximity factors of 1000 hours at one meter from an unshielded source.

There is, in my opinion, a significant amount of work to be done to improve the data quality for input to a risk analysis. The data in the NMSS General License Database System, at least in the printout we received, show some invalid values and other questionable entries (e.g., ^3H gamma gauges) as discussed in Appendix E, the revised Task 3 Report. Beyond the question of data validity, there is a need for detailed data on numbers of devices by source type (the Regulatory Guide 10.10 Device Code is not adequate), isotope(s), dates placed in service, activities, and design. These data should be available for each of the 500,000 or so sources now in use. One

Mr. Steven L. Baggett
June 3, 1994
Page 2

problem that occurs is the difference between the ORAU mandate of considering a subset of sources up to a maximum of 20 millicuries and the NMSS General License Database System, which includes all sources regardless of activity. This puts into question the validity of a probabilistic risk analysis based on these data.

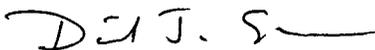
There is a continued difficulty of trying to match the ORAU Categories with the Device Codes of Regulatory Guide 10.10, which makes it difficult to use the ORAU analyses with the figures in the database summary you sent us. Our Task 3 review of the AEOD incident database should be re-done using the Regulatory Guide 10.10 Device Codes, if possible, to aid in using the rest of the available data.

A full risk analysis for generally-licensed devices should include a deeper analysis of which accident data to apply to which circumstances. If possible, it would be useful to subdivide risk analysis categories by source vulnerability, e.g., those sources designed to emit alpha particles as contrasted with those that are doubly-encapsulated in stainless steel and are, thus, "Special Form" as defined by the Department of Transportation.

Dr. Robin L. Hill and I feel that we have produced two very useful innovations in performing this project: the distributions of time-and-proximity factors and of fractions-taken-in. We would be pleased to use these results in future risk analysis work for the NRC.

It has been a pleasure working on this project. During our close-out period to meet our record-keeping requirements, please feel free to contact us for questions and comments.

Yours truly,



Daniel J. Strom, Ph.D., C.H.P.
Staff Scientist
Operational Health Physics Group
Health Protection Department

DJS:bga

Enclosures

cc: Mr. Daniel T. Huang, Mail Stop 8-A-23
File/LB

TECHNICAL LETTER REPORT:
TASK 7, FINAL REVIEW OF THE 1987 REPORT BY
OAK RIDGE ASSOCIATED UNIVERSITIES,
"IMPROPER TRANSFER/DISPOSAL SCENARIOS FOR
GENERALLY LICENSED DEVICES"

PNNL-11905

NRC JOB CODE L2536
PNL No. 20278

D. J. Strom¹
R. L. Hill¹
J. S. Dukelow²
G. R. Cicotte¹

¹Health Protection Department
²Nuclear Systems and Concepts Department
Pacific Northwest Laboratory
Richland, Washington 99352

June 3, 1994

Prepared for the U.S. Nuclear Regulatory Commission
Under Contract DE-AC06-76RLO 1830

1.0 EXECUTIVE SUMMARY

Task 7 of the project "Review of Improper Transfer/Disposal Scenarios for Generally Licensed Devices Study" requires that, "after Tasks 5 and 6 are completed, PNL will prepare a draft report based on the results of the individual reviews that documents its findings, conclusions and recommendations. The report will present a critical evaluation of the methods, data and assumptions used in the ORAU [Oak Ridge Associated Universities; now Oak Ridge Institute for Science and Engineering] study (Stabin et al. 1987). Any limitations or inadequacies will be described, as well valid insight and conclusions reached by the ORAU team. The report will specifically address the suitability of the 1987 ORAU report as a basis for revising regulatory requirements or guidance in 1993. It will provide the Staff with recommendations on how the contents of the ORAU report can be used in formulation of policy or regulation, as well as additional facts and information to support their decisions." "Within 2 months of receiving NRC staff comments on the draft report, PNL will respond to the comments and provide a final report to the Project Manager..."

As part of Task 5 of this project, Roger Cloutier, Kermit Paulson, Mike Stabin, and Evelyn Watson (the "ORAU Team") attended a meeting with Robin Hill, George Cicotte, Jim Dukelow, and Dan Strom of the Pacific Northwest Laboratory (PNL) on October 5, 1993, in Oak Ridge, Tennessee. The meeting was a very productive discussion with the original authors and team of the 1987 ORAU report. All PNL questions were satisfactorily resolved, as is documented in the Appendices to this final report.

The PNL reviewers conclude that the 1987 ORAU Report provides a good start on assessing worst-case consequences of improper transfer and disposal scenarios for generally licensed devices. For use as a basis for regulatory decision making, the principal shortcomings of the ORAU Report identified by the PNL review are:

- The 1987 ORAU Report does not include probabilities of the scenarios occurring on a per source, per year basis.
- The 1987 ORAU Report does not include a complete enough enumeration of the numbers of sources, and the distributions of source activities, in each category.
- The 1987 ORAU Report does not include the probabilistic distributions of outcomes (as opposed to the worst case outcomes) needed to realistically assess the probable human health consequences of such scenarios.

The original ORAU Team was constrained from considering many of the above issues by the limited scope of work of their project. The PNL reviewers repeat that the ORAU Report is a good start on the collection of data to support regulatory decision making.

As in the preliminary PNL review, the PNL reviewers conclude that the 1987 ORAU report is not an adequate basis for 1993 regulatory decision making, and that members of the NRC

Staff may need to use additional facts and information.

2.0 SUMMARY OF CHANGES SINCE 1987

Since 1987, there have been changes in several areas that impact the current relevance of the 1987 ORAU Report. These include changes in recommended limits on dose to the public, significant changes and developments in probabilistic dose assessment methodologies, and changes in dosimetric quantities and models that affect risk assessments.

2.1 CHANGES IN, AND CREATION OF, PUBLIC DOSE LIMITS

The original work under review in this project is referred to as the "ORAU Report" (Stabin et al. 1987). Since that report was prepared, NRC has instituted a dose limit for the public that is in concert with recommendations of radiation protection advisory groups.

For doses to the public (when such doses arise from licensee activities), the NRC has implemented a limit of 0.1 rem (0.001 Sv) total effective dose equivalent (TEDE) per year (10 CFR Part 20.1301(a)(1)). In addition, the NRC now specifies that the provisions of the U.S. Environmental Protection Agency's 40 CFR Part 190 apply to licensee activities.

Parenthetically, it is noted that the U.S. Department of Energy has lowered its limits for exposure to the public (10 CFR 835) to 0.1 rem (0.001 Sv) TEDE.

Since 1987, both the NCRP (1993) and the ICRP (1991) have lowered their recommended limits for members of the general public, to the same value of 0.1 rem (0.001 Sv) for the quantity *effective dose*, which is similar to TEDE although not identical. The NCRP and ICRP recommendations are based on new risk findings of the National Academy of Sciences (NAS 1988, 1990) and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR 1988).

With the establishment of a limit on TEDE to members of the public of 100 mrem (1 mSv; 10 CFR 20.1301(a)), or 2% of the occupational dose limit, there is less margin for error in dose assessments for improper transfer and disposal of generally licensed devices. The new, lower 10 CFR 20 limits for the public should be used in assessing impacts of improper transfer or disposal of generally licensed devices.

2.2 CHANGES IN PROBABILISTIC RISK METHODOLOGY

Since 1987, many changes have occurred in probabilistic risk methodologies. Recent summaries of these techniques are provided by IAEA (1989), Finkel (1990), and Morgan and Henrion (1990). In addition, the advent of user-friendly Monte Carlo simulation software for probabilistic health risk analysis, such as Crystal Ball (™Decisioneering, Inc., Denver, CO), makes it feasible to perform probabilistic risk assessments for this kind of work.

Distributions of source activities, and distributions of consequence severities, and probabilities of incidents occurring should be used to predict likely outcomes, with worst case outcomes being found in the extreme values of the resultant distributions.

2.3 CHANGES IN DOSIMETRIC QUANTITIES AND MODELS

Since 1987, quantities such as effective dose or total effective dose equivalent has replaced "dose equivalent" in most radiation protection recommendations and standards. Many improvements in internal dosimetry have occurred, as detailed in Appendix B.

3.0 THE ADEQUACY OF THE 1987 ORAU REPORT AS A BASIS FOR 1993 DECISION MAKING

The 1987 ORAU Report, in the judgment of the PNL reviewers, is no longer adequate for 1993 decision making. For use as a basis for regulatory decision making, the principal shortcomings of the ORAU Report identified by the PNL review are:

- The 1987 ORAU Report does not include probabilities of the scenarios occurring on a per source, per year basis.
- The 1987 ORAU Report does not include a complete enough enumeration of the numbers of sources, and the distributions of source activities, in each category.
- The 1987 ORAU Report does not include the probabilistic distributions of outcomes (as opposed to the worst case outcomes) needed to realistically assess the probable human health consequences of such scenarios.

The original ORAU Team was constrained from considering many of the above issues by the limited scope of work of their project. The PNL reviewers repeat that the ORAU Report is a good start on the collection of data to support regulatory decision making.

Additional shortcomings include

- changes in dose quantities (e.g., the introduction of effective dose equivalent from external irradiation) and in regulations (e.g., public limits on total effective dose equivalent) have taken place; and
- several potentially significant scenarios, such as an intact source out of a shield, and potentially significant consequences, such as doses to workers (rather than the public), have been omitted.

These topics are supported in individual reviewer comments in the Appendices to this report.

4.0 ADDITIONAL FACTS AND INFORMATION THAT MIGHT BE USED BY THE NRC STAFF TO DECIDE ON THE NEED FOR REGULATORY ACTION

As in the preliminary PNL review, the PNL reviewers conclude that the 1987 ORAU report is not an adequate basis for 1993 regulatory decision making, and that members of the NRC Staff may need to use additional facts and information.

4.1 AREAS IN WHICH ADDITIONAL FACTS AND INFORMATION ARE NEEDED

NRC Staff should consider

- the annual *rates* of incidents involving improper transfer or disposal by source category. Rate assessments require improved data (from both NRC-regulated and Agreement States) on numbers of sources, numbers of incidents, and dosimetric consequences (both individual and collective) of incidents;
- probability distributions of severity and of occurrence for various accident scenarios. There should be assessments of the magnitudes of doses and the sizes of the exposed populations that are likely to result from each instance of improper transfer or disposal of these devices. Such assessments can be based on historical incidents such as the Juarez, Mexico; Goiânia, Brazil; Korea-to-USA; and Indiana, Pennsylvania incidents. These distributions for various accident scenarios are needed to develop both the individual and collective dose estimates;
- a state-of-the-art probabilistic risk assessment with predictions of the impact of improper transfer and disposal scenarios on individual and collective TEDE. Such a risk assessment should include "worst case" scenarios only as limits of distributions on a probabilistic basis, not as simple point estimates;
- the impact of proposed changes in regulations on the benefits and economics of use of generally licensed devices, and on the reduction in risk to users and the public. Such assessments are optimization studies or regulatory impact analyses; and
- the need for much better data on numbers of devices. For a complete risk analysis, no *assumptions* should have to be made about the numbers of devices, their isotopes, their activities, Device Codes, design, and date placed into service: these numbers should be used directly or in categories with sufficient detail (e.g., 1600 ¹³⁷Cs gamma gauges of design XYZ placed in service at a rate of 100 per year beginning in 1978...) to perform the risk analysis.

4.2 NEW WORK BY PNL REVIEWERS FOR DISTRIBUTIONS OF ACCIDENT CONSEQUENCES

Pacific Northwest Laboratory reviewers have provided additional information in three areas.

4.2.1 Incident Rates on a Per Source, Per Year Basis

Appendix E contains the Revised Task 3 Technical Letter Report: Evaluation Of Historical Sealed Source Device Experience. This report incorporates additional data provided by NRC/NMSS on sealed source registrations. Incident rates, on a per source, per year basis, can be calculated as detailed in Appendix G of this report, using data from Appendix E of this report.

4.2.2 Proposed Probabilistic Framework for Risk Analysis

Appendix F contains the Revised Task 6 Technical Letter Report on the Development of Additional Probability and Risk Information. A proposed framework for risk analysis is included in that report. This framework could be used to support regulatory decision making. The decision makers would have to make judgements regarding acceptable levels, expressed in probabilistic terms, of collective effective dose equivalent, individual total effective dose equivalent, and individual local (or skin) dose equivalent for each category of sources analyzed.

For example, one category of source may have a once-in-20-years probability that a member of the public may receive a TEDE in excess of 0.1 rem from the practice of generally licensing such sources. Another category may have a higher probability, or even a virtual certainty, of one or more members of the public exceeding this 0.1 rem TEDE every year, but with a very large benefit to other members of the public. These are admittedly tough decisions, but, in the view of the PNL review team, the probabilistic information on which to base them should be made available to the decision makers.

4.2.3 Exposure Probabilities in Accidents

The revised Task 6 Report in Appendix F contains analyses of 42 historical accidents involving external doses to workers or the public, and gives distributions of Time-and-Proximity Factors for individual whole body, individual local (or skin), and collective doses.

Distributions of Time-and-Proximity Factors from historical accidents can be used in probabilistic risk analyses for both whole-body and local irradiation from external sources. An analysis of 42 accidents for which source identity and strength are available show that the average accident victim gets a whole body dose equal to that from being at 1 meter from the accident's unshielded source for an hour. The average accident is characterized by a value of 46 hours at a meter. In other words, the population-weighted average is about 1 hour at a meter, while the accident-weighted average is 46 hours at a meter. Clearly, the accidents with large numbers of victims (e.g., Goiânia and Juarez) dominate the former average. The maximum value seen for whole-body doses is about 700 hours at a meter (from the 1972 Texas child-abuse case). The average, geometric mean, and maximum values for local irradiation are 3100, 60 and 24,000 hours at a meter, respectively.

Such distributions should be used in probabilistic risk analyses to determine likely distributions of risks or doses from improper transfer and disposal scenarios for generally licensed devices.

Accidents that were terminated due to the appearance of clinical symptoms of acute irradiation have less value for risk analyses than accidents that were terminated by other means, or never terminated.

The current NRC Office of Analysis and Evaluation of Operational Data (AEOD) Nuclear Regulatory Event Report (NRRER) incident database does not contain the kinds of information needed to perform analysis of accidents for Time-and-Proximity Factors. It is recommended that the database either be modified to include this information, or a separate database be created. There is a great deal of work to be done to refine these preliminary analyses, extend them to additional accidents, and develop the logical framework for extrapolating to other kinds of sources and scenarios.

For intakes of radioactive materials, 60 historical accidents have been characterized by distributions of individual Fractions-Taken-In, that is, the fraction of the activity in the original source that was taken in by each individual involved in the accident. For most accidents involving radioactive sources, the fraction taken in is zero (0). For one accident, that at Goiânia, Brazil, in 1987, hundreds of persons had intakes, including 194 cleanup workers and at least 77 members of the public. The former were characterized by Fractions-Taken-In on the order of 10^{-12} , and the latter by Fractions-Taken-In averaging 5×10^{-6} . These values are far below the maximum possible value of 1, or of the value of 0.3 adopted as a worst case in the ORAU Report. In no accident has a value greater than 0.01 been seen.

5.0 PRELIMINARY RISK ANALYSIS

A full implementation of the risk analysis described in Section 2 is beyond the scope of work of the current project. However, a sample risk analysis for ^{137}Cs gamma gauges is given in Appendix G.

6.0 CONCLUSIONS

The PNL reviewers conclude that the 1987 ORAU Report provides a good start on assessing worst-case consequences of improper transfer and disposal scenarios for generally licensed devices. For use as a basis for regulatory decision making, the principal shortcomings of the ORAU Report identified by the PNL review are:

- The 1987 ORAU Report does not include probabilities of the scenarios occurring on a per source, per year basis.

- The 1987 ORAU Report does not include a complete enough enumeration of the numbers of sources, and the distributions of source activities, in each category.
- The 1987 ORAU Report does not include the probabilistic distributions of outcomes (as opposed to the worst case outcomes) needed to realistically assess the probable human health consequences of such scenarios.

The PNL reviewers conclude that the 1987 ORAU report is not an adequate basis for 1993 regulatory decision making, and that the NRC Staff should consider additional facts and information in the areas described above. In particular, the use of worst case scenarios with unrealistically high exposure factors tends to make the consequences of improper transfer and disposal seem worse than they would probably be.

An additional conclusion is that existing databases are not adequate for performing modern, probabilistic risk analyses. It would be desirable to collect and store in database format dosimetric information (including quantitative measurements or estimates of intakes) for all individuals involved in accidents. Such a data base would have to have an entry for each exposed individual, rather than simply one line of data per accident.

7.0 ACKNOWLEDGEMENTS

The PNL reviewers are pleased to acknowledge the help and support of numerous colleagues in the preparation of this report. At NRC headquarters, significant technical help and feedback has been given by Steven L. Baggett and Sterling Bell. The ORAU team of authors of the 1987 report, Roger Cloutier, Kermit Paulson, Mike Stabin, and Evelyn Watson, were very helpful in discussions held concerning that report. At PNL, original project manager James R. Jamison and technical contributor Peter C. Olsen have helped, as have technical reviewers Paul S. Stansbury and Eva Eckert Hickey. We are grateful to Betty Anderson and Rebecca Webster for clerical support.

8.0 REFERENCES

10 CFR 20. 1991. U. S. Nuclear Regulatory Commission (USNRC), "Standards for Protection Against Radiation." U.S. Code of Federal Regulations. 56 FR 23360-23474. Amended and corrected 56 FR 61352 (3 December 1991) and 57 FR 57877 (8 December 1992).

10 CFR 30. 1993. U.S. Nuclear Regulatory Commission (USNRC), "Rules of General Applicability to Domestic Licensing of Byproduct Material." U.S. Code of Federal Regulations, February 26, 1993.

10 CFR 31. 1992. U.S. Nuclear Regulatory Commission (USNRC), "General Domestic Licenses for Byproduct Material." U.S. Code of Federal Regulations, November 30, 1992.

10 CFR 32. 1992. U.S. Nuclear Regulatory Commission (USNRC), "Specific Domestic Licenses to Manufacture or Transfer Certain Items Containing Byproduct Material." U.S. Code of Federal Regulations, November 30, 1992.

10 CFR 835. 1993. U.S. Department of Energy (DOE), "Occupational Radiation Protection." U.S. Code of Federal Regulations. *Federal Register*, in press.

40 CFR 190. U.S. Environmental Protection Agency (EPA). U.S. Code of Federal Regulations.

Ayers, J. 1985. "General License Study - Analysis of Hazard." Internal memo to Division Director through channels, March 1985, attributed to J. Ayers. 8 pp. Washington, DC: Sealed Source Safety Section, U.S. Nuclear Regulatory Commission.

Baggett, S. 1987. *General License Study Report*. vii + 127 pp. Washington, DC: Sealed Source Safety Section, U.S. Nuclear Regulatory Commission.

Dean, C. M., M. S. Lawrence, and H. D. Lester. 1991. *Report on Survey of General Licensees Under 10 CFR 31.5*. NRC FIN D 2554-0. ICF Inc., Fairfax, Virginia.

Finkel, A. M. 1990. *Confronting Uncertainty in Risk Management. A Guide for Decision-Makers*. Center for Risk Management, Resources for the Future, Washington, DC.

International Atomic Energy Agency (IAEA). 1989. *Evaluating the Reliability of Predictions Made Using Environmental Transport Models*. IAEA Safety Series No. 100, IAEA, Vienna, Austria.

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

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

International Commission on Radiological Protection (ICRP). 1991. *1990 Recommendations of the International Commission on Radiological Protection*. ICRP Publication 60, Pergamon Press, Oxford.

Morgan, M. G. and M. Henrion. 1990. *Uncertainty. A Guide to Dealing with Uncertainty in Quantitative and Policy Risk Analysis*. Cambridge University Press, New York.

National Academy of Sciences (NAS), National Research Council. 1988. *Health Risks of Radon and Other Internally Deposited Alpha-Emitters: BEIR IV*. National Academy Press, Washington, DC.

National Academy of Sciences (NAS), National Research Council. 1990. *Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V*. National Academy Press; Committee on the Biological Effects of Ionizing Radiation, Washington, DC.

"ORAU Report:" see Stabin et al. 1987.

Stabin, M., K. Paulson, and S. Robinson. 1987. *Improper Transfer/Disposal Scenarios for Generally Licensed Devices*. "The ORAU Report" produced under NRC FIN B0299. Oak Ridge Associated Universities, Oak Ridge, Tennessee.

Unger, L. M., and D. K. Trubey. 1981. *Specific Gamma-Ray Dose Constants for Nuclides Important to Dosimetry and Radiological Assessment*. ORNL RSIC-45. United States Government Printing Office, Washington, DC.

United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). 1988. *Sources, Effects, and Risks of Ionizing Radiation*. United Nations Publications, New York.

APPENDIX A: FINAL REVIEW OF 1987 ORAU REPORT BY D.J. STROM,
INCORPORATING COMMENTS FROM VISIT TO ORAU TEAM, OCTOBER 4, 1993.

For this appendix, the following conventions are used:

- The original reviewer's comments made in the Task 4 draft letter report are shown in italics.
- The comments and results received during the meeting with the authors of the 1987 ORAU report are given in standard text format following specific Task 4 comments.

A.1. GENERAL COMMENTS

1. *Effective dose equivalent (EDE), committed effective dose equivalent (CEDE; which is used for internal doses), and total effective dose equivalent (TEDE) should be used, not simply dose equivalent.* The ORAU team was in agreement on this point.
2. *Which sources are normal form, and which are special form (IAEA/DOT classification)? This affects transport and fate of radionuclides in many accident scenarios. This should be determined for sources in future analyses. ORAU team did not consider this. However, Dodd et al. 1989 have considered this as important.*
3. *Probabilistic risk assessment methodologies have changed dramatically since 1987. This reviewer would suggest using making better use of historical accidents to evaluate probabilities of dose relationships for accidents. If risk is defined as (probability) \times (severity), then probability distributions for severity and for probability of occurrence should be given for various accident scenarios. There should be assessments of how likely a given scenario is to happen per source or device, and what kinds of doses to how many people are likely to result. Distributions of doses to individuals and the numbers of individuals receiving doses for various accident scenarios are needed to develop individual and collective dose estimates. The ORAU team made it explicitly clear that they had been requested not to consider the probabilities of accidents happening, but rather what the scenarios and consequences would be if an incident of improper transfer or disposal did occur. In our discussions with the ORAU team, they made it clear that they understood a risk analysis to consist of three parts: 1) the probability that an event will occur per source; 2) the number of sources; and 3) what the consequences are if an event does occur. They said that their statement of work limited them to the third part of this, even though they had addressed, to some extent, the second part.*
4. *Specific equations for the various models used are not present. This reviewer would prefer to see an equation for each dose that is arrived at. The ORAU Team thought this was a good idea.*

y denote the year of incident reporting and source use;
 $n(C,y)$ denote the number of incidents reported for that source category for that year;
 $N(C,y)$ denote the number of sources of category C in use in year y ; and
 $p(C)$ denote the fraction of incidents involving sources in a given category that are reported (presumed to be independent of year).

Then

$$R(C, y_i) = \frac{n(C, y_i)}{N(C, y_i) p(C)}$$

and, letting \bar{R} denote the average rate,

$$\bar{R}(C) = \frac{1}{I} \sum_{i=1}^I R(C, y_i) .$$

To fully address the rates at which accidents occur in each category, it would be necessary to have values of the above variables by year for a period of time long enough to give confidence in the rates and their fluctuations. In particular, the variable $p(C)$ is problematic, since estimation of under-reporting is always difficult to ascertain.

In the 1987 ORAU report, it seems that the incidents reported represent an unknown fraction of incidents. Furthermore, it is not clear whether those incidents are for all sources, agreement state sources, what fraction of agreement states, since there may be some reporting bias. For example, data may not be available from all Agreement States or may not be available in the same level of detail. The degree of extrapolation needed to get rates should be specified. Reports probably underestimate incidents, and the underestimation is probably worst for smaller sources. Again, the ORAU team agreed that this was an important point, but that it was outside the scope of their work.

5. p4 ¶2 §2.3 line 2: Using the "largest source:" will result in a high-biased estimate of risk. Worst-case analysis was requested by NRC in 1987, according to the ORAU team.
6. p4 ¶2 §2.3: Between "intact, shielded" (Case 1) and "wide dispersal" (Case 2) there should also be "source intact but unshielded." The latter is a likely outcome, e.g., in the Indiana, PA ¹⁹²Ir incident and the bulk of the radiography incidents.

Discussions with the ORAU Team led us to create Table A-1 of exposure potentials for various scenarios. The question of whether the ingestion of an intact source is "internal" exposure is moot; ingestion may or may not result in uptake of radioactive material from the gastrointestinal tract.

Table A-1. Exposure potential for external, contact, and internal irradiation for seven general device and source scenarios.

Scenario	Exposure Potential		
	External	Contact	Internal
1. Source inside device	x		
2. Source out of device or shielding compromised (e.g., shutter open)	xx		
3. Source in device and source leaking	x		x
4. Source removed from device and source not leaking	xx	xx	
5. Source removed from device and source leaking	xx	xx	x
6. Source removed from device and source dispersed	xx	xx	xx
7. Source removed from device and intact source ingested	x	xx	x

The accidents involving the most people, such as the Juarez, Mexico accident in 1983-4 and the Goiânia, Brazil accident in 1987, have been in scenario 6. Many fatalities and acute radiation injuries have resulted from scenario 2 and 4 accidents.

7. *p4 ¶2 §2.3: Average dose equal to 1/2 maximum (NUREG 1980) does not agree with history. Doses are likely to be lognormally distributed (e.g., uranium in urine, occupational doses (UNSCEAR 1977), etc.) The 1/2 value was taken from Buckley et al. (1980).*
8. *p4 ¶5: "...population dose equivalents were not calculated." Collective dose equivalent can and should be calculated, using a probabilistic basis. While there is concern for the maximally-exposed individual, the overall detriment under a linear, no-threshold dose-response hypothesis is proportional to the collective dose equivalent. The size of the population to be used in collective dose calculations should be addressed. Collective dose was considered for incineration scenarios.*
9. *p4 ¶6: "Gamma ray dose constants were taken from Unger and Trubey..." All work should be in EDE rate constants; see attached. For doses derived from values in roentgens, Γ values in roentgens should be used for calculating. If new risk factors (e.g., ICRP 1991, UNSCEAR 1988, NAS 1990) are used, then appropriate quantities and units should be used. This may make a significant difference for external*

exposures to some low energy photon emitters.

10. pp4-5: Mexican ^{60}Co Accident (Andrews 1963): *The most highly exposed boy's exposure corresponded to 687 hours at 1 meter, a high very time-and-proximity factor, with average being 446, geometric mean 408, GSD 1.7. However, the ORAU Report assumption (Exec. Summary) of 20 weeks at 100 cm (1 m) corresponds to 3360 h at a meter. (Note: Generalization of high-dose accidents is limited because of the censorship that occurs due to fatality. Presumably, with no fatalities, the source would be there for a very long time.) See TASK 6, development of additional probability and risk information. Both whole body exposures and localized (often extremity) exposures should be considered in a risk analysis, since the latter may result in deterministic (formerly non-stochastic) effects even though the former may not.*

11. p6 ¶2: *"These values are given with the qualification that the listed values must be increased by 25-45 percent to account for electron production in the stainless steel walls assumed to encapsulate the source." This "buildup" is not founded in either experiment or theory. To verify that there is no need for this correction, one could perform Monte Carlo calculations using the MCNP code for photon emissions from stainless steel encapsulated sources of ^{137}Cs , ^{60}Co , and ^{192}Ir . Indeed, the correction that is needed is one to effective dose equivalent rather than air kerma ("exposure is the ionization equivalent of collision kerma in air" - Attix, 1980) or "free-air dose." Table A-2 shows the percent (effective dose equivalent)/(ambient dose equivalent) [i.e., $\text{EDE}/\text{H}^*(10)$] for various nuclides, expressed as a %, by irradiation geometry.*

Table A-2. EDE/H*(10) by nuclide and irradiation geometry using ICRP 51 conversions of photon fluence vs. energy to EDE and H*(10) (Strom, 1993).

Nuclide	(AP)	(PA)	(ROT)	(ISO)	(LAT)
Cd-109	33.2%	14.6%	16.4%	13.6%	8.4%
I-125	42.6%	22.0%	21.9%	18.0%	12.4%
Xe-133	61.2%	40.2%	36.0%	29.3%	23.9%
Tl-201	72.6%	52.5%	46.1%	37.7%	33.2%
Co-57	78.9%	61.6%	53.2%	42.9%	38.8%
Tc-99m	81.4%	64.1%	55.3%	44.9%	40.5%
I-131	86.8%	73.7%	64.9%	54.4%	51.7%
Ir-192	87.3%	74.1%	65.3%	54.6%	51.9%
Ra-226	87.5%	78.6%	70.7%	61.7%	60.9%
Cs-137	87.8%	77.3%	69.0%	58.7%	57.2%
Al-26	89.7%	81.9%	73.7%	64.8%	64.5%
Co-60	90.0%	82.8%	74.7%	65.8%	65.7%
Na-24	91.6%	86.3%	78.5%	71.0%	72.1%

12. p5 ¶3: Maximum contact time of 3 hours: where did this "hypothesis" come from? ORAU Team consensus arrived at this value.
13. p5 ¶4: The use of internal dose assessments based on committed effective dose equivalent methods of ICRP 30 is good. No comment.
14. pp5-6: Brodsky was not cited (Brodsky, A. 1980. Resuspension Factors and Probabilities of Intake of Material in Process (or "Is 10^{-6} a Magic Number in Health Physics?"). Health Phys. 39(6):992-1000.). Intake should depend on mass involved. Intakes of significant masses are not plausible (e.g., for ^{238}U and ^{232}Th). This is addressed above under worst case scenarios.
15. p6 §2.3.2: An important scenario is missing, namely, the scenario of an intact, but unshielded, source. This has happened repeatedly in industrial radiography settings. See Table A-2, above.
16. p6 last ¶: The value of "5 devices per year" needs clarification. Is 5 devices per year of each kind or 5 devices per year of all kinds? An ORAU Team consensus.
17. p7 ll: There is no justification for the value of a nearby population of 73,000.

Buckley et al., p. D-31.

18. *p9 ¶2: average dose half as large as maximum dose? Unjustified based on accident histories. Review NUREG 1980. Buckley et al.*
19. *p14 Incineration studies: this seems incomplete, but may be adequate. Some incineration references are missing, e.g., Hamrick'a & Watson's work. There are doubtless other NTIS reports or DOE reports, in addition to the refereed literature. Jim Tripodes, who has hosted the recent incineration conferences, should be contacted. The question to be addressed is releases from low-tech incinerators, not incinerators with high-tech, scrubbed effluents. The ORAU team did a fairly thorough search.*

In conclusion, many of the major shortcomings that this reviewer found in the ORAU Report stemmed from the assumption that it was to be a complete risk analysis, when in fact it was a worst-case analysis starting from the assumption that a device had already gotten out of control. Other issues, as discussed above, remain unresolved.

APPENDIX B: R.L. HILL'S REVIEW OF THE 1987 ORAU REPORT,
"IMPROPER TRANSFER/DISPOSAL SCENARIOS
FOR GENERALLY LICENSED DEVICES"

For this appendix, the following conventions are used:

- The original reviewer's comments made in the Task 4 draft letter report are shown in italics.
- The comments and results received during the meeting with the authors of the 1987 ORAU report are given in standard text format following specific Task 4 comments. Major conclusions from this review are given in the subsection following the specific comments.
- New information pertaining to aspects related to internal dosimetry are given in a separate section at the end of this appendix.

B.1 GENERAL COMMENTS

G1. *This document should be a complete, stand-alone document where the methods are fully described in the document. As it stands now, NUREG/CR-1775 is referred to in order to obtain descriptions of many of the methods used in the study. The assumptions used from NUREG/CR-1775 for determination of external dose should be reviewed for appropriateness and consistency. For instance, on Page 4 of the NUREG/CR report, it is stated that a point source is assumed, while in Appendix A, a line source is assumed. Also, in Appendix A of the NUREG/CR report, the internal doses are calculated using dose conversion factors based on ICRP 2 methodology, which is not the case for the GLD study.*

The authors of the 1987 ORAU report indicated that, while NUREG/CR-1775 was heavily relied upon, the internal dose calculations were done using ICRP 30 methodology.

G2. *It appears that a lot of "short-cuts" were taken in this study, i.e., referring to another document for methods and not including worker dose calculations, food pathways, or intruder-type scenarios. A more in depth analysis is needed in order to encompass all probable scenarios that may lead to a public dose.*

The authors of the 1987 ORAU report indicated that they include the worker doses as part of the doses to members of the public.

APPENDIX B: R.L. HILL'S REVIEW OF THE 1987 ORAU REPORT,
"IMPROPER TRANSFER/DISPOSAL SCENARIOS
FOR GENERALLY LICENSED DEVICES"

For this appendix, the following conventions are used:

- The original reviewer's comments made in the Task 4 draft letter report are shown in italics.
- The comments and results received during the meeting with the authors of the 1987 ORAU report are given in standard text format following specific Task 4 comments. Major conclusions from this review are given in the subsection following the specific comments.
- New information pertaining to aspects related to internal dosimetry are given in a separate section at the end of this appendix.

B.1 GENERAL COMMENTS

G1. This document should be a complete, stand-alone document where the methods are fully described in the document. As it stands now, NUREG/CR-1775 is referred to in order to obtain descriptions of many of the methods used in the study. The assumptions used from NUREG/CR-1775 for determination of external dose should be reviewed for appropriateness and consistency. For instance, on Page 4 of the NUREG/CR report, it is stated that a point source is assumed, while in Appendix A, a line source is assumed. Also, in Appendix A of the NUREG/CR report, the internal doses are calculated using dose conversion factors based on ICRP 2 methodology, which is not the case for the GLD study.

The authors of the 1987 ORAU report indicated that, while NUREG/CR-1775 was heavily relied upon, the internal dose calculations were done using ICRP 30 methodology.

G2. It appears that a lot of "short-cuts" were taken in this study, i.e., referring to another document for methods and not including worker dose calculations, food pathways, or intruder-type scenarios. A more in depth analysis is needed in order to encompass all probable scenarios that may lead to a public dose.

The authors of the 1987 ORAU report indicated that they include the worker doses as part of the doses to members of the public.

- G3. *Better presentation and summary of the resulting dose estimates are needed. It would be much easier to reach conclusions on the study if all results were located in one location, such as a collection of several summary tables.*

The ORAU authors agreed that a better summary of the data would greatly enhance the readability of the report.

- G4. *The difference between MEI and average individual, and maximum individual and realistic individual needs to be better defined.*

The ORAU authors agreed that more realistic values for the MEI and average individual doses need to be used.

- G5. *Two important pathways/scenarios missing from this study that may have potentially large impacts on the reported dose estimates are food pathways and worker scenarios.*

The authors of the 1987 ORAU report indicated that the worker doses were lumped in as part of the public doses; and that, since they were not experts in using the dispersion and groundwater models, they relied heavily with methods used in NUREG/CR-1775, which did not include food pathways.

- G6. *Better estimates for parameters used to model incineration are needed.*

The authors of the 1987 ORAU report indicated that at the time the report was prepared, most, if not all, of the available literature on the topic was based on incineration of medical wastes. They used what information could be obtained from the available literature.

- G7. *More up-to-date computer codes are available for modeling the doses from plumes, incineration and landfills (i.e., GENII, GENII-S, CAP88-PC, etc.).*

The authors of the 1987 ORAU report they realize that they were not experts in using the dispersion and groundwater models, and that they probably did not used the most up-to-date models for the 1987 time frame. They stated that now they realize the project would have benefitted from having specialists in these areas involved on the project.

- G8. *An uncertainty and sensitivity analysis is needed since the values for most parameters used in this modeling exercise have wide or largely unknown ranges.*

The authors of the 1987 ORAU report stated that they were not asked to perform an uncertainty and sensitivity analysis for that report, and they probably could not have done such an analysis with the data and analytical tools available when the report was being prepared.

B.2. SPECIFIC COMMENTS

S1. Page i, last ¶: *It appears from the information presented here, that the four cases considered in this report for internal doses are 30% intake for inhalation, 30% intake for ingestion, inhalation after incineration, and ingestion after leached from a landfill. The fractional intake values need to be substantiated.*

The authors of the 1987 ORAU report stated that the 30% values was taken as an educated guess or a benchmark value. However, they could not provide data to support the selected value.

S2. Page ii, last ¶: *A metal recycling scenario was only considered for ^{60}Co . For a more realistic assessment, any of the GLD materials considered in this study could possibly find their way to a melter. Thus, all nuclides involved for GLDs should be considered in the dose assessments.*

The authors of the 1987 ORAU report agreed that the portion of the report dealing with recycling should be updated.

S3. Page ii, last ¶: *A justification (i.e., calculated dose estimates) is needed for the statement that for metal recycle, "...dose equivalents received by members of the general public who purchase contaminated products would most likely not exceed 500 mrem/yr (0.005 Sv/yr) in most cases". This might not be the case since in IAEA Safety Series 111-P-1.1 (IAEA 1992), where generic exposure and pathway analyses were used for recycle of steel, a limiting dose of $8.8\text{E-}5$ Sv per Bq/g in the scrap (0.33 mrem per pCi/g) was estimated for ^{60}Co .*

No response was given for this comment.

S4. Page 1, 4th ¶: *It is stated that "potential" scenarios are developed and assessments provided for "realistic" and "maximum dose equivalent to individuals" (MEI). However, in the current regulatory environment (for the NRC), rulemaking is not necessarily based on "worst-case" but rather on "prudently conservative" assumptions for the dose assessments. To this end, both deterministic assessments (based on the most realistic data available from literature, etc.) and stochastic uncertainty and sensitivity analyses are needed. The Latin Hypercube Monte Carlo method for uncertainty and sensitivity (U&S) analyses was published in 1964 by staff at the Sandia National Laboratory and is a possible approach to performing U&S analyses.*

B.2. SPECIFIC COMMENTS

S1. Page i, last ¶: *It appears from the information presented here, that the four cases considered in this report for internal doses are 30% intake for inhalation, 30% intake for ingestion, inhalation after incineration, and ingestion after leached from a landfill. The fractional intake values need to be substantiated.*

The authors of the 1987 ORAU report stated that the 30% values was taken as an educated guess or a benchmark value. However, they could not provide data to support the selected value.

S2. Page ii, last ¶: *A metal recycling scenario was only considered for ^{60}Co . For a more realistic assessment, any of the GLD materials considered in this study could possibly find their way to a melter. Thus, all nuclides involved for GLDs should be considered in the dose assessments.*

The authors of the 1987 ORAU report agreed that the portion of the report dealing with recycling should be updated.

S3. Page ii, last ¶: *A justification (i.e., calculated dose estimates) is needed for the statement that for metal recycle, "...dose equivalents received by members of the general public who purchase contaminated products would most likely not exceed 500 mrem/yr (0.005 Sv/yr) in most cases". This might not be the case since in IAEA Safety Series 111-P-1.1 (IAEA 1992), where generic exposure and pathway analyses were used for recycle of steel, a limiting dose of $8.8\text{E-}5$ Sv per Bq/g in the scrap (0.33 mrem per pCi/g) was estimated for ^{60}Co .*

No response was given for this comment.

S4. Page 1, 4th ¶: *It is stated that "potential" scenarios are developed and assessments provided for "realistic" and "maximum dose equivalent to individuals" (MEI). However, in the current regulatory environment (for the NRC), rulemaking is not necessarily based on "worst-case" but rather on "prudently conservative" assumptions for the dose assessments. To this end, both deterministic assessments (based on the most realistic data available from literature, etc.) and stochastic uncertainty and sensitivity analyses are needed. The Latin Hypercube Monte Carlo method for uncertainty and sensitivity (U&S) analyses was published in 1964 by staff at the Sandia National Laboratory and is a possible approach to performing U&S analyses.*

The authors of the 1987 ORAU report stated that they were not asked to perform an uncertainty and sensitivity analysis for that report, and they probably could not have done such an analysis with the data and analytical tools available when the report was being prepared.

