TECHNICAL ASSISTANCE TO THE U.S. ENVIRONMENTAL PROTECTION AGENCY ON 40 CFR PART 191

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CHAPTER 1

INTRODUCTION

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INTRODUCTION

Over the past several years, the U.S. Environmental Protection Agency (EPA) has been working on a revision to its environmental standard for management and disposal of spent nuclear fuel, high-level and transuranic radioactive wastes (40 CFR Part 191) in response to the 1987 remand by the U.S. Court of Appeals. In a December 20, 1991 management meeting between the U.S. Department of Energy (DOE) and the EPA, the DOE volunteered to provide technical assistance to the EPA in developing supporting technical justification for revising sections of 40 CFR Part 191. In a January 7, 1992 letter from M. Oge (EPA) to R. Berube (DOE), the EPA accepted the offer and requested technical assistance in several specific areas. Those areas were: human intrusion, the three-bucket approach, multimode release limits, collective dose, TRU waste equivalence unit, uncertainty propagation, and Carbon-14. The DOE envisioned that this technical assistance would consist of a six-month effort of comprehensive technical analyses and computer modeling exercises that could provide the technical foundation for any proposed revision. However, due to time constraints resulting from the EPA's 40 CFR Part 191 repromulgation schedule, the technical studies were compressed and preliminary working papers were provided to EPA on May 12, 1992, approximately six weeks after the initiation of the contractor's efforts. EPA and the U.S. Nuclear Regulatory Commission (NRC) reviewed the working papers and provided comments to DOE. During this period, the DOE contractors were finalizing the technical analyses and modeling exercises. Based on the EPA/NRC comments and the results on the contractor's studies, certain sections of the working papers have been revised or augmented with additional information. The Technical Assistance Document is considered a final product at this time.

In its efforts to develop a technical foundation for the changes in the seven identified areas, DOE found that it was not possible in some cases to construct a completely rigorous technical foundation on which to base any revision. DOE believes this occurred because these tasks attempted to correct a fundamentally flawed standard through a series of relatively minor changes. DOE believes that the changes discussed in this document are the minor adjustments necessary to make the standard nominally workable. However, they do not correct the underlying fundamental flaws. In an effort to accommodate EPA's structure and approach, much of the language from the 1985 40 CFR Part 191 Final Rule was retained in the technical writeups of the various chapters. This was done only for ease and clarity of presentation and does not indicate a Departmental endorsement.

In order to guide its contractors in performing the technical studies, the DOE developed task assignments containing statements of work for each area. These task assignments and responsible organizations are:

- Task 1: Human Intrusion
  Responsible Organization: Sandia National Laboratory

  Develop the specifics of an approach that separates human intrusion from the complementary cumulative distribution function (CCDF). Information developed from this task can be found in Chapter 3 of this document.

- Task 2: Three-Bucket Approach
  Responsible Organization: Sandia National Laboratory

  Analyze the NRC's suggested "three-bucket approach" (and EPA's modification of NRC's approach), evaluate its usefulness in alleviating problems with the probabilistic analysis, and determine the implementability of the approach. Information developed from this task can be found in Chapter 4 of this document.

- Task 3: Multimode Release Limits
  Responsible Organization: Sandia National Laboratory

  Develop the concept of a multi-column release limit table to cover the possible release modes for generic repositories, including methods for computing limits for each mode and methods for implementation. Information developed from this task can be found in Chapter 5 of this document.

- Task 4: Collective Dose
  Responsible Organization: Sandia National Laboratory

  Evaluate the feasibility and develop the concept of a collective dose option to the release limits approach, including the implementability of such an option. Information developed from this task can be found in Chapter 6 of this document.

- Task 5/6: TRU Waste Equivalence Unit
  Responsible Organization: Sandia National Laboratory

  Develop a fundamental criteria for disposal of TRU waste and a waste unit that is equivalent to HLW, based on a comparable acceptable collective risk. (This task was originally started as two tasks and later combined because of similarities in scope.) Information developed from this task can be found in Chapter 7 of this document.

- Task 7: Uncertainty Propagation
  Responsible Organization: CRWMS M&O (TESS)

  Conduct the necessary analyses and evaluations to provide a defensible estimate of the uncertainty in repository performance predictions as a function of time, for periods between 1,000 and 100,000 years. Information developed from this task can be found in Chapter 8 of this document.

- Task 8: Carbon-14
  Responsible Organization: CRWMS TMSS (SAIC)

  Develop further information concerning Carbon-14 releases from unsaturated media, including costs of compliance with the present standard, and develop an alternative requirement for regulating such releases. Information developed from this task can be found in Chapter 9 of this document.

For each of these tasks, information was developed to support a possible revision of the standard. Four types of material were developed for each task and are presented in this document:

1. Statement of the Problem
2. Recommended Approach
3. Supplementary Information
4. Technical Support Documentation

The Statement of the Problem identifies the concern about the standard that is being addressed in the sections that follow. The Recommended Approach provides example regulatory language to illustrate how the proposed revision might be incorporated into the standard. The Supplementary Information provides a general discussion of the technical and regulatory justification for the proposed revision in a format that is similar to the information that would be required in the Federal Register supplementary information text for the repromulgated standard. The Technical Support Documentation provides the details of the technical analysis that support the proposed revision; this type of information would be needed for the Background Information Document (BID) that the EPA would prepare as part of the repromulgation process.

Since the DOE intends that the recommendations in this document be considered as a whole, the suggested revisions to the standard resulting from each task have been consolidated, and are presented in Chapter 2.

CHAPTER 2

SUMMARY OF RECOMMENDED CHANGES

CHAPTER 2

SUMMARY OF RECOMMENDED CHANGES

2.1 OVERVIEW

Chapters 3 through 9 of this document contain recommended changes to EPA's environmental standard for the management and disposal of spent nuclear fuel, high-level, and transuranic wastes (40 CFR Part 191). Each chapter presents and discusses a separate set of changes in order to describe each recommendation clearly. The DOE intends, however, for the recommendations be considered as a whole. In formulating each recommendation, the DOE has considered its effect on the other recommendations. Furthermore, the intentions of the DOE can be understood fully only if the recommendations are thought of as constituting a single overall recommendation. The recommendations contained in each of the chapters that follow are summarized below:

- Chapter 3 describes a formulation of the containment requirements that eliminates some difficulties with the inclusion of human-initiated events and processes in the demonstration of compliance. The recommendation allows for such processes and events to be separated from the CCDF. The DOE intends that this formulation be a part of each option for demonstrating compliance with the containment requirements. These options, three in all, are discussed in item 3 below.

- Chapter 4 describes the DOE concerns with the proposed "three-bucket approach" to demonstrating compliance with the containment requirements. The DOE recommends that this approach remain as an option in the next issuance of the standard as a Proposed Rule allowing additional time for review and analysis.

- Chapters 5 and 6 describe additional options for the containment requirements. These options are: (a) a multimode option that includes limits for all release modes to be considered in the containment requirements (land, well, river, and ocean), and (b) a collective dose option that would apply to population doses resulting from the same four release modes. The DOE recommends that both of these options appear in the standard in addition to the current requirement, after it has been modified according to the recommendation for human intrusion in item 1 above. The DOE recommends that the standard allow the DOE to choose any one of three options for the demonstration of compliance. Furthermore, the DOE recommends that the standard also allow the DOE to choose the use of a combination of two of these options in generating the CCDF: the DOE may elect to use a combination of the original (but reworded) release limit option and the collective dose option (described in Chapter 6), or a combination of the multimode release limit option (described in Chapter 5) and the collective dose option.