- S5. Page 3, 1st ¶: *The radiotoxicity classification from the 1973 IAEA document cited is perhaps out of date. More recently, the IAEA has used the following classification which may be substituted: alpha emitters, photon emitters with large external dose conversion factors (DCF), no-photon emission with moderate internal DCFs, and other low dose radionuclides (IAEA Safety Series 111-P-1.1, 1992).*

The authors of the 1987 ORAU report stated that they used this radiotoxicity classification to narrow down from 600 categories to 16. It was used because the authors were familiar with it at the time the report was being prepared.

- S6. Page 3, 3rd ¶: *The probabilities assigned were "educated guesses". This brings up another reason to perform an uncertainty and sensitivity analysis (U&S) for the assessment. The upper and lower bounds for parameter values can be researched and used in the U&S analysis. This way, a range of possible dose estimates can be reported instead of point estimates.*

See response to comment S4.

- S7. Page 4, Sect.2.3: *For the dose assessments, only 'intact device' and 'dispersed device over wide areas' were considered. Shouldn't damaged device with limited dispersion be considered since the concentrations available for intake or exposure would be higher for limited dispersion (i.e., less dilution)?*

The authors of the 1987 ORAU report agreed with this assessment.

- S8. Page 4, 4th ¶: *Since only several people are assumed to be exposed to intact devices, no population doses are calculated. Is this valid, or should population dose be considered?*

The authors of the 1987 ORAU report contend that contact with an 'intact' source would be limited to a very small number of people. However, as shown in the appendix to the report that deals with additional risk information, data from actual accidents of mishandled sources indicate that a great many people can be exposed from an 'intact' source. (See Tables 3.3.1 and 3.3.2. of that appendix.)

S9. Page 5, last ¶: *It is very hard to justify the intake fractions assumed in this report. Using DOT-based scenarios as a source of these values is not entirely appropriate. More recent incidents or accidents involving improperly handling or disposal of specifically licensed sealed sources can be used to update the "realistic" parameter values used.*

See response to S1, and the internal dosimetry section of the additional risk information in Appendix B.

S10. Page 6, 2.3.2: *For extensively damaged devices with wide dispersion and many people involved, the following scenarios were described: Incinerator with release to the environment, incinerator with subsequent burial in landfill, metal recycling into consumer products, and metal recycling into construction materials. One very significant set of dose scenarios that were left out of this study were the dose to workers at the incinerators or melters. Generic exposure and pathway analyses performed by PNL for US DOE for incineration of hazardous materials and for smelting associated with recycling and reuse of contaminated materials have shown that, in many cases, the limiting scenarios and doses were found to be those specified for workers.*

The ORAU authors stated that worker doses were considered as and lumped in with the doses to members of the public.

S11. Page 6, 2.3.2, ¶ *Information on incineration from NUREG/CR-1775 was used as last basis for this study. Assumptions that are needed are stated to be (1) the number of devices incinerated per year, (2) the fraction of activity released by the incineration process, and (3) the fraction of activity released that escapes into stack emissions. Five devices incinerated per year was chosen as the value used in this study. All other dose calculations in the report are based on one device. The assumed values taken for the three major parameters need to be justified, especially the "arbitrarily chosen" value of 5 devices incinerated per year. Also, several parameter values that may have a major impact on the resulting dose estimates that were not discussed are the throughput of the incinerator, the feed composition, the incineration temperature, and the final population size considered.*

The arbitrarily chosen value of 5 devices/year was chosen only as a benchmark value for this study. No data to support this selection was given.

S12. Page 6, 2.3.2, ¶ *Dose calculation methods similar to those from NUREG/CR-1775 last were used in this study. The methods stated in NUREG/CR-1775 do not state that food pathways were considered in the analysis of landfills, thus it is assumed that they were not considered in this study either. Other studies have calculated estimates of doses to the public from radionuclides present in landfills. In some cases, the highest potential doses were found to result from intruder scenarios and well drilling scenarios some years after a landfill has closed. Both external and internal doses result from scenarios describing a residence being built on a landfill, where excavation is used for a basement and the soil is spread out over the yard and used for residential gardens. Doses can also result when a core is brought up from well-drilling on the site and dispersed on the surface. For large scale landfill sites, animal pathways may be needed (i.e., radionuclide uptake by plants, plants eaten by animals, animals used as food for humans living nearby).*

More recent computer codes can be used to perform the exposure and pathway analyses. For instance, the GENII code and its many modules have been used extensively for dose calculations using many different types of scenarios associated with landfills and incinerators. External and internal doses can be obtained; air, water, and plant and animal food pathways are included. The GENII-S code incorporates a Latin Hypercube uncertainty and sensitivity analysis capability into the GENII code. The EPA's code, CAP88-PC, can be used for modeling the transport and fate of radioactive air emissions and doses to the public surrounding incinerators. This code also includes the food pathways in the dose calculations. Use of such codes should provide more "realistic" dose estimates.

The authors of the 1987 ORAU report they realize that they were not experts in using the dispersion and groundwater models, and that they probably did not used the most up-to-date models for the 1987 time frame. They stated that now they realize the project would have benefitted from having specialists in these areas involved on the project.

S13. Page 7, 2nd ¶: *The incineration of medical wastes was used to select values for the last two parameters described in the comment above. The incineration process used for medical wastes may be quite different than that used for municipal wastes (i.e., which is a more likely occurrence for GLDs). Different fractions of the radionuclides can be release through stack emissions under the different incineration conditions. Thus, more research into appropriate partitioning values is needed.*

The IAEA (IAEA Safety Series 111-P-1.1 [IAEA 1992]) has taken one

approach to this problem of lack of knowledge about specific radionuclide partitioning by conservatively assuming that 100% of the initial radioactivity is retained in all three resultant phases. For incinerator scenarios, these phases are slag, fly ash, and flue gases in stack emissions. This triple accounting approach will maximize the potential importance of the scenarios associated with each possibility. A similar approach should be considered as a possible way to estimate the effect of radionuclide partitioning when no or limited partitioning data is available.

A different approach that may be used is a slight variation of the IAEA method. An assumption is made that 100% of the radionuclides will end up in each of the three resulting incinerator phases. But, instead of summing the dose results from all three resultant phases, only the maximum dose for a given resultant phase is chosen and reported.

The ORAU authors stated that most of the literature available at the time the report was prepared dealt with medical incinerators. Thus, the only parameter values they felt comfortable in using were taken from those pertaining to the available literature.

S14. Page 7, last ¶: A metal recycling scenario was only considered for ⁶⁰Co. However, for a more realistic assessment, any of the materials could possibly find their way to a melter. Thus, all nuclides involved for GLDs should be considered in the dose assessments.

The ORAU authors agreed that the recycling section of the 1987 ORAU report needs to be updated.

S15. Page 8, 2nd ¶: Inhalation of suspended particles is not considered as an exposure pathway for the landfill burial scenario. As mentioned above in other comments, this scenarios should be included for both worker scenarios, intruder scenarios, and well-drilling scenarios. For these scenarios, inhalation may be the major exposure pathway leading to internal doses for some of the radionuclides considered.

The ORAU authors contend that they relied heavily on NUREG/CR-1775 as a general basis for their report. Since this particular exposure scenario was not include in that report, they did not feel in necessary to include it in the 1987 ORAU report.

S16. Page 8, last ¶: Values for leach rates from landfills are given. More up-to-date information on nuclide-specific leach rates needs to be reviewed and

considered.

See response to S12.

S17. Page 9, 2nd ¶: It is arbitrarily assumed that the dose to the average individual is one-half that of the MEI. The difference in dose to the average individual versus the dose to the MEI should be dependent on the radionuclide, its half-life, the uptake pathways considered, hold-up time through the scenarios, time in a given location, amount of a given food eaten per year, etc., and generally is not a constant number as assumed. A more in-depth development of scenarios and pathways is needed in this study.

The ORAU authors agreed that the conversion to average individual dose from MEI dose should be a factor of 1/10 or less.

S18. Page 30, last ¶: For internal dose calculations, a fraction of 0.3 (30%) was chosen as the maximum amount of the initial activity that can be taken in. Since this value can vary with radionuclide, the value(s) need to be validated and changed as needed.

See response to S1.

S19. Page 30, last ¶: It is stated that hydrogen gas is usually converted to tritiated water in the atmosphere, and thus is not considered in the dose calculations. Results of recent research indicate that this is not the case; elemental tritium gas (HT) is not readily converted to tritiated water in the atmosphere, but rather enzymes present in soil microbes are necessary for this conversion to readily occur. HT gas release experiments performed in Canada and France indicate that a significant portion of the dose resulting from a release of HT gas to the atmosphere is the secondary plume of tritiated water released from the soil after the primary plume of HT has contacted the soil. Also, a small portion of the HT gas that is inhaled is absorbed into the blood, and is converted to tritiated water by gut flora. The significant portion of the dose from inhalation of HT gas is associated with the tritiated water that is formed from the HT by this mechanism and should be considered. (NOTE: The ICRP has not yet accepted these items in its dose commitment scheme.)

Alternatively, a review of the paper "Maximum Permissible Amounts of Accidentally Released Tritium from an Environmental Experiment to Meet Dose Limits for Public Exposure", by Taeschner, Bunnenberg, and Gulden, in Fusion Technology, August 1991, reporting on the 1986

French experiments found that the authors contend that the kinetics of the reaction,



avored the formation of tritiated water (HTO) in all environments with sufficient mass of water available, including that found in moist air. Following an HT release under dry air conditions, the absorption of HT in the soil, conversion to HTO, and re-release to the air will dominate the overall exposure to tritium (the dose due to exposure to the initial HT plume will be several orders of magnitude lower). Under moist conditions, the overall exposure will consist of roughly equal parts exposure to the initial HTO plume (following rapid conversion of HT to HTO in the atmosphere) and exposure to HTO absorbed in the soil and reemitted to the atmosphere over the following few days.

No response was made by the ORAU authors on this comment.

S20. Page 32,
2nd ¶:

The "realistic individual" doses from tritium are calculated assuming uptake of 10%. This value needs to be researched and updated. This comments applies generally to all radionuclides considered in this study.

No response was made by the ORAU authors on this comment.

S21. Page 114:

No scenarios are considered for the class J devices with natural uranium or thorium. However, the doses associated with these two radionuclides could be significant and should be determined.

No response was made by the ORAU authors on this comment.

B.3. MAJOR CONCLUSIONS PERTAINING TO THE REVIEW OF THE 1987 ORAU REPORT

The meeting with the authors of the 1987 ORAU report proved to be a significant help in identifying the constraints under which the study was done. The report was designed to address only the development of scenarios in which the public could receive doses from the improper transfer or disposal of generally licensed devices. It was their understanding that they were not charged with the task of establishing the total number of reported incidents with these devices or with calculating doses specific to actual incidents involving these devices. As a result, we now understand that many of the parameter values in question during this review were only used as benchmark values by the ORAU authors with the knowledge that more definitive values would be needed for the next phase of the analysis.

One item not addressed above is that while the committed dose equivalents to organs and

tissues (the units used in the ORAU report) is important, doses should be reported in units of committed effective dose equivalent (CEDE) and total effective dose equivalent (TEDE = sum of external and internal doses). This would have made comparison of the resultant doses between the different devices much easier.

Even with these caveats, it is still very difficult to justify the use of 0.30 as the maximum fraction-taken-in value used in the 1987 ORAU report for calculation of internal doses. First, if only worst case scenarios were to be considered, the maximum fraction-taken-in should be 1.0, that is assume that all of the source is taken into the body. Second, as discussed in Appendix B, in the review of reports of over 60 actual accidents, in no case did an intake fraction ever reach the maximum 30% level used in the ORAU study. Actual observed values ranged from 2E-4 to 2E-8 for individuals involved in the accidents, and ranged from 7E-11 to 2E-15 for the Goiania cleanup workers (IAEA 1998b). If the doses for the various scenarios were recalculated using a defensible range of values for the fraction-taken-in, with all other parameter values the same, the CEDE and TEDE will be significantly lower than those that would result from using 0.30 value.

B.4. GENERAL LISTING OF NEW DEVELOPMENTS IN INTERNAL DOSIMETRY SINCE THE 1987 GLD REPORT

- *There still is no regulatory limit established for collective dose to either workers or the public.*
- *The revision to NRC 10CFR20 published in Federal Register and goes into effect January 1, 1994. This updates the methodological basis for the report dose limits from ICRP 2 (1959) to ICRP 26/30 (1977;1979). The dose limits in units of CEDE and TEDE.*
- *Age-dependence of doses is being looked at more closely*
- *Refinements to biokinetic models (eg. 1992 Leggett Am model, etc.) have been made*
- *New lung models are being developed by ICRP and NCRP subcommittees*
- *ICRP60/61 methodologies (1991) changed recommended dose limits to 20 mSv per year for occupational exposures and 1 mSv in a year for exposures to the public. The ICRP 26/30 limits were 50 mSv per year for occupational exposures.*

APPENDIX C: COMMENTS ON THE ORAU REPORT BY J. S. DUKELOW

Original Review Comments appear in the Appendix in italics. Information gained during the October 1993 visit to Oak Ridge to confer with the authors of the ORAU report appears interspersed in standard non-italic font. Additional relevant information gained since the October trip appears in a separate section at the end of this Appendix, also in non-italic font.

C.1. GENERAL COMMENTS

Page i -- The appropriate "worst case" assumption for exposure to the external radiation field probably ought to be direct contact of the encapsulated source with the body for some number of hours. This case is plausible and is likely to produce more severe consequences than the report's assumption of 20 weeks at 100 cm (particularly for alpha and beta sources).

The report assumes a "worst case" for ingestion as ingestion or inhalation of 30% of the radioactive. No justification is given for using 30%; an obvious "worse" than worst case is inhalation or ingestion of 100% of the radioactive material. Assuming 100% is not unreasonable if one remembers that some of the previous sealed source incidents involved sources falling into the hands of small children.

We still consider these comments to be reasonable. For many of the sealed source accidents described in the body of the present report, the victim "finding" the sealed source has put it in his pocket. For the ingestion cases, several of the accidents have involved small children finding and playing with sealed sources or sealed source material. A soft beta sealed source would require a delicate "shield window", which would be unlikely to survive stomach acid. On the other hand, the analysis provided in the text of the present report establishes that these worst case assumptions are not at all representative of exposures resulting in the known sealed source accidents.

Page 2 -- Some additional explanation is needed of the interior structure of Figure 1. For instance, one could argue that the "incinerator" block belongs on the downstream side of "salvage dealer" in addition to or instead of the upstream side. One of the things a salvage dealer might do with materials not deemed of interest for metal recovery is incineration. On the other hand, the pathway "trash handler to incinerator to salvage dealer" presumes that the trash handler sends to the salvaged dealer incinerator ashes containing intact or dispersed sources, all of which seems implausible.

Page 3 -- The "probabilities" described in Section 2.2 are properly called "conditional probabilities". Also, Section 2.2 ought to say that the probability of reaching a final state by way of a particular pathway is the product of all the pathway segment conditional probabilities and that the probability of reaching the final state is the sum of all those pathway probabilities.

The fact that the pathway segment conditional probabilities are all (or mostly) "educated

guess" deserves more emphasis, as well as some description of the basis used to make those educated guesses.

Page 5 -- There is a discussion of increasing published dose rates by 25-45 per cent to account for "electron production in the stainless steel walls" of the capsule (presumably Compton scattering, pair production, and photoelectric absorption). Is this the entire rationale or should it include: "bremsstrahlung radiation resulting from the deceleration of beta particles, electrons, and positrons in the stainless steel"?

The second paragraph refers to dose rates at a "depth of 7 mg/cm²". Cember's phrase "density thickness" is more understandable; at any rate, some explanation should be provided for the reader who is not a health physicist.

The last two paragraphs of Section 2.3.1 discuss internal dose resulting from "intact" sources. How does this differ from the internal dose resulting from dispersed sources? Wilmot's 1981 report on spent fuel transportation accidents may not be a good guide for release fractions for improper handling of sealed sources because of the significant differences in the barriers to release and in the purposeful nature of behavior that can be assumed in sealed source mishandling incidents.

Page 7 -- No basis is given for the assumption that the population near the incinerator is 73000.

Page 9 -- The population of the US is 250M, so the average population served by a landfill is $250000000/18500 = 13500$.

Page 13, Table 4 -- The table would be clearer if the last column were "Fraction" instead of "Percentage". Thus, for Class A-1, the last column would show 5.0×10^{-5} .

Table 4 suffers from a significant censoring problem. It does not reflect those sources that have been mishandled in some fashion, but for which the mishandling has not been detected. This is significant, because in known incidents, the detection was either accidental or announced by the severe radiological consequences. Thus, the portion of the sample space that leads to low consequences will be significantly under-represented in Table 4, which is nonetheless used to assign probabilities to event initiators.

The text of the present report discusses the censoring issue in more detail and proposes a methodology for estimating the amount of censoring in the available data, and thus, for arriving at estimates of the initiating event probabilities for various types of diversion or mishandling of Generally Licensed Sources.

C.2. WHAT IS NEEDED FOR REGULATORY APPLICATION, BUT MISSING FROM THE ORAU REPORT?

The main thing missing from the ORAU report, but required for regulatory application is a set of realistic estimates of the risk associated with the production and use of generally licensed sources. To obtain these realistic estimates of risk, we will need to make some realistic estimates of the frequencies associated with initiating events, the conditional probabilities associated with the transitions from state to state in the ORAU block diagrams, and best estimates of the mishandling incident source terms. Wilmot (1981) does not strike me as a good basis for those best estimate source terms, but I don't have an alternative to offer at the moment.

C.3 ADDITIONAL INFORMATION GAINED SUBSEQUENT TO OCTOBER MEETING WITH ORAU AUTHORS

The present report now provides a detailed proposal for the estimation of initiating event probabilities, taking the data censoring into account. These probabilities, together with best-estimate calculations of the public health consequences of a diverted/mishandled source ending up in a particular "final status", provide the basis for calculation of the public health risk associated with Generally Licensed Sources of a particular type.

Although we had criticized the ORAU report for not including the initiating event probabilities and realistic estimates of the public health consequences of a diverted/mishandled source reaching a specific "final status", it became clear during the October meeting that the ORAU report authors were aware of this difficulty. They had been specifically constrained by the Scope of Work on their project not to estimate those quantities. Such constraints are not unreasonable in a first-cut analysis of whether any changes are needed in the regulation of generally licensed devices; had all of the worst case consequences calculated by the ORAU authors been acceptably low, no further analysis would be needed to support a decision to maintain current regulation of generally licensed devices. With some of the worst case consequences being unacceptably high, it may be necessary to collect the additional data needed to calculate the risk, i.e., the probability-weighted consequences, of generally licensed device diversion/mishandling.

APPENDIX D: COMMENTS ON 1987 ORAU REPORT BY G.R. CICOTTE

D.1. REGULATORY CHANGES

The USNRC made no significant changes in its regulations affecting the quantities or categories of generally licensed radioactive sources since 1987. There have also been other changes in the reporting, record keeping, and enforcement portions of the applicable regulations. These are summarized as follows:

- *The new 10 CFR 20 (56 FR 23360-23472, May 21, 1991) resulted in changes to the incident reporting requirements of 10 CFR 30, 10 CFR 31, and 10 CFR 32, invoking a choice between the old part 20 (§20.1-20.601) and the new part 20 (§20.1001-20.2401).*
- *As part of the new part 20, general licensees were specifically limited to doses of 10% of the limits in either 10 CFR 20.101(a), i.e., 125 mrem (1.25 mSv) per calendar quarter, or 10 CFR 20.1201(a), i.e., 500 mrem (5 mSv) per calendar year.*
- *Record retention requirements were clarified (53 FR 19240-19246, May 27, 1988)*
- *The new incident reporting requirements summarized in 56 FR 40757-40767 were imposed on general licensees, by invoking 10 CFR 30.50 in 10 CFR 31.2, and specifically on general licensees for Am-241 calibration and reference sources, as specified in 10 CFR 31.8.*
- *Enforcement authority was placed on general licensees, specifically authorizing injunctions and orders deemed necessary by NRC, and providing criminal penalties for willful violations. (57 FR 55063-55075)*

D.2. EFFECTS ON CONCLUSIONS IN ORAU REPORT OF CHANGED AND EXISTING REGULATIONS

The changes in the regulations should result in reporting of a greater fraction of the incidents which occur. The ratios of incident causes appear to have changed as described below.

The regulations require that in order to use generally licensed devices in accordance with 10 CFR 31.5, the source must not be likely to result in a dose in excess of 125 mrem (1.25 mSv) per quarter, which correlates to 500 mrem (5 mSv) per year. The Executive Summary of the ORAU Report states in part that: ". . . dose equivalents received by members of the general public who purchase contaminated products would most likely not exceed 500 mrem/yr (0.005 Sv/yr) in most cases." No further action would be necessary to protect the public if the goal is to meet current regulatory limits as called out in 10 CFR 20.101(a) or 10 CFR 20.1201(a). The change to the goal dose of 1 mSv per year to a member of the public may result in the need to regulate the manufacture of licensed sources to assure that exceeding 1 mSv per year

is unlikely.

D.3. OTHER CONSIDERATIONS RELATED TO THE ORAU REPORT

The ORAU Report addressed the following generally licensed uses:

1. *Certain measuring, gauging, or controlling devices as authorized by 10 CFR 31.5 - Po-210, Am-241, Ra-226, H-3, Kr-85, Co-60, Cs-137, Sr-90, Tl-204, Ru-106, Pm-147, C-14, Pb-210, Ni-63, Cm-244, Cd-109, and Fe-55.*
2. *Luminous safety devices for use in aircraft as authorized by 10 CFR 31.7 - H-3 and Pm-147.*
3. *Calibration or reference sources as authorized by 10 CFR 31.8 - Am-241.*

The ORAU Report did not appear to address the following generally licensed uses:

1. *Isotopes [byproduct material for certain in vitro clinical or laboratory testing] as authorized by 10 CFR 31.11(a)(1) through (7).*
2. *Ice detection devices containing Sr-90 as authorized by 10 CFR 31.10(a).*
3. *Beta- and/or gamma-emitting materials in measuring, gauging or controlling devices containing radioisotopes other than those specifically listed in Table 1 of the ORAU Report, authorized in 10 CFR 31.5.*
4. *Alpha-emitting materials in measuring, gauging or controlling devices containing radioisotopes other than U-238, Am-241, Ra-226, or Pu-239, as authorized in 10 CFR 31.5.*

It may be appropriate to consider whether scenarios for items 1 and 2 above should be addressed in any report developed for the same purpose as the ORAU Report.

Generally licensed sources authorized in 10 CFR 31.5 are constrained to meet certain dose limits, even if damaged or misused. The general license in 10 CFR 31.5 requires that the sources must have been manufactured or produced through adherence to a specific license issued pursuant to the conditions specified by 10 CFR 32.51. 10 CFR 32.51 requires that individual sources must not cause doses to any person [not necessarily just a member of the public] in excess of the organ doses listed in 10 CFR 32.24. The organ dose limits have not changed since June 13, 1969, and are summarized below:

<i>Part of Body</i>	<i>normal use of one unit (rem)</i>	<i>normal use of all units in one location (rem)</i>	<i>failure - low probability of dose (rem)</i>	<i>failure - negligible probability of dose (rem)</i>
<i>Whole body; head and trunk; active blood-forming organs; gonads; or lens of eye.....</i>	<i>0.001</i>	<i>0.01</i>	<i>0.5</i>	<i>15</i>
<i>Hands and forearms; feet and ankles; localized areas of skin averaged over areas no larger than 1 square centimeter.....</i>	<i>0.015</i>	<i>0.15</i>	<i>7.5</i>	<i>200</i>
<i>Other organs.....</i>	<i>0.003</i>	<i>0.03</i>	<i>1.5</i>	<i>50</i>

D.4. REFINEMENT OF SELECTED SCENARIOS IN THE ORAU REPORT

The ORAU Report hypothesizes a maximum skin contact time of three hours (2.3.1). The limited time was based on the assumption that most of the devices cannot be carried in an individual person's pocket due to size considerations. The great majority of reported incidents involving generally licensed devices were related to portable static eliminators (task 3 report), which are often small enough to be carried in a pocket, e.g., recording equipment static eliminator brushes.

The dispersed source scenario assumes that dispersed material in a landfill would, if previously incinerated, leach into surface or ground water supplies at a rate of 1.0 per year (2.3.2). Since most landfills now have leaching requirements which result in much greater time for breakthrough and there is some natural filtration by the landfill material itself, inclusion of this consideration could reasonably be expected to reduce the available dose significantly.

The metals recycling section of the ORAU Report assumes continuous contact over a year for a postulated maximum dose of 360 millirem (3.6 mSv) based on a distance from contact which reduces the dose to 0.1 of that calculated for continuous contact. In addition, Table 6 of the ORAU Report states that whole body exposures to all categories are based on 20 weeks at 100 cm distance. A better approach might be to assume continuous contact over a reasonable fraction of the time, with the appropriate distance factor. Two possibilities are household products:

members of the public." The new limit is 100 mrem (1 mSv) total effective dose equivalent (TEDE). Thus, many of the conclusions in the ORAU study related to the old limit would need to be reassessed relative to the new limit. Estimates meeting all the following (or similar) criteria would need to be addressed through additional restrictions on generally licensed quantities or categories:

- a. The estimated TEDE exceeds the limit of 1 mSv.*
- b. The probability of occurrence of the scenario is accurate within a margin which indicates the limit would be exceeded with a specified confidence level, e.g., when the estimate exceeds the limit by a factor of 2, the accuracy is within a factor of 2 to a specified confidence of 95%.*
- c. The dose estimate for the pathway itself is accurate within a margin which indicates the limit would be exceeded with a specified confidence level, e.g., when the estimate exceeds the limit by a factor of 2, the accuracy is within a factor of 2 to a specified confidence of 95%.*
- d. The combined effect of b. and c. is accurate within a margin which indicates the limit would be exceeded with a specified confidence level, e.g., when the estimate exceeds the limit by a factor of 2, the accuracy of b. and c. combined is within a factor of 2 to a specified confidence of 90%.*
- e. Any arbitrary assumptions made in the estimate have been supplanted by estimates based on actual information.*

D.7 ADDITIONAL INFORMATION OBTAINED IN MEETING WITH ORAU TEAM AND SUBSEQUENTLY

The ORAU team considered the pathways represented in the probability networks used in their report to represent a condensation of hundreds or thousand of possible pathways. Their intent was to display only the pathways responsible for the "first" 95% or so of the risk.

When asked why Po-210 was considered a significant hazard, given its short half-life (138 days), the ORAU team responded that it was considered an ingestion hazard due to the large number of Po-210 source and the loose control exercised over them.

The ORAU team indicated that sensitivity analysis was not part of the scope of their project.

When asked what they intended by the term "intact" sources, the ORAU team indicated that it varied somewhat, on a case-by-case basis, but that it was generally synonymous with "localized". External dose was not calculated for dispersed sources, with the exception of recycled material, such as scrap.

The ORAU team indicated that the "average" dose for intact sources was arbitrarily defined to be half of the worst-case dose.

The ORAU team was asked how population statistics were applied to external sources. They indicated that maximum and average population figures were applied only to dispersed sources (i.e., no external dose was assumed for populations, except for the case of recycled material).

The ORAU team was asked to describe any special considerations applicable to their dose calculations: 1) absorption was not used, 2) Maximum external dose was assumed to be 0.3 times the worst case for an intact source, 3) the most likely external dose was assumed to be 10^{-6} times the worst case for an intact source, 4) encapsulated source doses were calculated based on NCRP 40, and 5) non-encapsulated source dose were calculated based on beta dose, using the methodology of Kocher and Eckerman.

The assumptions in Section 2.3.2 of the ORAU report are based on NUREG/CR-1775.

The symbol nC in Section 2.3.2.2 of the ORAU report refers to nanoCoulombs; the distance 0.1 is in meters.

Re Section 2.3.2.3: Intrusion at landfills was not considered. The value $e^{-\lambda t}$ refers to the mean leach rate.

Re Section 2.3.1: The 50 year dose equivalent from ICRP 30 did not refer to the source remaining in place.

APPENDIX E TO
FINAL REVIEW OF THE 1987 REPORT BY
OAK RIDGE ASSOCIATED UNIVERSITIES,
"IMPROPER TRANSFER/DISPOSAL SCENARIOS FOR
GENERALLY LICENSED DEVICES:"

REVISED TASK 3 TECHNICAL LETTER REPORT:
EVALUATION OF HISTORICAL SEALED SOURCE DEVICE EXPERIENCE

NRC JOB CODE L2536
PNL No. 20278

D. J. Strom
G. R. Cicotte

Health Protection Department
Pacific Northwest Laboratory
Richland, Washington 99352

first draft August 27, 1993
revised February 4, 1994
revised May 31, 1994

Prepared for the U.S. Nuclear Regulatory Commission
Under Contract DE-AC06-76RLO 1830

1.0 EXECUTIVE SUMMARY

Data and records provided to the Pacific Northwest Laboratory (PNL) by Steven L. Baggett and Sterling Bell of the U.S. Nuclear Regulatory Commission (NRC) have been reviewed to establish the historical experience of sealed source device use and reported events of improper transfer or disposal.

2.0 SUMMARY OF FILES AND REPORTS USED IN THIS WORK

Documents supplied to PNL are listed in Section 6.0, References. The original work under review in this project is referred to as the "ORAU Report" (Stabin et al. 1987).

Computer database files (in dBase III format for DOS) received and reviewed by PNL are listed in Attachment 1.

3.0 NUMBER OF SEALED SOURCE LICENSES AND DEVICES OF EACH TYPE EXISTING IN EACH YEAR FOR WHICH RECORDS ARE READILY AVAILABLE

3.1 REVIEW OF SEALED SOURCE DEVICE REGISTRY

The Sealed Source Device Registry (SSDR) provided basic information about the design, construction, uses and authorized maximum activity for each type of device. Devices are categorized in the fashion of Table 1 of the ORAU Report, included here as Attachment 2.

For detailed, source-specific risk analyses, it would be necessary to determine the numbers of sources in each category by isotope, date placed in service, and activity. Use of design information regarding shielding would help to determine external doses in cases of improper transfer or disposal in which the source was not removed from the shield. Review in this level of detail was beyond the scope of the present project.

3.2 REVIEW OF NUMBER OF GENERALLY-LICENSED DEVICES

Four hardcopy reports from "General License Database System" were provided to the PNL reviewers attached to a letter from Steven L. Baggett to Daniel J. Strom dated November 2, 1993 (Baggett 1993). Two of the reports were entitled "General License Database System Report for Peer Review." The first of these, dated 10/20/93, was 6 pages long and contained sealed source registrations in the years 1987-1992. The second, dated 10/20/93, was 2 pages long and contained additional sealed source registrations in the years 1991-1992. Column headings on the two reports were year, isotope, device code, number of devices, number of general licensees, total activity, and average activity/device. (Device codes are shown in Table Task-3-2. Activities are in millicuries (mCi).)

These two reports were keyed and analyzed as a review of historical data for a revision of the Task 3 report. The ORAU report was limited to devices of less than 20 mCi, but the "General License Database System Report for Peer Review" printouts contained many sources of activities significantly greater than 20 mCi, so these data are difficult to use directly in a risk analysis.

These two printouts are summarized over the 6-year period in Tables Task-3-1A, Task-3-1B, and Task-3-1C. In each table, the Device Codes are those identified in the printouts as being licensed under 10 CFR 31.5 and listed in Appendix C of Regulatory Guide 10.10 (NRC 1987), with "W7" denoting self-luminous sources licensed under 10 CFR 31.7. The tables are identical except for the sort order. Rows are labeled by nuclide and device code. Rows in Table Task-3-1A are sorted by nuclide (alphabetically, the way the data were received) and within nuclide by Device Code; rows in Table Task-3-1B are sorted by Device Code and within Device Code by nuclide in order of increasing atomic number; and rows in Table Task-3-1C are sorted by fraction of total ingestion ALIs contributed by a nuclide-Device Code combination.

In order to get an idea of the steady-state activity for sources, the activities of sources for years 1987-91 were decay-corrected to 1992. For some sources, e.g., 138-day ^{210}Po , this means that the 1987 sources had essentially disappeared. Each table shows the total number of devices registered between 1987 and 1992, the decay-corrected sum of the activities by nuclide and device code.

To rank the relative hazards of the nuclide-device code combinations, the ingestion ALIs for each row were divided by the total number of ingestion ALIs contained in sources registered during the 6-year span. Using the decay correction, there were 3.37×10^9 ingestion ALIs in these 325,681 sources in 1992. The fractions are expressed in parts per million (ppm) to make the numbers easier to compare.

Table Task-3-1C shows that the ^{241}Am D (gamma gauge) sources account for nearly half of the ingestion ALIs (489,184 ppm or 49%), with ^3H W self-luminous sources accounting for 26% of the ALIs (using the admittedly incorrect, that is, too high by one or more orders of magnitude, ALI for $^3\text{H}_2\text{O}$, not the unlisted ALI for ^3H -labeled luminous materials). Promethium-147 E gauges account for 9%, followed by ^{244}Cm U devices (4.1%), ^{252}Cf E and U devices (2.9% and 2.6%), and ^{210}Po static eliminators (2.5%). Tritium self-luminous light sources licensed under 10 CFR 31.7 account for about 1%, as do ^3H gas sources.

Also shown in the tables are crude risk assessment numbers in the last 5 columns: the average number of ingestion ALIs per device; the average activity per device; the average source strength, Γ A, in rems per hour at 1 meter from an unshielded source with the average activity; the committed effective dose equivalent from ingesting 1/10,000 (10^{-4}) of the source; and the dose equivalent one would receive by spending 1000 hours at 1 meter from an unshielded source. A justification for use of the factors, 10^{-4} fraction-taken-in and 1000 hours used in the last two columns, is given in the main body of this report.

A number of difficulties were found in the SDDR printouts. These included invalid isotopes ("K85," "KR-95," "KR84 4," "CE137," "CS137 SR"), invalid device codes (GAUGE, blank, V, and 31.7U), and an invalid number (0) of general licensees that occurs on 6 occasions. The obvious corrections were made for these cases (K85 changed to Kr-85; zeros changed to ones for general licensees). There were also some puzzling uses of isotopes which may be explainable as keying errors: ^3H , ^{55}Fe , and ^{90}Sr gamma gauges; ^{55}Fe , ^{60}Co , ^{137}Cs , ^{210}Po , and ^{241}Am beta gauges; ^{63}Ni and ^{137}Cs neutron sources; Device Code I sources with activities less than 30 mCi; and ^{90}Sr gas sources. There was no basis for correcting these latter errors, so they remain as reported when included in the Tables Task-3-1A and 1B.

Table Task-3-1A. Summary of "General License Database System Report for Peer Review." Sorted by Atomic Number, Device Code.

Nuclide	Device Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALIs decayed to 1992 (ppm)	Average Ing. ALIs/Device	Average Activity/Device (mCi)	Average Source Strength, Gamma*A of 10^-4 at 1 m from Source (rem/h @1m)	Dose from Ingestion of 10^-4 at 1 m from Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
H-3	D	Gamma Gauges	1176	1.38E+07	1,364	3,909	11,727	-	2.0	-
H-3	E	Beta Gauges	2390	2.47E+07	2,438	3,439	10,317	-	1.7	-
H-3	N	Ion Gen, Chromatog	1378	5.62E+06	556	1,360	4,079	-	0.68	-
H-3	O	Ion Gen, Static Elim.	1392	1.51E+07	1,494	3,617	10,852	-	1.8	-
H-3	R	Gas Sources	10771	1.02E+08	10,066	3,150	9,451	-	1.6	-
H-3	S	Foil Sources	25	1.25E+03	0.12	17	50	-	0.0083	-
H-3	T	Other	20	2.93E+02	0.029	5	15	-	0.0024	-
H-3	W	Self-Lum Light Src	242505	2.66E+09	262,841	3,654	10,961	-	1.8	-
H-3	W7	31.7 Self-Lum Light Src	18519	1.07E+08	10,596	1,929	5,786	-	1.0	-
C-14	E	Beta Gauges	6	6.00E-01	0.0020	1	0.10	-	5.55E-04	-
C-14	N	Ion Gen, Chromatog	10	1.00E-02	0.000033	0	0.0010	-	5.55E-06	-
C-14	R	Gas Sources	42	2.10E+00	0.0069	1	0.050	-	2.78E-04	-
C-14	T	Other	33	7.25E+00	0.024	2	0.22	-	0.0012	-
C-14	W	Self-Lum Light Src	267	3.67E+01	0.12	2	0.14	-	7.64E-04	-
C-14	Y	Calibrators	24	1.20E-01	0.00040	0	0.0050	-	2.78E-05	-
Sc-46	N	Ion Gen, Chromatog	167	4.91E+02	0.16	3	2.9	0.0034	0.0016	3.4
Ti-44	N	Ion Gen, Chromatog	13	6.43E+02	0.64	165	49	0.0072	0.082	7.2
Fe-55	D	Gamma Gauges	57	1.10E+03	0.036	2	19	-	0.0011	-
Fe-55	E	Beta Gauges	13	4.64E+02	0.015	4	36	-	0.0020	-
Fe-55	N	Ion Gen, Chromatog	18	7.67E+02	0.025	5	43	-	0.0024	-
Fe-55	T	Other	12	5.60E+02	0.018	5	47	-	0.0026	-
Fe-55	U	X-Ray Fluorescence	811	6.51E+04	2	9	80	-	0.0045	-
Co-60	D	Gamma Gauges	41	4.13E+03	2	201	101	0.14	0.10	138
Co-60	E	Beta Gauges	3	3.52E+01	0.021	23	12	0.016	0.012	16
Ni-63	E	Beta Gauges	22	2.17E-01	7.15E-06	0	0.010	-	5.48E-07	-
Ni-63	H	Gen Neut Src Apps	22	3.30E+02	0.011	2	15	-	8.33E-04	-
Ni-63	N	Ion Gen, Chromatog	3451	2.59E+05	9	8	75	-	0.0042	-
Ni-63	O	Ion Gen, Static Elim.	56	8.28E+02	0.027	2	15	-	8.21E-04	-

Table Task-3-1A continued. Sorted by Atomic Number and Device Code.