  In addition, it is recommended that none of these options (or combination of options) be used to regulate gaseous radionuclide releases. In order to be consistent with other EPA regulations that address similar releases from other facilities, these gaseous releases should be regulated as part of the individual protection requirements in 40 CFR Part 191, as discussed in item 6 below.

Implementation of the multimode release limit or collective dose options discussed in Chapters 5 and 6 will result in the need to obtain more information regarding site characteristics. Such site characterization activities may prove to be prohibitively expensive. When compliance demonstrations require the input of more parameters (i.e., going from releases to collective dose), uncertainty is increased. The goal of the site characterization activity is to reduce the uncertainty in these parameters. By specifying acceptable values for some parameters (i.e., providing a standard biosphere) site characterization costs can be lower. Even though this cost may be viewed as a disadvantage, these proposed options have the advantage of allowing site-specific considerations to be taken into account while at the same time retaining the generic nature of the standard. It is also important to note that each of the three resulting options for the containment requirements has its advantages and disadvantages. For that reason, the revised standard should not require the use of any particular option but should leave the choice up to the Department. Table 2-1 provides a comparison of the various containment options being recommended.

- Chapter 7 describes the DOE recommendation of a new equivalence unit for TRU waste, which can be used as the fundamental criterion for disposal of TRU waste. This is based upon the same acceptable level of risk that was used for spent fuel and HLW, and upon the same concept of a reference-size repository. The DOE intends that this recommendation be a part of all options for demonstrating compliance.

- Chapter 8 discusses the propagation of uncertainty as it relates to demonstration of compliance for different time periods. These discussions support the DOE recommendation that the time period for individual and groundwater protection be limited to 1,000 years after disposal, as it was in the 1985 standard. Furthermore, the discussions in Chapter 8 support the recommendation that assessments of cumulative radionuclide releases or collective doses should not be required for time periods greater than 10,000 years or, in the case of individual doses, time periods greater than 1,000 years.

- Chapter 9 describes the DOE recommendation for dealing with releases of radionuclides in gaseous form, with special focus on Carbon-14. In order to be consistent with the manner in which the EPA regulates similar releases from other facilities, the DOE recommends that gaseous releases from a repository be governed by the limits established in 40 CFR Part 191 for individual protection, with some modifications. This recommendation was developed in conjunction with the recommendations for containment, individual protection, and groundwater protection. The DOE intends that this recommendation be considered in conjunction with any revision of the requirements that govern those three topics.

The DOE intends that these changes be considered as a whole, since they are interrelated. To assist the EPA in this, the rest of this chapter presents a consolidation of all the changes. For the most part, the changes refer to the 1985 standard. However, there are several instances where reference is made to some provisions being considered by the EPA that are contained in Draft Federal Register Notice, dated 2/3/92.

Table 2-1. Comparison of Present Single Generic Release Limits and Alternatives

Characteristic	Present Single Generic Release Limits	Alternative		
		Multimode Generic Release Limits	Collective Dose Standard (without release limit option)	Collective Dose Option (with release limit option)
Uniform Biosphere	Yes	Yes	Only if standard biosphere specified	Only if standard biosphere specified
Uses Appropriate Release Modes	No	Yes	Yes	Yes
Uniform Assessment of All Repositories and Pathways	No	Yes	Yes	Yes
All Repository Components in Evaluations	No	Yes	Yes	Yes
Inaccuracies Due to Generic Derivations	Major	Minor	None	None
Corrections for Repository Locations	No	Yes	Yes	Yes
Traceable to Fundamental Criteria	No	Yes	Yes	Yes
Site Specific	No, but risk nonuniform	No, with nearly uniform risk	No	No
Additional Site Characterization	No	Moderate	Extensive	None to Extensive
Compatible with 191 Format	Yes	Yes	Yes	Yes
Philosophy Change	No	No	Extensive	Moderate
PA Change	No	Moderate	Extensive	None to Extensive
Status	Complete	Minor derivations	Minor derivations	Minor derivations

2.2 RECOMMENDED CHANGES

The changes below reflect an outline for Subpart B of 40 CFR Part 191 that is similar to the 1985 standard, with some modification of the appendices. Other outline changes being considered, as reflected in the Draft Federal Register Notice (2/3/92), are not addressed here. To assist the reader in understanding the recommended changes, the modified outline is shown below:

Subpart B - Environmental Standards for Disposal

191.11 Applicability
191.12 Definitions.
191.13 Containment requirements.
191.14 Assurance requirements.
191.15 Individual protection requirements.
191.16 Groundwater protection requirements.
191.17 Alternative provisions for disposal.
191.18 Effective date.
Appendix A Table for Subpart B
Appendix B Alternative Tables for Subpart B
Appendix C Calculation of Annual Committed Effective Dose
Appendix D Guidance for Implementation of Subpart B

The following new definitions should be added to Section 191.12, Definitions:

"Point of compliance" means the location, for a given release mode, where radionuclides enter the biosphere. At this location, cumulative releases over 10,000 years are calculated for comparison to the multimode release limits table. In calculating cumulative releases over 10,000 years, the points of compliance are as follows:

Release Mode	Point of Compliance
Land	Location where radioactive material released from the repository is brought directly to the land surface.
Well	Any wellhead outside the controlled area from which groundwater containing radionuclides released from the repository is withdrawn for irrigation or supplying drinking water.
River	Location(s) of existing discharge of groundwater containing radionuclides released from the repository to a river.
Ocean	Location where river-water or groundwater containing radionuclides released from the repository discharges to an ocean.

"Release mode" means one of four potential ways in which radionuclides are transported from the lithosphere to the biosphere, resulting in exposure to humans. The release modes are: land (contaminated solids deposited on the land surface, such as volcanic materials); well (contaminated groundwater pumped to the land surface); river (all fresh surface waters); and ocean.

"Biosphere" means the zone of the Earth extending from (and including) the surface into the surrounding atmosphere.

Section 191.13, Containment requirements, should be revised to read as follows:

191.13 Containment requirements.

The Department shall demonstrate compliance with either subsection (a) or (b) of this section. If subsection (a) is chosen, the Department may select either of the two methods of release calculations permitted for compliance demonstration.

(a) Disposal systems for radioactive waste shall be designed to provide a reasonable expectation that the cumulative releases of radionuclides in the solid or liquid phases, calculated by performance assessments either to the accessible environment (for Table 1 in Appendix A) or to the biosphere through all applicable release modes (for Tables 2 and 3 in Appendix B), for 10,000 years after disposal from all significant natural processes and events that may affect the disposal system shall:

(1) Have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix A) or Tables 2 and 3 (Appendix B); and

(2) Have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 (Appendix A) or Tables 2 and 3 (Appendix B); or

(b) Disposal systems for radioactive waste shall be designed to provide a reasonable expectation, based upon performance assessments, that the collective (population) effective dose, calculated using the weighing factors in Appendix C, caused by releases of radionuclides in the solid or liquid phases to the accessible environment for 10,000 years after disposal from all significant natural processes and events that may affect the disposal system shall:

(1) Have a likelihood of less than one chance in 10 of exceeding 2.5 million person-rem (25,000 person-sieverts); and

(2) Have a likelihood of less than one chance in 1,000 of exceeding 25 million person-rem (250,000 person-sieverts).

Dose limits are based upon a repository containing the equivalent of 100,000 MTHM of spent nuclear fuel and high-level waste or 20 MCi of transuranic waste.

(c) Potential radionuclide releases to the accessible environment resulting from human-initiated events and processes shall be treated separately from potential radionuclide releases due to natural processes and events. Disposal systems for spent nuclear fuel or for high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from intermittent and inadvertent exploratory drilling for resources into the disposal system shall not exceed ten times the quantities calculated

according to Table 1 (Appendix A) or Tables 2 or 3 (Appendix B). The performance assessments on which this expectation shall be based shall assume that drilling occurs. The assessments shall also assume that drilling technology, reasons for drilling, and societal structure remain the same as are present today. No human-initiated events and processes, due to the occurrence of drilling, which have a probability of occurrence less than one chance in 1,000 over 10,000 years shall be considered in the assessment.

(d) {the paragraph designated (b) in the 1985 standard} Performance assessments need not provide complete assurance that the requirements of 191.13(a) or (b) will be met. . .that compliance with 191.13(a) or (b) will be achieved.

The "three-bucket approach" alternative for the containment requirements, as proposed in Sections 191.12(x) and (y) of the Draft Federal Register Notice (2/3/92), should be included in the proposed rule.

Section 191.15, Individual protection requirements, should be revised to read as follows:

191.15 Individual protection requirements.

a) Disposal systems for radioactive waste shall be designed to provide a reasonable expectation that, for 1,000 years after disposal, undisturbed performance of the disposal system shall not cause the annual committed effective dose received through all potential pathways from the disposal system to any member of the public in the accessible environment to exceed 25 millirems (250 microsieverts). The annual committed effective dose for gases released through the atmospheric pathway shall not exceed 10 millirems.

The time period for assessments of individual and groundwater protection should be no more than 1,000 years after disposal (as in Sections 191.15 and 191.16 of the 1985 standard), rather than 10,000 years (as proposed in Sections 191.14 and 191.23 of the Draft Federal Register Notice of 2/3/92).

The revised standard should not include requirements for projection of potential releases, collective doses, or individual doses out to 100,000 years after disposal because of the increase in uncertainty, as proposed in Sections 191.12(c) and 191.14(b) of the Draft Federal Register Notice (2/3/92).

Appendix A should be revised to reflect the change in the reference size repository (from 10^3 to 10^5 MTHM) and the new TRU waste unit (20 MCi).

Appendix A: Table for Subpart B

TABLE 1 - RELEASE LIMITS FOR CONTAINMENT REQUIREMENTS
{See Table 1 at end of chapter}

Application of Table 1

Note 1: *Units of Waste*. The Release Limits in Table 1 apply to the amount of wastes in any one of the following:

(a) An amount of spent nuclear fuel containing 100,000 metric tons of heavy metal (MTHM) exposed to a burnup between 25,000 megawatt-days per metric ton of heavy metal (MWd/MTHM) and 40,000 MWd/MTHM;

(b) The high-level radioactive wastes generated from reprocessing each 100,000 MTHM exposed to a burnup between 25,000 MWd/MTHM and 40,000 MWd/MTHM;

(c) Each 10,000,000,000 curies of gamma or beta-emitting radionuclides with half-lives greater than 20 years but less than 100 years (for use as discussed in Note 5 or with materials that are identified by the Commission as high-level radioactive waste in accordance with part B of the definition of high-level waste in the NWPA);

(d) Each 100,000,000 curies of other radionuclides (i.e., gamma or beta-emitters with half-lives greater than 100 years or any alpha-emitters with half-lives greater than 20 years) (for use as discussed in Note 5 or with materials that are identified by the Commission as high-level radioactive waste in accordance with part B of the high-level waste in the NWPA); or

(e) An amount of transuranic (TRU) wastes containing twenty million curies of radionuclides.

Note 2: *Release Limits for Specific Disposal Systems*. To develop Release Limits for a particular disposal system, the quantities in Table 1 shall be adjusted for the amount of waste included in the disposal system compared to the various units of waste defined in Note 1. For example:

(a) If a particular disposal system contained the high-level wastes from 50,000 MTHM, the Release Limits for that system would be the quantities in Table 1 multiplied by .5 (50,000 MTHM divided by 100,000 MTHM).

(b) If a particular disposal system contained two million curies of alpha-emitting transuranic wastes, the Release Limits for that system would be the quantities in Table 1 multiplied by .1 (two million curies divided by twenty million curies).

(c) If a particular disposal system contained both the high-level wastes from 50,000 MTHM *and* 2 million curies of alpha-emitting transuranic wastes, the Release Limits for that system would be the quantities in Table 1 multiplied by .6:

$$\frac{50{,}000 \text{ MTHM}}{100{,}000 \text{ MTHM}} + \frac{2{,}000{,}000 \text{ curies TRU}}{20{,}000{,}000 \text{ curies TRU}} = .6$$

Note 3: {same as 1985 standard}
Note 4: {same as 1985 standard}
Note 5: {same as 1985 standard}
Note 6: {same as 1985 standard}

A new Appendix B, similar to Appendix A, should be created as follows:

Appendix B - Alternative Multimode Tables for Subpart B

TABLE 2 - CUMULATIVE RELEASE LIMITS FOR 10,000 YEARS FOR MULTIPLE RELEASE MODES (CURIES)
{See Table 2 at end of chapter}

TABLE 3 - CUMULATIVE RELEASE LIMITS FOR 10,000 YEARS FOR MULTIPLE RELEASE MODES (BEQUERELS)
{See Table 3 at end of chapter}

Application of Tables 2 and 3

Note 1: {same as in Appendix A}
Note 2: {same as in Appendix A}
Note 3: {same as in Appendix A}
Note 4: {same as in Appendix A}
Note 5: {same as in Appendix A}

Note 6: *Use of Site Adjustment Factors*. The Agency assumed, in deriving the release limits for the river and well releases in Tables 2 and 3, that the entire drainage system of all rivers (for river releases) and all aquifers (for well releases) is contaminated by the released radionuclides. Site Adjustment Factors (SAFs) should be used with Tables 2 and 3 to account for specific site locations. The following are examples of how SAFs might be developed for the surface flow system and other geologic and hydrologic components of a geologic disposal system.

Example 1--River Releases: For the river column, the release limits are calculated assuming that the entire drainage of all rivers is contaminated. For an actual site, only the downstream section of the tributary that is fed by groundwater passing through the repository is contaminated. To correct for this, a Site Adjustment Factor for the river release mode (SAF_R) is used as a multiplier to adjust the risk factors. The Reciprocal Site Adjustment Factor ($RSAF_R$), with which the release limits are multiplied, is calculated as follows:

$$RSAF_R = \frac{\sum_{i=1}^{n} (L_{C(i)} * F_{C(i)}) + \sum_{j=1}^{n} (L_{U(j)} * F_{U(j)})}{\sum_{i=1}^{n} (L_{C(i)} * F_{C(i)})}$$

This approximation represents the sums of the products of all tributary lengths and flow rates divided by the equivalent sums of contaminated tributaries. "L" is the length of the river segments and "F" is the volumetric flow rate of that segment. The subscripts "C" and

"U" refer to contaminated and uncontaminated segments, respectively. The release limits in Tables 2 and 3 are then multiplied by this ratio to provide a site-specific release limit for the river release mode.