Nuclide	Dev- ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALLIs decayed to 1992 (ppm)	Average Ing. ALLIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ at 1 m from Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
Ni-63	S	Foil Sources	6	6.99E+01	0.0023	1	12	-	6.47E-04	-
Ni-63	T	Other	12	1.19E-01	3.93E-06	0	0.010	-	5.52E-07	-
Ni-63	U	X-Ray Fluorescence	8	3.91E+01	0.0013	1	4.9	-	2.72E-04	-
Ni-63	W	Self-Lum Light Src	19	1.20E+05	4	702	6,320	-	0.35	-
Kr-85	D	Gamma Gauges	291	1.20E+05	0	0	413	0.00E+00	-	0.00E+00
Kr-85	E	Beta Gauges	623	2.43E+06	0	0	3,905	0.0061	-	6.1
Kr-85	N	Ion Gen, Chromatog	135	6.94E+04	0	0	514	8.05E-04	-	0.80
Kr-85	O	Ion Gen, Static Elim.	128	3.44E+02	0	0	2.7	4.21E-06	-	0.0042
Kr-85	R	Gas Sources	2	2.11E+03	0	0	1,054	0.0017	-	1.7
Kr-85	T	Other	11	2.20E+03	0	0	200	3.13E-04	-	0.31
Kr-85	U	X-Ray Fluorescence	13	5.44E+01	0	0	4.2	6.55E-06	-	0.0065
Sr-90	D	Gamma Gauges	104	2.28E+04	226	7,324	220	-	3.7	-
Sr-90	E	Beta Gauges	1153	3.94E+05	3,895	11,387	342	-	5.7	-
Sr-90	I	Calib Src A>30 mCi	1	5.00E-01	0.0049	17	0.50	-	0.0083	-
Sr-90	R	Gas Sources	135	2.44E+02	2	60	1.8	-	0.030	-
Sr-90	T	Other	182	8.63E+01	0.85	16	0.47	-	0.0079	-
Sr-90	Y	Calibrators	2	2.41E-03	0.000024	0	0.0012	-	2.01E-05	-
Ru-106	D	Gamma Gauges	2	4.00E+01	0.059	100	20	-	0.050	-
Ru-106	E	Beta Gauges	1	9.65E-04	1.43E-06	0	0.0010	-	2.41E-06	-
Cd-109	D	Gamma Gauges	44	5.27E+01	0.052	4	1.2	2.21E-04	0.0020	0.22
Cd-109	E	Beta Gauges	85	4.60E+02	0.45	18	5.4	0.0010	0.0090	1.0
Cd-109	N	Ion Gen, Chromatog	8	2.65E+01	0.026	11	3.3	6.11E-04	0.0055	0.61
Cd-109	T	Other	10	4.90E+01	0.048	16	4.9	9.03E-04	0.0082	0.90
Cd-109	U	X-Ray Fluorescence	567	6.25E+02	0.62	4	1.1	2.03E-04	0.0018	0.20
I-129	T	Other	24	2.96E+01	0.0029	0	1.2	1.55E-04	2.05E-04	0.16
Cs-137	D	Gamma Gauges	2493	2.20E+06	6,533	8,833	883	0.34	4.4	337
Cs-137	E	Beta Gauges	31	6.05E+05	1,795	195,228	19,523	7.5	98	7,455
Cs-137	H	Gen Neut Src Apps	45	3.56E+03	11	791	79	0.030	0.40	30

Table Task-3-1A continued. Sorted by Atomic Number and Device Code.

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALIs decayed to 1992 (ppm)	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Spending 1000 hours at 1 m from Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
Cs-137	I	Calib Src A>30 mCi	17	6.49E+02	1.9	382	38	0.015	0.19	15
Cs-137	N	Ion Gen, Chromatog	11	3.13E+02	0.93	284	28	0.011	0.14	11
Cs-137	T	Other	1060	1.40E+06	4,139	13,163	1,316	0.50	6.6	503
Cs-137	U	X-Ray Fluorescence	18	2.60E+03	8	1,445	144	0.055	0.72	55
Cs-137	W	Self-Lum Light Src	5	2.25E+03	7	4,503	450	0.17	2.3	172
Cs-137	Y	Calibrators	109	4.64E+01	0.14	4	0.43	1.62E-04	0.0021	0.16
Ba-133	I	Calib Src A>30 mCi	93	1.53E+00	0.00023	0	0.016	7.51E-06	4.12E-06	0.0075
Ba-133	N	Ion Gen, Chromatog	5	8.79E-02	0.000013	0	0.018	8.01E-06	4.39E-06	0.0080
Ba-133	T	Other	644	5.88E+02	0.087	0	0.91	4.16E-04	2.28E-04	0.42
Pm-147	D	Gamma Gauges	11	6.20E+03	0.46	141	564	1.51E-06	0.070	0.0015
Pm-147	E	Beta Gauges	2685	1.09E+09	80,974	101,660	406,639	0.0011	51	1.1
Pm-147	T	Other	1	5.90E+02	0.044	147	590	1.58E-06	0.074	0.0016
Pm-147	W	Self-Lum Light Src	1	4.42E+04	3	11,055	44,220	1.18E-04	5.5	0.12
Eu-152	T	Other	56	1.08E+00	0.00040	0	0.019	1.44E-05	1.21E-05	0.014
Tl-204	E	Beta Gauges	538	4.05E+06	601	3,764	7,528	0.0084	1.9	8.4
Po-210	E	Beta Gauges	98	3.85E+01	4	131	0.39	2.07E-09	0.065	2.07E-06
Po-210	O	Ion Gen, Static Elim.	27235	2.55E+05	25,251	3,125	9.4	4.94E-08	1.6	4.94E-05
Po-210	U	X-Ray Fluorescence	535	9.23E+03	912	5,749	17	9.09E-08	2.9	9.09E-05
Bi-210	E	Beta Gauges	1	2.79E-89	1.04E-92	0	2.79E-89	-	1.75E-92	-
Ra-226	I	Calib Src A>30 mCi	1	2.00E-02	0.0030	10	0.020	2.85E-05	0.0050	0.029
Ra-226	S	Foil Sources	53	1.24E+00	0.18	12	0.023	3.34E-05	0.0058	0.033
Ra-226	T	Other	12	1.20E-01	0.018	5	0.010	1.42E-05	0.0025	0.014
Pu-238	D	Gamma Gauges	2	5.84E+01	19	32,425	29	0.0023	16	2.3
Pu-238	U	X-Ray Fluorescence	12	3.52E+02	116	32,620	29	0.0023	16	2.3
Am-241	D	Gamma Gauges	2261	1.32E+06	489,184	729,319	583	0.18	365	183
Am-241	E	Beta Gauges	82	7.89E+04	29,240	1,202,034	962	0.30	601	302
Am-241	H	Gen Neut Src Apps	3	1.36E+03	504	565,999	453	0.14	283	142
Am-241	I	Calib Src A>30 mCi	9	2.17E-01	0.080	30	0.024	7.55E-06	0.015	0.0076

Table Task-3-1A continued. Sorted by Atomic Number and Device Code.

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALIs decayed to 1992 (ppm)	Average Strength, Source Gamma*A (rem/h @1m)	Dose from Spending 1000 hours at 1 m from Source (rems)	Average Activity/ Device (mCi)	Average Source Strength, Ingestion of 10^-4 of Source (rem/h @1m)	Dose from Spending 1000 hours at 1 m from Source (rems)
Am-241	N	Ion Gen, Chromatog	1	1.30E-01	0.048	4.07E-05	0.041	0.13	4.07E-05	0.081
Am-241	O	Ion Gen, Static Elim.	107	1.47E+03	545	0.0043	4.3	14	0.0043	8.6
Am-241	S	Foil Sources	2	2.60E-01	0.10	4.07E-05	0.041	0.13	4.07E-05	0.081
Am-241	T	Other	7	2.00E+01	7	8.96E-04	0.90	2.9	8.96E-04	1.8
Am-241	U	X-Ray Fluorescence	350	6.97E+04	25,830	0.062	62	199	0.062	124
Am-241	Y	Calibrators	6	6.79E-05	0.000025	3.55E-09	3.55E-06	0	1.13E-05	7.08E-06
Cm-244	D	Gamma Gauges	16	2.16E+03	642	0.0087	8.7	135	0.0087	68
Cm-244	U	X-Ray Fluorescence	239	1.35E+05	40,171	0.036	36	567	0.036	283
Cf-252	H	Gen Neut Src Apps	16	1.60E+00	0.24	4.18E-06	0.0042	0.10	4.18E-06	0.025
			<u>325681</u>							

Table Task-3-1B. Summary of "General License Database System Report for Peer Review." Sorted by Device Code and Nuclide.

Nuclide	Device Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALIs	Average Ing. ALIs/Device	Average Activity/Device (mCi)	Average Source Strength, Gamma*A of 10^-4 at 1 m from Source (rem/h @1m)	Dose from Ingestion of 10^-4 at 1 m from Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
H-3	D	Gamma Gauges	1176	1.38E+07	1,364	3,909	11,727	-	2.0	-
Fe-55	D	Gamma Gauges	57	1.10E+03	0.036	2	19	-	0.0011	-
Co-60	D	Gamma Gauges	41	4.13E+03	2	201	101	0.14	0.10	138
Kr-85	D	Gamma Gauges	291	1.20E+05	0	0	413	1.53E-04	-	1.53E-01
Sr-90	D	Gamma Gauges	104	2.28E+04	226	7,324	220	-	3.7	-
Ru-106	D	Gamma Gauges	2	4.00E+01	0.059	100	20	-	0.050	-
Cd-109	D	Gamma Gauges	44	5.27E+01	0.052	4	1.2	2.21E-04	0.0020	0.22
Cs-137	D	Gamma Gauges	2493	2.20E+06	6,533	8,833	883	0.34	4.4	337
Pm-147	D	Gamma Gauges	11	6.20E+03	0.46	141	564	1.51E-06	0.070	0.0015
Pu-238	D	Gamma Gauges	2	5.84E+01	19	32,425	29	0.0023	16	2.3
Am-241	D	Gamma Gauges	2261	1.32E+06	489,184	729,319	583	0.18	365	183
Cm-244	D	Gamma Gauges	16	2.16E+03	642	135,307	135	0.0087	68	8.7
H-3	E	Beta Gauges	2390	2.47E+07	2,438	3,439	10,317	-	1.7	-
C-14	E	Beta Gauges	6	6.00E-01	0.0020	1	0.10	-	5.55E-04	-
Fe-55	E	Beta Gauges	13	4.64E+02	0.015	4	36	-	0.0020	-
Co-60	E	Beta Gauges	3	3.52E+01	0.021	23	12	0.016	0.012	16
Ni-63	E	Beta Gauges	22	2.17E-01	7.15E-06	0	0.010	-	5.48E-07	-
Kr-85	E	Beta Gauges	623	2.43E+06	0	0	3,905	0.0061	-	6.1
Sr-90	E	Beta Gauges	1153	3.94E+05	3,895	11,387	342	-	5.7	-
Ru-106	E	Beta Gauges	1	9.65E-04	1.43E-06	0	0.0010	-	2.41E-06	-
Cd-109	E	Beta Gauges	85	4.60E+02	0.45	18	5.4	0.0010	0.0090	1.0
Cs-137	E	Beta Gauges	31	6.05E+05	1,795	195,228	19,523	7.5	98	7,455
Pm-147	E	Beta Gauges	2685	1.09E+09	80,974	101,660	406,639	0.0011	51	1.1
Tl-204	E	Beta Gauges	538	4.05E+06	601	3,764	7,528	0.0084	1.9	8.4
Po-210	E	Beta Gauges	98	3.85E+01	4	131	0.39	2.07E-09	0.065	2.07E-06
Bi-210	E	Beta Gauges	1	2.79E-89	1.04E-92	0	2.79E-89	-	1.75E-92	-
Am-241	E	Beta Gauges	82	7.89E+04	29,240	1,202,034	962	0.30	601	302
Ni-63	H	Gen Neut Src Apps	22	3.30E+02	0.011	2	15	-	8.33E-04	-

Table Task-3-1B continued. Sorted by Device Code and Nuclide.

Nuclide	Dev - ice	Device Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALIs	Average Ing. Device ALIs/Device	Average Activity/Device (mCi)	Average Source Strength, Gamma*A of 10^-4 at 1 m from Source (rem/h @1m)	Dose from Ingestion of Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
Cs-137	H	H	Gen Neut Src Apps	45	3.56E+03	11	791	79	0.030	0.40	30
Am-241	H	H	Gen Neut Src Apps	3	1.36E+03	504	565,999	453	0.14	283	142
Cf-252	H	H	Gen Neut Src Apps	16	1.60E+00	0.24	50	0.10	4.18E-06	0.025	0.0042
Sr-90	I	I	Calib Src A>30 mCi	1	5.00E-01	0.0049	17	0.50	-	0.0083	-
Cs-137	I	I	Calib Src A>30 mCi	17	6.49E+02	1.9	382	38	0.015	0.19	15
Ba-133	I	I	Calib Src A>30 mCi	93	1.53E+00	0.00023	0	0.016	7.51E-06	4.12E-06	0.0075
Ra-226	I	I	Calib Src A>30 mCi	1	2.00E-02	0.0030	10	0.020	2.85E-05	0.0050	0.029
Am-241	I	I	Calib Src A>30 mCi	9	2.17E-01	0.080	30	0.024	7.55E-06	0.015	0.0076
H-3	N	N	Ion Gen, Chromatog	1378	5.62E+06	556	1,360	4,079	-	0.68	-
C-14	N	N	Ion Gen, Chromatog	10	1.00E-02	0.000033	0	0.0010	-	5.55E-06	-
Sc-46	N	N	Ion Gen, Chromatog	167	4.91E+02	0.16	3	2.9	0.0034	0.0016	3.4
Ti-44	N	N	Ion Gen, Chromatog	13	6.43E+02	0.64	165	49	0.0072	0.082	7.2
Fe-55	N	N	Ion Gen, Chromatog	18	7.67E+02	0.025	5	43	-	0.0024	-
Ni-63	N	N	Ion Gen, Chromatog	3451	2.59E+05	9	8	75	-	0.0042	-
Kr-85	N	N	Ion Gen, Chromatog	135	6.94E+04	0	0	514	8.05E-04	-	0.80
Cd-109	N	N	Ion Gen, Chromatog	8	2.65E+01	0.026	11	3.3	6.11E-04	0.0055	0.61
Cs-137	N	N	Ion Gen, Chromatog	11	3.13E+02	0.93	284	28	0.011	0.14	11
Ba-133	N	N	Ion Gen, Chromatog	5	8.79E-02	0.000013	0	0.018	8.01E-06	4.39E-06	0.0080
Am-241	N	N	Ion Gen, Chromatog	1	1.30E-01	0.048	162	0.13	4.07E-05	0.081	0.041
H-3	O	O	Ion Gen, Static Elim.	1392	1.51E+07	1,494	3,617	10,852	-	1.8	-
Ni-63	O	O	Ion Gen, Static Elim.	56	8.28E+02	0.027	2	15	-	8.21E-04	-
Kr-85	O	O	Ion Gen, Static Elim.	128	3.44E+02	0	0	2.7	4.21E-06	-	0.0042
Po-210	O	O	Ion Gen, Static Elim.	27235	2.55E+05	25,251	3,125	9.4	4.94E-08	1.6	4.94E-05
Am-241	O	O	Ion Gen, Static Elim.	107	1.47E+03	545	17,184	14	0.0043	8.6	4.3
H-3	R	R	Gas Sources	10771	1.02E+08	10,066	3,150	9,451	-	1.6	-
C-14	R	R	Gas Sources	42	2.10E+00	0.0069	1	0.050	-	2.78E-04	-
Kr-85	R	R	Gas Sources	2	2.11E+03	0	0	1,054	0.0017	-	1.7
Sr-90	R	R	Gas Sources	135	2.44E+02	2	60	1.8	-	0.030	-

Table Task-3-1B continued. Sorted by Device Code and Nuclide.

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALIs	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10^-4 at 1 m from Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
H-3	S	Foil Sources	25	1.25E+03	0.12	17	50	-	0.0083	-
Ni-63	S	Foil Sources	6	6.99E+01	0.0023	1	12	-	6.47E-04	-
Ra-226	S	Foil Sources	53	1.24E+00	0.18	12	0.023	3.34E-05	0.0058	0.033
Am-241	S	Foil Sources	2	2.60E-01	0.10	162	0.13	4.07E-05	0.081	0.041
H-3	T	Other	20	2.93E+02	0.029	5	15	-	0.0024	-
C-14	T	Other	33	7.25E+00	0.024	2	0.22	-	0.0012	-
Fe-55	T	Other	12	5.60E+02	0.018	5	47	-	0.0026	-
Ni-63	T	Other	12	1.19E-01	3.93E-06	0	0.010	-	5.52E-07	-
Kr-85	T	Other	11	2.20E+03	0	0	200	3.13E-04	-	0.31
Sr-90	T	Other	182	8.63E+01	0.85	16	0.47	-	0.0079	-
Cd-109	T	Other	10	4.90E+01	0.048	16	4.9	9.03E-04	0.0082	0.90
I-129	T	Other	24	2.96E+01	0.0029	0	1.2	1.55E-04	2.05E-04	0.16
Cs-137	T	Other	1060	1.40E+06	4,139	13,163	1,316	0.50	6.6	503
Ba-133	T	Other	644	5.88E+02	0.087	0	0.91	4.16E-04	2.28E-04	0.42
Pm-147	T	Other	1	5.90E+02	0.044	147	590	1.58E-06	0.074	0.0016
Eu-152	T	Other	56	1.08E+00	0.00040	0	0.019	1.44E-05	1.21E-05	0.014
Ra-226	T	Other	12	1.20E-01	0.018	5	0.010	1.42E-05	0.0025	0.014
Am-241	T	Other	7	2.00E+01	7	3,572	2.9	8.96E-04	1.8	0.90
Fe-55	U	X-Ray Fluorescence	811	6.51E+04	2	9	80	-	0.0045	-
Ni-63	U	X-Ray Fluorescence	8	3.91E+01	0.0013	1	4.9	-	2.72E-04	-
Kr-85	U	X-Ray Fluorescence	13	5.44E+01	0	0	4.2	6.55E-06	-	0.0065
Cd-109	U	X-Ray Fluorescence	567	6.25E+02	0.62	4	1.1	2.03E-04	0.0018	0.20
Cs-137	U	X-Ray Fluorescence	18	2.60E+03	8	1,445	144	0.055	0.72	55
Po-210	U	X-Ray Fluorescence	535	9.23E+03	912	5,749	17	9.09E-08	2.9	9.09E-05
Pu-238	U	X-Ray Fluorescence	12	3.52E+02	116	32,620	29	0.0023	16	2.3
Am-241	U	X-Ray Fluorescence	350	6.97E+04	25,830	248,772	199	0.062	124	62
Cm-244	U	X-Ray Fluorescence	239	1.35E+05	40,171	566,579	567	0.036	283	36
H-3	W	Self-Lum Light Src	242505	2.66E+09	262,841	3,654	10,961	-	1.8	-

Table Task-3-1C. Summary of "General License Database System Report for Peer Review." Sorted by "Fractions of Total Ing. ALIs."

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices,		Sum of Activity, decayed to		Fractions of Total Ing. ALIs		Average Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10^-4 at 1 m from Source (rems)	Dose Spending 1000 hours at 1 m from Source (rems)
			'87-'92	'92	1992 (mCi)	1992 (ppm)	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)			
Am-241	D	Gamma Gauges	2261	1.32E+06	489,184	729,319	583	0.18	365	183	
H-3	W	Self-Lum Light Src	242505	2.66E+09	262,841	3,654	10,961	-	1.8	-	
Pm-147	E	Beta Gauges	2685	1.09E+09	80,974	101,660	406,639	0.0011	51	1.1	
Cm-244	U	X-Ray Fluorescence	239	1.35E+05	40,171	566,579	567	0.036	283	36	
Am-241	E	Beta Gauges	82	7.89E+04	29,240	1,202,034	962	0.30	601	302	
Am-241	U	X-Ray Fluorescence	350	6.97E+04	25,830	248,772	199	0.062	124	62	
Po-210	O	Ion Gen, Static Elim.	27235	2.55E+05	25,251	3,125	9.4	4.94E-08	1.6	4.94E-05	
H-3	W7	31.7 Self-Lum Light Src	18519	1.07E+08	10,596	1,929	5,786	-	1.0	-	
H-3	R	Gas Sources	10771	1.02E+08	10,066	3,150	9,451	-	1.6	-	
Cs-137	D	Gamma Gauges	2493	2.20E+06	6,533	8,833	883	0.34	4.4	337	
Cs-137	T	Other	1060	1.40E+06	4,739	13,163	1,316	0.50	6.6	503	
Sr-90	E	Beta Gauges	1153	3.94E+05	3,895	11,387	342	-	5.7	-	
H-3	E	Beta Gauges	2390	2.47E+07	2,438	3,439	10,317	-	1.7	-	
Cs-137	E	Beta Gauges	31	6.05E+05	1,795	195,228	19,523	7.5	98	7,455	
H-3	O	Ion Gen, Static Elim.	1392	1.51E+07	1,494	3,617	10,852	-	1.8	-	
H-3	D	Gamma Gauges	1176	1.38E+07	1,364	3,909	11,727	-	2.0	-	
Po-210	U	X-Ray Fluorescence	535	9.23E+03	912	5,749	17	9.09E-08	2.9	9.09E-05	
Cm-244	D	Gamma Gauges	16	2.16E+03	642	135,307	135	0.0087	68	8.7	
Tl-204	E	Beta Gauges	538	4.05E+06	601	3,764	7,528	0.0084	1.9	8.4	
H-3	N	Ion Gen, Chromatog	1378	5.62E+06	556	1,360	4,079	-	0.68	-	
Am-241	O	Ion Gen, Static Elim.	107	1.47E+03	545	17,184	14	0.0043	8.6	4.3	
Am-241	H	Gen Neut Src Apps	3	1.36E+03	504	565,999	453	0.14	283	142	
Sr-90	D	Gamma Gauges	104	2.28E+04	226	7,324	220	-	3.7	-	
Pu-238	U	X-Ray Fluorescence	12	3.52E+02	116	32,620	29	0.0023	16	2.3	
Pu-238	D	Gamma Gauges	2	5.84E+01	19	32,425	29	0.0023	16	2.3	
Cs-137	H	Gen Neut Src Apps	45	3.56E+03	11	791	79	0.030	0.40	30	
Ni-63	N	Ion Gen, Chromatog	3451	2.59E+05	9	8	75	-	0.0042	-	
Cs-137	U	X-Ray Fluorescence	18	2.60E+03	8	1,445	144	0.055	0.72	55	

Table Task-3-1C continued. Sorted by "Fractions of Total Ing. ALIs."

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALIs decayed to 1992 (ppm)	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10^-4 at 1 m from Source (rems)	Dose Spending 1000 hours at 1 m from Source (rems)
Am-241	T	Other	7	2.00E+01	7	3,572	2.9	8.96E-04	1.8	0.90
Cs-137	W	Self-Lum Light Src	5	2.25E+03	7	4,503	450	0.17	2.3	172
Ni-63	W	Self-Lum Light Src	19	1.20E+05	4	702	6,320	-	0.35	-
Po-210	E	Beta Gauges	98	3.85E+01	4	131	0.39	2.07E-09	0.065	2.07E-06
Pm-147	W	Self-Lum Light Src	1	4.42E+04	3	11,055	44,220	1.18E-04	5.5	0.12
Co-60	D	Gamma Gauges	41	4.13E+03	2	201	101	0.14	0.10	138
Sr-90	R	Gas Sources	135	2.44E+02	2	60	1.8	-	0.030	-
Fe-55	U	X-Ray Fluorescence	811	6.51E+04	2	9	80	-	0.0045	-
Cs-137	I	Calib Src A>30 mCi	17	6.49E+02	1.9	382	38	0.015	0.19	15
Cs-137	N	Ion Gen, Chromatog	11	3.13E+02	0.93	284	28	0.011	0.14	11
Sr-90	T	Other	182	8.63E+01	0.85	16	0.47	-	0.0079	-
Ti-44	N	Ion Gen, Chromatog	13	6.43E+02	0.64	165	49	0.0072	0.082	7.2
Cd-109	U	X-Ray Fluorescence	567	6.25E+02	0.62	4	1.1	2.03E-04	0.0018	0.20
Pm-147	D	Gamma Gauges	11	6.20E+03	0.46	141	564	1.51E-06	0.070	0.0015
Cd-109	E	Beta Gauges	85	4.60E+02	0.45	18	5.4	0.0010	0.0090	1.0
Cf-252	H	Gen Neut Src Apps	16	1.60E+00	0.24	50	0.10	4.18E-06	0.025	0.0042
Ra-226	S	Foil Sources	53	1.24E+00	0.18	12	0.023	3.34E-05	0.0058	0.033
Sc-46	N	Ion Gen, Chromatog	167	4.91E+02	0.16	3	2.9	0.0034	0.0016	3.4
Cs-137	Y	Calibrators	109	4.64E+01	0.14	4	0.43	1.62E-04	0.0021	0.16
H-3	S	Foil Sources	25	1.25E+03	0.12	17	50	-	0.0083	-
C-14	W	Self-Lum Light Src	267	3.67E+01	0.12	2	0.14	-	7.64E-04	-
Am-241	S	Foil Sources	2	2.60E-01	0.10	162	0.13	4.07E-05	0.081	0.041
Ba-133	T	Other	644	5.88E+02	0.087	0	0.91	4.16E-04	2.28E-04	0.42
Am-241	I	Calib Src A>30 mCi	9	2.17E-01	0.080	30	0.024	7.55E-06	0.015	0.0076
Ru-106	D	Gamma Gauges	2	4.00E+01	0.059	100	20	-	0.050	-
Cd-109	D	Gamma Gauges	44	5.27E+01	0.052	4	1.2	2.21E-04	0.0020	0.22
Cd-109	T	Other	10	4.90E+01	0.048	16	4.9	9.03E-04	0.0082	0.90
Am-241	N	Ion Gen, Chromatog	1	1.30E-01	0.048	162	0.13	4.07E-05	0.081	0.041

Table Task-3-1C continued. Sorted by "Fractions of Total Ing. ALIs."

Nuclide	Dev - ice	Code	Device Definition	Total Number of Devices, '87-92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALIs decayed to 1992 (ppm)	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h of Source @1m)	Dose from Ingestion of 10 ⁻⁴ at 1 m from Source	Dose from Spending 1000 hours at 1 m from Source
Pm-147	T		Other	1	5.90E+02	0.044	147	590	1.58E-06	0.074	0.0016
Fe-55	D		Gamma Gauges	57	1.10E+03	0.036	2	19	-	0.0011	-
H-3	T		Other	20	2.93E+02	0.029	5	15	-	0.0024	-
Ni-63	O		Ion Gen, Static Elim.	56	8.28E+02	0.027	2	15	-	8.21E-04	-
Cd-109	N		Ion Gen, Chromatog	8	2.65E+01	0.026	11	3.3	6.11E-04	0.0055	0.61
Fe-55	N		Ion Gen, Chromatog	18	7.67E+02	0.025	5	43	-	0.0024	-
C-14	T		Other	33	7.25E+00	0.024	2	0.22	-	0.0012	-
Co-60	E		Beta Gauges	3	3.52E+01	0.021	23	12	0.016	0.012	16
Fe-55	T		Other	12	5.60E+02	0.018	5	47	-	0.0026	-
Ra-226	T		Other	12	1.20E-01	0.018	5	0.010	1.42E-05	0.0025	0.014
Fe-55	E		Beta Gauges	13	4.64E+02	0.015	4	36	-	0.0020	-
Ni-63	H		Gen Neut Src Apps	22	3.30E+02	0.011	2	15	-	8.33E-04	-
C-14	R		Gas Sources	42	2.10E+00	0.0069	1	0.050	-	2.78E-04	-
Sr-90	I		Calib Src A>30 mCi	1	5.00E-01	0.0049	17	0.50	-	0.0083	-
Ra-226	I		Calib Src A>30 mCi	1	2.00E-02	0.0030	10	0.020	2.85E-05	0.0050	0.029
I-129	T		Other	24	2.96E+01	0.0029	0	1.2	1.55E-04	2.05E-04	0.16
Ni-63	S		Foil Sources	6	6.99E+01	0.0023	1	12	-	6.47E-04	-
C-14	E		Beta Gauges	6	6.00E-01	0.0020	1	0.10	-	5.55E-04	-
Ni-63	U		X-Ray Fluorescence	8	3.91E+01	0.0013	1	4.9	-	2.72E-04	-
Eu-152	T		Other	56	1.08E+00	0.00040	0	0.019	1.44E-05	1.21E-05	0.014
C-14	Y		Calibrators	24	1.20E-01	0.00040	0	0.0050	-	2.78E-05	-
Ba-133	I		Calib Src A>30 mCi	93	1.53E+00	0.00023	0	0.016	7.51E-06	4.12E-06	0.0075
C-14	N		Ion Gen, Chromatog	10	1.00E-02	0.000033	0	0.0010	-	5.55E-06	-
Am-241	Y		Calibrators	6	6.79E-05	0.000025	0	1.13E-05	3.55E-09	7.08E-06	3.55E-06
Sr-90	Y		Calibrators	2	2.41E-03	0.000024	0	0.0012	-	2.01E-05	-
Ba-133	N		Ion Gen, Chromatog	5	8.79E-02	0.000013	0	0.018	8.01E-06	4.39E-06	0.0080
Ni-63	E		Beta Gauges	22	2.17E-01	7.15E-06	0	0.010	-	5.48E-07	-
Ni-63	T		Other	12	1.19E-01	3.93E-06	0	0.010	-	5.52E-07	-

Table Task-3-1C continued. Sorted by "Fractions of Total Ing. ALIs."

Nuclide	Device Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Fractions of Total Ing. ALIs decayed to 1992 (ppm)	Average Ing. ALIs/Device	Average Activity/Device (mCi)	Average Source Strength, Gamma*A of 10^-4 at 1 m from Source (rem/h @1m)	Dose from Ingestion of Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
Ru-106	E	Beta Gauges	1	9.65E-04	1.43E-06	0	0.0010	-	2.41E-06	-
Bi-210	E	Beta Gauges	1	2.79E-89	1.04E-92	0	2.79E-89	-	1.75E-92	-
Kr-85	D	Gamma Gauges	291	1.20E+05	0	0	413	0.00E+00	-	0.00E+00
Kr-85	E	Beta Gauges	623	2.43E+06	0	0	3,905	0.0061	-	6.1
Kr-85	N	Ion Gen, Chromatog	135	6.94E+04	0	0	514	8.05E-04	-	0.80
Kr-85	O	Ion Gen, Static Elim.	128	3.44E+02	0	0	2.7	4.21E-06	-	0.0042
Kr-85	R	Gas Sources	2	2.11E+03	0	0	1,054	0.0017	-	1.7
Kr-85	T	Other	11	2.20E+03	0	0	200	3.13E-04	-	0.31
Kr-85	U	X-Ray Fluorescence	13	5.44E+01	0	0	4.2	6.55E-06	-	0.0065
			<u>325681</u>							

Table Task-3-2 shows the total number of sources in each Device Code category. Since many of these contain activities greater than 20 mCi, the numbers of sources used as a basis for the ORAU Report would be less than these numbers reported here.

Table Task-3-2. Summary of All Database (Baggett 1993).

Device Code ^a	Device Definition ^b	Number of Devices ^a
31.5D	Gamma Gauges	24,679
31.5E	Beta Gauges	18,336
31.5H	General Neutron Source Applications	110
31.5I	Calibration Sources, A > 30 mCi	317
31.5N	Ion Generators, Chromatography	9,474
31.5O	Ion Generators, Static Eliminators	37,084
31.5R	Gas Sources	11,146
31.5S	Foil Sources	827
31.5T	Other	5,277
31.5U	X-Ray Fluorescence	3,763
31.5W	Self-Luminous Light Source	310,667
31.5Y	Calibrators	577
31.7W	Self-Luminous Light Source	71,315
TOTAL		493,572

^aAttachment to letter dated Nov. 2, 1993, from S.L. Baggett to D.J. Strom, entitled "Summary of all Database."

^bU.S. Nuclear Regulatory Commission Regulatory Guide 10.10, Appendix C (NRC 1987).

Taylor (1989) estimated that there were "approximately 30,000 general licenses in non-Agreement States using about 400,000 devices, and about twice this many in Agreement States." Using Taylor's overall factor of 3 and the figure of 493,572 provided by NMSS, one is led to conclude that about $3 \times 493,572$, or roughly 1.5 million generally licensed sources exist in the USA.

Registration rates for the various Device Codes are shown in Table Task-3-3. Production figures are presented in the "Enclosure 1"¹ and ORAU report (see Attachment 2, Table 1, from the

¹The data identified as "Enclosure 1" data, entitled "Estimated number of generally licensed devices and materials," dated 4/21/87, have been ascribed by S.L. Baggett to "part of an older Commission paper or NRC staff report." The exact provenance of this data table is uncertain, except that it was ultimately supplied to PNL by NMSS for this review.

ORAU Report). Production appears to have remained relatively constant for the major categories of sources identified in the ORAU report. Thus, a constant annual production was assumed to estimate the total number of remaining sources (with the assumption of decay for short-lived sources), as shown in Table Task-3-4. There is remarkable agreement between the 1987-92 figures and the data reported in earlier work.

The data in Tables Task-3-3 and Task-3-4 do not include isotopes, dates, and activities, and thus, are of very limited use in risk analysis.

The 1987 production and accumulation of total quantities of generally licensed sources is given in the "Enclosure 1" table, as shown in Table Task-3-5. A correspondence between the "Enclosure 1" categories and the ORAU categories is given where possible.

Table Task-3-3. Annual device registration rates based on 6-year averages (1987-1992) from "General License Database System Report for Peer Review."

Code	Device Definition	per year	per 6 years
D	Gamma Gauges	1083	6498
E	Beta Gauges	1289	7731
H	General Neutron Source Applications	14	86
I	Calibration Sources, A > 30 mCi	20	121
N	Ion Generator, Chromatography	866	5197
O	Ion Generator, Static Eliminator	4820	28918
R	Gas Sources	1825	10950
S	Foil Sources	14	86
T	Other	347	2084
U	X-Ray Fluorescence	426	2553
W	Self-Luminous Light Source	40466	242797
W7	31.7 Self-Luminous Light Source	3087	18519
Y	Calibrators	24	141
TOTALS		54280	325681

Table Task-3-4. Annual Production Rate and Numbers of Sources in Non-Agreement States from the "Enclosure 1" table dated 4/21/87 and from Table 4 of the 1987 ORAU Report by ORAU Report Category.

ORAU Category	"Encl. 1" 4/21/87 Annual Production Rate	"Encl. 1" 4/21/87 Total as of 1987	ORAU 1987 Table 4
A-1 Static Eliminators: Hand-Held/Portable/Small Brushes (10 CFR 31.3)	7000	20000	20000
A-2 Static Eliminators or Detectors: In Equipment or Process Line (Very High Toxicity Sources) (10 CFR 31.5)	80000	160000	170000
A-3 Static Eliminators or Detectors: In Equipment or Process Line (Low Toxicity Sources) (10 CFR 31.5)	120	9600	
B Gamma Gauges (10 CFR 31.5)	337	16000	4200
C-1 Beta Gauges: Backscatter Type (10 CFR 31.5)	800	7000	8000
C-2 Beta Gauges: Transmission Type (10 CFR 31.5)	-	-	
D Gas Chromatographs (10 CFR 31.5)			8000
E-1 X-Ray Fluorescence Analyzers: Very High Toxicity Sources (10 CFR 31.5)	90	720	720
E-2 X-Ray Fluorescence Analyzers: Moderate Toxicity Sources (10 CFR 31.5)	-	-	-
F Sources for Checking Detector Operation or Calibration and Analytical Reference Sources (10 CFR 31.5)	-	-	-
G-1 Self-Luminous Devices (10 CFR 31.5)	20000	180000	180000
G-2 Self-Luminous Devices in Aircraft (10 CFR 31.7)	7600	90000	90000
H Analytical Instruments Containing Small Calibration or Reference Sources (10 CFR 31.5)	> 600 (Liq Scint)	7000 (Liq Scint)	7000
I Sources for Checking Detector Operation or Calibration and Analytical Reference Sources (10 CFR 31.8)	60	480	2000
J Small Quantities of Source Material (10 CFR 40.22)	70 kg	2000 kg	-
K Sources for Checking Detector Operation or Calibration and Analytical Reference Sources (10 CFR 70.19)	Pu-239	NA	-
TOTAL (except J,K)	-	506745	487920

Table Task-3-5. Estimated Number of Generally Licensed Devices and Materials ("Enclosure 1," 4/21/87). ORAU Categories are given where there is a clear correspondence between the "Enclosure 1" categories (i.e., Regulatory Guide 10.10; NRC 1987) and the ORAU categories.

CFR Sec.	Dev-ice Code	Device Type	Number of Devices Sold per Year	Total Number of Devices	ORAU Category (if applicable)
31.3		Static Eliminator	7000	20000	A-1
31.5		Aerosol Neutralizer	120	9600	A-3
31.5	E	Beta Backscatter Gauge	800	7000	C-1
31.5	N	Electron Capture Detector	900	8000	D
31.5		Electrostatic Voltmeter	890	3000	H
31.5		Fill Level Gauge	600	4200	B
31.5		Fuel Densitometer Emitter	200	945	B
31.5	D	Gauging Devices (Part I)	337	16000	B
31.5		In Flight Blade Inspection Systems	200	1000	B
31.5		Liquid Scintillation Spectrometers	600	7000	H
31.5	W	Self-Luminous Exit Signs	20000	180000	G-1
31.5		Static Eliminators/Meters	80000	160000	A-2
31.5		X-Ray Fluorescence Spectrometer	90	720	E-2
31.7		Self-Luminous Aircraft Signs	7600	80000	G-2
31.8		Calibration or Reference Sources	60	480	
40.22		Source Material (Depleted Uranium)	140	2000	J
TOTAL			119397	497945	

The data provided by NMSS (Baggett 1993) shows 71,315 devices in Device Code 31.7W, self-luminous devices in aircraft. This is in reasonably good agreement with the 10 CFR 31.7 entry in the "Enclosure 1" Table (80,000) and with the ORAU estimate for Category G-2 (90,000).

The "Enclosure 1" Table, produced in 1987, shows 497,945 devices, and the NMSS table, produced 6.5 years later in 1993, shows 493,572. This is a remarkable coincidence with the ORAU Table 4 Total of 487,920 devices. One could safely conclude that there are about a half a

million generally licensed devices in existence under NRC General License, with perhaps twice that many in Agreement States (Taylor 1989).

In general, it has not been possible to establish a correspondence among the ORAU categories, the categories in the "Enclosure 1" Table of 4/21/87, and the data provided by NMSS (Baggett 1993). This has been confirmed in the letter dated March 30, 1994, from S.L. Baggett to D.J. Strom. To apply the ORAU risk analysis methods, it will be necessary to resolve what numbers of devices of what designs, sources, and activities are in use. This will require significant additional investigation and direct access to the database.

4.0 TYPES, FREQUENCIES, AND RELATIVE SEVERITIES OF IMPROPER TRANSFER/DISPOSAL OCCURRENCES

NMSS inspection report summaries (Piner 1990, Wheaton 1993) and the operational experience reports and bulletins issued by the NRC's Office of Analysis and Evaluation of Operational Data (AEOD) provided information on the types, frequencies, and relative severities of improper transfer/disposal occurrences having actual or potential public dose implications. These data were accessed through written reports and the Nuclear Regulatory Event Report (NRER) database files listed in Attachment 1. These files covered event reports during the period 1980 through 1992.

The NRER database was searched for occurrences of anything leading to events concerning generally licensed materials. Such events were found with a code of "GEN" or "GL" in the LIC_NO field, and in a couple of other instances. Many of the entries, however, either did not involve generally-licensed devices or materials, or involved materials whose range of activities exceeded the activity limits in the ORAU Report. The latter groups were categorized as "L," and tabulated with the other ORAU Report Categories. Several hundred entries under "NL" (not licensed) were also reviewed, revealing no events involving generally-licensed devices.

Results of the survey are summarized in Table Task-3-6 (number of incidents) and Table Task-3-7 (number of sources or devices). Table Task-3-6 shows incident rates by rows labeled with ORAU Category. Column entries are numbers of incidents of each kind that the PNL reviewers found in the NRER database. Since a given incident may involve more than one source, Table Task-3-7 shows number of devices involved in the incidents listed in Table-3-6 by rows labeled with ORAU Category. Many event reports covered multiple sources, including various nuclides (^{60}Co , ^{210}Po and ^{210}Pb in one case, and ^{241}Am , ^{210}Po , and ^{85}Kr in another case). By far the most common occurrences were in Category A2, static elimination sources, with ^{210}Po being the most commonly-involved nuclide.

In the tables, an incident can be classified in more than one category of outcome, such as lost and recovered, or damaged and leaking.

In most cases, the NRER database gives no information concerning severity. Sometimes a description like "significant event" occurs. On only a rare occasion is a dose to a person mentioned. Occasionally a description like "completely destroyed in a fire resulting in contamination" occurs.

Table Task-3-6. Number of incidents involving generally-licensed devices. Data are from the NRER database files listed in the attachment, for the period 1980-1992. Category "L" involves sources whose activities exceed the upper limits on activity in the ORAU Report.