Example 2--Well Releases: The derivation of the release limits for the well release mode using world average parameters assumes all groundwater from the recharge area to the locations where it enters surface waters is contaminated. For an actual site, wells up-gradient of the repository do not produce contaminated water. In addition, during the 10,000-year regulation period, the contaminated plume may not reach the discharge location, thus some uncontaminated water may also be withdrawn down-gradient from the repository.

A method for approximating the ratio of contaminated to total available water can be applied by dating the water at the repository (A_1), at the point it is expected that the radionuclides will reach in 10,000 years (A_2), and at the location where groundwater discharges to a river (A_3). With these ages, the Site Adjustment Factor for the well release mode (SAF_W) may then be calculated and used as a multiplier to adjust the risk factors. Calculation of the Reciprocal Site Adjustment Factor ($RSAF_W$) is done by dividing the age of the water at the river by the difference in the ages of the water at the repository and at the farthest point of migration in 10,000 years, or:

$$RSAF_W = \frac{A_3}{A_2 - A_1}$$

However, if it is found that the contaminated plume will reach a river within 10,000 years the formula becomes:

$$RSAF_W = \frac{A_3}{A_3 - A_1}$$

Release limits in Tables 2 and 3 are then multiplied by one of these ratios (the $RSAF_W$s) to provide a site specific release limit for the well release mode. The use of SAFs and the parameters to be considered in calculating SAFs shall be determined by the Department.

Note 7: *Points of Compliance*. In calculating cumulative releases over 10,000 years, the points of compliance are as follows:

Release Mode	Point of Compliance
Land	Location where radioactive material released from the repository is brought directly to the land surface.
Well	Any wellhead outside the controlled area from which groundwater containing radionuclides released from the repository is withdrawn for irrigation or supplying drinking water.
River	Location(s) of existing discharge of groundwater containing radionuclides released from the repository to a river.
Ocean	Location where river-water or groundwater containing radionuclides released from the repository discharges to an ocean.

Note 8: *Uses of Release Limits to Determine Compliance with 191.13*. Once release limits for a particular disposal system have been determined in accordance with Notes 1 through 7, these release limits shall be used to determine compliance with the requirements of 191.13 as follows. In cases where a mixture of radionuclides is projected to be released to the accessible environment, the limiting values shall be determined as follows: For each radionuclide in the mixture, determine the ratio between the cumulative release quantity projected over 10,000 years and the limit for that radionuclide for each applicable release mode as determined from Tables 2 or 3 and Notes 1 through 7. The sum of such ratios for all the radionuclides in the mixture may not exceed one with regard to 191.13(a)(1) and may not exceed ten with regard to 191.13(a)(2).

For example, if all release modes (L,W,R, and O referring to land, well, river, and ocean release modes) are used in the example, if radionuclides a and b are projected to be released in amounts Q_a and Q_b, and if the applicable release limits are RL_a and RL_b, then the cumulative releases over 10,000 years shall be limited so that the following relationship exists:

$$Q_{L,a}/RL_{L,a} + Q_{L,b}/RL_{L,b} + \ldots + Q_{W,a}/RL_{W,a} + Q_{W,b}/RL_{W,b} + \ldots +$$

$$Q_{R,a}/RL_{R,a} + Q_{R,b}/RL_{R,b} + \ldots + Q_{O,a}/RL_{O,a} + Q_{O,b}/RL_{O,b} + \ldots +$$

$$Q_{O,n}/RL_{O,n} < 1.$$

A new Appendix C, Calculation of Annual Committed Effective Dose, should be created. This Appendix could contain the information that was in Appendix B of the Draft Federal Register Notice (2/3/92). However, the information in that Appendix, which is based on ICRP 60, has yet to be fully accepted by the United States. Consideration should be given to returning to the information contained in Appendix A of Working Draft 3 (4/25/91) until ICRP 60 has been accepted.

The existing Appendix B from the 1985 standard should be renamed Appendix D. The following should be inserted between the second and third sentences of the first paragraph:

> Quantitative evaluations for these predictions compare predicted releases with either Table 1 of Appendix A or Tables 2 and 3 of Appendix B. If the multimode release limits in Tables 2 and 3 of Appendix B are used, the presence or absence of the four possible release modes (land, well, river, and ocean) to be considered in the containment requirements must be determined. The fifth release mode, for atmospheric releases, is considered under the individual protection requirements. Site Adjustment Factors for the well and river release modes, to be determined by the Department, may be calculated to account for differences between the actual site-specific availability of water and the original assumption that the entire drainage system is available and contaminated.

The following paragraph in the renamed Appendix D should be revised to read as follows:

> *Compliance with Section 191.13.* The Agency assumes that, whenever practical, the Department. . .compliance with 191.13(a) or (b) into a "complementary cumulative distribution function" that. . .for each disposal system considered. Section 191.13 contains options for comparing results of performance assessments with release limits and dose limits. The complementary cumulative distribution function may represent both summed release fractions and summed dose fractions. It is appropriate to apply dose standards to specific events or processes for which the release limits are inappropriate. The predicted doses for each event may then be normalized relative to the dose limits set by the Agency in the same manner as predicted releases. The dose fraction then replaces the summed release fraction for that event in the complementary cumulative distribution function. The Agency assumes that. . .this single distribution meets the requirements of 191.13(a) or (b).

The following paragraph should be added to the renamed Appendix D. This discussion of "future states" provides the Department with a means of addressing some of the uncertainties that could result from predicting conditions 10,000 years into the future:

> *Future States.* Uncertainties about the future involving conditions that are unknowable can only be dealt with by making assumptions and recognizing that these may, or may not, correspond to a future reality. The Agency believes that speculation concerning future conditions should not be the focus of the compliance-determination process. Therefore, it would be appropriate for assessments made for Part 191 to proceed under the assumption that many future conditions related to humans or to interactions between humans and their environment will remain the same as those of today's world. Factors in this

category include human physiology and nutritional needs, level of knowledge and technical capability, the state of medical knowledge, societal structure and behavior, patterns of water use, and pathways through the accessible environment that are affected by or result from human interactions with the accessible environment. In some instances, consideration of these factors may be specific to the region in which a disposal site is located (e.g., population distributions or patterns of water and land use). In contrast, the Agency would not find it appropriate to include in this category the future states of geologic, hydrologic, and climatic conditions that may be estimated by examining the geologic record. Additionally, the Agency would find inappropriate the assumption that national or world populations will remain unchanged; however, assuming future world populations that cannot reasonably be sustained by current abilities to produce, distribute, and consume food would also be inappropriate. For this reason, future world populations in excess of 10 billion people need not be assumed for evaluations under 191.13. For standardization, a "reference person" is assumed to ventilate (breathe) at a rate of _____ m^3/sec and to ingest _____ liters/day of drinking water; _____ kg/day of fish; _____ kg/day of mollusks; _____ kg/day of aquatic invertebrates; _____ kg/day of water plants; _____ kg/day of leafy vegetables; _____ kg/day of root vegetables; _____ kg/day of grains; _____ kg/day of fruit; _____ kg/day of meat; _____ kg/day of poultry; _____ kg/day of eggs; and _____ liters/day of milk.

Some standardization of current conditions unrelated to particular sites can be attained by providing parameters for a "reference person." A physiological model of "reference man" is available from the International Commission on Radiological Protection (see attached table) [ICRP 23, 1975]. Values for other parameters need to be determined. In addition, the Nuclear Energy Agency initiated a BIOsphere MOdel Validation Study (BIOMOVS) in 1985. The first phase of the study examined environmental assessment models for selected contaminants and exposure scenarios. The second phase of the study, which began in 1991, has as one of its objectives the development of a reference biosphere model that could be used in performance assessments of radioactive waste repositories. Although this phase is not complete, preliminary results of the study may provide an additional means for standardizing current conditions that could be used as guidance for future states. The provisional reference biosphere(s) should be formulated by October 1992, but the guidance for using the reference biosphere(s) is not expected until 1996.

The following paragraphs in the renamed Appendix D should be revised to read as follows:

Consideration of Inadvertent Human Intrusion into Geologic Repositories. The most speculative potential disruptions of a mined geologic repository are those associated with inadvertent human intrusion. Some types of intrusion would have virtually no effect on a repository's containment of waste. On the other hand, it is possible through speculation to conceive of intrusions (involving widespread societal loss of knowledge regarding radioactive wastes) that could result in major disruptions that no reasonable repository selection or design precautions could alleviate.

Neither the Agency nor any other regulatory body has identified a reliable, defensible basis for predicting future human behavior and for estimating the probabilities of possible

human actions. Therefore, the Agency does not require an estimate of the probabilities that various human actions will affect a repository. Nevertheless, the implementing agencies are required to consider these actions in making their determination that there is reasonable expectation of compliance with the standard. Instead of estimating the probability of drilling, it shall be assumed that drilling occurs and the consequences of such drilling estimated. These assessments may be supplemented by a description of the natural and engineered features of the disposal system that reduce the likelihood and consequences of human intrusion. The Agency believes that the most productive consideration of inadvertent intrusion concerns those realistic possibilities that may be usefully mitigated by repository design, site selection, or use of passive controls (although passive institutional controls should not be assumed to completely rule out the possibility of intrusion). In calculating the consequences of drilling, the implementing agencies can assume that passive institutional controls or the intruders' own exploratory procedures are adequate for the intruders to soon detect, or be warned of, the incompatibility of the area with their activities.

Frequency and Severity of Inadvertent Human Intrusion into Geologic Repositories by Exploratory Drilling. In the calculations supplied in compliance with paragraph 191.13(c), the implementing agencies need not assume intrusion scenarios more severe than inadvertent and intermittent intrusion by exploratory drilling for resources. The implementing agency need not assume any drilling for the resources that are provided by the disposal system itself. The implementing agencies should describe qualitatively the effects of each particular disposal system's site, design, and passive institutional controls in mitigating the potential effects of such inadvertent exploratory drilling. Descriptions of such inadvertent and intermittent exploratory drilling over 10,000 years need not assume that more than 30 boreholes per square kilometer of repository area will be drilled in that time at geologic repositories in proximity to sedimentary rock formations or that more than 3 boreholes per square kilometer will be drilled in that time at repositories in other geologic formations. Furthermore, when the discussions treat the consequences of inadvertent and intermittent exploratory drilling, the implementing agency need not assume that those consequences are more severe than (1) direct release to the land surface. . .the permeability of a carefully sealed borehole.

TABLE 1 - RELEASE LIMITS FOR CONTAINMENT REQUIREMENTS
[Cumulative releases to the accessible environment for 10,000 years after disposal]

Radionuclide	Release limit per 100,000 MTHM or other unit of waste (see notes) (curies)
Americium-241 or -243	10,000
Carbon-14	10,000
Cesium-135 or -137	100,000
Iodine-129	10,000
Neptunium-237	10,000
Plutonium-238, -239, -240, or -242	10,000
Radium-226	10,000
Strontium-90	100,000
Technetium-99	1,000,000
Thorium-230 or -232	1,000
Tin-126	100,000
Uranium-233, -234, -235, -236, or -238	10,000
Any other alpha-emitting radionuclide with a half-life greater than 20 years	10,000
Any other radionuclide with a half-life greater than 20 years that does not emit alpha particles	100,000

Table 2. Cumulative Release Limits for 10,000 years (curies per 100,000 MTHM) for Multiple Release Modes

Release Limit (curies per 100,000 MTHM)				
Nuclide	River	Well	Ocean	Land
C-14	TBD[a]	TBD	TBD	TBD
Ni-59	2E+07	9E+06	TBD	1E+09
Sr-90	4E+04	2E+04	4E+07	3E+07
Zr-93	7E+06	3E+06	3E+07	4E+07
Tc-99	3E+06	1E+06	6E+08	2E+10
Sn-126	1E+04	4E+03	9E+03	7E+05
I-129	1E+04	5E+03	4E+06	3E+05
Cs-135	1E+05	6E+04	2E+07	2E+06
Cs-137	9E+04	8E+04	2E+06	5E+07
Sm-151	1E+08	4E+07	TBD	1E+10
Pb-210	8E+03	4E+03	TBD	7E+06
Ra-226	6E+03	3E+03	TBD	2E+05
Ra-228	4E+04	2E+04	TBD	6E+07
Ac-227	1E+04	6E+03	7E+03	8E+06
Th-229	3E+04	1E+04	6E+03	5E+04
Th-230	2E+03	8E+02	TBD	3E+03
Th-232	3E+03	1E+03	TBD	3E+03
Pa-231	7E+03	3E+03	2E+04	4E+04
U-233	5E+04	2E+04	1E+06	1E+06
U-234	5E+04	2E+04	TBD	2E+06
U-235	5E+04	2E+04	1E+06	1E+06
U-236	5E+04	2E+04	TBD	2E+06
U-238	5E+04	2E+04	TBD	1E+06
Np-237	1E+04	8E+03	7E+04	8E+06
Pu-238	2E+04	1E+04	TBD	3E+06
Pu-239	2E+04	8E+03	2E+04	2E+05
Pu-240	2E+04	8E+03	2E+04	2E+05
Pu-241	5E+05	2E+05	TBD	4E+08
Pu-242	2E+04	8E+03	TBD	2E+05
Am-241	2E+04	8E+03	5E+03	1E+06
Am-243	2E+04	8E+03	5E+03	4E+05
Cm-245	1E+04	4E+03	3E+03	1E+05
Cm-246	2E+04	8E+03	TBD	3E+05

[a] To be determined

Table 3. Cumulative Release Limits for 10,000 years (TBq per 100,000 MTHM) for Multiple Release Modes

Release Limit (TBq per 100,000 MTHM)				
Nuclide	River	Well	Ocean	Land
C-14	TBD*	TBD	TBD	TBD
Ni-59	8E+05	3E+05	TBD	5E+07
Sr-90	2E+03	7E+02	2E+06	1E+06
Zr-93	2E+05	1E+05	9E+05	2E+06
Tc-99	1E+05	4E+04	2E+07	7E+08
Sn-126	4E+02	1E+02	3E+02	3E+04
I-129	5E+02	2E+02	1E+05	9E+03
Cs-135	5E+03	2E+03	6E+05	6E+04
Cs-137	3E+03	3E+03	8E+04	2E+06
Sm-151	4E+06	2E+06	TBD	6E+08
Pb-210	3E+02	1E+02	TBD	2E+05
Ra-226	2E+02	1E+02	TBD	7E+03
Ra-228	2E+03	7E+02	TBD	2E+06
Ac-227	6E+02	2E+02	2E+02	3E+05
Th-229	1E+03	4E+02	2E+02	2E+03
Th-230	7E+01	3E+01	TBD	1E+02
Th-232	1E+02	4E+01	TBD	1E+02
Pa-231	3E+02	1E+02	6E+02	2E+03
U-233	2E+03	7E+02	4E+04	5E+04
U-234	2E+03	8E+02	TBD	6E+04
U-235	2E+03	7E+02	4E+04	4E+04
U-236	2E+03	8E+02	TBD	6E+04
U-238	2E+03	7E+02	TBD	5E+04
Np-237	5E+02	3E+02	3E+03	3E+05
Pu-238	9E+02	4E+02	TBD	1E+05
Pu-239	7E+02	3E+02	6E+02	6E+03
Pu-240	8E+02	3E+02	6E+02	7E+03
Pu-241	2E+04	7E+03	TBD	1E+07
Pu-242	8E+02	3E+02	TBD	6E+03
Am-241	7E+02	3E+02	2E+02	4E+04
Am-243	6E+02	3E+02	2E+02	2E+04
Cm-245	4E+02	2E+02	1E+02	5E+03
Cm-246	7E+02	3E+02	TBD	1E+04