ORAU Cat.	Incidents*	Lost	Stolen	Land-fill	Scrap-yard	Damaged	Leaking	Recovered	Misuse
A1	3	2				1			
A2	49	28	2	5	1	9	12	2	1
A3	1							1	
B	6	4	1		1	1		1	1
C1	1						1		
C2	1	1		1					
D	1			1					
E1	2	2							
E2	2	1				1			
F									
G1	11	1	5		2	2	2	1	
G2	1	1							
H	3	1			1		1	1	
I									
J	3	1					1		
K									
L	30	8	4	1	5	6	2	5	4
Total	114	50	12	8	10	20	19	11	6
Total w/o L	84	42	8	7	5	14	17	6	2

*Incidents may be less than row total because some reports are cataloged multiple times, e.g., "stolen and recovered."

Table Task-3-7. Number of devices involved in the incidents listed in Table Task-3-6. Data are from the NRER database files listed in the attachment, for the period 1980-1992 (there were no entries for 1980-82, so this is a 10-year period). Category "L" involves sources whose activities exceed the upper limits on activity in the ORAU Report.

ORAU Cat.	Devices*	Lost	Stolen	Land-fill	Scrap-yard	Damaged	Leaking	Recovered	Misuse
A1	14	9				5			
A2	89	54	2	5	3	28	27	4	3
A3	1								
B	9	6	1		1	2		1	3
C1	1						1		
C2	1	1		1					
D	1			1					
E1	2	2							
E2	2	1				1			
F									
G1	57	1	43		2	9	9	1	
G2	1	1							
H	5	1			1		3	1	
I									
J	4	2					1		
K									
L	113	12	51	1	7	35	34	6	7
Total	300	90	97	8	14	80	75	13	13
Total w/o L	187	78	46	7	7	45	41	7	6

*Devices may be less than row total because some reports are cataloged multiple times, e.g., "stolen and recovered."

There were 41 incidents tabulated in the ORAU report (Table 4 in Stabin et al. 1987, p. 13). Our analysis, displayed in Tables Task-3-6 and Task-3-7, shows 114 incidents involving 300 devices over a 10-year period, for rates of 11.4 incidents per year and 30 device-incidents per year. The ORAU incidents were based on only a few years worth of data, and included data from agreement state reports. It is also unclear whether the ORAU Report included numbers of devices involved (see Table Task-3-7.) Thus, the two totals are not strictly comparable. If the "L" incidents listed in Table Task 3-7 are not counted, the rates become 8.4/year and 18.7/year.

Incident breakdown by nuclide and ORAU source category is shown in Table Task-3-8. It is difficult to justify the relevance of the "L" category for this study. Ignoring L incidents, many of

which may be improperly classified in the database, the bottom row gives the total number of incidents for each nuclide. The nuclides, ^{210}Po and ^{241}Am , dominate with 44 and 17 occurrences, respectively, out of a total of 84 incidents. Clearly, row totals show that incidents are dominated by static elimination sources (49 of 84 generally-licensed incidents), with self luminous devices following at 11.

Table Task-3-8. NRER Incident Breakdown by Nuclide and Source Category. Table entries are number of incidents per 10 years.

ORAU Cat.	^{241}Am	^{109}Cd	^{60}Co	^{137}Cs	DU		^{85}Kr	^{63}Ni	^{147}Pm	^{210}Po	^{90}Sr	Z	Total
					^{238}U	^3H							
A1										3			3
A2	4									40		5	49
A3							1						1
B	1			5									6
C1									1				1
C2											1		1
D								1					1
E1	2												2
E2		2											2
F													0
G1						5	6						11
G2									1				1
H				1				1				1	3
I													0
J					3								3
K													0
L	10		2	6		5			3	1		3	30
Total	17	2	2	12	3	10	7	2	5	44	1	9	114
Total w/o L	7	2	0	6	3	5	7	2	2	43	1	6	84

5.0 CONCLUSIONS AND NEED FOR FURTHER WORK

The NRER data base gives a good idea of how many incidents have been reported. It would be good to find a formal method of estimating the number of unreported incidents, and for this a survey of field regulatory personnel may be the best approach.

A better study of the actual numbers of sources in use and in storage would be desirable. A more formal survey of manufacturers would provide a good basis for risk estimates for short-lived nuclides, such as Po-210, which appears to dominate the numbers of sources.

The readily available information examined here should provide an adequate basis for an order-of-magnitude risk analysis, which would be better than many risk analyses in non-radiation fields.

6.0 REFERENCES

Ayers, J. 1985. "General License Study - Analysis of Hazard." Internal memo to Division Director through channels, March 1985, attributed to J. Ayers. 8 pp. Washington, DC: Sealed Source Safety Section, U.S. Nuclear Regulatory Commission.

Baggett, S. 1987. *General License Study Report*. vii + 127 pp. Washington, DC: Sealed Source Safety Section, U.S. Nuclear Regulatory Commission.

Baggett, S. L. 1993. Attachments to letter addressed to Daniel J. Strom, dated November 2, 1993. The four hardcopy reports from "General License Database System" included one computer printout of six pages (plus a cover sheet) dated 10/22/93; one computer printout of two pages dated 10/20/93; one undated typewritten page titled "Summary of All Database;" and one undated 33-page report listing license numbers, model numbers, type numbers, and registration numbers for sources.

Dean, C. M., M. S. Lawrence, and H. D. Lester. 1991. *Report on Survey of General Licensees Under 10 CFR 31.5*. NRC FIN D 2554-0. ICF Inc., Fairfax, Virginia.

"ORAU Report:" see Stabin et al. 1987.

Piner, E., and B. Smith. 1990. *Conformance with Regulatory Requirements*. Also known as "Inspection Report Summaries." 50 pp. Washington, DC: Sealed Source Safety Section, U.S. Nuclear Regulatory Commission.

Sealed Source Safety Section and Policy and Procedures Section, U.S. Nuclear Regulatory Commission. 1992. *Program Code Descriptions Used in NRC Licensing and Inspection Programs*. 56 pp. Revision 2. Washington, DC: U.S. Nuclear Regulatory Commission.

Stabin, M., K. Paulson, and S. Robinson. 1987. *Improper Transfer/Disposal Scenarios for Generally Licensed Devices*. "The ORAU Report" produced under NRC FIN B0299. Oak Ridge Associated Universities, Oak Ridge, Tennessee.

Taylor, J. M. 1989. *Staff Initiatives on the General License Program*. Memo to the Commissioners of the U.S. NRC. SECY-89-289. Includes 4 attachments. Washington, DC: U.S. Nuclear Regulatory Commission.

Wheaton, A. 1993. *Analysis of Selected Incidents*. Also known as "Operational Experience Reports." Revision 2. Previous editions by E. Piner and B. Smith. Washington, DC: Sealed Source Safety Section, U.S. Nuclear Regulatory Commission.

U.S. Nuclear Regulatory Commission (NRC). 1987. *Guide for the Preparation of Applications for Radiation Safety Evaluation and Registration of Devices Containing Byproduct Material*. Regulatory Guide 10.10. U.S.N.R.C., Washington, DC.

TASK 3-ATTACHMENT 1: DOS FILES RECEIVED AND REVIEWED BY PNL

The following list contains the most recent versions of files reviewed by PNL staff for this task.

ADDRESS	DBF	176066	06-03-93	3:33p	D:\NMSS\SSD
B-BITXT	CSV	413152	04-07-93	4:56p	D:\NMSS\INCIDENT
BYCODE	NTX	14336	06-03-93	2:41p	D:\NMSS\SSD
CATALOG	CAT	439	03-17-92	10:17a	D:\NMSS\NRER
COMPANY	NTX	67584	06-03-93	2:41p	D:\NMSS\SSD
CUSTOM	DBF	25149	06-03-93	12:17p	D:\NMSS\SSD
CUSTOM1	NTX	5120	06-15-93	4:49p	D:\NMSS\SSD
CUSTOM2	NTX	10240	06-15-93	4:49p	D:\NMSS\SSD
EXP	DBF	299458	05-03-93	4:29p	D:\NMSS\NRER
EXP	XLS	158633	08-25-93	10:23a	D:\NMSS\NRER
LIC	NDX	49664	06-24-92	4:48p	D:\NMSS\NRER
NRER	ZIP	926512	01-14-93	11:08a	D:\NMSS\NRER
NRER0	DBF	186138	05-14-91	10:40a	D:\NMSS\NRER
NRER1	DBF	250880	07-19-92	8:47p	D:\NMSS\NRER
NRER2	DB	587469	04-20-93	9:34a	D:\NMSS\NRER
NRER2	DBF	306150	01-11-93	2:05p	D:\NMSS\NRER
NRER2	DBT	195584	01-11-93	2:05p	D:\NMSS\NRER
NRER8	DBF	119808	05-23-90	2:08a	D:\NMSS\NRER
NRER80-4	DBF	939378	08-20-92	6:07p	D:\NMSS\NRER
NRER85-7	DBF	769898	05-03-93	4:25p	D:\NMSS\NRER
NRER9	DBF	178818	01-04-80	1:48a	D:\NMSS\NRER
PRINUSE	DBF	1398	12-21-91	9:57a	D:\NMSS\SSD
REGNUM	NTX	73728	06-03-93	2:41p	D:\NMSS\SSD
REGNUM1	NTX	55296	06-03-93	2:41p	D:\NMSS\SSD
REGNUM2	NTX	47104	06-03-93	2:41p	D:\NMSS\SSD
SSD	DBF	2706829	06-03-93	3:33p	D:\NMSS\SSD
SSDS	EXE	324608	06-15-93	4:59p	D:\NMSS\SSD
TEMP	DBF	1155	04-26-93	7:18a	D:\NMSS\SSD
TEMP	NTX	152576	08-20-93	1:16p	D:\NMSS\SSD
UNTITLED	CAT	439	03-17-92	10:17a	D:\NMSS\NRER
YR86	DBF	38778	03-17-92	10:40a	D:\NMSS\NRER

TASK 3 ATTACHMENT 2: Table 1 from the ORAU Report (Stabin et al. 1987).

TABLE 1 CLASSES OF DEVICES FOR SCENARIO DEVELOPMENT*

APPLICABLE REGULATORY SECTION**	CLASS	DEVICE	RADIONUCLIDES AND MAXIMUM ACTIVITIES
31.3	A-1	Static Eliminators: Hand-Held/Portable/ Small Brushes	Po-210 - 0.50 mCi (18.5 MBq)
31.5	A-2	Static Eliminators or Detectors: In Equipment or Process Line (Very High Toxicity)	Po-210 - 100 mCi (3700 MBq) Am-241 - 0.0005 mCi (0.0185 MBq) Ra-226 - 0.0005 mCi (0.0185 MBq)
31.5	A-3	Static Eliminators or Detectors: In Equipment or Process Line (Low Toxicity)	H-3 - 250 mCi (9250 MBq) Kr-85 - 2 mCi (74 MBq)
31.5	B	Gamma Gauges	Co-60 - 10 mCi (370 MBq) Cs-137 - 20 mCi (740 MBq) Am-241 - 20 mCi (740 MBq) Ra-226 - 10 mCi (370 MBq)
31.5	C-1	Beta Gauges: Backscatter Type	Sr-90 - 0.025 mCi (0.925 MBq) Tl-204 - 0.10 mCi (3.7 MBq) Ru-106 - 0.025 mCi (0.925 MBq) Pm-147 - 0.050 mCi (1.85 MBq) C-14 - 0.050 mCi (1.85 MBq) Pb-210 - 0.010 mCi (0.37 MBq)
31.5	C-2	Beta Gauges: Transmission Type	Sr-90 - 20 mCi (740 MBq)
31.5	D	Gas Chromatographs	Ni-63 - 20 mCi (740 MBq) H-3 - 1000 mCi (37 GBq)
31.5	E-1	X-Ray Fluorescence Analyzers (Very High Toxicity)	Am-241 - 30 mCi (1100 MBq) Cm-244 - 100 mCi (3700 MBq)

TASK 3 ATTACHMENT 2 (continued)

TABLE 1 CLASSES OF DEVICES FOR SCENARIO DEVELOPMENT* - CONTINUED

APPLICABLE REGULATORY SECTION	CLASS	DEVICE	RADIONUCLIDES AND MAXIMUM ACTIVITIES
31.5	E-2	X-Ray Fluorescence Analyzers (Moderate Toxicity)	Cd-109 - 20 mCi (740 MBq) Fe-55 - 100 mCi (3700 MBq)
31.5	F	Calibration or Reference Sources	Cs-137 - 0.10 mCi (3.7 MBq) Co-60 - 0.01 mCi (0.37 MBq) Ra-226 - 0.004 mCi (0.15 mBq) Sr-90 - 0.001 mCi (0.037 MBq)
31.5	G-1	Self-Luminous Devices	H-3 - 5000 mCi (185 GBq) Kr-85 - 1700 mCi (62.9 GBq) C-14 - 0.10 mCi (3.7 MBq)
31.7	G-2	Self-Luminous Devices in Aircraft	H-3 - 5000 mCi (185 GBq) Pm-147 - 300 mCi (11 GBq)
31.8	H	Analytical Instruments Containing Small Calibration or Reference Sources	Cs-137 - 0.040 mCi (1.5 MBq) Ni-63 - 15 mCi (555 MBq)
31.8	I	Calibration or Reference Sources	Am-241 - 0.005 mCi (0.185 MBq)
40.22	J	Small Quantities of Source Material	U-238 and Th-232 - 15 pounds at any one time, no more than 150 pounds per calendar year
70.19	K	Calibration or Reference Sources	Pu-239 - 0.005 mCi (0.185 MBq)

* See Appendix for device descriptions
 ** Code of Federal Regulations, Title 10

APPENDIX F TO
FINAL REVIEW OF THE 1987 REPORT BY
OAK RIDGE ASSOCIATED UNIVERSITIES,
"IMPROPER TRANSFER/DISPOSAL SCENARIOS FOR
GENERALLY LICENSED DEVICES:"

TECHNICAL LETTER REPORT:
TASK 6, DEVELOPMENT OF ADDITIONAL PROBABILITY AND RISK
INFORMATION

NRC JOB CODE L2536
PNL No. 20278

D. J. Strom¹
R. L. Hill¹
J. S. Dukelow²

¹Health Protection Department
²Nuclear Systems and Concepts Department
Pacific Northwest Laboratory
Richland, Washington 99352

January 31, 1994
revised February 4, 1994
revised June 2, 1994

Prepared for the U.S. Nuclear Regulatory Commission
Under Contract DE-AC06-76RLO 1830

1.0 EXECUTIVE SUMMARY

Task 6 of the project "Review of Improper Transfer/Disposal Scenarios for Generally Licensed Devices Study" is entitled "Development of Additional Probability and Risk Information." For this task, ... "the individual reviewers will develop additional probability and risk information in their area of expertise to assist the NRC staff's decision making regarding the need for regulatory action. This information may include refined estimates of probabilities associated with selected disposal scenarios, assessments of the sensitivity of consequence and risk calculations to different assumptions and inputs, and quantitative estimates of individual and population risk resulting from selected improper disposal activities."

Two areas of additional risk information have been developed by the PNL reviewers. The first area of additional risk information that was identified in PNL's preliminary review of the 1987 ORAU Report is the need for a mathematical framework or "formula" for the risk of radiological accidents. This framework should address two items of methodology missing from the ORAU Report; the probabilities of initiation of accident sequences, and the use of historically-derived probability distributions of accident consequences (which include worst cases as their extremes).

The second area of additional risk information is a quantitative characterization of relevant historical accidents with sources, whether generally licensed or not. The quantitative characterization results in three numerical factors for each person involved in each accident for whom intake, whole body dose, and local dose values are published. The numerical values are "fraction taken in," "whole body time-and-proximity factor," and "local dose time-and-proximity factor." These factors depend only on human behavior and accident circumstances, not on the amount, kind, and quantity of radioactive material involved in the accident. The factors can be used to predict more realistically the radiological consequences of future accidents than the use of "worst case" factors. Furthermore, over 10 accidents involved exposures to two or more people, resulting in *distributions*, rather than point estimates, of values. Such distributions can be used as inputs to modern probabilistic risk calculations.

2.0 PROBABILISTIC MATHEMATICAL FRAMEWORK

The first area of additional risk information that was identified in PNL's preliminary review of the 1987 ORAU Report is the need for a mathematical framework or "formula" for the risk of radiological accidents. This framework uses individual and collective radiation dose as surrogates for risk, and considers both the magnitude and probability of occurrence of various doses. The PNL reviewers have identified two probability considerations as missing from the ORAU Report. The probabilities of initiation of accident sequences, and the likely (rather than worst case) consequences are included. Using historically-derived probability distributions of accident consequences (which include worst cases as their extremes) enhances the realism of risk estimates calculated from postulated accidents.

2.1 ASSESSMENT OF THREE KINDS OF DOSES FOR USE AS SURROGATES FOR HUMAN HEALTH RISKS

Risk is conventionally defined as

$$Risk = Probability \times Severity . \quad (1)$$

There may be several separate components to the probability term: probability of an accident happening, probability of a given dose resulting when the accident happens, and probability of that dose resulting in a stochastic health effect, for example. For human health risks due to radiation exposure, various dose quantities multiplied by suitable health risk coefficients may be used as surrogates *severity* (ICRP 1991). At low doses, *severity* may connote the *likelihood* of a severe effect such as cancer occurring in an individual. At higher doses exceeding thresholds for deterministic effects, "severity" has a more conventional meaning for an individual, such as how serious a burn is.

Severity in Equation 1, for incidents involving sealed sources, can be defined both as individual doses and collective doses, that is, the sum of all doses accruing to all individuals in a given incident.

An individual tissue or organ dose equivalent, if below 50 rems, carries no risk of deterministic (formerly "non-stochastic") health effects (such as radiation burns, developmental abnormalities, etc.), but represents some degree of risk for stochastic effects (i.e., cancer and heritable ill-health). Such individual tissue or organ dose equivalents from internal and external exposure can be combined to form a total effective dose equivalent (TEDE), which is a modern surrogate for stochastic risk in individuals. Current risk estimates are on the order of 4×10^{-2} per sievert (4×10^{-4} per rem) for adverse stochastic health outcomes in workers, and perhaps 5×10^{-2} per sievert (5×10^{-4} per rem) in the general public (ICRP 1991).

Under the linear, non-threshold dose response hypothesis for stochastic effects used for radiation protection purposes, individual TEDE values can be summed to make collective total effective dose equivalent ("collective dose"). Collective dose is a surrogate for collective risk of adverse stochastic health outcomes in populations.

For improper transfer and disposal scenarios for generally licensed devices, it is also necessary to consider the possibility of individual tissue or organ doses that exceed thresholds for deterministic effects. Such doses result in certain injury, whether sub-clinical, mild, severe, or fatal, to the individual receiving the dose. Doses above a few tenths of a sievert (a few tens of rems) should be expressed in absorbed dose units, i.e., grays (or rads), specifying the radiation type, since the relative biological effectiveness of high linear energy transfer (LET) radiation (e.g., neutrons and α particles) at such dose levels is significantly less than the quality factor used for limitation of stochastic effects (ICRP 1991). Furthermore, no clinical effects may occur whatsoever from protracted irradiation

significantly exceeding the traditional deterministic threshold of 0.5 Gy, since significant repair can occur between damaging events on a microscopic scale. For this reason, *committed* doses of long-lived, tenaciously-retained radionuclides will be poor predictors of deterministic effects.

Thus there are three dose endpoints that should be considered in a risk analysis: Distributions of individual TEDEs, the collective TEDE, and distributions of individual tissue or organ dose above thresholds, such as 0.5 Gy for acute irradiation and perhaps 1 Gy or more for protracted irradiation.

$$\begin{aligned}
 \text{Individual Stochastic Risk} &= \text{Probability} \times \text{TEDE} \\
 \text{Collective Stochastic Risk} &= \text{Probability} \times \text{Collective TEDE} \\
 \text{Individual Deterministic Risk} &\propto \text{Probability of Dose Above a Threshold} \\
 &\quad \times \text{Dose Effect Function}
 \end{aligned}
 \tag{2}$$

The probabilities of various doses being received from a given improper transfer/disposal scenario are related both to the probability of the scenario occurring and the probability distribution of doses resulting from the scenario. Finally, risks are summed over all scenarios.

Ecological risk is the risk to ecosystems, habitats, and potential loss of access and usability of land and environmental resources. Although improper transfer and disposal scenarios for generally licensed sources may result in ecological risks, they are not considered here.

2.2 PROBABILISTIC RISK METHODOLOGY

Since 1987, many changes have occurred in probabilistic risk methodologies. Recent summaries of these techniques are provided by IAEA (1989), Finkel (1990), and Morgan and Henrion (1990). In addition, the advent of user-friendly Monte Carlo simulation software for probabilistic health risk analysis, such as Crystal Ball (™Decisioneering, Inc., Denver, CO), makes it feasible to perform probabilistic risk assessments for this kind of work.

2.1.1 Risk Networks

The draft ORAU report contains, for each class of Generally Licensed Sources, a "risk network" connecting the Initial Events (listed in Table 2 of the report) with the Final Status of Device conditions (listed in Table 3). Figure 1 is an example risk network from the draft report. Each of these networks starts with the assumption that a device of that class has been improperly transferred/disposed. Along the left side of the network is a collection of Initial Events, or states that the device can be found in after improper transfer/disposal. Along the right side of the network are the Final Status of Device conditions. In between is a collection of transition conditions and a collection of paths leading (from left to right) from the Initial Events, perhaps through one or more of the transition conditions, ending in one of

CLASS A-2 EQUIPMENT STATIC ELIMINATORS

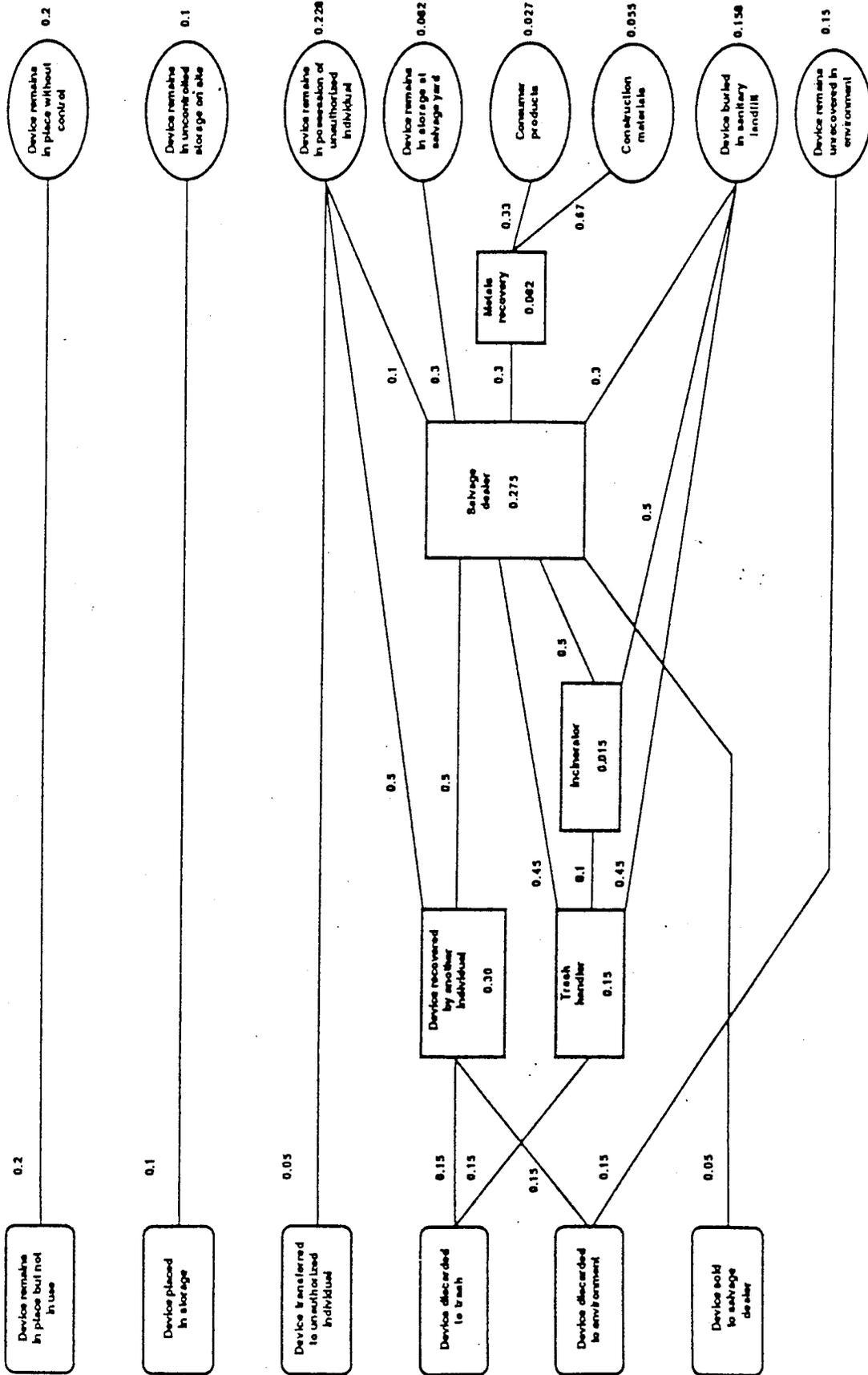


Figure 2.2.1. An example of a risk network from the ORAU Report (Figure 3).

the Final Status of Device conditions. Associated with each of the path segments between two of the condition boxes is the conditional probability of the transition from the left box (i.e., the left end of the path segment) to the right box. Finally, along the right side of the network, associated with each of the Final Status of Device conditions is the conditional probability of ending up in that state, obtained by adding up the probabilities associated with each of the distinct paths through the risk network that terminate in that state.

2.2.1 Adequacy of ORAU Report Risk Networks

We feel that the conditional probabilities assigned (using engineering judgment) to the path segments by the authors of the ORAU draft report are generally reasonable. There is, however, a structural aspect of these risk networks with which we take issue: all of the networks have a pathway leading from "trash handler" to "incinerator" and thence on to either "salvage dealer" (with conditional probability 0.5) or to the Final Status of Device condition "Device buried in sanitary landfill" (with conditional probability 0.5). This seems to presuppose that the trash handler sifts the incinerator ashes for metal/ceramic slag which is sent on to the salvage dealer, and that half the time (i.e., probability 0.5) the remains of the source are incorporated in that slag. The probability 0.5 of "source-in-slag" seems high. In addition, for those cases in which the bulk of the source is volatilized in the incinerator and released to the atmosphere, the box "incinerator" has effectively become a Final Status of Device condition for which health consequences should be assessed. Several potentially significant scenarios, such as an intact source out of a shield, and potentially significant consequences, such as doses to workers (rather than the public), have been omitted. Finally, one could argue that the incinerator box ought to be "downstream" of the salvage dealer, with incineration as one of the salvage dealer's options for dealing with items that incorporate both salvageable (i.e., metals) and non-salvageable materials.

What is missing from this picture, if we desire to estimate the risks associated with the improper transfer/disposal of various types of Generally Licensed Devices? Missing is the probability of entering the risk network in the first place (i.e., the probability that a device of that type will be improperly transferred/disposed) and the consequences associated with each of the Final Status of Device conditions.

2.2.2 Censoring of Scenario Probabilities by Under-Reporting

Table 4 of the draft ORAU report contains a computation of the fraction of devices of various types that have been improperly transferred/disposed, based on very limited data. These values cannot be directly used as improper transfer/disposal probabilities because of an obvious censoring problem. Improperly transferred/disposed devices show up in Table 4 only if the improper transfer/disposal was detected in some fashion. This detection could occur if an appropriately labelled device was found somewhere it didn't belong, if an inspection of records and device inventory discovers that a device is missing, or if someone's medical symptoms can be tied directly to the improper transfer/disposal of a specific device. If a device was buried in a landfill or incinerated improperly, that improper transfer/disposal

is unlikely to be detected, since inspections and audits of holders of General Licenses seem to sample only a small fraction of the total population of General Licensees.

2.2.3 Correcting Probabilities for Under-Reporting of Incidents

One way of dealing with this censoring would be to obtain an estimate for the probability $P(d)$ of the event d , where d denotes the detection, in some fashion, of the improper transfer/disposal of a particular Generally Licensed Device.

We can then take the fraction of devices of that class known to be improperly transferred/disposed, F (from Table 4), and increase it by the factor $1/P(d)$ to obtain the total fraction of devices of that type improperly transferred/disposed. That is, we assume that F is telling us only about that subset of the set of improperly transferred/disposed devices for which the improper transfer/disposal is detected in some fashion; that subset is only the $P(d)$ -the part of the whole set of improperly transferred/disposed devices.

In other cases (see, for instance, the discussion of industrial process line static eliminators in Section 3.2 of the ORAU report), there is enough information to directly estimate the fraction of devices that are improperly transferred/disposed, without reference to Table 4. In this case, all of the devices are out on lease and the distributor simply offers to charge for an additional year's lease if the device isn't returned for legally authorized disposal.

Suppose we are going to estimate $P(d)$. We might consider d to be the union or "sum" of two events:

- $a =$ the event that the improper transfer/disposal of the source is detected by observation of the label, by radiation detection, by inspection of records and inventory, etc.; and
- $b =$ the event that radiation sickness or injury is recognized and tied back to the improper transfer/disposal of the source.

We can then calculate

$$P(d) = P(a) + P(b) - P(ab), \quad (3)$$

where ab is the set intersection of the events a and b . If $P(a)$ and/or $P(b)$ are relatively small, say 0.1 or less, then the term $P(ab)$ will be second order and can be ignored.

The probability $P(a)$ will depend on a variety of factors, including the durability and intrusiveness of the labelling, the likelihood of "fortuitous" radiation detection, the pervasiveness of inspections and audits, the unwieldiness of the individual sources, etc. By fortuitous radiation detection, we mean something similar to discovery of high indoor radon levels in the Reading Prong because a Pennsylvania nuclear plant worker set off one of the portal detectors as he was arriving for work or the outside world's first knowledge of the

Chernobyl accident when area detectors outside a Swedish nuclear plant started alarming. We could also reasonably expect $P(a)$ to vary by type of source.

Reviewing the history of improper transfer/disposal of sealed sources, even "strong", specifically licensed sealed sources used in radiography or radiotherapy, we see dozens of incidents in which a sealed source improper transfer/disposal is first detected by recognition of radiation sickness or injury, followed by an investigation to determine the cause of that sickness. These detections are noted by the entry "Med" for "medical" detection of the incident with the subsequent re-establishment of control over the source.

We know of no such incidents involving Generally Licensed sources or equivalent amounts of radioactivity. Because of this, $P(b)$ is likely to be significantly smaller than $P(a)$ for most generally licensed sources. It would be reasonable to model $P(b)$ as directly proportional to the worst-case and/or the average human health consequences of improper transfer/disposal of that type of source.

We propose a model for $P(b)$ of the following type:

$$P(b) = k_a \cdot f(H_{T,max}) + (1 - k_a) \cdot g(H_{T,avg}), \quad (4)$$

where $H_{T,max}$ = dose to the critical organ for the maximally exposed individual;
 $H_{T,avg}$ = dose to the critical organ for the average individual;

$$f(H_{T,max}) = \begin{cases} 0 & \text{if } H_{T,max} \leq H_{th}; \\ \frac{H_{T,max} - H_{th}}{H_{cr} - H_{th}} & \text{if } H_{th} < H_{T,max} < H_{cr}; \\ 1 & \text{if } H_{T,max} \geq H_{cr}, \end{cases} \quad (5)$$

H_{the} is the average individual threshold dose for developing clinical symptoms;

H_{cr} is the critical threshold dose for certain diagnosis of clinical symptoms;

$g(H_T)$ is defined similarly to f , using values $H_{the,pop}$ appropriate for failure to detect and $H_{cr,pop}$ for sure detection of the incident in a whole population exposed to a given dose H_T ; and

k_a = an appropriately defined weighting factor for balancing between detection based on a symptomatic maximally-exposed individual and detection based on symptoms in a population; $0 \leq k_a \leq 1$.

2.3 RECOMMENDATIONS FOR IMPLEMENTING THE PROBABILISTIC RISK METHODOLOGY

To carry out the program defined in the preceding paragraphs, we need to do the following:

- 1) For each Generally Licensed sealed source type x , reassess the improper transfer/disposal fractions, F_x , defined in Table 4 of the ORAU draft report, on the basis of additional data, as available. The annual *rates* of incidents of improper transfer or disposal as a function of source category, including probabilities of incidents not being reported were not adequately assessed in the ORAU Report. Improved data will be available from the revised Task 3 report.
- 2) For each Generally Licensed sealed source type x (i.e., A-1, A-2, A-3, B, etc.) and for each radionuclide that can be utilized for devices of that type, produce estimates of either $P(Mx)$ directly or of $P(ax)$ and $P(bx)$. If $P(ax)$ and $P(bx)$ are estimated, then $P(dx) = P(ax) + P(bx)$ and $P(Mx) = F_x/P(dx)$. If Mx is the event that a device of Type x is improperly transferred/disposed, we can use the probability $P(Mx)$ to "enter" the corresponding risk network (whether that probability is estimated directly or obtained by estimating the probability $P(d)$ and factoring up the fraction, F , from Table 4). The probability $P(Mx)$ will propagate through the network and each of the Final Status probabilities (from the ORAU draft) will be multiplied by $P(Mx)$ to give the probability that a particular device of that class will end up in that Final Status of Device. We can then multiply that probability by the associated health consequences, $C_{x,fs}$, for device type x and that Final Status fs , to obtain the risk with a single device of that type ending up in that particular Final Status. Those risks can be summed for all of the Final Statuses to obtain the risk associated with a single device of that type, which can then be multiplied by the number of such devices extant to obtain the total risk to the public associated with that type of device.
- 3) For each Final Status of Device end state, produce estimates of the human health consequences resulting from a device ending up in that state, either using information in the draft ORAU report or additional information, as necessary.

- 4) Define the risk associated with a particular type x of device in a particular end-state as:

$$R_{x,fs} = P(M_x) \cdot p_{x,fs} \cdot (\text{consequences associated with the device } x \text{ final state } fs), \quad (6)$$

where $p_{x,fs}$ is the end state probability from the corresponding ORAU draft report risk network.

- 5) Define the total risk, R_x , associated with a single device of type x as the sum of all the $R_{x,fs}$ over the set of final states fs . Finally, the total risk associated with devices of type x is the product of the number of devices of type x and R_x .
- 6) We can do a simple uncertainty analysis by replacing all of the estimated probabilities and the probabilities given in the ORAU draft report by probability distributions and using random variable arithmetic to propagate those distributions through the calculations described above. The quick and dirty part would use just the mean and

variance of the distribution (or the mean and variance of the log-transformation of the distribution) and the Central Limit theorem to replace sums of random variables by the normal distribution with the appropriate mean and variance and replace products of random variables by the lognormal distribution with the appropriate mean and variance. These approximations give good results for random variables with a central tendency and arithmetic calculations consisting of several sums and products (roughly speaking the more sums and products, the better the approximation).

3.0 QUANTITATIVE ASSESSMENTS OF HISTORICAL IMPROPER TRANSFER/DISPOSAL INCIDENTS AND ACCIDENTS FOR INPUT TO PROBABILISTIC RISK METHODOLOGY

Many new risk assessment tools have been developed since 1987 when the ORAU Report was finalized. Furthermore, the PNL reviewers believe that the probabilistic nature of risk assessment results rather than worst case scenarios need to be emphasized. We have analyzed historical accidents involving sources to yield quantitative characterizations of human behavior in accident situations as input to probabilistic risk assessments. Nuclear weapons, nuclear fuel cycle, and criticality accidents and accidents involving accelerators or x-ray machines have not been analyzed, since it is difficult to determine the relevance of these to improper transfer/disposal scenarios for generally licensed devices.

3.1 DISTRIBUTIONS OF RESULTS RATHER THAN POINT ESTIMATES

We have performed a quantitative characterization of relevant historical accidents with sources, whether generally licensed or not, for those accidents for which necessary data are available in the literature. The quantitative characterization results in three numerical factors for each person involved in each accident for whom intake, whole body dose, and local dose values are published. The numerical values are "fraction taken in," "whole body time-and-proximity factor," and "local dose time-and-proximity factor." For the most part, these factors depend only on human behavior and accident circumstances, not on the amount, kind, and quantity of radioactive material involved in the accident. The factors can be used to more realistically predict the radiological consequences of future accidents than the use of "worst case" factors. Furthermore, over 10 accidents involved exposures to two or more people, resulting in *distributions*, rather than point estimates, of values. Such distributions can be used as inputs to modern probabilistic risk calculations.

3.2 TIME-AND-PROXIMITY FACTORS FOR WHOLE-BODY AND LOCAL EXTERNAL IRRADIATION

There is a need for risk analysis for accidents involving single radionuclide radioactive sources (as opposed to nuclear reactor or nuclear weapons accidents). Such risk analysis requires knowledge of the probabilities and severities of such accidents. Historically, accidents have involved anywhere from one to several thousand people.

Information for risk analysis of small, generally-licensed sources can be derived from accidents, usually involving large sources, that have already happened.

The objectives of this study were to

- Develop generalized, quantitative descriptions of human behavior and interactions with radiation sources from study of historical accidents;
- Evaluate applicability, advantages, disadvantages, and limitations of the approach; and
- Identify additional information needed to apply these risk estimates to quantitative risk assessments for informed regulatory decision making.

Historical records of accidental human interactions with radiation sources are used to develop distributions of external radiation exposure factors. These factors retain information about human behavior from the accidents, but are independent of source strength or the radionuclide(s) involved. For external exposures, we define a "time-and-proximity" exposure factor, F_p , with and without provision for breach of source shielding, for each person involved in an accident. The distributions of exposure factors can be used to predict ranges of possible radiation doses to individuals from a variety of accident scenarios involving different radionuclides, activities, and device designs. Use and limitations of the exposure factors and their distributions are discussed below.

One limitation is the possibility that the exposure factor distributions based on accident information are altered when the accidents are discovered. For example, deaths or symptoms of acute radiation syndrome may lead to an investigation, discovery of an accident, and termination of exposure. Distributions of factors from such an accident may be of limited applicability to accidents that go undiscovered.

3.2.1 Mathematical Description of Time and Proximity Factors

For an unshielded point source, dose equivalent H depends on exposure time t (hours), distance r (meters), source strength A (activity in Ci) and isotope (through Γ in rem/hr m^2 /Ci or Sv/h m^2 /Bq):

$$\begin{aligned} H &= \int \dot{H} dt \\ &= \int \frac{\Gamma A dt}{r(t)^2} \\ &= \Gamma A \frac{t}{r^2}. \end{aligned} \tag{7}$$

For each individual i , exposed in an incident for whom the dose equivalent H is known, the

source isotope and activity are known, one can calculate a time-and-proximity factor for the incident:

$$F_{p,i} = \frac{H_i}{\Gamma A} = \frac{t}{r^2}. \quad (8)$$

This factor is independent of both source strength and radionuclide involved. Distributions of time-and-proximity factors can then be applied to similar accidents to determine external exposures, even those involving a radioactive source of different isotope and activity. One simple definition of F_p is the number of hours one would have to spend at one meter from the source to receive a dose equivalent of H.

If the source remained partially or wholly shielded, then an additional factor should be introduced:

$$F_{p,i} = \frac{H_i}{F_s \Gamma A} = \frac{t}{F_s r^2}. \quad (9)$$

where F_s is the fraction transmitted through a shield, a number less than 1. F_s can be taken as the dose rate at 1 m from the source (in its shield) to the unshielded dose rate at 1 m.

When sources are sub-divided, the full activity is still used in the calculations because the human interaction is what we want to characterize, not the immediate source. So doses from the 1984 Mexican accident, for example, are attributable to the entire source.

3.2.2 Theoretical Limits Are Not Useful

The theoretical upper limit on F_p is

$$\frac{1 \text{ lifetime}}{(\text{very close})^2}, \quad (10)$$

where (very close) represents a small distance from the source, e.g., 0.01 m. For a weak source, this represents about 10^9 to 10^{10} hours at one meter, a quantity so large as to be useless. However, in tens of historical accident cases, a person (and in two of the worst accidents, in Mexico in 1962 [5 fatalities] and Morocco in 1984 [8 fatalities], a child) has found an industrial radiography source and put it in a "hip pocket." In many cases, the "very close" is 1 cm or so, and exposure times have been up to several months. When large sources are involved, such as industrial radiography sources, these cases result in local radiation burns. The ratio of the average bone marrow dose to the dose at the site of the radiation burn is dependent on "how far the bone marrow is away from the hip pocket." In many cases, the bone marrow to burn site dose ratio is over 100, sometimes over 1000. In other words, one needs to look at truly potential exposures which may result in a high dose,

but are realistic.

3.2.3 Historically-Derived Values of Time-and-Proximity Factors

Table 3.2.1 shows an analysis of 42 incidents involving external exposure. Accidents are characterized by year of occurrence, by nuclide, source type and activity. The number of people involved (broken down as public, workers, and cleanup workers) is given. Each accident is characterized by whether it involved brief or protracted whole-body irradiation, brief or protracted localized irradiation, whether there were intakes, whether the source was removed from the shield, and whether the source was damaged. References to the literature are given. Specific dose equivalent rate constants are tabulated, along with dose rates at 1 meter from the sources. The table contains average and geometric mean values of whole-body time-and-proximity factors, as well as minimum, maximum, and standard deviation and geometric standard deviation values, where appropriate. Also tabulated are maximum values of skin dose or local irradiation time-and-proximity factors.

These 42 incidents were chosen because of the availability of data on source identity, source activity, whole body and/or local doses to individuals, incident descriptions. The NRC's Office of Analysis and Evaluation of Operational Data (AEOD) incident database does not, in general, contain the information needed for this kind of analysis. Nuclides include ^{241}Am (1 accident), ^{60}Co (20), ^{137}Cs (4; note that Goiânia is listed 4 times for various analyses), ^{131}I (1), and ^{192}Ir (16). Accidents included 21 industrial radiography sources, 8 sterilization facilities, 5 teletherapy sources, 3 experimental sources, 2 brachytherapy sources, 1 defense incident, and 1 "radiation station." Members of the public were involved in 14 incidents, in numbers ranging from 1 person to 4000 persons, totalling roughly 7000. Four accidents (Mexico 1983 [4000], Morocco 1984 [28], Brazil 1987 [2800], and Pennsylvania 1992 [94]) account for virtually all of this total 7000 people. There were 29 incidents involving exposure at work, and 6 incidents involved exposure to cleanup or recovery workers.

The weighted average of the whole body is 1.28 hours at a meter. This is far below the 3360 hours exposure time assumed as a worst case in the ORAU Report (i.e., 20 weeks \times 7 days per week \times 24 hours per day = 3360 [page i, Stabin et al. 1987]). Ignoring the Texas child-abuse case of 1972, whole-body time-and-proximity factors ranged from essentially zero to 686 hours at a meter, the latter deriving from the 1962 Mexican incident (which formed the basis for the ORAU Report value of 3360). For only accidents exposing the public, time-and-proximity factors averaged 117 hours at a meter (averaging over accidents) and 1.37 hours at a meter (using a weighted average over all accident victims, for which accidents with many victims dominate the average).

The 16 available maximum local time-and-proximity factors formed a highly skewed distribution ranging from 0.05 to 24000 hours at a meter, with an average of 3100, a standard deviation of 6900, and a geometric standard mean of 91. The average of 3100 is comparable to the 1962 Mexico ^{60}Co accident whole body factor.

Four accidents in particular are especially interesting, due to their nature and the potential for generalization to improper transfer and disposal scenarios for generally licensed devices. These are the November 1992 Indiana, Pennsylvania accident; the 1983-84 Mexican ^{60}Co accident, and the 1987 Goiânia, Brazil accident, and the 1990 Korean source shipment accident.

The November 1992 Indiana, Pennsylvania case of a medical misadministration is particularly instructive for improper transfer and disposal scenarios for generally licensed devices (NRC 1993a, NRC 1993b). Doses are listed in the Appendix. This was a case of complete loss of control of the source, and the persons exposed did not know about it until the source was essentially "out of harm's way." The ^{192}Ir source activity was 3.7 Ci ($1.37\text{E}11$ Bq). The specific dose equivalent rate constant, Γ , is 0.48 rem/hr m^2/Ci . The central (i.e., halfway between the minimum and maximum doses) estimates of doses ranged from 0.0006 to 18.9 rems, with an average of 3.0 ± 4.1 rems, a geometric mean of 0.29 rems $\times 25$ (i.e., a GSD of 25). The time-and-proximity factors ranged from 0.0003 to 8.6 for the 85 cases that the NRC (1993b) evaluated. A preliminary investigation of the distribution of time-and-proximity factors for this incident shows that, while the distribution is very broad and skewed, it is not lognormal (Figure 3.2.1). There were a number of persons who evidently received roughly the same doses due to similar duties involving patient care, so these may need to be treated separately from doses to the others involved.

The 1983-84 Mexican ^{60}Co accident (IAEA 1989) was also a complete loss of control of a large source ($1.7\text{E}13$ Bq (450 Ci) of ^{60}Co ; $\Gamma = 3.6\text{E}-13$ Sv/hr m^2/Bq (1.3 rem/hr m^2/Ci)). This incident resulted in an estimated 4000 persons exposed. From the data, the largest F_p was 1.2, the mean of lognormal fits (with GSDs of 13.8 (uniform weighting) and 22.5 (Finney weighting)) were 0.004 to 0.008. In this situation, symptoms of acute radiation syndrome would have prevented much larger F_p values, since the $F_p = 1.2$ corresponds to 700 rems.

The 1987 Goiânia, Brazil accident presents four different populations for analysis, and it is difficult to determine whether some individuals may appear in more than one of these populations. Depending on how many persons are included, the time-and-proximity factors for the public plus the 9 workers involved with the source average 0.20, 0.11, or 0.0034 hours at a meter, with the former population including 97 persons, the next, 249 persons, and the last, 2812 persons. The maximum whole-body time-and-proximity factor was 1.46, and minimum factors on the order of 10^{-9} . The maximum local irradiation time-and-proximity factor was 4.2 hours at a meter. Figure 3.2.2 shows a histogram and cumulative density function for whole-body time-and-proximity factors from the Goiânia accident for the 46 highest dose cases, with a roughly logarithmic horizontal axis.

The 1992 Korean Accident (NUREG 1405, NRC 1990) includes dose estimates for 24 individuals. These are clearly lognormally-distributed as shown in Figure 3.2.3. Doses are listed in the Appendix. The average time-and-proximity factor was 1.3 ± 3.7 hours at a meter, with a geometric mean of 0.11 $\times 7.9$ (GSD = 7.9), and a range of 0.0063 to 14.7.

A Time-and-Proximity Factor analysis has been performed for 40 accidents involving 231 individual doses. The complete results (231 lines of data) are presented in the Appendix. The results are shown in Figure 3.2.4 and given in Table 3.2.2.

3.2.4 Censoring of factors for high-dose accidents by the appearance of clinical symptoms of acute irradiation

In many cases, accidents were discovered by the appearance of clinical symptoms of acute irradiation. In many cases, persons stopped receiving any more dose because they died. In such cases, it can be concluded that the accident did not run its normal course. A plot of time-and-proximity factors versus source strength, ΓA , in sieverts per hour at 1m from an unshielded source, is shown in Figure 3.2.5. It shows a decreasing relationship between the time-and-proximity factor and source strength. In most cases, the incidents involving a source with activity greater than 10,000 Ci resulted in fatalities. For many of the strong source accidents, medical symptoms were the first sign of an accident. The column in Table 3.2.1 labeled "How Terminated" shows "Med" if medical symptoms appeared, and "Rad" if the accident was discovered by other means, usually by radiation measurements or by discovery of missing sources through inventory or malfunction.

Only a few accidents appear to have run their full course, that is, delivered all the dose they ever would have. The 1992 Indiana, PA accident and the 1990 source shipment from Korea fall into this category. In each case, the accident was essentially over when it was discovered. These accidents are particularly valid for assessment of improper transfer and disposal scenarios.

3.2.5 Collective Time-and-Proximity Factors for Accidents

The collective dose can be determined using a Collective Time-and-Proximity Factor for an accident. Values of Time-and-Proximity Factors are given in Table 3.2.6 for the accidents at Goiânia, Brazil; Indiana, PA; and the Korea-USA incident based on collective dose due to external irradiation (UNSCEAR 1993; NRC 1992; NRC 1990). Notice that the Indiana, PA accident was an order of magnitude more serious than the Brazilian accident, and the Korea-USA shipment incident was a factor of 3 more serious, in terms of Time-and-Proximity Factors. That is, had the Brazilian source been involved in the Indiana or Korea accidents, the collective doses would have been 10 times and 3 times higher, respectively, than they were at Goiânia. Using a similar rationale, if any of these three accidents had occurred with a small, generally licensed source, the collective dose would have been lowest for the Goiânia-like accident, and highest for the Indiana, PA-like accident.

3.2.6 Conclusions

Distributions of Time-and-Proximity Factors from historical accidents can be used in probabilistic risk analyses for both whole-body and local irradiation from external sources. An analysis of 42 accidents for which source identity and strength are available show that the

average accident victim gets a whole body dose equal to that from being at 1 meter from the accident's unshielded source for an hour. The average accident is characterized by a value of 46 hours at a meter. In other words, the population-weighted average is about 1 hour at a meter, while the accident-weighted average is 117 hours at a meter. Clearly, the accidents with large numbers of victims (e.g., Goiânia and Juarez) dominate the former average. The maximum value seen for whole-body doses is about 700 hours at a meter. The average, geometric mean, and maximum values for local irradiation are 3100, 60 and 24,000 hours at a meter, respectively.

Such distributions should be used in probabilistic risk analyses to determine likely distributions of risks or doses from improper transfer and disposal scenarios for generally licensed devices.

Accidents that were terminated due to the appearance of clinical symptoms of acute irradiation have less value for risk analyses than accidents that were terminated by other means, or never terminated.

The current AEOD incident database does not contain the kinds of information needed to perform this analysis. It is recommended that the database either be modified to include this information, or a separate database be created. There is a great deal of work to be done to refine these preliminary analyses, extend them to additional accidents, and develop the logical framework for extrapolating to other kinds of sources and scenarios.

Table 3.2.1. Radiation Accidents involving external exposures. See text.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	#Public	#Workers	#Cleanup Workers	How Terminated	WB-brief	WB-protected	Local-brief	Local-protected	Intakes	Removed from Shield?	Source Damaged?	Reference	Gamma (Sv/h [m ² /Bq])
ALG78	78	Ir-192	IndRad	25	9.25E+11	7	0	0	Med	1	1				1		Jammet et al. 1980a, 1980b	1.60E-13
AUS70	70	Ir-192	IndRad	22	8.14E+11	0	2	0	Rad	1							Brown and McNeill 1971	1.60E-13
BAN85	85	Ir-192	IndRad	50	1.85E+12	0	1	0	Med	1							Jalil and Molla 1989	1.60E-13
BRA87	87	Cs-137	Tele	1400	5.18E+13	88	9		Med	1	1	1		1	1	1	Ramalho et al. 1988 Tables 1&2 (97 persons for whom cytogenetic dosimetry was done)	9.25E-14
BRA87	87	Cs-137	Tele	1400	5.18E+13	240	9		Med	1							IAEA 1988: lognormal fit to 50 individual doses from Fig. 9 and 199 zeroes (249 total, p. 117)	9.25E-14
BRA87	87	Cs-137	Tele	1400	5.18E+13	2803	9		Med	1	1	1		1	1	1	Lushbaugh et al. IAEA-CN-51-92 p401	9.25E-14
BRA87	87	Cs-137	Tele	1400	5.18E+13			583	Med		1						IAEA 1988 p. 116 External doses for 583 Cleanup Workers	9.25E-14
CA79	79	Ir-192	IndRad	28	1.04E+12	0	11	0	Rad	1							Ross 1980	1.60E-13
CZE66	66	I-131	Medical	2.25	8.33E+10	0	0	16	Rad		1						Carach et al. 1967	7.63E-14
CZE73	73	Co-60	Tele	2973	1.10E+14	0	2	0	Rad	1							Klener et al. 1986	3.70E-13
FRG68	68	Ir-192	IndRad	7.8	2.89E+11	0	1	0	Med	1		1					Chone et al. 1970	1.60E-13
FRG72	72	Ir-192	IndRad	29.7297	1.10E+12	0	1	0	Rad	1							UNSCEAR 1988 p. 416	1.60E-13
FRG81	81	Co-60	Tele	2594.59	9.60E+13	0	2	0	Rad	1	1						Stephan et al. 1983	3.70E-13
IND68	68	Ir-192	IndRad	1.4	5.18E+10	0	1	0	Rad	1		1					Annamalal et al. 1978	1.60E-13
ISR90	90	Co-60	Steril	340541	1.26E+16	0	1	0	Med	1		1					IAEA 1993	3.70E-13
ITA75	75	Co-60	Steril	36000	1.33E+15	0	1	0	Rad	1	1						Parmentier et al. 1980	3.70E-13
JOH59	59	Co-60	IndRad	1.75	6.48E+10	1	0	0	Rad	1	1						Elliott 1960	3.70E-13
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	5	1	0	Rad		1	1					Hirashima et al. 1980	1.60E-13
KOR90	90	Ir-192	IndRad	4	1.48E+11	0	19	5	Rad	1	1						NUREG-1405	1.60E-13
KY76	76	Ir-192	IndRad	78	2.89E+12	0	1	0	Rad	1		1					Jacobson et al. 1977	1.60E-13
LA78	78	Ir-192	IndRad	100	3.70E+12	0	1	0	Med								Scott 1980	1.60E-13

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	#Public	#Workers	#Cleanup Workers	How Terminated	WB-brief	WB-protected	Local-brief	Local-protected	Intakes	Removed from Shield?	Source Damaged?	Reference	Gamma (Sv/h [m ² /Bq])
MEX62	62	Co-60	IndRad	5	1.85E+11	6	0	0	Med	1	1	1	1	1	1	1	Andrews 1963	3.70E-13
MEX83	83	Co-60	Tele	430	1.59E+13	4000	100 ?		Rad						1	1	Lister 1984; IAEA 1989	3.70E-13
MOR84	84	Ir-192	IndRad	16.2162	6.00E+11	28	0	0	Med	1	1	1	1	1	1	1	Marshall 1984	1.60E-13
NJ74	74	Co-60	Steril	120000	4.44E+15	0	1	0	Rad	1							Barlotta 1980	3.70E-13
NJ77	77	Co-60	Steril	500000	1.85E+16	0	1	0	Rad	1							Barlotta 1980	3.70E-13
NOR82	82	Co-60	Steril	65720	2.43E+15	0	1	0	Med	1							Flatby et al. 1983	3.70E-13
NY83	83	Co-60	IndRad	25	9.25E+11	0	10	0	Rad	1	1	1	1	1	1	1	NRC I&E Notice 83-16	3.70E-13
PA92	92	Ir-192	Brachy	3.7	1.37E+11	94	0	0	Rad	1	1	1	1	1	1	1	NUREG-1480	1.60E-13
PRC63	63	Co-60	IndRad	10	3.70E+11	7	0	0	Med						1	1	Gen-yao et al. 1980	3.70E-13
PRC80	80	Co-60	Steril	53000	1.96E+15	0	1	0	Med	1					1	1	Gen Yao Ye et al. 1990	3.70E-13
PRC85	85	Cs-137	IndRad?	10	3.70E+11	3	0	0	Med	1	1	1	1	1	1	1	Gen Yao Ye et al. 1990	9.25E-14
PRC86	86	Co-60	(1)	6888	2.55E+14	2	0	0	?	1							Gen Yao Ye et al. 1990	3.70E-13
PRC87	87	Co-60	Steril	89000	3.29E+15	0	1	0	?	1							Gen Yao Ye et al. 1990	3.70E-13
PRC92	92	Co-60	Expt	12	4.44E+11	18	0	0	Med	1	1	1	1	1	1	1	Nenot 1993	3.70E-13
SAF77	77	Ir-192	IndRad	6.75676	2.50E+11	3	0	0	?Med	1	1	1	1	1	1	1	UNSCEAR 1988 p. 416	1.60E-13
SAL89	89	Co-60	Steril	18000	6.66E+14	0	3	0	?Med	1	1	1	1	1	1	1	Littlefield et al. 1991	3.70E-13
SCO69	69	Ir-192	IndRad	25	9.25E+11	0	1	0	Rad	1	1	1	1	1	1	1	Harrison et al. 1973	1.60E-13
TN71	71	Co-60	Expt	7700	2.85E+14	0	1	0	Rad	1	1	1	1	1	1	1	Wade 1972, Vodopick and Andrews 1980	3.70E-13
TRK76	76	Co-60	Tele	2260	8.36E+13	0	5	0	Rad	1	1	1	1	1	1	1	Yalcintas et al. 1980	3.70E-13
TX72	72	Cs-137	IndRad	4	1.48E+11	1	0	0	Med	1	1	1	1	1	1	1	Collins 1980	9.25E-14
UK77	77	Ir-192	IndRad	21.6216	8.00E+11	0	1	0	Rad	1	1	1	1	1	1	1	UNSCEAR 1988 p. 416	1.60E-13
UK81	81	Cs-137	Brachy	0.12	4.44E+09	0	0	1	Rad	1	1	1	1	1	1	1	Heaton and Murray 1982	9.25E-14
WA76	76	Am-241	Dfnse	343.337	1.27E+13	0	1	10	Rad	1	1	1	1	1	1	1	McMurray 1983	7.57E-15
WI61	61	Co-60	Expt	200	7.40E+12	0	1	0	Rad	1	1	1	1	1	1	1	Rossi et al. 1962	3.70E-13
																	(1) "Radiation Station"	

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	#Public	#Workers	#Cleanup Workers	How Terminated	WB-brief	WB-protected	Local-brief	Local-protected	Intakes:	Removed from Shield?	Source Damaged?	Reference	Gamma (Sv/h [m ² /Bq])
Minimum				0.12	4.44E+09	0	0	0										7.57E-15
Maximum				500000	1.85E+16	4000	100	583										3.7E-13
Average				2.8E+04	1.029E+15	166	4.545	15										2.38E-13
Std Dev				9.1E+04	3.358E+15	726.9	15.24	90.97										1.24E-13
GeoMean				1.9E+02	6.85E+12													0
Median				50	1.85E+12	0	1	0										1.6E-13
Mode				1400	5.18E+13	0	1	0										3.7E-13
Number				45	45	44	44	41		31	15	16	5	3	28	8		45

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	TIME AND PROXIMITY FACTORS, F(t,p)						Whole Body DOSES (Sv)				Skin DOSE				
	Gamma*Act (Sv/h @ 1m)	Average (h @ 1m)	Std Dev (h @ 1m)	Minimum (h @ 1m)	Maximum (h @ 1m)	Geometric Mean (h @ 1m)	Geo. Std. Deviation	Average	Standard Deviation	Minimum	Maximum	Geometric Mean	Geo. Std. Deviation	Max Local F(t,p) (h @ 1m)	Max Local Dose (Sv)
ALG78	0.148	88.108	88.176	6.7568	270.27	47.297	4	13.04	13.05	1	40	7	4	675.68	100
AUS70	0.13024	1.6892		1.2285	3.5319	2.083		0.22		0.16	0.46	0.271293			
BAN85	0.296	8.4459		6.7568	10.135	8.2753		2.5		2	3	2.44949		81.081	24
BRA87	4.7915	0.2024	1.9827	0.0209	1.4609	0.0605	4.7	0.97	9.5	0.1	7	0.29	4.7	4.1741	20
BRA87	4.7915	0.1119	3.2975	2E-06	1.4609	0.0038	13	0.536	15.8	1.02E-05	7	0.0182	13.49	4.1741	20
BRA87	4.7915	0.0034	0.6261	1E-09	1.4609			0.0163	3	7.00E-09	7			4.1741	20
BRA87	4.7915	0.0002	0.0003	5E-06	0.0033	0.0001	2.8	1.07E-03	1.48E-03	2.426E-05	0.016	6.2E-04	2.82	4.1741	20
CA79	0.16576	0.7264	1.5601	0.0121	5.2787	0.1647	5.8	0.1204	0.2586	0.002	0.875	0.0273	5.83	24131	4000
CZE66	0.00635	0.0236		0.0079	0.0394	0.0176		0.00015		0.00005	0.00025	0.000112			
CZE73	40.7	0.0184		0.0025	0.0344	0.0092		0.75		0.1	1.4	0.374166		2.457	100
FRG68	0.04618	32.484		21.656	43.313	30.627		1.5		1	2	1.414214		4331.3	200
FRG72	0.176	1.7045						0.3							
FRG81	35.52	0.0084		0.0056	0.0113	0.008		0.3		0.2	0.4	0.282843			
IND68	0.00829	156.85						1.3						15685	130
ISR90	4662	0.0032		0.0021	0.0043			15		10	20				
ITA75	492.84	0.0284						14						0.0487	24
JOH59	0.02396	1.0435						0.025						375.67	9
JPN71	0.03114	13.167	15.222	3.2114	42.711	8.4138	2.7	0.41	0.474	0.1	1.33	0.262	2.66		
KOR90	0.02368	1.2965	3.7162	0.0063	14.717	0.1098	7.9	0.0307	0.088	0.00015	0.3485	0.0026	7.87		
KY76	0.46176	2.0573		0.8663	3.2484	1.6775		0.95		0.4	1.5	0.774597		32.484	15
LA78	0.592	0.0845						0.05						168.92	100

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	Gamma*Act. (Sv/h @ 1m)	TIME AND PROXIMITY FACTORS, F(t,p)						Whole Body DOSES (Sv)					Skin DOSE		
		Average (h @ 1m)	Std Dev (h @ 1m)	Minimum (h @ 1m)	Maximum (h @ 1m)	Geometric Mean (h @ 1m)	Geo. Std. Deviation	Average	Standard Deviation	Minimum	Maximum	Geometric Mean	Geo. Std. Deviation	Max Local F(t,p) (h @ 1m)	Max Local Dose (Sv)
MEX62	0.06845	446.17	184.51	175.31	686.63	408.04	1.7	30.54	12.63	12	47	27.93	1.7		
MEX83	5.8867	0.0041	0.1276	8E-09	1.1891	0.0001	14	0.024	0.751	5.0E-08	7	7.7E-04	13.8		
MOR84	0.096	123.02	75.417	10.417	260.42	92.188	2.6	11.81	7.24	1	25	8.85	2.6		
NJ74	1642.8	0.0025						4.1							
NJ77	6845	0.0003						2.1							
NOR82	899.707	0.0445						40							
NY83	0.34225	0.0044		0.0015	0.0058	0.0029		0.0015		0.0005	0.002	0.001			
PA92	0.0219	1.3313	1.8718	0.0003	8.6286	0.1329	25	0.02916	0.041	5.6E-06	0.189	0.00291	25.3		
PRC63	0.1369	170.2	227.17	14.609	584.37	75.237	4.1	23.3	31.1	2	80	10.3	4.1		
PRC80	725.57	0.0072				0.0072		5.22				5.22			
PRC85	0.03423	262.97	29.218	233.75	292.18	261.88		9	1	.8	10	8.962809	1.12		
PRC86	94.2967	0.0323		0.0276	0.0371	0.032		3.05		2.6	3.5	3.016621			
PRC87	1218.41	0.0011		0.0011	0.0012	0.0012		1.35		1.35	1.46	1.403923			
PRC92	0.16428	22.583	33.479	1.887	121.74	11.748	2.9	3.71	5.5	0.31	20	1.93	2.89		
SAF77	0.04	11.925	14.825	2.5	29	6.75	3.6	0.477	0.593	0.1	1.16	0.27	3.63	2500	100
SAL89	246.42	0.0215	0.0108	0.013	0.0337	0.0198	1.6	5.3	2.67	3.2	8.3	4.89	1.62		
SCO69	0.148	4.0541						0.6						1351.4	200
TN71	105.413	0.0247						2.6							
TRK76	30.9394	0.0001	0.0001	2E-06	0.0003	3E-05	10	0.00346	0.00411	0.000047	0.00978	0.000945	10.1		
TX72	0.01369	730.46						10							
UK77	0.128	0.7813						0.1							
UK81	0.00041	0.487						0.0002							
WA76	0.09617	0.052						0.005							
WI61	2.738	0.9131						2.5							

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	TIME AND PROXIMITY FACTORS, F(t,p)										Whole Body DOSES (Sv)				Skin DOSE	
	Gamma*Act (Sv/h @ 1m)	Average (h @ 1m)	Std Dev (h @ 1m)	Minimum (h @ 1m)	Maximum (h @ 1m)	Geometric Mean (h @ 1m)	Geo. Std. Deviation	Average	Standard Deviation	Minimum	Maximum	Geometric Mean	Geo. Std. Deviation	Max Local F(t,p) (h @ 1m)	Max Local Dose (Sv)	
Minimum	0.00041	0.0001	0.0001	1E-09	0.0003	3E-05	1.6	0.00015	0.00148	7E-09	0.00025	0.000112	1.12	0.0487	9	
Maximu	6845	730.46	227.17	233.75	686.63	408.04	25	40	31.1	12	80	27.93	25.3	24131	4000	
Average	3.8E+02	46	38	17	82	34	6.7	4.62	5.76	1.57	10.21	3.07	6.37	3084.5	317.6	
Std Dev	1.2E+03	133	67	53	176	90	6.3	8.47	8.19	3.08	17.94	5.79	6.26	6848.8	984.1	
GeoMea	1.3E+00	0.32						0.43		0.018	1.42	0.19	4.45	91.2	59.7	
Median	3.0E-01	0.49	3.51	0.01	3.25	0.1213	4.1	0.95	1.835	0.16	2	0.332083	4	125	62	
Mode	4.8E+00			6.76	1.46			2.5		0.1	7			4.2	100	
Number	45	45	18	29	29	28	16	45	18	29	29	28	17	16	16	

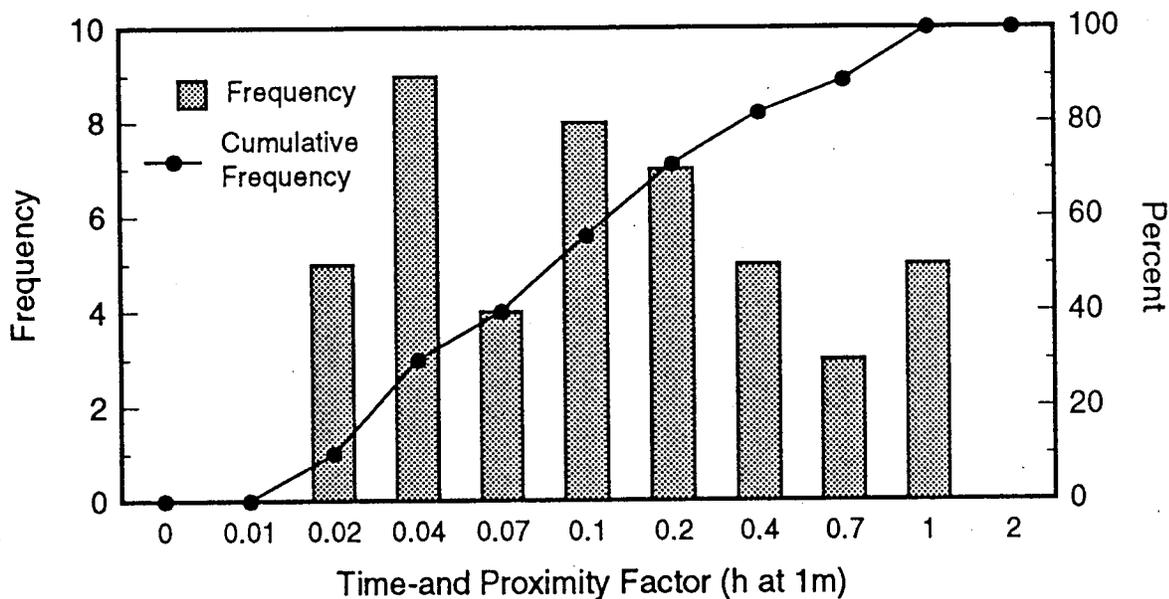


Figure 3.2.1. Frequency distribution of time-and-proximity factors from the 46 highest dose cases from the 1987 Goiânia accident on a roughly logarithmic horizontal scale. (IAEA, 1988a; Figure 9)

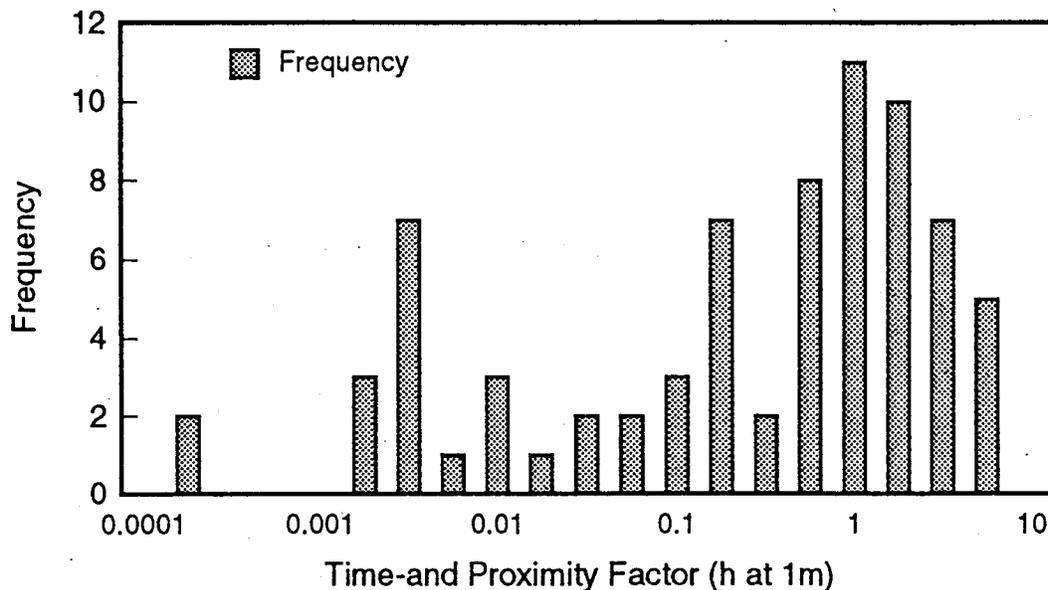


Figure 3.2.2. Frequency Distribution of Time-and-Proximity Factors for the 1992 Indiana, PA ^{192}Ir therapy misadministration accident on a logarithmic horizontal scale. (NRC, NUREG-1480, 1993b)

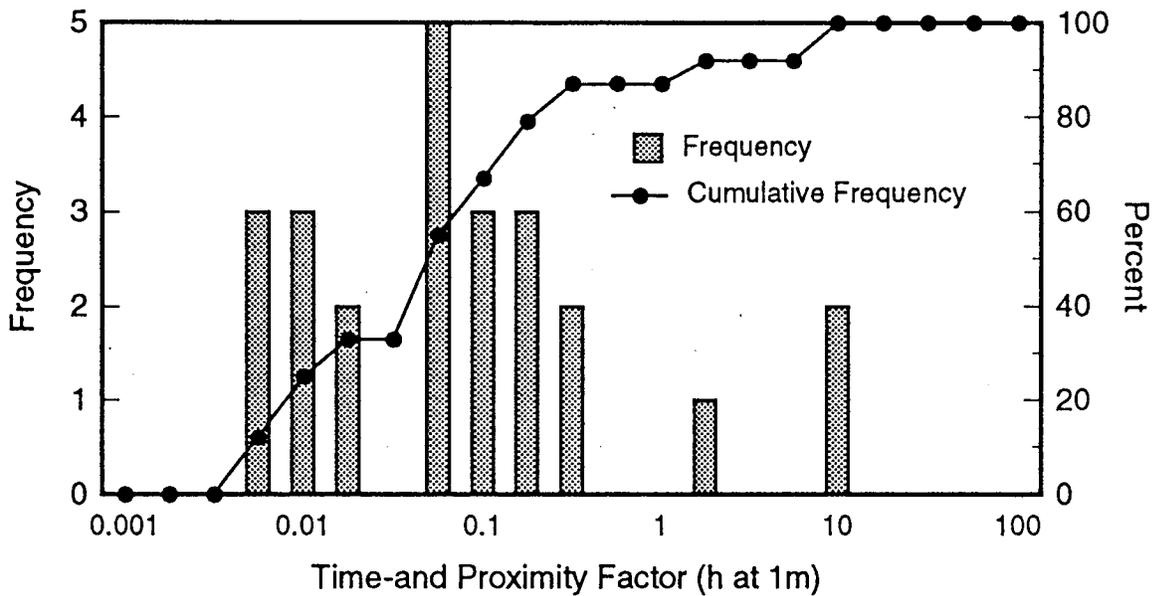


Figure 3.2.3. Frequency and Cumulative Frequency of Time-and-Proximity Factors for the 1990 Korea-USA ¹⁹²Ir source shipment accident on a logarithmic horizontal scale. (NRC, NUREG-1405, 1990)

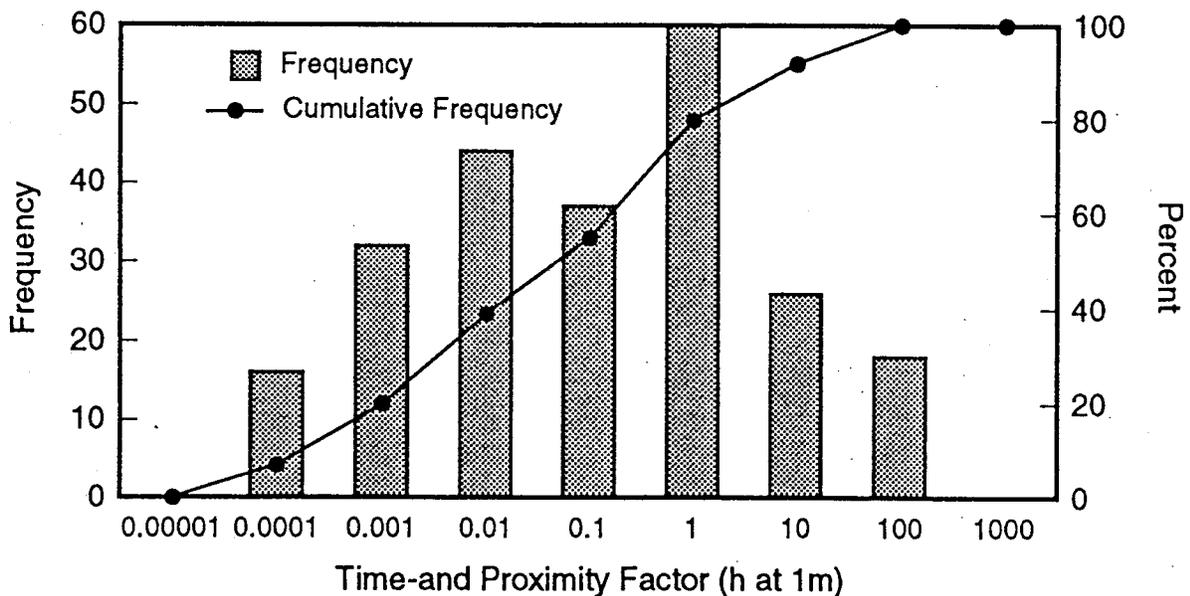


Figure 3.2.4. Analysis of 40 radiation accidents involving sealed sources, involving 231 individuals for whom doses were available. The Goiânia and Juarez accidents are excluded because the large numbers of victims of those accidents would dominate the graphs.

Table 3.2.2. Summary of Whole Body Time-and-Proximity Factors from a survey of 40 accidents involving sealed sources. The accidents at Goiânia, Brazil, and Juarez, Mexico are not included. The very large numbers of persons involved would dominate the summary statistics.

Quantity	Value	Units
Number of accidents	40	-
Number of individual doses	231	-
Average	30	hours at a meter
Standard Deviation	100	hours at a meter
Geometric Mean	0.37	hours at a meter
Geometric Standard Deviation	43	-
Median	0.46	hours at a meter
Mode	0.00068	hours at a meter
Minimum	0.000112	hours at a meter
Maximum	730	hours at a meter

Table 3.2.3. Collective Time-and-Proximity Factors for several sealed source accidents.

Population Group	Collective External Dose (Person-Gy)	Source Activity (Bq)	Source Strength (Sv/h at 1 m)	Collective Time-and-Proximity Factor (hours at 1 m)
All persons in Goiânia	56.3*	5.09E13	4.71	11.96
4 who died in Goiânia	14.9*	5.09E13	4.71	3.17
85 persons Indiana, PA	2.5	1.37E11	0.022	113
24 persons, Korea - USA	0.74	9.2E10	0.024	31.2

*Collective doses from ¶172, UNSCEAR 1993. These doses are probably too low by a factor of 2, from comparison with Figure 9 of IAEA 1988b (74 person-Gy listed there among 50 most highly exposed).

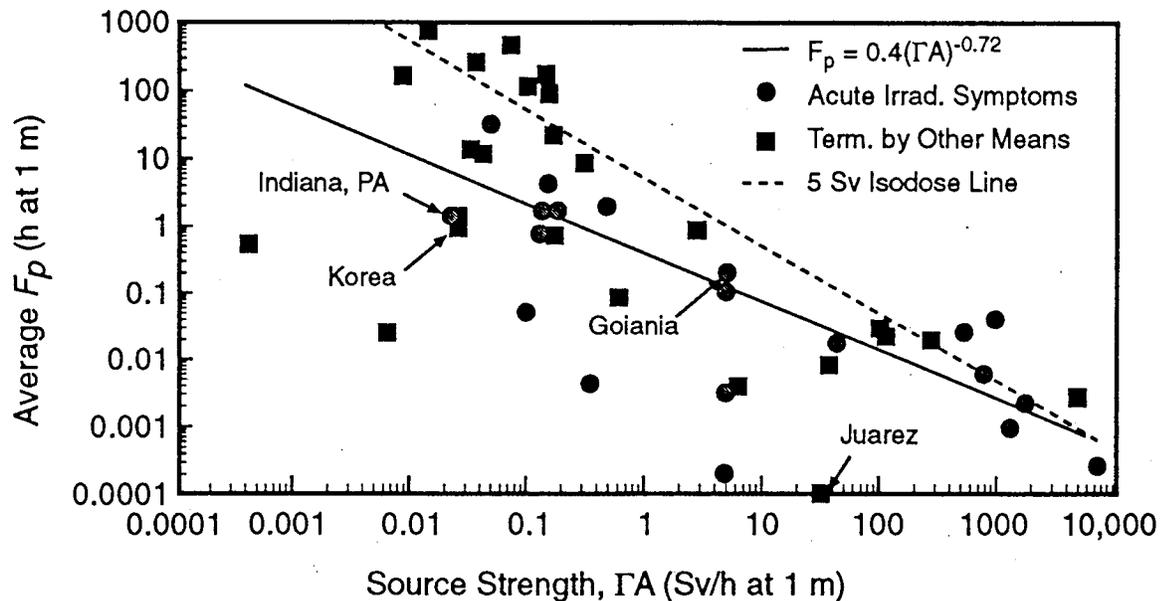


Figure 3.2.5. Average whole-body time-and-proximity factors (F_p) for 41 sealed source accidents as a function of source strength on a logarithmic horizontal scale. Most points shown above the 5-Sv (500 rem) isodose line were fatal accidents.

3.3 INTERNAL EXPOSURES

There are substantial differences between cases involving only external irradiation and cases where internal contamination has resulted. First, it is relatively rare for internal contamination to occur in non-reactor-type accidents. Second, early clinically recognizable symptoms of the internal radiation exposure are rarely seen in internal contamination cases. In most instances, any deterministic effects, and certainly the stochastic effects, will display themselves only at a time in the distant future, if at all. As a result, accidents may go unreported. Third, medical treatment procedures can be used to induce decorporation of the internal radionuclide burden, resulting in a reduction in radiation dose.