* To be determined

EXAMPLE TABLE

REFERENCE MAN: SUMMARY OF PHYSIOLOGICAL DATA

		Page
Carbon dioxide exhaled	1000 g/day	340
Dietary intake (nutrients)		
Protein	95 g/day	351
Carbohydrates	390 g/day	351
Fat	120 g/day	351
Dietary intake (major elements)		
Carbon	300 g/day	352
Hydrogen	350 g/day	352
Nitrogen	16 g/day	352
Oxygen	2600 g/day	352
Sulfur	1 g/day	352
Elements (summary of model values for daily balance)	See section O	
Energy expenditure	3000 kcal/day	338
Feces, weight of	135 g/day	353
Feces, components of		
Water	105 g/day	353
Solids	30 g/day	353
Ash	17 g/day	353
Fats	5 g/day	353
Nitrogen	1.5 g/day	353
Other substances	6.5 g/day	353
Feces, major elements in		
Carbon	7 g/day	353
Hydrogen	13 g/day	353
Nitrogen	1.5 g/day	353
Oxygen	100 g/day	353
Human milk, composition of	See Table 128	361
Intake of milk	300 ml/day	357
Lung capacities		
Total capacity	5.6 l	345
Functional residual capacity	2.2 l	345
Vital capacity	4.3 l	345
Dead space	160 ml	345
Lung volume and respiration		
Minute volume, resting	7.5 l/min	346
Minute volume, light activity	20 l/min	346
Air breathed, 8 h light work activity	9600 l	346
Air breathed, 8 h nonoccupational activity	9600 l	346
Air breathed, 8 h resting	3600 l	346

Metabolic rate	17 cal/min-kg W	341
Nasal secretion, composition of (major elements)		
Water	95-97 g/100 ml	365
Calcium	11 g/100 ml	365
Chlorine	495 g/100 ml	365
Potassium	69 g/100 ml	365
Sodium	295 g/100 ml	365
Oxygen Inhaled	920 g/day	340
Saliva, composition of	See Table 130	364
Sweat, composition of	See Table 129	362
Urine values		
Volume	1400 ml/day	354
Specific gravity	1.02	354
pH	6.2	354
Solids	60 g/day	354
Urea	22 g/day	354
"Sugars"	1 g/day	354
Bicarbonates	0.14 g/day	354
Urinary loss of major elements		
Nitrogen	15 g/day	354
Hydrogen	160 g/day	354
Oxygen	1300 g/day	354
Carbon	5 g/day	354
Water balance (gains)		
Total fluid intake	1950 ml/day	360
Milk	300 ml/day	360
Tap water	150 ml/day	360
Other	1500 ml/day	360
In food	700 ml/day	360
By oxidation of food	350 ml/day	360
Total	3000 ml/day	360
Water balance (losses)		
In urine	1400 ml/day	360
In feces	100 ml/day	360
Insensible loss	850 ml/day	360
In sweat	650 ml/day	360
Total	3000 ml/day	360

[1]All sections reference ICRP 23, 1975

CHAPTER 3

HUMAN INTRUSION

CHAPTER 3

HUMAN INTRUSION

3.1 STATEMENT OF THE PROBLEM

In the 1985 EPA standard, processes and events initiated by human actions are treated in much the same way as naturally occurring processes and events. That is, the consequences of human actions must be included in the calculations that examine compliance with the numerical, probabilistic containment requirements. This provision creates difficulties that arise because it forces a demonstration of compliance to estimate the probabilities and the consequences of human-initiated phenomena that may occur during the next 10,000 years. There is no reliable basis for estimating human behavior over so long a period. Consequently, assumptions about the human activities that may occur at a repository site and about their probabilities are difficult to defend, because they lack a firm technical foundation. An analysis of compliance may well be so heavily dominated by such assumptions that it fails to reveal the adequacy, or inadequacy, of the isolation characteristics offered by a repository site. Speculation about future human activity should therefore not be the focus of the compliance determination process.

On the other hand, the human-initiated events and processes should not be ignored in that process. They clearly should be part of an evaluation of the adequacy of a proposed repository system. The problem, then, is to construct and propose a treatment of such phenomena that guarantees their consideration in determining compliance but does not skew the process toward rejection of adequate sites on the basis of indefensible assumptions.

3.2 RECOMMENDED APPROACH

The following material suggests a way that section 191.13 of the 1985 version of 40 CFR Part 191 might be written to avoid the problems with putting human intrusion into the quantitative, probabilistic comparison with limits. The same material, perhaps with minor changes, may be used if the standard also allows for alternative approaches to the demonstration of compliance.

191.13 Containment requirements.

a) Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation, based on performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant natural processes and events that may affect the disposal system shall:

   1. Have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix A); and

   2. Have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 (Appendix A).

b) Potential radionuclide releases to the accessible environment resulting from human-initiated events and processes shall be treated separately from potential radionuclide releases due to natural processes and events. Disposal systems for spent nuclear fuel or for high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from intermittent and inadvertent exploratory drilling for resources into the disposal system shall not exceed ten times the quantities calculated according to Table 1 (Appendix A). The performance assessments on which this expectation shall be based shall assume that drilling occurs. The assessments shall also assume that drilling technology, reasons for drilling, and societal structure remain the same as are present today. No human-initiated events and processes, due to the occurrence of drilling, which have a probability of occurrence less than one chance in 1,000 over 10,000 years shall be considered in the assessment.

c) {the paragraph designated (b) in the 1985 version, unchanged} Performance assessments need not provide complete assurance that the requirements . . .

If the EPA includes in its next version of the standard some alternatives to the original section 191.13, (e.g., the "four-column" approach or either of the two optional containment requirements suggested in the draft Federal Register notice dated 2/3/92), similar changes should be made.