3.3.1 Theoretical Basis for Fraction-Taken-In

Radioactive material may gain entry to the body by inhalation, ingestion, or absorption through skin or wounds. The amount of radioactive material, or fraction of the total radioactive material available, that enters the body is a major factor in the radiation dose resulting from the intake. The approach taken in this report is to perform a quantitative, probabilistic characterization of historical radiation accidents in which the fraction-taken-in is known or can be calculated and then to generalize the results to generally licensed radioactive

devices.

It should be noted that since, in many cases, it is generally not known that an accident has occurred, the majority of early biological data needed to calculate radiation doses from the incident will not have been collected. In some cases, the lack of early biological data has hampered the use of historical records for determining the fraction-taken-in values.

For internal contamination cases, Brodsky (1980) has proposed the fraction-taken-in approach to characterizing internal contamination cases. Depending on how the data are reported in the literature, the fraction-taken-in may be available directly, or may be calculated from reported doses. In the former case,

$$F_{t,i} = \frac{A_i}{A}, \quad (11)$$

where $F_{t,i}$ denotes the fraction-taken-in for the i th person;
 A_i denotes the activity taken in by the i th person; and
 A denotes the source activity.

If activities taken in are not published, the fraction-taken-in may be calculated from 50-year committed effective dose equivalent to the i th person using

$$H_{50,i} = \frac{A F_{t,i}}{ALI} \cdot 5 \text{ rems}, \quad (12)$$

where ALI is the 1980 stochastic annual limit on intake (EPA 1988); and
5 rems is the 50 year committed effective dose equivalent resulting from an intake of
1 ALI .

Solving for $F_{t,i}$, we have

$$F_{t,i} = \frac{H_{50,i}}{5 \text{ rems}} \frac{ALI}{A}. \quad (13)$$

where $H_{50,i}$ denotes the 50 year committed effective dose equivalent received by individual i ;
 ALI denotes the annual limit on intake; and
5 rems is the committed effective dose equivalent associated with an intake of a 1980
 ALI .

If 1991 ALIs (based on 2 rems effective dose) are used, $F_{t,i}$ is

$$F_{t,i} = \frac{E_i}{2 \text{ rems}} \frac{ALI}{A}, \quad (14)$$

where E_i denotes the effective dose to the i th person from the incident.

3.3.2 Results of Analysis of Accidents for Fraction-Taken-In

The activity of the radioactive source at the time of the accident is an essential piece of information. The information needed for this analysis was generally unavailable in the NRC/AEOD incident database. A literature review of available data on accidents involving radioactive materials was conducted. The results of this review are presented in Table 3.3.1. The historical cases reported in this table are only those for which the source activity data were obtainable from the literature sources available, and are the same as those included in Table 3.2.1. For each incident, the locate, date, nuclide, source activity, incident type, and reference were reported, as well as whether or not external or internal exposure(s) were involved, the number of deaths resulting from such exposure(s) and the calculated fraction-taken-in (using Eq. 11).

Of the 60 radiation accidents reviewed and reported in Table 3.3.1, only 17 incidents, or (28%), involved cases of internal contamination. The low percentage of internal contamination cases is most probably due to the fact that many of the reported radiation accidents involved sealed, radiographic sources that maintained their structural integrity during the accident. As a result, no radioactive material was available for intake into the body, and the fraction-taken-in is zero. Of the 4364 individuals involved in the accidents listed in Table 3.3.1, 90 individuals¹ (2.1%) received internal contamination as a result of the indicated accident. The calculated fraction-taken-in for these individuals ranges from two in ten-thousand to twenty in a billion (2E-4 to 2E-8). In only one case of an industrial accident involving an unsealed source of tritium (Lloyd 1986) was the fraction-taken-in greater than this range (in that case it was 2E-2).

In most cases, the distribution of the fraction-taken-in, and thus the resultant dose, is lognormally distributed. This is best illustrated in Figure 3.3.1 using data from the 1987 Goiânia, Brazil accident which involved a 1,375 Ci ¹³⁷Cs source (Brandao-Mello 1987). The log probability analysis of the 20 highest-exposed individuals indicated in Table 3.3.1 indicates that the distribution of fraction-taken-in values is lognormal with a mean of 2.6E-6 and a geometric standard deviation (GSD) of 5.4 (Figure 3.3.1). When a total of 77 individuals who had internal contamination from this accident were considered (IAEA 1988b), the distribution of fraction-taken-in values was lognormal with a mean of 6.2E-6 and a GSD of 21 (Figure 3.3.2). The committed effective dose equivalents determined for these 77 individuals are also lognormally distributed with a mean of 0.3 Gy and a GSD of 11.2 (Figure 3.3.3). When 194 of the cleanup workers at Goiânia were evaluated, their fraction-taken-in values were estimated to range from 7E-11 to 2E-15.

In the most recent publication of the UNSCEAR report (UNSCEAR 1993), the collective dose for all persons involved in the Goiânia incident and for the 4 individuals who died as a

¹ For the 77 cases of internal contamination resulting in the Goiânia accident (Brandao-Mello 1991b), only 20 highest internal exposures are listed.

result of the incident were given. As shown in Table 3.3.2, this data can then be used as a direct application of Eq. 13. By using the reported collective internal doses of 3.7 and 2.3, respectively, the resulting fraction-taken-in values are $9\text{E-}6$ and $5\text{E-}6$, respectively. The estimated collective doses for this incident are within the range of values calculated for individuals.

Figure 3.3.4 provides an illustration of the frequency of values for fraction-taken-in for the 60 radiation incidents reviewed. In addition, both the 0.3 maximum value and the realistic range ($10\text{E-}6$ to $10\text{E-}5$) used in the ORAU report are include on the plot. It is interesting to note that the so called 'realistic' range of values used in the ORAU report (obtained from published literature on transportation accidents) falls in the range of values observed from actual accidents. However, the maximum value of 0.3 used by ORAU falls above the observed range. Thus, the ORAU approach to calculation the doses from their selected scenarios might be applicable today, if two things are done: 1) the more realistic values for fraction-taken-in should be used, and 2) CEDE and TEDE should be calculated instead of just organ doses.

3.3.3 Conclusions and Generalizations of Results to Improper Transfer and Disposal Scenarios for Generally Licensed Devices

When considering all reported non-reactor-type radiation accidents listed in Table 3.3.1, the fraction-taken-in was found to range from $2\text{E-}4$ to $2\text{E-}8$. The internal contamination cases upon which this range of values is based resulted largely from either industrial accidents involving unsealed sources of radioactive material or intentional destruction of licensed radiation sources.

When generalizing the results of this historical review, it is difficult to support the "arbitrarily chosen" value of 0.3 for the fraction-taken-in for generally licensed devices used in the ORAU Report. Thus, the most defensible values to use for the fraction-taken-in are distributions in the range of $2\text{E-}4$ to $2\text{E-}8$, except for cases involving sealed tritium sources in generally licensed devices.

TABLE 3.3.1. Summary of Radiation Accidents Used to Determine Fraction-Taken-In Values

CODE	LOCATION	DATE	NUCLIDE	SOURCE ACTIVITY	ISOTOPES	INCIDENT TYPE	EXTERNAL INVOLVED	INTERNAL INVOLVED	# INVOLVED	# DEATHS	FRACTION TAKEN IN	REFERENCE
UK44	UK	1944	Ra-226	0.005	Cl	Brachytherapy	X	—	2	0	0	Smith 1982
JOH59	JOH	1959	Co-60	2	Cl	Industrial Accident	X	—	1	0	0	Elliott 1960
WI61	WI	1961	Co-60	200	Cl	Experimental Accident	X	—	1	0	0	Ross 1962
MEX62	Mexico	1962	Co-60	5	Cl	Industrial Accident	X	—	5	4	0	Andrews 1963
PA63	Phil. PA	1963	S-35	1.27	Cl	Explosion-Vial	X	X	1	0	1.0E-05	Maass 1963
PRC63	PR of China	1963	Co-60	10	Cl	Lost sources	X	—	6	2	0	Mettler 1990
USA64	U.S.A	1964	Am-241	0.05	Cl	Lost source	X	X	2 ^a	0	1.8E-06	Cohen 1979
USA64	U.S.A	1964	Am-241	0.05	Cl	Lost source	X	X	2 ^b	0	7.2E-07	Cohen 1979
FRG64	Germany	1964	H-3	?	Cl	Tritiated Paint Accident	—	X	4	1	?	Mettler 1990
CZE66	Czechoslovakia	1966	I-131	2.25	Cl	Medical/Aircraft accident	X	X	16	0	0	Carach 1968
NY67	Univ of Rochester	1967	Ir-192	2	Cl	Hot cell leak	X	X	2 ^a	0	1.4E-04	Cool 1982
NY67	Univ of Rochester	1967	Ir-192	2	Cl	Hot cell leak	X	X	2 ^b	0	2.1E-04	Cool 1982
USA67	USA	1967	Ir-192	70	Cl	Industrial Accident	X	—	2	0	0	Bhusan 1973; Maxfield 1969
FRG68	Germany	1968	Ir-192	8	Cl	Industrial Accident	X	—	1	0	0	Chone 1970
IND68	India	1968	Ir-192	1	Cl	Industrial Accident	X	—	1	0	0	Annamalai 1978
ARG68	Argentina	1968	Cs-137	14	Cl	Industrial Accident	X	—	1	0	0	Beninson 1969
FRG68	Germany	1968	Ir-192	7	Cl	Industrial Accident	X	—	1	0	0	Schneider 1969
IND69	India	1969	Ir-192	1	Cl	Industrial Accident	X	—	1	0	0	Annamalai 1978
SCO69	SCO	1969	Ir-192	25	Cl	Industrial Accident	X	—	1	0	0	Harrison 1973
UK69	UK	1969	Ir-192	24	Cl	Industrial Accident	X	—	1	0	0	Harrison 1973
AUS70	AUS	1970	Ir-192	22	Cl	Industrial Accident	X	—	2	0	0	Brown and McNeill 1971
UK70	UK	1970	Ir-192	22	Cl	Industrial Accident	X	—	1	0	0	Purrott 1972
JPN71	Chiba, Japan	1971	Ir-192	5	Cl	Industrial Accident	X	—	6	0	0	Kumatori 1977; Mettler 1980
TN71	Oak Ridge, TN	1971	Co-60	7,700	Cl	Experimental Accident	X	—	1	0	0	Volopick 1974; Preston 1974; Fry 1990
UK71	UK	1971	Ir-192	5	Cl	Industrial Accident	X	—	1	0	0	Purrott 1973
BUL72	Bulgaria	1972	Cs-137	53	Cl	Sukkle	X	—	1	1	0	Oliviera 1987
TX72	Texas	1972	Cs-137	4	Cl	Intentional accident	X	—	1	0	0	Collins 1980
FRG72	Germany	1972	Ir-192	30	Cl	Industrial Accident	X	—	1	0	0	Sagell 1975
CZE73	Czechoslovakia	1973	Co-60	2,973	Cl	Med. Therapy Accident	X	—	2	0	0	Klener 1986
ME74	Middle East	1974	Ir-192	7	Cl	Industrial Accident	X	—	1	0	0	Purrott 1976
NV74	EPA-Las Vegas	1974	Hg-203	0.03	Cl	Accid. med. volatilization	?	X	2 ^a	0	3.0E-06	Brown 1975
NV74	EPA-Las Vegas	1974	Hg-203	0.03	Cl	Accid. med. volatilization	?	X	2 ^b	0	2.5E-05	Brown 1975

TABLE 3.3.1. (cont.)

CODE	LOCATION	DATE	NUCLIDE	SOURCE ACTIVITY	UNITS	INCIDENT TYPE	EXTERNAL INVOLVED	INTERNAL INVOLVED	# INVOLVED	# DEATHS	FRACTION TAKEN IN	REFERENCE
NJ74	Parsippany, NJ	1974	Co-60	120,000	Ci	Medical Steril. Accident	X	—	1	0	0	Stedley 1976; Fry 1990
IRQ75	Iraq	1975	Ir-192	62	Ci	Industrial Accident	X	—	1	0	0	Lloyd 1979
ITA75	Italy	1975	Co-60	36,000	Ci	Industrial Accident	X	—	2	1	0	Jammet 1980; Parmentier 1980; Oliveria 1987
USA76	USA	1976	Ir-192	95	Ci	Industrial Accident	X	—	1	0	0	NUREG-0322 1977 / NUREG-0090-6 1977
USA76	USA	1976	Co-60	165	Ci	Industrial Accident	X	—	1	0	0	NUREG-0322 1977 / NUREG-0090-6 1977
WA76	Hanford	1976	Am-241	340	Ci	Explosion-Glovebox	X	X	1	0	3.2E-06	Held 1979; McMurray 1983; Fry 1990
KY76	Kentucky	1976	Ir-192	78	Ci	Industrial Accident	X	—	1	0	0	Jacobson 1977
TRK76	Turkey	1976	Co-60	2,260	Ci	Med. Therapy Accident	X	—	5	0	0	Yalcintas 1980
UK77	UK	1977	Ir-192	22	Ci	Industrial Accident	X	—	1	0	0	Lloyd 1978
SA77	South Africa	1977	Ir-192	7	Ci	Industrial Accident	X	—	1	0	0	Lloyd 1978; Basson 1980
NJ77	Parsippany, NJ	1977	Co-60	500,000	Ci	Medical Steril. Accident	X	—	1	0	0	Stedley 1979; Fry 1990
ALG78	Algeria	1978	Ir-192	25	Ci	Lost Industrial source	X	?	22	5	0	Mettler 1990
LA78	Louisiana	1978	Ir-192	100	Ci	Industrial accident	X	—	1	0	0	Scott 1980
CA79	California	1979	Ir-192	28	Ci	Industrial accident	X	—	11	0	0	Ross 1990
LA79	Louisiana	1979	Ir-192	?	Ci	Industrial accident	X	—	3	0	0	Scott 1980
PRC80	PR of China	1980	Co-60	53,000	Ci	Industrial accident	X	—	1	0	0	Ye 1990
MA80	BU Med Center	1980	Tc-99m	0.04	Ci	Boiled vial to dryness	—	—	1	0	0	Evdokimoff 1980
UK81	UK	1981	Cs-137	0.12	Ci	Brachytherapy	X	—	1	0	0	Heaton and Murray 1982
NOR82	Norway	1982	Co-60	65,000	Ci	Industrial Use	X	—	1	1	0	Relian 1990
MEX83	Juarez, Mexico	1983	Co-60	450	Ci	Lost Industrial source	X	—	4000	5	0	Lister 1986; IAEA 1989b
MOR84	Morocco	1984	Ir-192	16	Ci	Lost Industrial source	X	X	28	8	?	Mettler 1990; Lushbaugh 1990
SW585	Switzerland	1985	H-3	50	Ci	Industrial accident	—	X	5 a	0	2.0E-02	Lloyd 1986
SW585	Switzerland	1985	H-3	50	Ci	Industrial accident	—	X	5 b	0	5.4E-04	Lloyd 1986
SW585	Switzerland	1985	H-3	50	Ci	Industrial accident	—	X	5 c	0	7.0E-05	Lloyd 1986
SW585	Switzerland	1985	H-3	50	Ci	Industrial accident	—	X	5 d	0	7.0E-05	Lloyd 1986
SW585	Switzerland	1985	H-3	50	Ci	Industrial accident	—	X	5 e	0	7.0E-05	Lloyd 1986
BAN85	BAN	1985	Ir-192	50	Ci	Industrial accident	X	—	1	0	0	Jalli and Molla 1989
PRC85	PR of China	1985	Cs-137	10	Ci	Lost Industrial Source	X	—	3	1	0	Ye 1990
PRC86	PR of China	1986	Co-60	6,888	Ci	Industrial accident	X	—	2	0	0	Ye 1990

TABLE 3.3.1. (cont.)

CODE	LOCATION	DATE	NUCLIDE	SOURCE ACTIVITY	UNITS	INCIDENT TYPE	EXTERNAL INVOLVED	INTERNAL INVOLVED	# INVOLVED	# DEATHS	FRACTION TAKEN IN	REFERENCE
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 a	1	2.0E-05	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 b	0	7.9E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 c	0	3.9E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 d	0	3.9E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 e	1	2.0E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 f	0	2.0E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 g	0	2.0E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 h	0	2.0E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 i	0	2.0E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 j	1	1.1E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 k	0	1.1E-06	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 l	0	9.8E-07	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 m	0	8.8E-07	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 n	0	5.9E-07	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 o	1	3.9E-07	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 p	0	2.0E-07	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 q	0	1.6E-07	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 r	0	1.1E-07	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 s	0	8.8E-08	Brandao-Mello 1991b
BRA87	Goiânia, Brazil	1987	Cs-137	1,375	Ci	Lost Medical Source	X	X	20 t	0	1.6E-08	Brandao-Mello 1991b
PRC89	PR of China	1989	Co-60	89,000	Ci	Industrial accident	X	--	1	0	0	Ye 1990
SAL89	El Salvador	1989	Co-60	18,000	Ci	Stuck sterilizer source	X	--	3	1	0	Hurtado 1991
ISR90	Israel	1990	Co-60	340,541	Ci	Stuck sterilizer source	X	--	1	0	0	Ricks 1991; IAEA 1993
KOR90	Korea	1990	Ir-192	4	Ci	Industrial accident	X	--	24	0	0	NUREG-1405 1990
PA92	Pennsylvania	1992	Ir-192	4	Ci	Brachytherapy	X	--	94	1	0	NUREG-1480 1993

TABLE 3.3.2 Calculation of Fraction-Taken-In Using Collective Internal Doses From the Goiânia Brazil Accident

Population Group	Collective Internal Dose (Person-Gy)	Source Activity (Bq)	ALI for Cs-137 (Bq)	Collective Fraction-Taken-In
All persons in Goiânia	3.7	5.09E13	6E6	8.7E-6
4 who died	2.3	5.09E13	6E6	5.4E-6

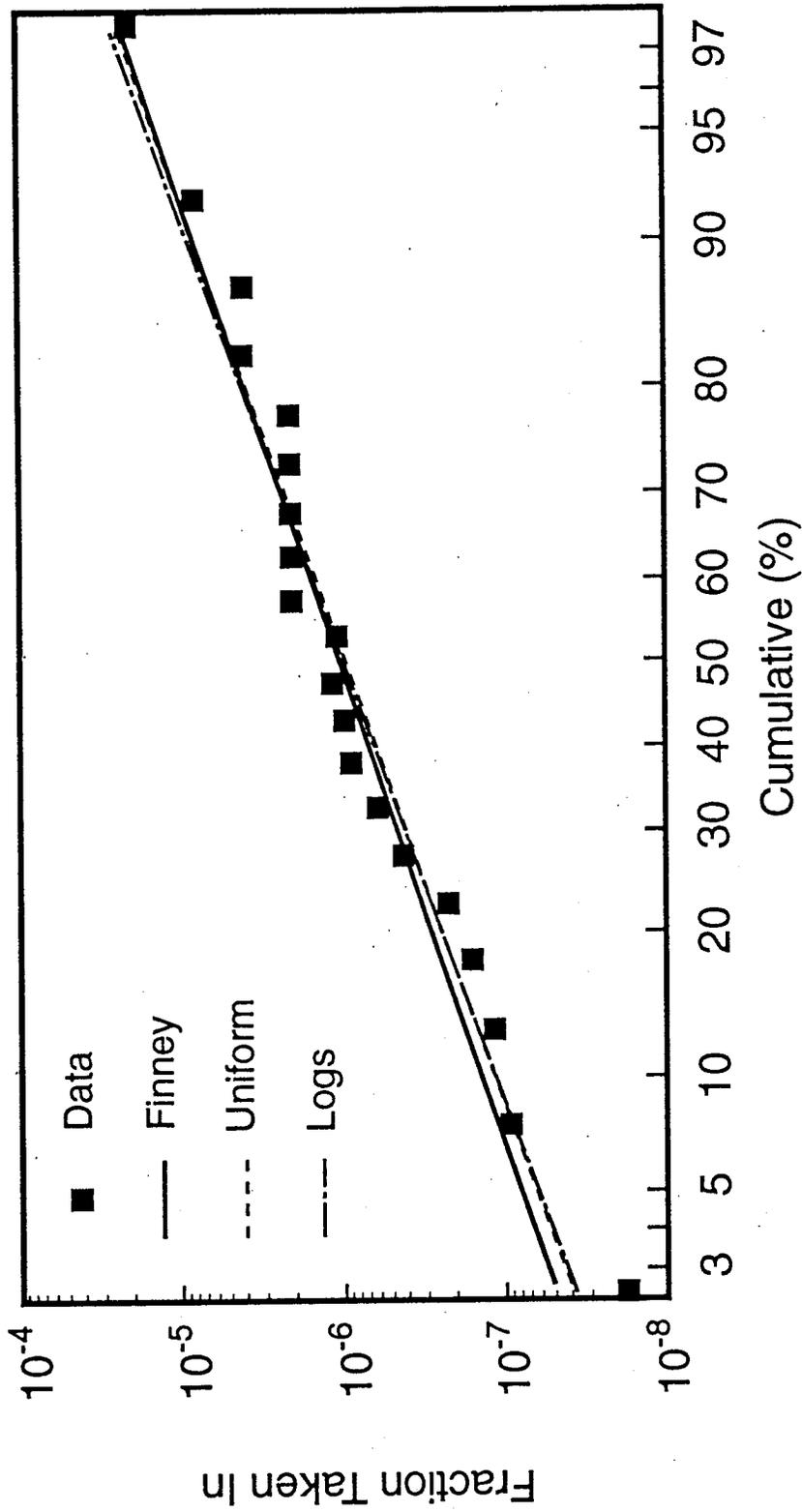


FIGURE 3.3.1. Log Probability Analysis of Fraction-Taken-In for 20 Highest-Exposed Individuals with Internal Contamination from the 1987 Goiânia ^{137}Cs Accident. Regression lines are uniform- or Finney-weighted.

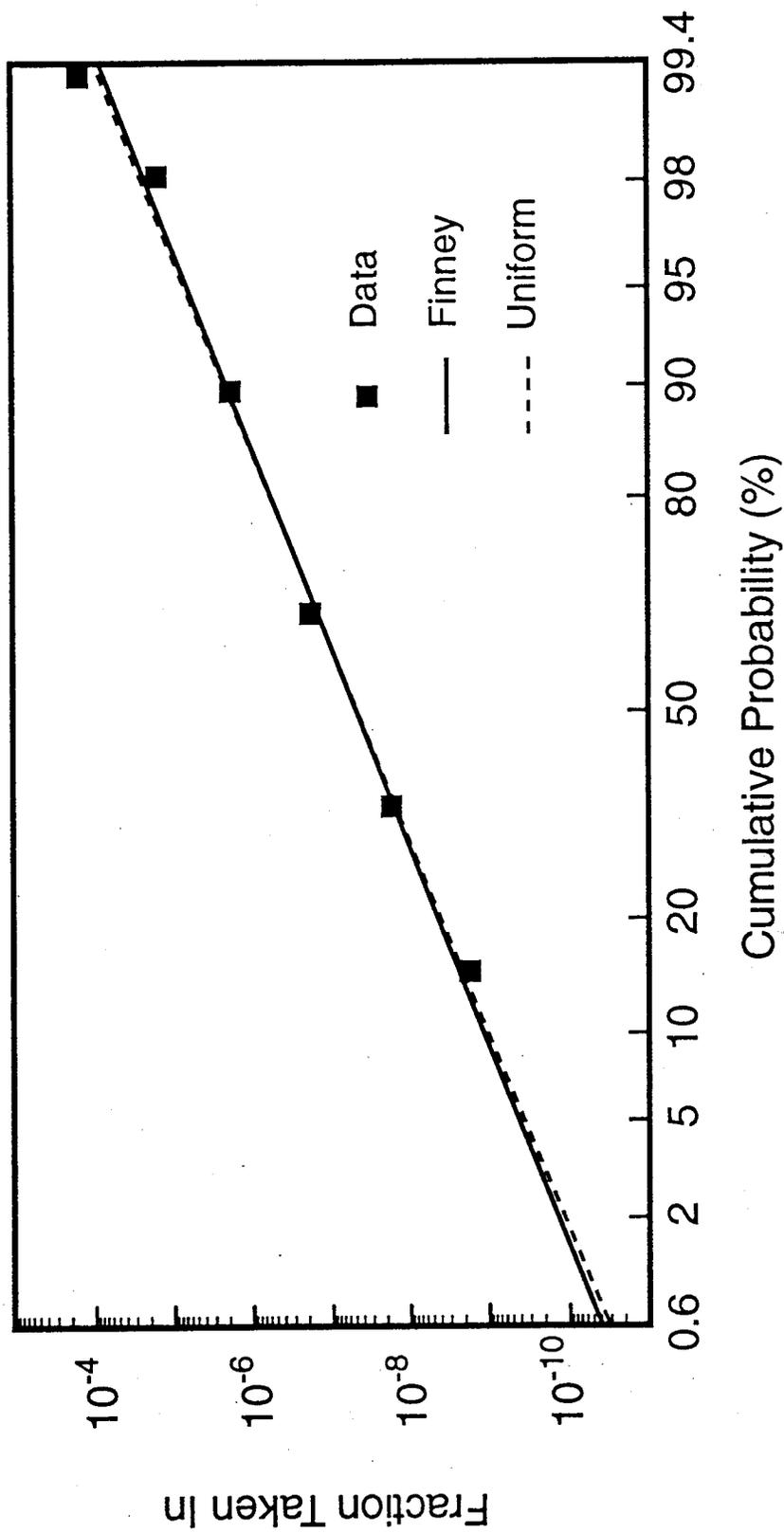


FIGURE 3.3.2. Log Probability Analysis of Fraction-Taken-In for 77 Individuals with Internal Contamination from the 1987 Goiânia ¹³⁷Cs Accident. Regression lines are uniform- or Finney-weighted. (IAEA, 1988a, from Figure 13)

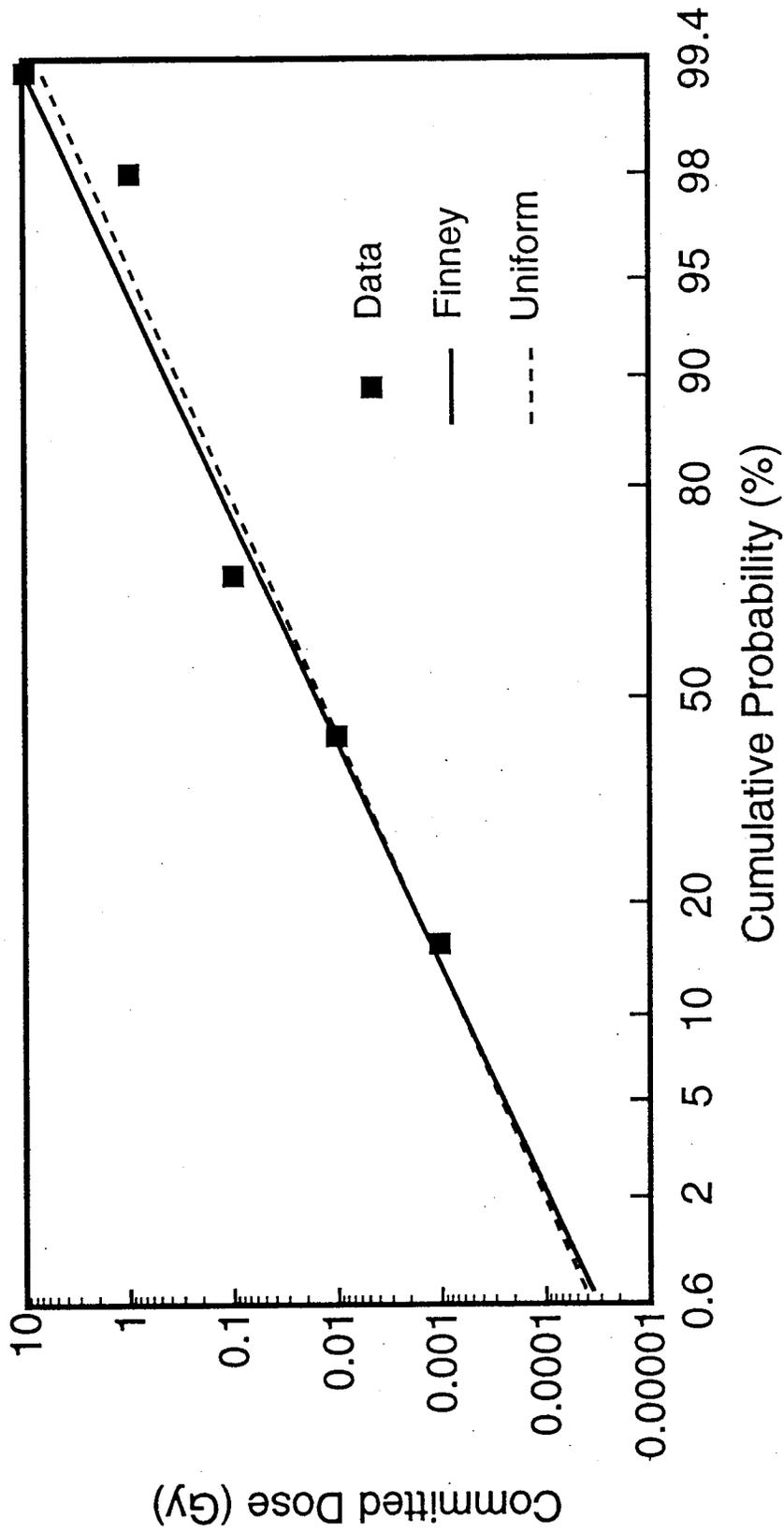


FIGURE 3.3.3. Log Probability Analysis of Committed Dose for 77 Individuals with Internal Contamination from the 1987 Goiânia ¹³⁷Cs Accident. Regression lines are uniform- or Finney-weighted. (IAEA, 1988a, from Figure 14)

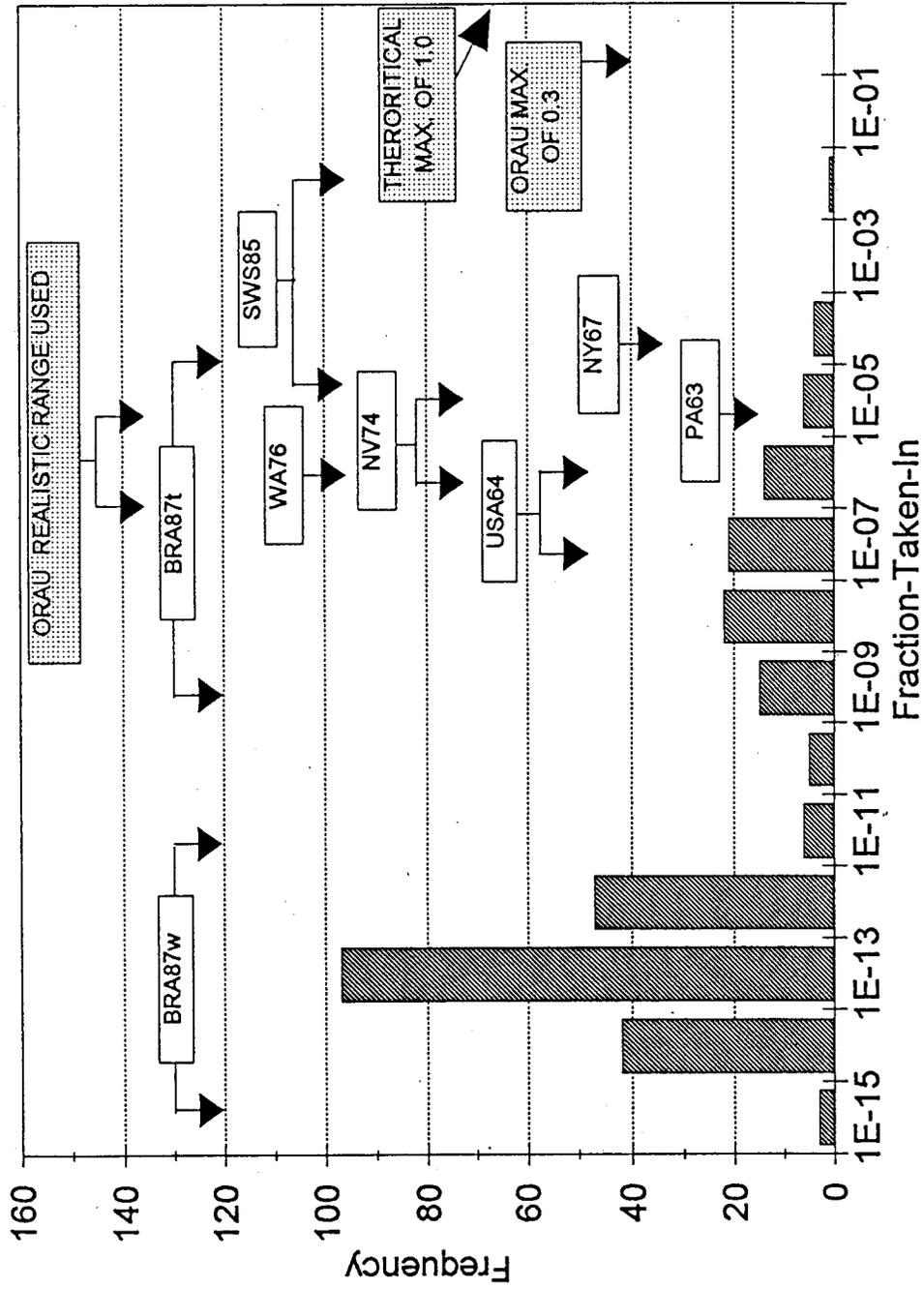


FIGURE 3.3.4. Frequency of Fraction-Taken-In For 60 Radiation Accidents

4.0 CONCLUSIONS

Work should be done in concert with NMSS staff to develop a risk framework that meets the NRC's regulatory decision making needs. Such a framework is proposed in this report, but should be fine-tuned to meet the staff's requirements.

Such a framework should include the probabilities of accidents occurring. The probabilities should be expressed on a per-source, per-year basis, and include summation over accident types and multiplication by the number of sources in use. Potential harm from accidents should be assessed using distributions of coefficients, not point estimates, derived from historical accidents, including fractions of source activity taken in, and probable external doses (both local and whole-body) based on analyses such as those presented above.

Time-and-proximity factors for 231 individuals involved in 40 accidents reviewed by PNL (excluding the large population accidents in Goiânia Brazil and Juarez Mexico) averaged 30 hours at a meter, with a standard deviation of 100 hours at a meter, a geometric mean of 0.37 hours at a meter with a geometric standard deviation of 43. Collective time-and-proximity factors were 12, 113, and 31 for the Goiânia, Indiana PA, and Korea-USA accidents, respectively.

Fractions-taken-in were found to be in the range of 2×10^{-8} to 2×10^{-4} for many accidents reviewed by PNL, many involving unsealed sources. Cleanup workers in Goiânia had fractions-taken-in in the range of 2×10^{-15} to 7×10^{-11} , while townspeople, including those directly involved in the incident, had fractions-taken-in averaging 6×10^{-6} . Roughly 98% of individuals in the 60 accidents surveyed had fractions-taken-in of zero.

5.0 REFERENCES

10 CFR 20. 1991. U. S. Nuclear Regulatory Commission (USNRC), "Standards for Protection Against Radiation." U.S. Code of Federal Regulations. 56 FR 23360-23474. Amended and corrected 56 FR 61352 (3 December 1991) and 57 FR 57877 (8 December 1992).

10 CFR 30. 1993. U.S. Nuclear Regulatory Commission (USNRC), "Rules of General Applicability to Domestic Licensing of Byproduct Material." U.S. Code of Federal Regulations, February 26, 1993.

10 CFR 31. 1992. U.S. Nuclear Regulatory Commission (USNRC), "General Domestic Licenses for Byproduct Material." U.S. Code of Federal Regulations, November 30, 1992.

10 CFR 32. 1992. U.S. Nuclear Regulatory Commission (USNRC), "Specific Domestic Licenses to Manufacture or Transfer Certain Items Containing Byproduct Material." U.S. Code of Federal Regulations, November 30, 1992.

10 CFR 835. 1993. U.S. Department of Energy (DOE), "Occupational Radiation Protection." U.S. Code of Federal Regulations, December 14, 1993.

40 CFR 190. U.S. Environmental Protection Agency (EPA). U.S. Code of Federal Regulations.

Accidental Nuclear Excursion. Recuplex Operation 234-5 Facility. Final Medical Report. From the Health Operation, General Electric, Hanford Atomic Products Operation, Richland, Washington, April 7, 1962.

Allen, W. R. 1966. "Radiation Injury to the Hand." J. Kansas Med. Soc. 67:447-453.

Alves, R. N. 1990. "Contamination control in the 137 Cs accident in Goiânia. In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979." Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York, New York.

Andrews, G. A., B. W. Sitterson, A. L. Kretchmar, and M. Brucer. 1959. "Accidental Radiation Excursion at the Oak Ridge Y-12 Plant IV. Preliminary Report on Clinical and Laboratory Effects in the Irradiated Employees." Health Phys. 2:134-138.

Andrews, G. A. 1963. "Mexican ⁶⁰Co Radiation Accident." Isotopes Radiat. Technol. 1:2:200-201.

Andrews, G. A., K. F. Hübner, and S. A. Fry. 1980. "Report of 21-year Medical

Follow-up of Survivors of the Oak Ridge Y-12 Accident," pp. 59-79. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hilbner and S. A. Fry. Elsevier, New York.

Annamalai, M, P. S. Iyer, and T. M. R. Panicker. 1978. "Radiation Injury from Acute Exposure to an Iridium-192 Source: Case History." Health Phys. 35:387-389.

Auxier, J. A. 1965. "Nuclear Accident at Wood River Junction." Nucl. Saf. 6:3:298-300.

Barlotta, F. M. 1980. "The New Jersey Radiation Accidents of 1974 and 1977" 1980. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hahner and S. A. Fry. Elsevier, New York: pp. 151-160.

Baron, J. M., S. Yachnin, R. Polcyn, et al. 1969. "Accidental Radiogold (^{198}Au) Liver Scan Overdose with Fatal Outcome," pp. 399-414. In: Proceedings of a Symposium on the Handling of Radiation Accidents, International Atomic Energy Agency/World Health Organization, Vienna, Austria, May 19-23, 1969.

Basson J. K., A. P. Hanekom, F. C. Coetzee. 1980. "Health physics evaluation of an acute over-exposure to a radiography source." In: Radiation Protection: A Systematic Approach to Safety, Proceedings of the Fifth Congress of the International Radiation Protection Association, Jerusalem, 1980. Pergamon Press: 64-68.

Beck, H. L. 1982. "Special Composition of the Gamma-ray Exposure Rate Due to Noble Gases Released During a Reactor Accident." Health Phys. 43(3):335-343.

Becker, J. and T. Rosen. 1987. "Acute Radiodermatitis from Occupational Exposure to ^{192}Ir ." South. Med. J. 82:12:1561-1563.

Beninson, D., A. Placer, and Vander Elst E. 1969. "Estudio de un caso de irradiación humana accidental," pp. 415-429. In: Proceedings of a Symposium on the Handling of Radiation Accidents, International Atomic Energy Agency/World Health Organization, Vienna, Austria, 19-23 May 1969.

Berteig, L. and J. Flatby. 1984. "The Radiation Accident at Institute for Energy Technology, September 1982. Some Technical Considerations." J. Indust. Irradiat. Technol. 2:3 and 4:309-319, 1984.

Bertelli, L., J. L. Lipsztein, C. A. N. Oliveira, and D. R. Mello. 1990. "Internal ^{137}Cs Contamination in the Goiânia, Brazil Accident," In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident

Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York: pp. 342-252.

Bhushan, V. 1974. "Large Radiation Exposure." In: Proceedings of the Third International Congress of the International Radiation Protection Association. CONF-730907: Washington: 769-772.

Brahams, D. 1988. "Radiotherapy Overdose." Lancet.

Brandao-Mello, C. E. 1991a. "Personal Insights into the Goiânia Radiation Accident." Health Phys. 60:3-4.

Brandao-Mello, C. E., A. R. Oliveira, N. J. Valverde, R. Farina, and J. R. Cordeiro. 1991b. "Clinical and Hematological Aspects of Cs-137: The Goiânia Radiation Accident." Health Phys. 60:31-39.

"Brazilian Radiological Accident Traced to Abandoned Radiotherapy Device." 1987. Nuclear Waste News 303-304, October 22, 1987.

Breitenstein, B. D. Jr., and H. E. Palmer. 1989. "Lifetime Follow-up of the 1976 Americium Accident Victim." Radiat. Prot. Dosim. 26:317-322.

Breitenstein, B. D. Jr, S. A. Fry, and C. C. Lushbaugh. 1990. "DTPA Therapy: The U.S. Experience 1958-1987," pp. 397-403. In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.

Brodsky, A. 1980. Resuspension Factors and Probabilities of Intake of Material in Process (or "Is 10^{-6} a Magic Number in Health Physics?"). Health Phys. 39(6):992-1000.

Brown, J. K. and J. R. McNeill. 1971. "Biological Dosimetry in an Industrial Radiography Accident." Health Phys. 21:519-522.

Brown, K. W., J. C. McFarlane, and D. E. Bernhardt. 1975. "Accidental Inhalation of Mercury-203." Health Phys. 28:1-4.

Burson, Z. and C. C. Lushbaugh. 1990. "The 1983-1984 Ciudad Juarez, Mexico, ^{60}Co Accident," 13-23. In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.

- Carach, J., S. Csipka, M. Petrasove, T. Tronovec, and F. Minarik. 1968. "Recovery Operations Following An Aircraft Accident Involving Radioactive Cargo." Health Phys. 15:280-282.
- Catlin, R. J. and V. P. Bond. 1990. "Assessing the Risk to the General Population in Large Scale Radiation Accidents: A Review," pp. 291-315. In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.
- Catsaros, N., and A. Vassiliou. 1987. "An Assessment of the Average Shielding Factor for the Population of the Attica Basin Using the Shield-f Code." Radiat. Prot. Dosim. 21:97-102.
- Chone, B., G. Schneider, and D. Fehrentz. 1970. "¹⁹²Iridium-Kontaktbestrahlung und Folgeerscheinungen, English Summary." Strahlentherapie 140 1:113122.
- Cohen, N., T. L. Sasso, and M. E. Wrenn. 1979. "Metabolism of Americium 241 in Man; An Unusual Case of Internal Contamination of a Child and His Father." Science 206.
- Collins, V. P., and M.E. Gaulden. 1980. "A Case of Child Abuse by Radiation Exposure." In: The Medical Basis for Radiation Accident Preparedness. pp. 197-203. Eds. K.F. Hubner, S.A. Fry, Elsevier Science Publishing Co., New York.
- Collins, D. L. 1992. "Behavioral Differences of Irradiated Persons Associated with the Kyshtym, Chelyabinsk, and Chernobyl Nuclear Accidents." Military Med. 157:548-552.
- Collins, V. P. and M. E. Gaulden. 1980. "A Case of Child Abuse by Radiation Exposure," pp. 197-203. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Habner and S. A. Fry. Elsevier Science Publishing Co., New York.
- Cool, D. A. 1982. "Committed Dose Equivalents from an Accidental Inhalation of Insoluble Ir-192." Health Phys. 42.
- Cronkite, E. P. 1990. "Clinical Aspects of Accidents Resulting in Acute Total Body Irradiation (Continued)," pp. 75-78. In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.

- da Silva, C. J., J. U. Delgado, M. T. B. Luiz, P. G. Cunha, and P. D. de Barros. 1991. "Considerations Related to the Decontamination of Houses in Goiânia: Limitations and Implications." Health Phys. 60:87-90.
- de Oliveira, A. R. 1987. Un répertoire des accidents radiologiques 1945-1985. Radioprotection 22(2):89-135.
- Desrosiers, M. F. 1991. "In vivo Assessment of Radiation Exposure." Health Phys. 61:859-861.
- Dodd, B. and L. L. Humphries. 1988. "Hazards Assessment of Worst Case Transportation Accidents Having Typical Radioactive Material Shipments." Health Phys. 55:963-983.
- Dodd, B. and L. L. Humphries. 1989. "Errata -- Hazards Assessment of Worst Case Transportation Accidents Having Typical Radioactive Material Shipments." Health Phys. 56:974.
- ED. 1984. "NRPS - R166 'Doses in Radiation Accidents Investigated by Chromosome Aberration Analysis. XIV A Review of Cases Investigated: 1983' by D. C. Lloyd, J. S. Presser, J. E. Moquet and P. Finnion." J. Radiat. Prot. 4:189-190.
- ED. 1985. "Juarez Incident, 1984." J. Radiat. Prot. 5:145-147.
- Elliott, G. A. 1960. "Accidental Acute Irradiation from Cobalt-60." S. Afr. Med. J. 34:524-529.
- Evdokimoff, V. N. 1980. "Hot Lab Radiopharmaceutical Accident: Potential Airborne Release." Health Phys. 39:573-574.
- Farina, R., C. E. Brandao-Mello, and A. R. Oliveira. 1991. "Medical Aspects of Cs-137 Decorporation: The Goiânia Radiological Accident." Health Phys. 60:63-66.
- Finkel, A. M. 1990. Confronting Uncertainty in Risk Management. A Guide for Decision-Makers. Center for Risk Management, Resources for the Future, Washington, DC.
- Fisher, D. R., K. L. Kathren, M. J. Swint. 1991. "Modified Biokinetic Model for Uranium from Analysis of Acute Exposure to UF₆." Health Phys. 60:335-342.
- Flatby, J., T. Hendriksen, and H. Host. 1983. The Radiation Accident at the Institute for Energy Technology, Kjeller, Norway, September 2, 1982. Dosimetric Evaluations. National Institute of Radiation Hygiene, Statens Instituit for Stralehygiene Report.
- Fry, F. A. 1976. "Long-term Retention of Americium-241 Following Accidental Inhalation." Health Phys. 31:13-20.

Fry, S. A., L. G. Littlefield, C. C. Lushbaugh, et al. 1990. "Follow-up of Survivors of Serious Radiation Accidents in the United States," pp. 373-396. In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.

Fuqua, P. A., W. D. Norwood, and S. Marks. 1965. "Biologic Effects of Human Radiation Exposure; Report of a Criticality Accident." J. Occup. Med. 7:3:85-93.

Gilberti, M. V. 1980. "The 1967 Radiation Accident Near Pittsburgh, Pennsylvania, and a Follow-up Report," pp. 131-140. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Habner and S. A. Fry. Elsevier Science Publishing Co., New York.

Hammermaier, A., E. Reich, and W. Bogt. 1988. "An Evaluation of Chemiluminescence from Pharmaceuticals and Other Solids for Neutron Accident Dosimetry." Radiat. Prot. Dosim. 25:97-100.

Harrison, N. T., P. C. Escott, G. W. Dolphin, et al. 1974. "The Investigation and Reconstruction of a Severe Radiation Injury to an Industrial Radiographer in Scotland," pp. 760-768. In: Proceedings of the Third International Congress of the International Radiation Protection Association, Ed. W. S. Snyder. U.S. Atomic Energy Commission, Washington, D.C.

Hashizume, T. Y. Kato, T. Nakajima, H. Yamaguchi, and K. Fujimoto. 1972. "Emergent Dose Estimation of Non-Occupational Persons Accidentally Exposed to IR-192 Gamma Rays." Health Phys. 23:855-7.

Hasterlik, R. J. and L. D. Marinelli. 1956. "Physical Dosimetry and Clinical Observations on Four Human Beings Involved in an Accidental Critical Assembly Excursion." In: Proceedings of an International Conference on the Peaceful Uses of Atomic Energy. United Nations, Geneva, Switzerland.

Harrison, N. T., P. C. Escott, G. W. Dolphin. 1974. "The Investigation and Reconstruction of a Severe Radiation Injury to an Industrial Radiographer in Scotland." pp. 760-768. In: International Congress of the IRPA, Washington, 1973. CONF-730907.

Heaton, B., and A. A. Murray. 1982. "Experience Gained from the Recovery of Lost Radioactive Sources." J. Radiat. Prot. 2:34-35.

Heid, K. R., B. D. Breitenstein, H. E. Palmer, B. J. McMurray, N. Wald. 1979. The 1976 Hanford Americium Accident. TID-28938. Pacific Northwest Laboratory, Richland,

Washington.

Heid, K. R., B. D. Breitenstein, H. E. Palmer, et al. 1980. "The 1976 Hanford Americium Accident," pp. 345-355. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Habner and S. A. Fry. Elsevier Science Publishing Company, New York.

Hirashima, K., H. Sugiyama, T. Ishihara, et al. 1980. "The 1971 Chiba, Japan Accident: Exposure to Iridium 192," pp. 179-195. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hahner and S. A. Fry. Elsevier Science Publishing Company, New York.

Holly, F. E. and W. L. Beck. 1980. "Dosimetry Studies for an Industrial Radiography Accident," In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hahner and S. A. Fry. Elsevier Science Publishing Company, New York. pp. 265-277.

Hurtado, R. M., R. Secin, M. Marquez, R. C. Ricks, M. E. Berger. 1991. "The Radiological Accident in El Salvador: Psychological Aspects." In: The Medical Basis for Radiation Accident Preparedness III. Proceedings of the Third REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. R.C. Ricks, M. E. Berger, and F. M O'Hara. Elsevier Science Publishing Company, New York. pp. 187-191.

Hurst, G. S., R. H. Ritchie, F. W. Sanders, P. W. Reinhardt, J. A. Auxier, E. B. Wagner, A. D. Callihan, and K. Z. Morgan KZ. 1961. "Dosimetric Investigation of the Yugoslav Radiation Accident." Health Phys. 5:179-202.

International Atomic Energy Agency (IAEA). 1962. "The Vinca Dosimetry Experiment." Technical Report Series No.6, International Atomic Energy Agency. IAEA, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1969. "Handling of Radiation Accidents." Proceedings Series, IAEA, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1974. "Evaluation of Radiation Emergencies and Accidents: Selected Criteria and Data." Technical Report Series No. 152, IAEA, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1977. "Handling of Radiation Accidents 1977." Proceedings Series, IAEA, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1986. "Derived Intervention Levels for Application in Controlling Radiation Doses to the Public in the Event of a Nuclear Accident or Radiological Emergency: Principles, Procedures and Data." Safety Series No. 81, IAEA, Vienna, Austria.

International Atomic energy Agency (IAEA). 1988a. "Medical Handling of Accidentally Exposed Individuals." Safety Series No. 88, IAEA, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1988b. "The Radiological Accident in Goiânia." IAEA, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1989a. "Evaluating the Reliability of Predictions Made Using Environmental Transport Models." IAEA Safety Series No. 100, IAEA, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1989b. "Emergency Planning and Preparedness for Accidents Involving Radioactive Materials Used in Medicine, Industry, Research and Training." Safety Series No. 91. pp. 51-79. IAEA, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1990. "The Radiological Accident in San Salvador." IAEA, Vienna, Austria.

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

International Atomic Energy Agency (IAEA). 1993. "The Radiological Accident in Soreq." IAEA, Vienna, Austria.

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

International Commission on Radiological Protection (ICRP). 1991. "1990 Recommendations of the International Commission on Radiological Protection." ICRP Publication 60, Pergamon Press, Oxford.

Jacob, P. and R. Meckbach. 1987. "Shielding Factors and External Dose Evaluation." Radiat. Prot. Dosim. 21:79-85.

Jacob, P., H. G. Paretzke and H. Rosenbaum. 1988. "Organ Doses from Radionuclides on the Ground. Part II. Non-trivial Time Dependence." Health Phys. 55:37-49.

Jacobson, A., B. W. Wilson, T. E. Banks, and R. M. Scott. 1977. "¹⁹²Ir Over Industrial

Radiography." Health Phys. 32:291-293.

Jalil, A. and K. A. Rab Molla. 1989. "An Overexposure in Industrial Radiography Using An Ir-192 Radionuclide." Health Phys. 57:117-119.

Jammet, H., G. Mathe, B. Pendic, et al. 1959. "Etudes de six cas d'irradiation totale accidentelle." Revue Francaise d'Etudes Cliniques et Biologiques 4:210-225.

Jammet, H. P., R. Gongora, R. Le Go, et al. 1966. "Observation Clinique et Traitement d'un cas d'irradiation globale accidentelle." Proceedings of the First International Congress of Radiation Protection (IRP), IRP Health Physics, Roma, Italy. 5-9 September 1966.

Jammet, H., R. Gongora, P. Pouillard, et al. 1980. "The 1978 Algerian Accident: Four Cases of Protracted Whole-body Irradiation," pp. 113-129. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hahner and S. A. Fry. Elsevier Science Publishing Co., New York.

Jammet, H., R. Gongora, P. Jockey, et al. 1980. "The 1978 Algerian Accident: Acute Local Exposure of Two Children," pp. 229-245. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Habner and S. A. Fry. Elsevier Science Publishing Co., New York.

Johnson, J. R. 1978. "Summary of Bioassay and Thyroid Monitoring Results Following an Accidental Exposure to I-125." Health Phys. 34:106-107.

Johnson, J. R., D. W. Dunford, and G. H. Kramer. 1983. "Summary of a Strontium-89 Contamination Case." Radiat. Prot. Dosim. 5:247-249.

Kathren, R. L. and R. H. Moore. 1986. "Acute, Accidental Inhalation of U: A 38-year Follow-up." Health Phys. 51(5):609-619.

Kereiakes, J. G. 1978. "Handling of Radiation Accidents 1977" Med. Phys. 5:457.

Klener, V., R. Tuscany, J. Vejlupekova, J. Dvorak, and P. Vlkovic. 1986. "Long-term Follow-up After Accidental Gamma Irradiation from a Co-60 Source." Health Phys. 51:601-607.

Kirchmam, R., G. S. Gerber, E. Fagniard, C. M. Vandecasteele, and M. van Hees. 1986. "Accidental Release of Elemental Tritium Gas and Tritium Oxides: Models and In Situ Experiments on Various Plant Species." Radiat. Prot. Dosim. 16:107-110.

Kumatori, T., K. Hirashima, T. Ishihara, et al. 1977. "Radiation Accident Caused by an

Iridium-192 Radiographic Source," pp. 35-42. In: Proceedings of a Symposium on Handling Radiation Accidents. International Atomic Energy Agency, Vienna, Austria.

Lanzl, L. H., M. L. Rozenfeld, A. R. Tarlov. 1967. "Injury Due to Accidental High-Dose Exposure to 10 MeV Electrons." Health Phys. 13:241-251.

Le Grand, J., J. C. Croize, T. de Dorlodot, and Y. Roux. 1987. "Statistical Survey of the Housing Characteristics and Evaluation of Shielding Factors in the Surrounding of French Nuclear Sites." Radiat. Prot. Dosim. 21:87-95.

Levanon, I. and A. Pernick. 1988. "The Inhalation Hazard of Radioactive Fallout." Health Phys. 54:6:645-657.

Lipsztein, J. L., P. G. Cunha, and C. A. N. Oliveira. 1991a. "The Goiânia Accident: Behind the Scenes." Health Phys. 60:5-6.

Lipsztein, J. L., L. Bertelli, D. R. Melo, A. M. G. F. Azeredo, L. Juliao, and M. S. Santos. 1991b. "Application of in-vitro Bioassay for Cs-137 During the Emergency Phase of the Goiânia Accident." Health Phys. 60:43-49.

Lipsztein, J. L., L. Bertelli, C. A. N. Oliveira, and B. M. Dantas. 1991c. "Studies of Cs Retention in the Human Body Related to Body Parameters and Prussian Blue Administration." Health Phys. 60:57-61.

Lister, B. A. J. 1986. "Contaminated Mexican Steel Incident." J. Radiat. Prot. 6:48.

Littlefield, L. G., E. E. Joiner, S. P. Colyer, R. C. Ricks, C. C. Lushbaugh, and R. Hurtado-Monroy. 1991. "The 1989 San Salvador Co-60 Radiation Accident: Cytogenetic Dosimetry and Follow-up Evaluations in Three Accident Victims." Radiat. Prot. Dosim. 35:115-123.

Lloyd, D. C., R. J. Purrott, J. S. Prosser. 1978. Doses in radiation accident investigated by chromosome aberration analysis. VIII: A review of cases investigated - 1977. NRPB-R70, United Kingdom.

Lloyd, D. C., R. J. Purrott, J. S. Prosser. 1979. Doses in radiation accident investigated by chromosome aberration analysis. IX: A review of cases investigated - 1978. NRPB-R83. United Kingdom.

Lloyd, D. C., A. A. Edwards, J. S. Prosser, A. Auf der Maur, A. Etzweiler, U. Weickhardt, U. Gossi, L. Geiger, U. Noelpp, and H. Rosler. 1986. "Accidental Intake of Tritiated Water: A Report of Two Cases." Radiat. Prot. Dosim. 15:191-196.

Lubenau, J. O., J. S. Davis, D. J. McDonald, and T. M. Gerusky. 1969. "Analytical

X-ray Hazards: A Continuing Problem." Health Phys. 16:739-746.

Lubenau, J. O. and T. M. Gerusky. 1971. "Radiation Incidents Registry - Pennsylvania Experience." Health Phys. 21:605-7.

Lubenau, J. O. and D. A. Nussbaumer. 1986. "Comment of 'A Possible Hazard: Pressure Build-up in Sealed Ampoules of Radionuclides in Aqueous Solution'." Health Phys. 51:147-148.

Lushbaugh, C. C., S. A. Fry, K. F. Habner, and R. C. Ricks. 1980. "Total-body Irradiation: A Historical Review and Follow-up," pp. 59-79. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hahner and S. A. Fry. Elsevier Science Publishing Co., New York.

Lushbaugh, C. C. 1981. "Management of Persons Accidentally Contaminated with Radionuclides, NCRP Report 65" by National Council on Radiation Protection and Measurements. Med. Phys. 8:525-526.

Lushbaugh, C. C., S. A. Fry, R. C. Ricks, et al. 1986. "Historical Update of Past and Recent Skin Damage Radiation Accidents." In: Radiation Damage to Skin: Fundamental and Practical Aspects. Eds. H. Jammet, F. Daburon and G. B. Gerber. Brit. Inst. Radiol. (London):7-12.

Lushbaugh, C. C., R. C. Ricks, S. A. Fry. 1988. "Radiological Accidents. A Historical Review of Sealed Sources Accidents." Proceedings of an International Conference on Radiation Protection in Nuclear Energy. IAEA-CN-51-92, International Atomic Energy Agency, Vienna, Austria.

Lushbaugh, C. C., S. A. Fry, A. Sipe, R. C. Ricks. 1990. "An Historical Perspective of Human Involvement in Radiation Accidents." In: Radiation Protection Today - The NCRP at Sixty Years. Proceedings of the Twenty-Fifth Annual Meeting April 5-6, 1989. pp. 171-188. NCRP Proceedings No. 10, NCRP, Bethesda, Maryland.

Lushbaugh, C., G. Eisele, W. Burr Jr., K. Hubner, and B. Wachholz. 1991. "Current Status of BioLogical indicators to Detect and Quantify Previous Exposures to Radiation." Health Phys. 60(Supp 1):103-109.

Maass, A. R. and T. L. Flanagan. 1963. "Accidental Personnel Exposure to Elemental S-35." Health Phys. 9:731-740.

Majborn, B. 1984. "Estimation of Accidental Gamma Dose by Means of Thermoluminescence from Watch Jewels." Health Phys. 46:917-919.

- Marshall, E. 1984a. "Juarez: An Unprecedented Radiation Accident." Science 23:1152-1154.
- Marshall, E. 1984b. "Morocco Reports Lethal Radiation Accident." Science 225:395.
- Martin, J. B. 1991. "'The Radiological Accident in San Salvador' by IAEA." Health Phys. 61:578.
- Matetskos, C. J. and C. C. Lushbough. 1991. "The Goiânia Radiation Accident." Health Phys. 60:1.
- Matthews, J. D. 1970. "Accidental Extremity Exposures from Analytical X-ray Beams." Health Phys. 18:75-6.
- Maxfield, WS, and GH Porter. "Accidental radiation exposure from iridium-192 camera." 1969. In: Handling of Radiation Accidents. IAEA, Vienna: 459-467.
- McMurray, B. J. 1983. "1976 Hanford Americium Exposure Incident: Accident Description." Health Phys. 45(4):847-853.
- Mettler, F. A., and R. C. Ricks. 1990. "Historical Aspects of Radiation Accidents." In: Medical Management of Radiation Accidents. Ed. F.A. Mettler, C.A. Kelsey, R. C. Ricks. CRC Press, Boca Raton, Florida. pp. 17-32.
- Mexican Cobalt-60 Incident-1984. Aerial Measuring System (AMS) Surveys, Nenot JC. Overview of the Radiological Accidents in the world, Updated December 1989. Internat. J. Rad. Biol. 57:1073-1085, 1990.
- Minder, W. 1969. "Interne Kontamination mit Tritium." Strahlentherapie 137:700-704.
- Morgan, J. 1989. 'Medical Handling of Accidentally Exposed Individuals' IAEA Safety Series No. 88. J. Radiat. Prot. 9:232.
- Morgan, M. G. and M. Henrion. 1990. Uncertainty. A Guide to Dealing with Uncertainty in Quantitative and Policy Risk Analysis. Cambridge University Press, New York.
- Mossman, K. L. 1989. "Health Physics Annotated Bibliography" by Charles A. Willis. Health Phys. 57:790.
- National Academy of Sciences (NAS), National Research Council. 1988. Health Risks of Radon and Other Internally Deposited Alpha-Emitters: BEIR IV. National Academy Press, Washington, DC.
- National Academy of Sciences (NAS), National Research Council. 1990. Health Effects of

Exposure to Low Levels of Ionizing Radiation: BEIR V. National Academy Press; Committee on the Biological Effects of Ionizing Radiation, Washington, DC.

Nenot, J. C. 1990. Review: Overview of the Radiological Accidents in the World, Updated December 1989. Int. J. Radiat. Biol. 57(6):1073-1085.

Nenot, J. C. 1993a. Bilan des accidents d'exposition interne. Radioprotection 28(3):265-277.

Nenot, J. C. 1993b. Accident d'irradiation en Chine. Radioprotection 28(4):453-455.

Newman, H. F. 1990. "The Malfunction 115411 Accelerator Accidents 1985, 1986, 1987," pp. 165-171 In: The Medical Basis for Radiation Accident Preparedness II Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.

Newton, D., J. Rundo, and J. D. Eakins. 1981. "Long-term Retention of The-228 Following Accidental Intake." Health Phys. 40:291-298.

Oberhofer, M. and J. L. Bacelar Leao. 1988. The Radiological Accident at Goiânia. International Atomic Energy Agency, Australia.

Oliveira, A. R. 1987. "Un Repertoire des Accidents Radiologiques 1945-1985." Radioprotection 22(2):89-135.

Oliveira, A. R., C. E. Brandeo-Heilo, N. J. L. Valverde, R. Farina, and M. P. Curado. 1991. "Localized Lesions Induced by Cs-137 During the Goiânia Accident." Health Phys. 60:25-29.

Oliveira, A. R., J. G. Hunt, N. J. L. Valverde, C. E. Brandao-Mello, R. Farina. 1991. "Medical and Related Aspects of the Goiânia Accident: An Overview." Health Phys. 60:17-24.

Oliveira, A. R., N. J. Valverde, C. E. Brandao-Mello, et al. 1990. "Skin Lesions Associated with the Goiânia Accident," pp. 173-181. In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.

Oliveira, C. A. N., R. Farina, L. Bertelli, A. T. Natarajan, A. T. Ramalho, and B. M. Dantas. 1991. "Measurements of Cs-137 in Blood from Individuals Exposed During the Goiânia Accident." Health Phys. 60:41-42.

Oliveira, C. A. N., N. C. Lourerico, B. M. Dantas, and E. A. 1991. "Design and Operation of a Whole-body Monitoring System for the Goiânia Radiation Accident." Health Phys. 60:51-55.

"ORAU Report:" see Stabin et al. 1987.

Parmentier, N. C., J. C. Nenot, and H. J. Jammet. 1980. "A Dosimetric Study of the Belgian and Italian Accidents," pp. 105-112. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hubner and S. A. Fry. Elsevier Science Publishing Co., New York.

Parmentier, N. C., J. C. Nenot, and C. Parmentier. 1990. "Two Cases of Accidental Protracted Overexposure: Aspects of an Extensive Bone Marrow Study.," pp. 29-51. In: The Medical Basis for Radiation Accident Preparedness II Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.

Petkau, A. and S. D. Pleskach. "Case of Accidental Aspiration of Sr-90 (CL 2)." Health Phys. 22:87-90.

Popplewell, D. S., and G. J. Ham. 1989. "Distribution of Plutonium and Americium in Tissues from a Human Autopsy Case." J. Radiat. Prot. 9:159-164.

Purrott R. J., G. W. Dolphin, D. C. Lloyd. 1972. The study of chromosome aberration yield in human lymphocytes as an indicator of radiation dose. II: A review of cases investigated - 1970-71. NRPB-R5, United Kingdom.

Purrott R. J., G. W. Dolphin, D. C. Lloyd. 1973. The study of chromosome aberration yield in human lymphocytes as an indicator of radiation dose. II: A review of cases investigated - 1971-72. NRPB-R10, United Kingdom.

Purrott R. J., D. C. Lloyd, J. S. Prosser. 1976. The study of chromosome aberration yield in human lymphocytes as an indicator of radiation dose. VI: A review of cases investigated - 1975. NRPB-R41, United Kingdom.

Preston R. H., J. G. Brewen, N. Gengozian. 1964. "Persistence of radiation induced chromosome aberrations in marmoset and man." Radiat. Res. 60:516-524.

"Radiation Accident Grips Goiânia. News and Comment." Science 238:1028-1031, November 1987.

24.

Sanders, S. M. Jr. 1974. "Excretion of Am-241 and Cm-244 Following Two Cases of Accidental Inhalation." Health Phys. 27:359-65.

Schneider G. J., B. Chones, T. Blonnigen. 1969. "Chromosomal Aberrations in a Radiation Accident: Dosimetric and Hematological Aspects." Radiat. Res. 40:613-617.

Scott, L. M. and C. M. West. 1975. "Excretion of PO-210 Oxide Following Accidental Inhalation." Health Phys. 28:563-565.

Scott, E. B., Jr. 1980. "The 1978 and 1979 Louisiana Accidents: Exposure to Iridium 192," pp. 223-262. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for radiation Accident Preparedness. Eds. K. F. Hilbner and S. A. Fry. Elsevier/North Holland, New York.

Smith, E. E. 1982. "The Recovery on 29th March 1944 of a Lost 5mg Radium Tube." Radiat. Prot. 2:39-40.

Smith, H. 1983. "Dose - Effect Relationships for Early Response to Total Body Irradiation." J. Radiat. Prot. 3:5-10.

Smith, N. 1985. "Assessment of and Therapy Following Contamination Through the Lungs or Gastrointestinal Tract." J. Radiat. Prot. 5:15-20.

Sowby, D. 1989. 'The Radiological Accident in Goiânia' by IAEA. J. Radiat. Prot. 9:78.

Sowby, D. 1990. 'The Radiological Accident in San Salvador' by International Atomic Energy Agency. J. Radiat. Prot. 10:313.

Stabin, M., K. Paulson, and S. Robinson. 1987. Improper Transfer/Disposal Scenarios for Generally Licensed Devices. "The ORAU Report" produced under NRC FIN B0299. Oak Ridge Associated Universities, Oak Ridge, Tennessee.

Steidley, K. D. 1976. "A Co-60 Hot Cell Accident." Health Phys. 31:382-385.

Steidley, K. D, G. S. Zeik, and R. Ouellette. 1979. "Another Co-60 Hot Cell Accident." Health Phys. 36:437-441.

Steinhausler, F. 1987. "The Effect of Fall-out Deposition on Indoor Gamma Radiation Levels in Single-family Dwellings." Radiat. Prot. Dosim. 21:103-105.

Stephan, G., W. Hadnagy, C. H. Hammaier, and U. Imhof. 1983. "Biologically and

Ramalho, A. T., A. C. M. Nescimento, and A. T. Netarajan. 1988. "Dose Assessments by Cytogenetic Analysis in the Goiânia (Brazil) Radiation Accident." Radiat. Prot. Dosim. 23:151-154.

Ramalho, A. T., and A. C. H. Mascimento. 1991. "The Fate of Chromosomal Aberrations in Cs-137 Exposed Individuals in the Goiânia Radiation Accident." Health Phys. 60:67-70.

Reitan, J. P., P. Stavem, K. Kett, et al. 1990. "The ⁶⁰Co Accident in Norway, 1982: A Clinical Reappraisal," pp. 3-11 In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.

Ricks, R. C. 1991. "The Scope of the Problem." In: The Medical Basis for Radiation Accident Preparedness III. Proceedings of the Third REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. pp. 3-10. Eds. R.C. Ricks, M. E. Berger, and F. M O'Hara. Elsevier Science Publishing Company, New York.

Ricks, R. C., S. A. Fry, A. H. Sipe, M. E. Berger, F. H. Fong, C. C. Lushbaugh. 1992. "History of Radiation Accidents." In: The Biological Basis of Radiation Protection Practice. pp. 218-225. Eds. K. L. Mossman and W. A. Mills. Williams and Wilkins, Baltimore, Maryland.

Rosenthal, J. J., C. E. de Almeida, and A. H. Mendonca. 1991. "The Radiological Accident in Goiânia: The Initial Remedial Actions." Health Phys. 60:7-15.

Ross, J. P., F. E. Holly, H. A. Zarem, et al. 1990. "The 1979 Los Angeles Accident: Exposure to Iridium 192 Industrial Radiographic Source," pp. 205-221. In: The Medical Basis for Radiation Accident Preparedness. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hahner and S. A. Fry. Elsevier Science Publishing co., New York.

Rossi, E. C., A. A. Thorngate, and F. C. Larson. 1962. "Acute Radiation Syndrome Caused by Accidental Exposure to Cobalt-60." J. Lab. Clin. Med., 59:655-666.

Rubin, L. S. 1978. "The Riverside Radiation Tragedy." Columbus Monthly 52-66.

Saenger, E. L., J. G. Kereiakes, N. Wald, et al. 1974. "Clinical Course and Dosimetry of Acute Hand Injuries to Industrial Radiographer from Multicurie Sealed Gamma Sources." Proceedings of the Third International Congress of International Radiation Protection Association United States Atomic Energy Commission, Office of Information Services (Tech Div.), 1:773-782.

Sagell H. "Ein lehrreicher Strahlenunfall." 1975. Arbeitsmed. Sozialmed. Praventivmed:

Physically Recorded Doses After an Accidental Exposure to Co-60 Gamma Rays. Health Phys. 44:409-411.

Stott, A. N. B. 1980. "Co-60 Hot Cell Accidents." Health Phys. 39:363-364.

Straume, T., R. G. Langlois, J. Lucas, R. H. Jensen, W. L. Bigbee, A. T. Ramlho, and C. E. Brandao-Mello. 1991. "Novel Biodosimetry Methods Applied to victims of the Goiânia Accident." Health Phys. 60:71-76.

Taylor, D. M. 1989. "Biological Assessment of Occupational Exposure to Actinides -- Round Table Discussion." Radiat. Prot. Dosim. 26:391-394.

"The Medical Basis for Radiation Accident Preparedness." 1980. Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Habner, and S. A. Fry. Elsevier/North Holland, New York.

"The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979." 1990. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co, New York.

Thompson, R. C., editor-in-chief. 1983. "1976 Hanford Americium Exposure Incident: Multiple Reports." Health Phys. 45:4.

United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). 1988. Sources, Effects, and Risks of Ionizing Radiation. United Nations Publications, New York.

United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). 1993. Sources and Effects of Ionizing Radiation. United Nations Publications, New York.

U.S. Nuclear Regulatory Commission (NRC). 1977. Ninth Annual Occupational Radiation Exposure Report. NUREG-0322. NRC, Washington, DC.

U.S. Nuclear Regulatory Commission (NRC). 1977. Report to Congress on Abnormal Occurrences. NUREG-0090-6. NRC, Washington, DC.

U.S. Nuclear Regulatory Commission (NRC). 1985. Contaminated Mexican Steel Incident. Importation of Steel into the US that had been Inadvertently Contaminated with ⁶⁰Co as a Result of Scrapping of a Teletherapy Unit. NUREG-1103. NRC, Washington, DC.

U.S. Nuclear Regulatory Commission (NRC). 1990. Inadvertent Shipment of a Radiographic Source from Korea to Amersham Corporation, Burlington, Massachusetts. NUREG-1405. NRC, Washington, DC.

U.S. Nuclear Regulatory Commission (NRC). 1993a. The NRC Calendar XII(6), Week ending February 12; p. 1.

U.S. Nuclear Regulatory Commission (NRC). 1993b. Loss of an Iridium-192 Source and Therapy Misadministration at Indiana Regional Cancer Center Indiana, Pennsylvania, on November 16, 1992. NUREG-1480. NRC, Washington, DC.

Valverde, N. J., J. M. Cordeiro, A. R. Oliveira, and C. E. Brandao-Mello. 1990. "The Acute Radiation Syndrome in the ¹³⁷Cs Brazilian Accident, 1987," pp. 89-107 In: The Medical Basis for Radiation Accident Preparedness II - Clinical Experience and Follow-up Since 1979. Proceedings of the Second International REAC/TS Conference on the Medical Basis for Radiation Accident Preparedness. Eds. R. C. Ricks and S. A. Fry. Elsevier Science Publishing Co., New York.

Vandecasteele, C. M., J. P. Dehut, S. Van Laer, D. Deprins, and C. Myttenaere. 1980. "Long-term availability of Tc deposited on soil after accidental releases." Health Phys. 57:247-254.

Vodopick H., and G. A. Andrews. 1974. "Accidental radiation exposure." Arch. Environ. Health 28:53-56.

Vodopick, H. and G. A. Andrews. "The University of Tennessee Comparative Animal Research Laboratory Accident in 1971." 1980. In: The Medical Basis for Radiation Accident Preparedness, Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hahner and S. A. Fry. Elsevier/North Holland:141-149.

Wade, L., Jr. 1972. "Accidental ⁶⁰Co Exposure at the University of Tennessee Atomic Energy Commission Agricultural Research Laboratory." Nuclear Safety 13:304-308.

Webber, C. E. and J. W. Harvey. 1976. "Accidental Human Inhalation of Ruthenium Tetroxide." Health Phys. 30:352-355.

Widmer, D. J., K. W. Logan, S. M. Langhorst, and W. L. Kenedy. 1986. "Gamma Camera Measurement of Accidental Internal Radionuclide Deposition: Ir-192 and Sm-153." Health Phys. 51(3):349-351.

Yalcintas, M. G., T. D. Jones, H. R. Meyer, H. Ozer, and S. Unsal. 1980. "Estimation of Dose Due to Accidental Exposure to a Co-60 Therapy Source." Health Phys. 38:187-191.

Ye, G. Y., Y. Lui, N. Tien, B. Chiang, F. Chien, C. Xiae. 1980. "The People's Republic of China Accident in 1963." In: The Medical Basis for Radiation Accident Preparedness, Proceedings of the REAC/TS International Conference: The Medical Basis for Radiation Accident Preparedness. Eds. K. F. Hahner and S. A. Fry. Elsevier/North Holland:81-89.

Yoshizumi, T, T, and C. A. Chuprinko. 1989. "Accidental CT Scanning Without Al Filtration and Its Dosimetry." Health Phys. 56:253-254.

Zeevaert, T., C. M. Vandecasteele, and R. Kirchem. 1989. "Assessment of Dose to Man from Releases of Tc-99 in Fresh Water Systems." Health Phys. 57:337-343.

6.0 TASK 6 APPENDIX

Table Task-6-A1. Individual dose equivalent values for the Indiana, PA accident.

at Indiana Regional Cancer Center Indiana, Pennsylvania, on November 16, 1993, NUREG-1480 1.37E11 Bq (3.7 Ci) Ir-192				Gamma			
				(Sv/h @ [m ² /Bq])	t. (Sv/h @ 1m)		
				1.6E-13	0.021904		
				mSv	F_p		
				Low	High		
				High	Avg.		
				Avg.	(h @ 1 m)		
1	Table 6.3	Physician A	1.82	7.21	4.515	0.004515	0.206127
2	Table 6.3	TRTT-A			8.2	0.0082	0.374361
3	Table 6.3	RTT-B			1.1	0.0011	0.050219
4	Table 6.3	RTR			1.4	0.0014	0.063915
5	Table 6.3	Medical Physicist A			1.2	0.0012	0.054785
6	Table 6.3	Nurse A			6.3	0.0063	0.287619
7	Table 6.4	Patient C	0.09	0.1	0.095	0.000095	0.004337
8	Table 6.4	Patient D	0.09	0.1	0.095	0.000095	0.004337
9	Table 6.4	Patient E	0.09	0.1	0.095	0.000095	0.004337
10	Table 6.4	Patient F	0.09	0.1	0.095	0.000095	0.004337
11	Table 6.4	Patient G	0.09	0.1	0.095	0.000095	0.004337
12	Table 6.4	Patient H	0.09	0.1	0.095	0.000095	0.004337
13	Table 6.4	Patient I	0.09	0.1	0.095	0.000095	0.004337
14	Table 6.4	Patient J	0.17	0.99	0.58	0.00058	0.026479
15	Table 6.4	Phlebotomist	3.5	13.9	8.7	0.0087	0.397188
16	Table 6.4	Office Manager	0.04	0.09	0.065	0.000065	0.002967
17	Table 6.4	Medical Secretary	0.04	0.09	0.065	0.000065	0.002967
18	Table 6.4	Tumor Registrar	0.04	0.09	0.065	0.000065	0.002967
19	Table 6.5	Patient K	2.4	4.3	3.35	0.00335	0.15294
20	Table 6.5	Patient L	2.4	4.3	3.35	0.00335	0.15294
21	Table 6.5	Administrative Aide A	0.09	0.58	0.335	0.000335	0.015294
22	Table 6.5	Administrative Aide B	0.09	0.58	0.335	0.000335	0.015294
23	Table 6.5	Laboratory Employee	0.08	0.23	0.155	0.000155	0.007076
24	Table 6.6	Ambulance Driver	4.8	8.4	6.6	0.0066	0.301315
25	Table 6.6	Ambulance Aide	13.3	25.7	19.5	0.0195	0.890248
26	Table 6.8	RN A	25	42	33.5	0.0335	1.529401
27	Table 6.8	RN B	89	137	113	0.113	5.158875
28	Table 6.8	RN C	38	63	50.5	0.0505	2.305515
29	Table 6.8	RN D	3.6	5.5	4.55	0.00455	0.207725
30	Table 6.8	GPN A	88	136	112	0.112	5.113221
31	Table 6.8	LPN A	56	87	71.5	0.0715	3.264244
32	Table 6.8	LPN B	112	174	143	0.143	6.528488
33	Table 6.8	LPN C	56	87	71.5	0.0715	3.264244
34	Table 6.8	LPN D	28	43	35.5	0.0355	1.620709
35	Table 6.8	CNA A	103	148	125.5	0.1255	5.729547
36	Table 6.8	CNA B	52	74	63	0.063	2.876187
37	Table 6.8	CNA C	155	223	189	0.189	8.628561
38	Table 6.8	CNA D	52	74	63	0.063	2.876187
39	Table 6.8	CNA E	103	148	125.5	0.1255	5.729547
40	Table 6.8	CNA F	52	74	63	0.063	2.876187
41	Table 6.8	CNA G	52	74	63	0.063	2.876187
42	Table 6.8	CNA H	3.6	5.5	4.55	0.00455	0.207725
43	Table 6.8	CNA I	4.3	6.2	5.25	0.00525	0.239682
44	Table 6.8	Maintenance Man A	19	38	28.5	0.0285	1.301132
45	Table 6.8	Dietician	3.6	6.3	4.95	0.00495	0.225986

Table Task-6-A1, continued. Individual dose equivalent values for the Indiana, PA accident.