The following paragraph is to be added to Appendix B of the 1985 version:

Future States. Uncertainties about the future involving conditions that are unknowable can only be dealt with by making assumptions and recognizing that these may, or may

not, correspond to a future reality. The Agency believes that speculation concerning future conditions should not be the focus of the compliance-determination process. Therefore, it would be appropriate for assessments made for Part 191 to proceed under the assumption that many future conditions related to humans or to interactions between humans and their environment will remain the same as those of today's world. Factors in this category include human physiology and nutritional needs, level of knowledge and technical capability, the state of medical knowledge, societal structural and behavior, patterns of water use, and pathways through the accessible environment that are affected by or result from human interactions with the accessible environment. In some instances, consideration of these factors may be specific to the region in which a disposal site is located (e.g., population distributions or patterns of water and land use). In contrast, the Agency would not find it appropriate to include in this category the future states of geologic, hydrologic, and climatic conditions that may be estimated by examining the geologic record. Additionally, the Agency would find inappropriate the assumption that national or world populations will remain unchanged; however, assuming future world populations that cannot reasonably be sustained by current abilities to produce, distribute, and consume food would also be inappropriate. For this reason, future world populations in excess of 10 billion people need not be assumed for evaluations under 191.13. For standardization, a "reference person" is assumed to ventilate (breathe) at a rate of _____ m^3/sec and to ingest _____ liters/day of drinking water; _____ kg/day of fish; _____ kg/day of mollusks; _____ kg/day of aquatic invertebrates; _____ kg/day of water plants; _____ kg/day of leafy vegetables; _____ kg/day of root vegetables; _____ kg/day of grains; _____ kg/day of fruit; _____ kg/day of meat; _____ kg/day of poultry; _____ kg/day of eggs; and _____ liters/day of milk.

The above changes in paragraph 191.13 will require a change to the reference to 191.13 that appears in Appendix B of the 1985 version in the paragraph called *"Compliance with Section 191.13."* Two other references to 191.13 will not need to be changed. The revised paragraph will read as follows:

> The Agency assumes that . . . compliance with 191.13(a) into a "complementary cumulative distribution function" that indicates . . . a disposal system can be considered to be in compliance with 191.13 if this single distribution function meets the requirements of 191.13(a) and if the calculation of the consequences of exploratory drilling for resources required by 191.13(b) meets the requirements of 191.13(b).

Some sentences will need to be inserted into the paragraph in Appendix B called *"Consideration of Inadvertent Human Intrusion . . ."* This paragraph will then read as follows:

> *Consideration of Inadvertent Human Intrusion into Geologic Repositories.* The most speculative potential disruptions of a mined geologic repository are those associated with inadvertent human intrusion. Some types of intrusion would have virtually no effect on a repository's containment of waste. On the other hand, it is possible through speculation to conceive of intrusions (involving widespread societal loss of knowledge regarding radioactive wastes) that could result in major disruptions that no reasonable repository selection or design precautions could alleviate.

Neither the Agency nor any other regulatory body has identified a reliable, defensible basis for predicting future human behavior and for estimating the probabilities of possible human actions. Therefore, the Agency does not require an estimate of the probabilities that various human actions will affect a repository. Nevertheless, the implementing agencies are required to consider these actions in making their determination that there is reasonable expectation of compliance with the standard. Instead of estimating the probability of drilling, it shall be assumed that drilling occurs and the consequences of such drilling estimated. These assessments may be supplemented by a description of the natural and engineered features of the disposal system that reduce the likelihood and consequences of human intrusion. The Agency believes that the most productive consideration of inadvertent intrusion concerns those realistic possibilities that may be usefully mitigated by repository design, site selection, or use of passive controls (although passive institutional controls should not be assumed to completely rule out the possibility of intrusion). In calculating the consequences of drilling, the implementing agencies can assume that passive institutional controls or the intruders' own exploratory procedures are adequate for the intruders to soon detect, or be warned of, the incompatibility of the area with their activities.

The paragraph in Appendix B labeled *"Frequency and Severity of Inadvertent Human Intrusion . . .,"* is to be modified as follows (with the original wording continuing from the ellipsis at the end of this suggested wording):

Frequency and Severity of Inadvertent Human Intrusion into Geologic Repositories by Exploratory Drilling. In the calculations supplied in compliance with paragraph 191.13(b), the implementing agencies need not assume intrusion scenarios more severe than inadvertent and intermittent intrusion by exploratory drilling for resources. The implementing agencies need not assume any drilling for the resources that are provided by the disposal system itself. The implementing agencies should describe qualitatively the effects of each particular disposal system's site, design, and passive institutional controls in mitigating the potential effects of such inadvertent exploratory drilling. Descriptions of such inadvertent and intermittent exploratory drilling over 10,000 years need not assume that more than 30 boreholes per square kilometer of repository area will be drilled in that time at geologic repositories in proximity to sedimentary rock formations or that more than 3 boreholes per square kilometer will be drilled in that time at repositories in other geologic formations. Furthermore, when the discussions treat the consequences of inadvertent and intermittent exploratory drilling, the implementing agency need not assume that those consequences are more severe than: (1) direct release to the land surface . . .

3.3 SUPPLEMENTARY INFORMATION

The following material could be used as supplementary information in explaining why the rule is reasonable when written in the form on the preceding pages. This material could probably appear in the EPA's supplementary information just after its explanations of the probabilistic standard that is promulgated in paragraph 191.13(a).

> In developing the probabilistic standard, the Agency recognized that there is a fundamental difference between estimating the probabilities of future natural phenomena and estimating the probabilities of future human activities. Reasonable estimates of natural phenomena can often be based on evidence provided by the geologic record. Most of the natural phenomena that might be expected to affect a repository (e.g., fault movement, erosion, or diapirism) can be studied in records that extend back for millions of years. An extrapolation of that information through the next 10,000 years can be a reasonable basis for estimating the probabilities that those phenomena will occur. Although there will seldom be unanimous agreement among experts about the precise values of those probabilities, their reasonableness can be examined by reference to the geologic record. Believing that probabilities can be derived and defended in this way, the Agency deems appropriate the probabilistic standard required for natural phenomena in paragraph 191.13(a).
>
> On the other hand, there is no similarly reliable basis for estimating what human beings are likely to do in the next few thousand years, or even in the next few hundred years. The records of human activity are not nearly so long as the geologic record, and 10,000-year extrapolations would, for that reason alone, be less reliable than extrapolations from the geologic record. More important, the past few hundred years--the past few decades, in particular--have seen an enormous increase in the rates at which human societies and their associated technical abilities have changed. With such rapid changes in so short a time, extrapolation to 10,000 years would necessarily consist of speculation about whether these rates will continue. Neither the Agency nor other regulatory bodies have identified a reliable basis for such speculation, which the Agency consequently believes should not be the focus of the compliance-determination process.
>
> For these reasons, the Agency has not required a probabilistic treatment of human actions that may affect a repository. Nevertheless, the Agency believes that an implementing agency should carefully consider the effects of human actions in seeking reasonable expectation of compliance. Paragraph 191.13(b) therefore requires an evaluation of the consequences of exploratory drilling, which the Agency believes to be a reasonable representation of severe human-initiated phenomena that might affect a disposal system. The paragraph also requires that potential releases of radionuclides to the accessible environment, resulting from such intrusion, shall not exceed ten times the quantities in Table 1 of the rule. This limit is reasonable because, as originally developed, it applied to phenomena with likelihoods between 1 chance in 10 and 1 chance in 1000 over 10,000 years.

With this change in the rule, paragraph 191.13(b) avoids the problems associated with speculative, quantitative estimates that human intrusions will occur. It simply requires calculations made on the assumption that such intrusions do occur. It recognizes, however, that treating future human actions will require that some further assumptions be made. It guides these assumptions by stating further requirements that follow the Agency's more extensive guidance, in Appendix B, for the treatment of future states. The paragraph also recognizes that some phenomena occurring during and after the assumed intrusions occur stochastically. To keep from introducing speculation about phenomena of extremely low probability, the paragraph therefore limits the treatment of phenomena that occur during and after the assumed intrusions. The limitations are essentially the same as those applied to demonstrations of compliance under paragraph 191.13(a).