46	Table 6.8	Activities Director	4.8	22	13.4	0.0134	0.61176
47	Table 6.10	Relative A	54.4	166	110.2	0.1102	5.031045
48	Table 6.10	Relative B	29.2	42	35.6	0.0356	1.625274
49	Table 6.10	Relative C	36.5	64.5	50.5	0.0505	2.305515
50	Table 6.10	Relative D	23.5	36.5	30	0.03	1.369613
51	Table 6.10	Relative E	21.9	31.5	26.7	0.0267	1.218955
52	Table 6.10	Relative F	21.9	31.5	26.7	0.0267	1.218955
53	Table 6.10	Friend A	23.5	92.8	58.15	0.05815	2.654766
54	Table 6.11	Resident B	37.6	128	82.8	0.0828	3.780131
55	Table 6.11	Resident C	63.9	197	130.45	0.13045	5.955533
56	Table 6.11	Resident D	23.1	84	53.55	0.05355	2.444759
57	Table 6.11	Resident E	11.2	15.4	13.3	0.0133	0.607195
58	Table 6.11	Resident F	27.5	31.7	29.6	0.0296	1.351351
59	Table 6.11	Resident G	18.3	22	20.15	0.02015	0.919923
60	Table 6.11	Resident H	34.2	39.5	36.85	0.03685	1.682341
61	Table 6.11	Resident I	21.6	25.9	23.75	0.02375	1.084277
62	Table 6.11	Resident J	14.4	24.3	19.35	0.01935	0.8834
63	Table 6.11	Resident K	17.3	22.8	20.05	0.02005	0.915358
64	Table 6.11	Resident L	12.2	27.5	19.85	0.01985	0.906227
65	Table 6.11	Resident M	57.3	90.9	74.1	0.0741	3.382944
66	Table 6.11	Resident N	13.8	23.4	18.6	0.0186	0.84916
67	Table 6.13	Driver A			1.7	0.0017	0.077611
68	Table 6.13	Driver B	2.5	5.2	3.85	0.00385	0.175767
69	Table 6.13	Driver C	0.34	0.36	0.35	0.00035	0.015979
70	Table 6.13	Supervisor A	13	51.3	32.15	0.03215	1.467768
71	Table 6.13	Safety Technician A	34.3	89.5	61.9	0.0619	2.825968
72	Table 6.13	Safety Technician B	26.9	68.4	47.65	0.04765	2.175402
73	Table 6.13	Other BFI 1	0.013	0.017	0.0148	1.48E-05	0.000676
74	Table 6.13	Other BFI 2	0.013	0.017	0.0148	1.48E-05	0.000676
75	Table 6.13	Other BFI 3	0.013	0.017	0.0148	1.48E-05	0.000676
76	Table 6.13	Other BFI 4	0.013	0.017	0.0148	1.48E-05	0.000676
77	Table 6.13	Other BFI 5	0.013	0.017	0.0148	1.48E-05	0.000676
78	Table 6.13	Other BFI 6	0.013	0.017	0.0148	1.48E-05	0.000676
79	Table 6.13	Other BFI 7	0.013	0.017	0.0148	1.48E-05	0.000676
80	Table 6.13	Other BFI 8	0.013	0.017	0.0148	1.48E-05	0.000676
81	Table 6.13	Other BFI 9	0.013	0.017	0.0148	1.48E-05	0.000676
82	Table 6.13	Other BFI 10	0.013	0.017	0.0148	1.48E-05	0.000676
83	Table 6.13	Other BFI 11	0.013	0.017	0.0148	1.48E-05	0.000676
84	Table 6.13	Other BFI 12 11/29	0.004	0.007	0.00564	5.64E-06	0.000257
85	Table 6.13	Other BFI 13 11/29	0.004	0.007	0.00564	5.64E-06	0.000257
		Collective			2478	2.478	113.139
		Average			29.16	0.02916	1.33105
		Standard Deviation			40.75	0.041	1.860
		Minimum			0.00564	5.64E-06	0.000257
		Maximum			189.00	0.189	8.628561
		No. Individuals			85	85	85
		Geometric Mean			2.91	0.00291	0.13304
		Geometric Standard Deviation			25.3	25.3	25.3

Table Task-6-A2. Individual dose equivalents and Time-and-Proximity Factors for the 1990 Korea-USA ¹⁹²Ir Shipment Accident.

Inadvertent Shipment of a Radiographic Source from Korea to Amersham Corporation, Burlington, Massachusetts, NUREG-1405, 1990

Ir-192 Korea: 1/18//90 - 2/11/90

INC S/N 1062 USA: 2/11/90 - 3/8/90

date	74.6 days days	Act. (Ci)	Gamma Gamma*Ac (Sv/h t. (Sv/h @ [m ² /Bq] 1m)	
			1.60E-13	0.02368
1/18/90	55	4.00		
3/8/90	6	2.54		
3/14/90	0	2.40	2.49	

Source of data	Who	mrem	rem	Sv	F _p , hours at a meter
Table 5.2	Nova Truck Driver	40	0.04	0.0004	0.016892
Table 5.2	Nova Cargo Unloader 1	330	0.33	0.0033	0.139358
Table 5.2	Nova Cargo Unloader 2	330	0.33	0.0033	0.139358
Table 5.2	Nova Unloading Forklift Operator	50	0.05	0.0005	0.021115
Table 5.2	Nova Unloading Checker	70	0.07	0.0007	0.029561
Table 5.2	Nova Shipping clerk	230	0.23	0.0023	0.097128
Table 5.2	Nova Shipping Supervisor	230	0.23	0.0023	0.097128
Table 5.2	Nova Asst. Shipping Clerk	150	0.15	0.0015	0.063345
Table 5.2	Nova Receiving Clerk	400	0.4	0.004	0.168919
Table 5.2	Nova Loading Forklift Operator	200	0.2	0.002	0.084459
Table 5.2	Nova Cargo Loader 1	470	0.47	0.0047	0.19848
Table 5.2	Nova Cargo Loader 2	470	0.47	0.0047	0.19848
Table 5.2	Nova Loading Checker	220	0.22	0.0022	0.092905
Tables 5.3 & 5.4	Covenant Senior Driver	34850	34.85	0.3485	14.71706
Tables 5.3 & 5.4	Covenant Driver Trainee	27560	27.56	0.2756	11.63851
Table 5.4	Patriot Operator	5600	5.6	0.056	2.364865
Table 5.4	Patriot Warehouseman	1080	1.08	0.0108	0.456081
Table 5.4	USCS inspector	810	0.81	0.0081	0.342061
Table 5.4	Patriot Truck Driver	550	0.55	0.0055	0.232264
Table 5.5	Amersham Rad Tech A	20	0.02	0.0002	0.008446
Table 5.5	Amersham Rad Safety Specialist	30	0.03	0.0003	0.012669
Table 5.5	Amersham Rad Tech B	20	0.02	0.0002	0.008446
Table 5.5	Amersham Rad Safety Officer	15	0.015	0.00015	0.006334
Table 5.5	Amersham Hot Lab Supervisor	40	0.04	0.0004	0.016892
Collective		73765	73.765	0.73765	31.15
Average		3073.5	3.07	0.0307	1.30
Standard Deviation		8801.9	8.80	0.0880	3.72
Minimum		15	0.015	0.00015	0.00633
Maximum		34850	34.85	0.3485	14.72
No. Individuals		24	24	24	24
Geometric Mean		259.58	0.260	0.00260	0.110
Geometric Standard Deviation		7.87	7.87	7.87	7.87

Table Task-6-A3. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma (Sv/h [m ² /Bq])	Gamma*Act. (Sv/h @ 1m)	Whole Body Time-and-Proximity Factor (hours at a meter)
ALG78	78	Ir-192	IndRad	25	9.25E+11	1.60E-13	0.148	87.837838
ALG78	78	Ir-192	IndRad	25	9.25E+11	1.60E-13	0.148	89.527027
ALG78	78	Ir-192	IndRad	25	9.25E+11	1.60E-13	0.148	81.081081
ALG78	78	Ir-192	IndRad	25	9.25E+11	1.60E-13	0.148	74.324324
ALG78	78	Ir-192	IndRad	25	9.25E+11	1.60E-13	0.148	270.27027
ALG78	78	Ir-192	IndRad	25	9.25E+11	1.60E-13	0.148	6.7567568
ALG78	78	Ir-192	IndRad	25	9.25E+11	1.60E-13	0.148	6.7567568
AUS70	70	Ir-192	IndRad	22	8.14E+11	1.6E-13	0.13024	1.23
AUS70	70	Ir-192	IndRad	22	8.14E+11	1.6E-13	0.13024	3.53
BAN85	85	Ir-192	IndRad	50	1.85E+12	1.6E-13	0.296	8.45
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	0.0120656
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	0.0241313
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	0.0422297
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	0.0844595
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	0.1025579
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	0.11764
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	0.1870174
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	0.3619691
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	0.3921332
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	1.3875483
CA79	79	Ir-192	IndRad	28	1.04E+12	1.6E-13	0.16576	5.2787162
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.002
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.004
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.006
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.008
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.01
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.012
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.014
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.016
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.018
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.02
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.022
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.024
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.026
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.028
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.03
CZE66	66	I-131	Medical	2.25	8.33E+10	7.63E-14	0.006351975	0.032
CZE73	73	Co-60	Tele	2973	1.10E+14	3.7E-13	40.7	0.0025
CZE73	73	Co-60	Tele	2973	1.10E+14	3.7E-13	40.7	0.0344
FRG68	68	Ir-192	IndRad	7.8	2.89E+11	1.6E-13	0.046176	32.4
FRG72	72	Ir-192	IndRad	29.73	1.10E+12	1.6E-13	0.176	1.7
FRG81	81	Co-60	Tele	2595	9.60E+13	3.7E-13	35.52	0.0113
FRG81	81	Co-60	Tele	2595	9.60E+13	3.7E-13	35.52	0.0056
IND68	68	Ir-192	IndRad	1.4	5.18E+10	1.6E-13	0.008288	156.85
ISR90	90	Co-60	Steril	340541	1.26E+16	3.7E-13	4662	0.0032

Table Task-6-A3 continued. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma (Sv/h [m ² /Bq])	Gamma*Act. (Sv/h @ 1m)	Whole Body Time-and-Proximity Factor (hours at a meter)
ITA75	75	Co-60	Steril	36000	1.33E+15	3.7E-13	492.84	0.0284
JOH59	59	Co-60	IndRad	1.75	6.48E+10	3.7E-13	0.0239575	1.04
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	3.2113863
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	4.1748022
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	4.8170794
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	8.0284657
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	16.056931
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	42.711438
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0063345
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0084459
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0084459
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0126689
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0168919
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0168919
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0211149
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0295608
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0633446
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0844595
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0929054
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0971284
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0971284
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1393581
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1393581
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1689189
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1984797
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1984797
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.2322635
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.3420608
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.4560811
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	2.3648649
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	11.638514
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	14.717061
KY76	76	Ir-192	IndRad	78	2.89E+12	1.60E-13	0.46176	2.06
LA78	78	Ir-192	IndRad	100	3.70E+12	1.60E-13	0.592	0.0845
MEX62	62	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	175.31045
MEX62	62	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	686.63258
MEX62	62	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	511.32213
MEX62	62	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	419.28415
MEX62	62	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	438.27611
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	10.416667
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	26.041667
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	67.708333
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	83.88
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	106.5138
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	123.3923
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	139.457
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	156.6363

Table Task-6-A3 continued. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma (Sv/h [m ² /Bq])	Gamma*Act. (Sv/h @ 1m)	Whole Body Time-and-Proximity Factor (hours at a meter)
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.0639153
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.0776114
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.1529401
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.1529401
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.175767
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2061267
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2077246
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2077246
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2259861
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2396822
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2876187
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.3013148
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.3743608
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.3971877
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.607195
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.6117604
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.84916
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.8834003
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.8902484
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.9062272
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.9153579
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.9199233
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.0842768
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.2189554
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.2189554
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.3011322
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.3513514
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.3696129
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.4677684
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.529401
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.6207085
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.6252739
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.6823411
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.1754018
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.305515
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.305515
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.4447589
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.6547663
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.8259679
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.876187
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.876187
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.876187
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.876187
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	3.264244
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	3.264244
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	3.3829438
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	3.7801315

Table Task-6-A3 continued. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma (Sv/h [m ² /Bq])	Gamma*Act. (Sv/h @ 1m)	Whole Body Time-and-Proximity Factor (hours at a meter)
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.0310446
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.1132213
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.1588751
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.7295471
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.7295471
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.9555332
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	6.5284879
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	8.628561
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	14.609204
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	29.218408
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	43.827611
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	58.436815
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	292.18408
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	584.36815
PRC80	80	Co-60	Steril	53000	1.96E+15	3.70E-13	725.57	1.89
PRC85	85	Cs-137	IndRad?	10	3.70E+11	9.25E-14	0.034225	3.35
PRC86	86	Co-60	(1)	6888	2.55E+14	3.70E-13	94.29672	3.83
PRC87	87	Co-60	Steril	89000	3.29E+15	3.70E-13	1218.41	7.3
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	8.143138
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	8.677649
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	9.118739
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	9.521521
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	9.912457
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	10.31026
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	10.73358
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	11.20769
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	11.77738
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	12.55044
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	14
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	60.87168
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	91.30752
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	121.7434
SAF77	77	Ir-192	IndRad	6.76	2.50E+11	1.60E-13	0.04	2.5
SAF77	77	Ir-192	IndRad	6.76	2.50E+11	1.60E-13	0.04	4.25
SAF77	77	Ir-192	IndRad	6.76	2.50E+11	1.60E-13	0.04	29
SAL89	89	Co-60	Steril	18000	6.66E+14	3.70E-13	246.42	0.013
SAL89	89	Co-60	Steril	18000	6.66E+14	3.70E-13	246.42	0.018
SAL89	89	Co-60	Steril	18000	6.66E+14	3.70E-13	246.42	0.034
SCO69	69	Ir-192	IndRad	25	9.25E+11	1.60E-13	0.148	4.05
TN71	71	Co-60	Expt	7700	2.85E+14	3.70E-13	105.413	0.0247
TRK76	76	Co-60	Tele	2260	8.36E+13	3.70E-13	30.9394	0.0001118
TX72	72	Cs-137	IndRad	4	1.48E+11	9.25E-14	0.01369	730
UK77	77	Ir-192	IndRad	21.6	8.00E+11	1.60E-13	0.128	0.78
UK81	81	Cs-137	Brachy	0.12	4.44E+09	9.25E-14	0.0004107	0.487
WA76	76	Am-241	Dfnse	343	1.27E+13	7.57E-15	0.096165142	0.052
WI61	61	Co-60	Expt	200	7.40E+12	3.70E-13	2.738	0.913

Table Task-6-A3 continued. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma (Sv/h [m ² /Bq])	Gamma*Act. (Sv/h @ 1m)	Whole Body Time- and- Proximity Factor (hours at a meter)
						Minimum		0.0001118
						Maximum		730
						Average		30
						Std Dev		100
						GeoMean		0.37
						GSD		42.53
						Median		0.46
						Mode		6.76E-04
						Number		231

APPENDIX G: PRELIMINARY RISK ANALYSIS FOR SELECTED SOURCES

A full implementation of the risk analysis described in Section 2 of the Task 7 Final Report is beyond the scope of work of the current project. However, a sample risk analysis is given below.

G1.0 RADIOLOGICAL PROPERTIES OF RADIONUCLIDES REGISTERED BETWEEN 1987 AND 1992

Data provided to PNL by S. L. Baggett (Baggett 1993) included 24 valid radionuclide names. Table G-1 lists those radionuclides as row labels in column 1, in order of increasing atomic number. The specific dose equivalent rate constant¹ for each nuclide (or chain, in the case of ²²⁶Ra) is given in the next column in units of rems per hour at 1 m from 1 Ci (Unger and Trubey, 1981). The next two columns list the most restrictive ALI values for inhalation and ingestion, respectively. There is no ALI for ⁸⁵Kr. The half-life for each nuclide is given in the center column.

Since distributions of source strengths and numbers of sources in service *with activities below 20 millicuries (the limit of the ORAU report)* are not available, a source strength of 20 mCi was arbitrarily chosen as a reference value in Table G-1. Also, for simplicity, the "worst plausible case" reference value for the time-and-proximity factors, F_p , of 1000 hours at one meter was chosen (730 was the highest observed) for use in Table G-1. Similarly, a "worst plausible case" reference value of the fraction-taken-in, F_t , of 10^{-4} was chosen for use in Table G-1.

Column 7 shows the dose equivalent in rems for external exposure for 1000 hours at 1 meter away from an unshielded 20 mCi source. Columns 8 and 9 show the committed effective dose equivalent in rems for intakes of 10^{-4} of a 20 mCi source, i.e., $2 \mu\text{Ci}$, for inhalation and ingestion, respectively. Note that these columns can be interpreted as doses in millirems for 1 hour at a meter and 10^{-7} fraction-taken-in, both more central values from the PNL analysis presented in Appendix F, Task 6.

The sixth column of Table G-1 gives the ratio of the value in column 7 divided by the value in column 9. The column 6 values are independent of source strength. Column 6 values are a ratio of external exposure hazard (dose) to intake hazard (dose) under the 1000 hours at a meter and 10^{-4} scenarios. Figure G-1 is a bar plot of Column 6 values. For ⁸⁵Kr, there is no ALI even though it poses an external hazard; for ⁶³Ni, ⁵⁵Fe, ³H, ²¹⁰Bi, ¹⁰⁶Ru, ¹⁴C, and

¹Although it would be desirable to use effective dose equivalent rate constants, none have been published in the peer-reviewed literature. Calculations of such constants, based on the methods of the ICRP and ICRU, as shown in Appendix A of the Task 7 Final Report, lead to unrealistically high numbers for low-energy photon emitters if "bare" sources are assumed (e.g., ²⁴¹Am). Further research is needed in this area. Thus, the work of Unger and Trubey (1981), based on ANSI/ANS-6.1.1-1977, has been used for dose equivalent rate constants.

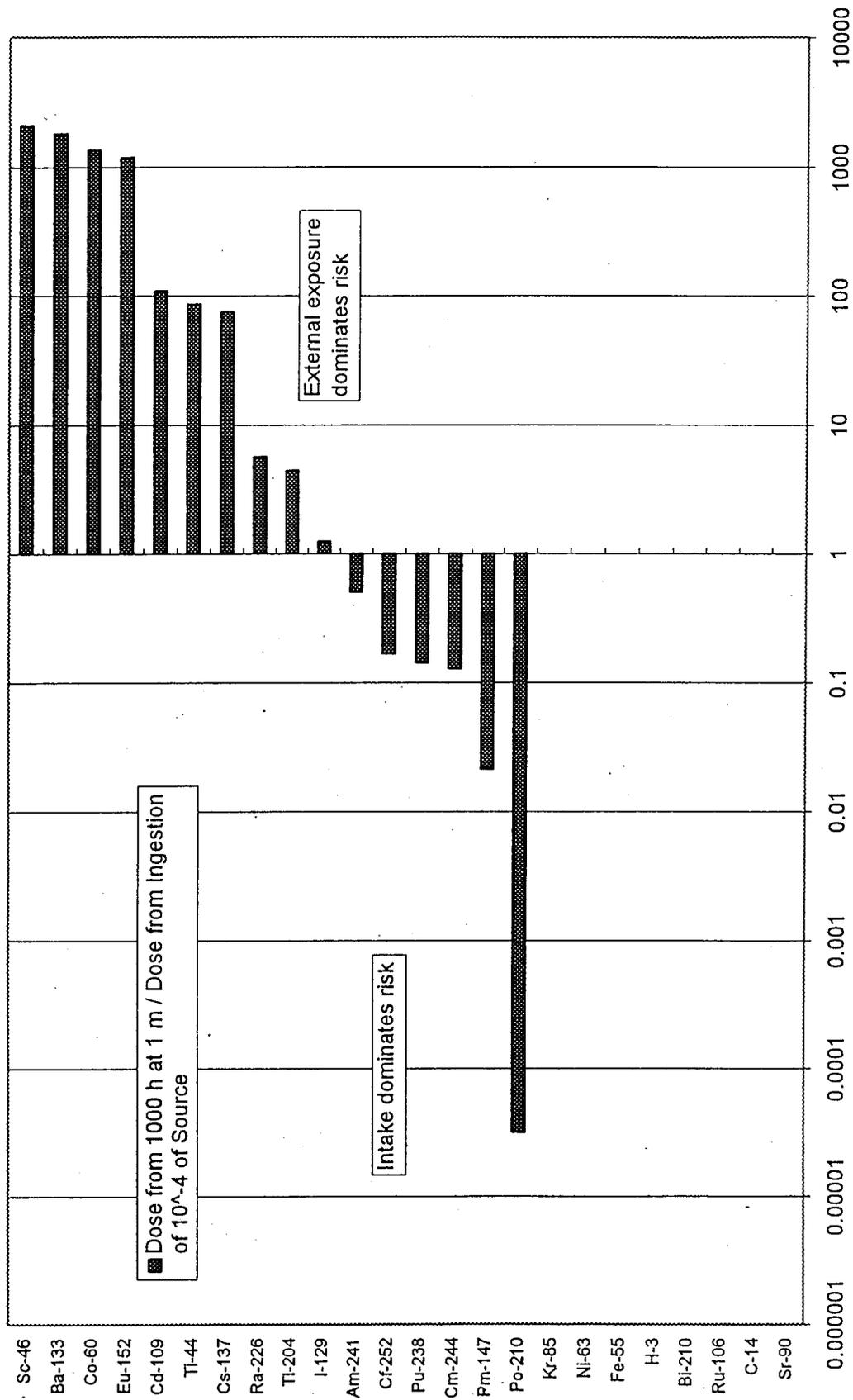


Figure G-1. Ratios of (Dose Equivalent from 1000 h at 1 m) divided by (Committed Effective Dose Equivalent from intake of 10^{-4} of a source). These dimensionless ratios are independent of source strength.

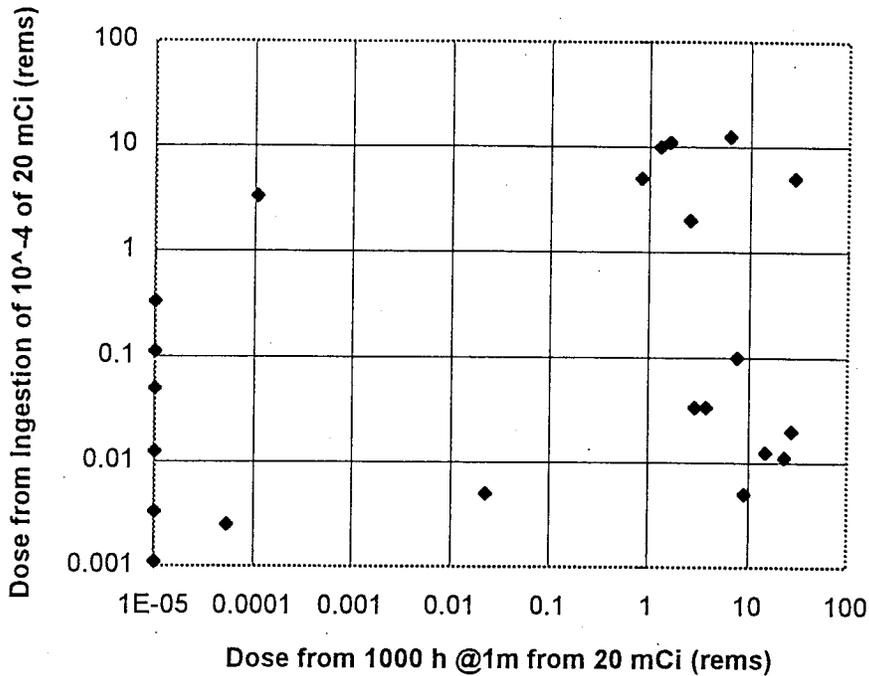


Figure G-2. Committed effective dose equivalent from ingestion of 10^{-4} of a 20 mCi source (vertical axis) as a function dose equivalent from external exposure for 1000 hours at 1 meter from an unshielded 20 mCi source. The relative doses are seen to be widely variable.

G2.0 NUMBER OF DEVICES

Numbers of sources are given in Table Task-3-1B for by Device Code. For this analysis, Device Code D, Gamma Gauges, is chosen for radionuclide ^{137}Cs , which includes 2493 sources with average activity 883 mCi. Table G-2 shows a possible (hypothetical) distribution of numbers of sources and activities to give the correct total and average. It would be best to have exact numbers from a full database.

Table G-2. Possible (hypothetical) distribution of numbers of sources and activities for ^{137}Cs gamma gauges for a total of 2493 sources and an average of 883 mCi per source, as shown in Table Task-3-1B.

Activity (mCi)	Number
1	500
10	500
100	559
1000	800
10,000	134
TOTAL	2493

G3.0 RATES OF DEVICE INVOLVEMENT

G3.1 RATES OF INCIDENTS

Table Task-3-8 shows that there were 6 incidents in ORAU Category B (Regulatory Guide 10.10 Device Code D) over a 10 year period, 5 incidents involving ^{137}Cs . Assuming that one out of two incidents is reported, this gives a rate of $(6 \text{ incidents reported}) \times (2 \text{ incidents occurring per incident reported}) \div (10 \text{ years}) = 1.2 \text{ incidents per year}$ for all Device Code D devices. Using the total number of Device Code D sources in Table Task-3-2, 24,679, and correcting for the fraction of Device Code D sources that are ^{137}Cs , one calculates $(1.2 \text{ incidents per year}) \times (2493 / 24,679) = 0.121 \text{ incident per year}$ for ^{137}Cs Device Code D devices.

One could argue from Table Task-3-8 that the rate for cesium sources is predominant, that is, cesium sources account for 5/6 of all incidents. Using this logic would yield a rate of $(1.2 \text{ incidents per year}) \times (5/6) = 1.0 \text{ incident per year}$ for ^{137}Cs Device Code D devices. This analysis uses the latter value.

G3.2 NUMBERS OF DEVICES PER INCIDENT

Table Task-3-7 shows 9 devices were involved in the 6 incidents listed in Table Task-3-6, for a rate of 1.5 device per incident.

G3.3 RATE OF DEVICE INVOLVEMENT IN INCIDENTS

The rate of device involvement in incidents is the product of the number of incidents per year and the number of devices per incident, $(1.0 \text{ incident per year}) \times (1.5 \text{ devices per}$

incident) = 1.5 devices involved in incidents per year.

G4.0 SEVERITY OF INCIDENTS

There are several aspects to severity.

G4.1 FRACTION OF INCIDENTS RESULTING IN LEAKING SOURCES

In Table Task-3-6, 19 out of 114 incidents resulted in leaking sources, although none of these involved Device Code D (ORAU Category B) gamma gauge sources. One could argue that such sources are made to withstand leaking. Alternatively, Table Task-3-7 shows that 75 of 300 devices involved in incidents leaked, primarily static eliminators and ORAU Category L sources. For this value, 0.125 of incidents result in a leaking source.

G4.2 NUMBER OF PERSONS INVOLVED IN INCIDENT

Most incidents involve 1 person, but more may be involved. For intakes, Table 3.1.1 for Task 6 showed the distribution of numbers of persons involved in accidents, as summarized in Table G-3. There are 4364 persons involved in the 60 incidents, using 77 persons (instead of the value of 20 in Table 3.3.1) for Goiânia.

Due to the difficulty of incorporating different numbers of persons with different fractions-taken-in and different time-and-proximity factors, the accompanying analysis considers only 1-person incidents.

Table G-3. The distribution of numbers of persons involved in incidents.

Number of Persons Involved in Incident	Frequency of Occurrence
1	34
2	9
3	3
4	1
5	3
6	2
11	1
16	1
22	1
24	1
28	1
77	1
94	1
4000	1

G4.3 DISTRIBUTION OF FRACTIONS-TAKEN-IN

These data are also taken from Table 3.3.1 of the Task 6 Report. The fractions-taken-in are 4355 values, expanded from Table 3.3.1, with the omission of the Swiss accident of 1985 (which involved an unsealed ^3H source and a, very high fraction-taken-in, namely, 0.02). The average fraction-taken-in was 1.34×10^{-7} , with a standard deviation of 4.17×10^{-6} . The latter two parameters were used to define a lognormal distribution.

G4.4 DISTRIBUTION OF TIME-AND-PROXIMITY FACTORS

These factors were modeled using a lognormal distribution with mean of 30 hours at 1 m and a standard deviation of 100 hours at 1 m, based on the results presented in Table Task-6-A3.

G4.5 DISTRIBUTION OF SOURCE REMOVAL FROM SHIELD

A scenario of the source removal from the shield was assumed to occur in 0.5 of incidents. This is an arbitrary figure that is probably an overestimate.

G5.0 PRELIMINARY PROBABILISTIC RISK ANALYSIS RESULTS

The output of a 1000-trial Crystal Ball simulation is attached at the end of this appendix.

External dose equivalent per year was modeled as $(\text{Incident Rate [devices/year]}) \times (\text{Source Activity}) \times (\text{Probability of Removal from Shield}) \times (\text{Time-and-Proximity Factor}) \times (\text{Specific Dose Equivalent Rate Constant})$, where the factors in italics were sampled from distributions described above. The mean dose was 6.81 rems per year from incidents (recall that many of these sources have activities of 10,000 millicuries, as described in Table G-2) with a maximum of 1056 rems. Had the source activities been limited to 20 mCi, the highest dose would have been 20/883 as large, or 24 rems.

Committed effective dose equivalent (CEDE) from ingestion intakes was modeled as $(\text{Incident Rate [devices/year]}) \times (\text{Source Activity}) \times (\text{Probability of Leakage}) \times (\text{Fraction-Taken-In}) \times (5 \text{ rems CEDE/ALI}) \times (1 \text{ ALI}/0.1 \text{ mCi})$, where the factors in italics were sampled from distributions described above. The mean dose was 0.4 mrem with a maximum of 154 mrem. Had the source activities been limited to 20 mCi, the highest dose would have been 20/883 as large, or 3.5 millirems.

G6.0 DISCUSSION

This analysis is preliminary and was made to demonstrate the proof-of-principle. The methods should be reviewed and refined, and calculations for extremity doses, non-stochastic effects, and collective doses should be made for each nuclide and each Device Code listed in Table Task-3-1. It should be determined whether there is a need to restrict the data to source strengths less than or equal to 20 mCi as was done in the scope of work for the ORAU Report. There is a need for detailed data on numbers of devices by source type (the Regulatory Guide 10.10 Device Code is not adequate), isotope(s), dates placed in service, activities, and design. These data should be available for each of the 500,000 or so sources now in use.

It was discovered that incorporating realistic scenarios (e.g., different fractions-taken-in and different time-and-proximity factors for each individual in multiple-person incidents) is more difficult than expected, the remaining funds did allow us to complete this extra task.

G7.0 ADDITIONAL REFERENCES FOR APPENDIX G

Baggett, S. L. 1993. Letter to D. J. Strom of Pacific Northwest Laboratory dated November 2, 1993. Washington, DC: Sealed Source Safety Section, U.S. Nuclear Regulatory Commission.

Unger, L. M., and D. K. Trubey. 1981. *Specific Gamma-Ray Dose Constants for Nuclides Important to Dosimetry and Radiological Assessment*. ORNL RSIC-45. United States

G8.0 SAMPLE PROBABILISTIC RISK ANALYSIS: CRYSTAL BALL REPORT

Forecast: External Dose

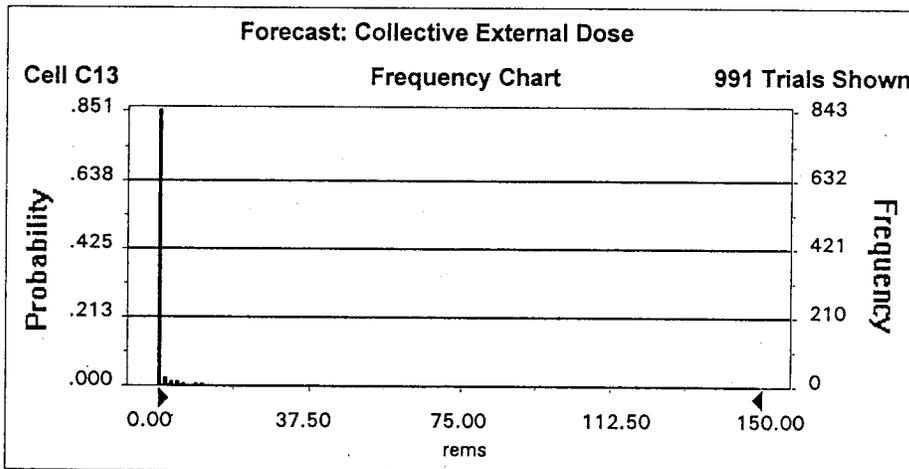
Cell: C13

Summary:

Display Range is from 0.00 to 150.00 rems
 Entire Range is from 0.00 to 1,056.46 rems
 After 1,000 Trials, the Std. Error of the Mean is 1.53

Statistics:

	<u>Value</u>
Trials	1000
Mean	6.81E+00
Median	0.00E+00
Mode	0.00E+00
Standard Deviation	4.84E+01
Variance	2.34E+03
Skewness	1.44E+01
Kurtosis	2.62E+02
Coeff. of Variability	7.10E+00
Range Minimum	0.00E+00
Range Maximum	1.06E+03
Range Width	1.06E+03
Mean Std. Error	1.53E+00



Forecast: Collective External Dose (cont'd)

Cell: C13

Percentiles:

<u>Percentile</u>	<u>rem/s</u>
0.0%	0.00
2.5%	0.00
5.0%	0.00
50.0%	0.00
95.0%	21.48
97.5%	50.46
100.0%	1,056.46

End of Forecast

Forecast: Intake Dose

Cell: C14

Summary:

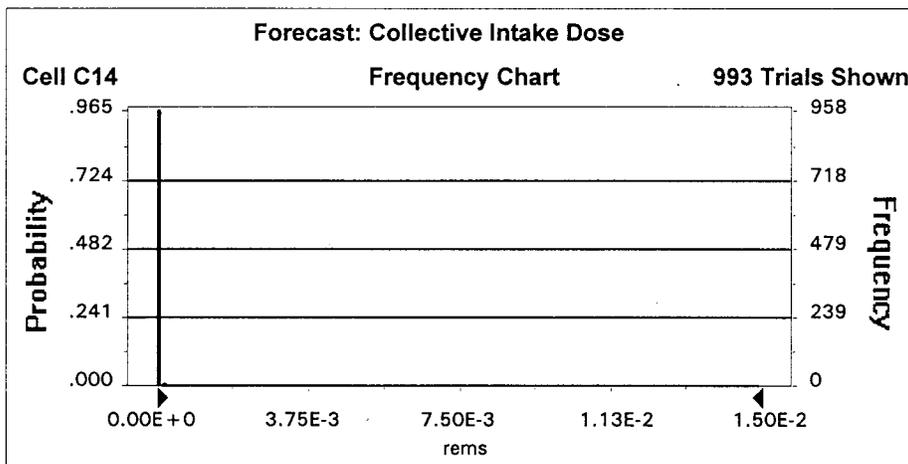
Display Range is from 0.00E+0 to 1.50E-2 rems

Entire Range is from 0.00E+0 to 1.54E-1 rems

After 1,000 Trials, the Std. Error of the Mean is 1.74E-4

Statistics:

	<u>Value</u>
Trials	1000
Mean	3.99E-04
Median	0.00E+00
Mode	0.00E+00
Standard Deviation	5.51E-03
Variance	3.04E-05
Skewness	2.33E+01
Kurtosis	6.22E+02
Coeff. of Variability	1.38E+01
Range Minimum	0.00E+00
Range Maximum	1.54E-01
Range Width	1.54E-01
Mean Std. Error	1.74E-04



Forecast: Collective Intake Dose (cont'd)

Cell: C14

Percentiles:

<u>Percentile</u>	<u>rem/s</u>
0.0%	0.00E+00
2.5%	0.00E+00
5.0%	0.00E+00
50.0%	0.00E+00
95.0%	4.95E-05
97.5%	6.15E-04
100.0%	1.54E-01

End of Forecast

Assumptions

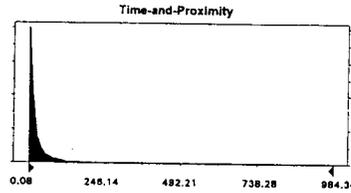
Assumption: Time-and-Proximity

Cell: C9

Lognormal distribution with parameters:

Mean 30.00
Standard Dev. 100.00

Selected range is from 0.00 to +Infinity
Mean value in simulation was 33.57



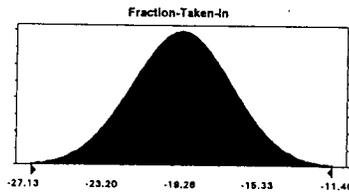
Assumption: Fraction-Taken-In

Cell: C8

Lognormal distribution with parameters:

Mean -19.26 (log space)
Standard Dev. 2.62 (log space)

Selected range is from -Infinity to +Infinity
Mean value in simulation was 0.00



Assumption: Probability of Leakage

Cell: C7

Custom distribution with parameters:

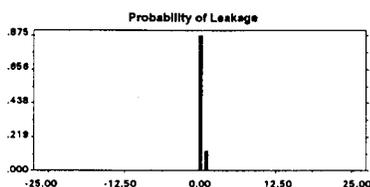
		<u>Relative Prob.</u>
Single point	0.00	0.875000
Single point	1.00	0.125000
Total Relative Probability		1.000000

Mean value in simulation was 0.14

Appendix G

Assumption: Probability of Leakage (cont'd)

Cell: C7



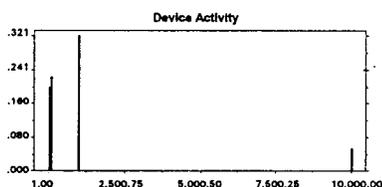
Assumption: Device Activity

Cell: C6

Custom distribution with parameters:

		<u>Relative Prob.</u>
Single point	1.00	0.200562
Single point	10.00	0.200562
Single point	100.00	0.224228
Single point	1,000.00	0.320899
Single point	10,000.00	0.053751
Total Relative Probability		1.000000

Mean value in simulation was 764.28



Assumption: Probability of Removal from Shield

Cell: C11

Custom distribution with parameters:

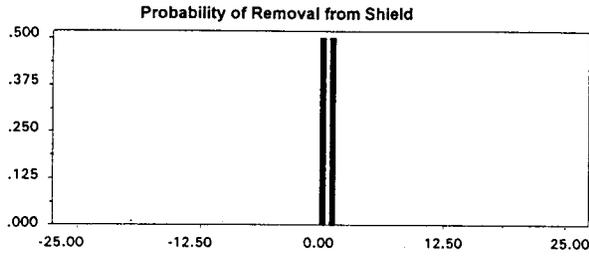
		<u>Relative Prob.</u>
Single point	0.00	0.500000
Single point	1.00	0.500000
Total Relative Probability		1.000000

Mean value in simulation was 0.47

Appendix G

Assumption: Probability of Removal from Shield (cont'd)

Cell: C11



Assumption: Number of Persons

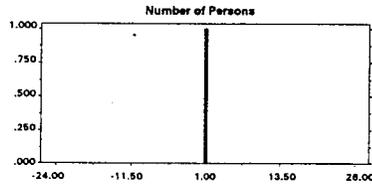
Cell: C10

Custom distribution with parameters:

Single point 1.00
Total Relative Probability

Relative Prob.
1.000000
1.000000

Mean value in simulation was 1.00



End of Assumptions