The requirement does not rule out the use of additional calculations that may produce useful insights into the future behavior of a repository system under intrusions by exploratory drilling. Further information about the Agency's intentions is furnished in Appendix B, which explains what the Agency would consider appropriate treatment of future states of nature and of human civilization.

3.4 TECHNICAL SUPPORT DOCUMENTATION

The following material is supporting information that could be cited as reasons for the DOE suggestions for the above revision. It could be part of a technical support document for the rule.

Many comments on 40 CFR Part 191 have pointed out the difficulties that arise when human activities are included with natural phenomena in the complementary cumulative distribution function (CCDF) that the Agency recommended in 1985 for examining compliance with paragraph 191.13(a). The difficulties also arise in alternative compliance methods that have been suggested for incorporation into the standard--i.e., the suggestions known as the "four-column" alternative, the collective-dose alternative, and the "three-bucket" alternative. Summarized broadly, these difficulties arise from the basic difficulty of guessing what future human societies will be able to do or will want to do. For example, to include the drilling of exploratory boreholes into a forgotten repository would require estimates of the consequences of the drilling and of the probability of its occurrence. Estimating the consequences would require speculation about how drilling would be done in the future. Given the rapid advances in drilling methods in the past hundred years, it would be extremely difficult to guess how drilling will be done thousands of years from now. Estimating the probability of drilling would be even more speculative; given that only 200 years ago deep drilling was a rare occurrence, it is hard to guess how often people will want to drill thousands of years from now.

Because there is no way to rigorously defend estimates of either the consequences or the probabilities of future human actions, the CCDF could easily be dominated by assumptions about these estimates. And there would be little possibility that the estimates could be limited to "reasonable" values, because there appears to be no defensible basis for deciding what will be "reasonable" in future societies.

A specific example of this possibility appears in a detailed preliminary performance assessment recently completed for the potential site at Yucca Mountain, Nevada (Reference 3-1). That study examined the effect of varying the number of boreholes that it assumed would penetrate the repository during the next 10,000 years. At the larger numbers of boreholes, the effects of natural release mechanisms (e.g., groundwater flow) were obscured by the effects of drilling. There was, of course, no basis other than assumption for choosing one number of boreholes over another--i.e., for deciding which CCDF is best representative of the site's future performance. (Although the EPA has provided suggestions that guide assumptions about numbers of boreholes, licensing activities are not bound to follow those suggestions, which appear in the guidelines that accompanied the 1985 version of the standard.) When CCDFs that include guesses about numbers of future boreholes are introduced into licensing activities, the licensing process may find itself focused on speculation about those numbers rather than on substantive issues of repository performance.

In other words, a CCDF dominated by guesses about future human behavior may obscure the more defensible estimates of the ability of a repository system to isolate waste through its natural characteristics and its engineered features. These characteristics and features are barriers on which geologic disposal relies, and it is important that the performance measure embodied in the standard reveal their effectiveness. The CCDF can do so if the obscuring effects of estimates about human actions are removed from it.

This line of reasoning suggests only that human actions should not be part of a standard that requires estimates of the probabilities of those actions. It would not be appropriate to eliminate human activities altogether from a determination that a repository system will isolate waste effectively. A simple way to remove the difficulties associated with estimating the probabilities of future human activities is to assume that the activities occur and to calculate their consequences on that assumption. This deterministic way of determining compliance must be supplemented, however, if the calculations are to be used alongside the probabilistic standard that governs natural phenomena and if they are to be kept from unconstrained speculation.

First, the human activities must be removed from paragraph 191.13(a) of the standard. That paragraph is built on the use of likelihoods as an integral part of the determination of compliance. Calculations that treat the likelihood of human intrusion deterministically could not be a part of that method. Because the Agency feels that calculations should be evaluated against a numerical standard, a limit on releases must, however, be established. A reasonable limit, which follows the reasoning behind the original release limits, would be 10 times the quantities in Table 1 of the current EPA standard. This limit is reasonable because, as originally developed, it applied to phenomena with likelihoods between 1 chance in 10 and 1 chance in 1000 over 10,000 years.

Second, the human activities must be constrained by rule. If the likelihoods of human-initiated intrusive activities are completely removed from consideration, there would be no restraint on what should be calculated. Clearly, a site with otherwise acceptable natural and engineered features should not be declared unacceptable simply because an unrealistic, highly improbable future human activity could inadvertently exhume some of the waste. For example, drilling on 2-foot centers would be an improbable future event that would probably exceed the release limits of any disposal system. It would be so improbable that it should not be part of a realistic appraisal of the system. But if its low probability of occurrence is ignored, an analysis of it would show releases that violate a standard that makes no allowance for likelihood. A reasonable way to constrain the human activities is to follow the EPA guidance that says exploratory drilling would be severe enough to adequately represent intrusive activities. Also reasonable would be the inclusion of the current EPA guidance on the number of boreholes that should be assumed for drilling in different types of rock. These constraints are compatible with the choice of a release limit 10 times the quantities in Table 1 of the original standard.

Third, the phenomena that occur after or during the assumed drilling must also be constrained. These phenomena occur stochastically for a number of reasons: e.g., natural variation in the properties of materials, randomness in natural processes, randomness in the location of exploratory boreholes, and uncertainties in data. Unless some constraints are placed on the likelihoods of these phenomena, an evaluation of releases could be dominated by speculative, highly unlikely events and processes. For example, the study of exploratory drilling described in Reference 3-1 used the guidance suggested by the EPA for number of boreholes (17 boreholes over 10,000 years); it also assumed today's drilling technology and methods. The study had to assume, however, a probability distribution for the times at which the boreholes were drilled and for the possibility that any particular borehole would actually penetrate a canister filled with radioactive waste. If performed iteratively for many thousands of times, a stochastic calculation like the one performed in Reference 3-1 would eventually produce, at extremely low probability, a simulation of a set of events in which each of the 17 boreholes penetrated a waste canister at an early time after the closure of a disposal system. Such a calculation would be unsuitable for

assessing the ability of the system to isolate waste, because the event it modeled would have extremely low likelihood, but the calculated releases would clearly violate the current limits.

The current standard avoids such unsuitable calculations by placing constraints on the natural events and processes that must be examined. These constraints may reasonably be applied to the events and processes that should be examined once an assumption is made that human intrusion occurs. Releases from phenomena with a likelihood of occurrence less than 1 chance in 1000 over 10,000 years are currently not compared with quantities stated in the standard. It is consistent with the original standard to accord the same treatment to the phenomena that occur during and after drilling into a disposal system. Paragraph 191.13(b) can therefore defensibly exclude such events and processes from comparison against the limits if their likelihoods of occurrence are less than 1 chance in 1000 over 10,000 years.

Little experience from other countries is available for guiding the U.S. development of the treatment for human intrusions. European nations have not come to consensus on an appropriate way to handle human intrusion in their analyses of waste isolation. They do, however, recognize that "such low-probability, high-consequence scenarios would be difficult to treat within the normal regulatory guidelines and might, therefore, need separate consideration . . . These issues will be treated within the NEA Working Group on Assessment of Future Human Actions . . ." (Reference 3-2). Because these nations do not currently plan to use a probabilistic standard like the EPA standard, the difficulties they perceive are somewhat different from those involved with including human intrusion in a CCDF. But they clearly intend to pay special attention to the problems of including human intrusion along with natural disruptions, even in nonprobabilistic assessments.

REFERENCES

3-1. Barnard, R. W., et al., "TSPA 1991: An Initial Total-System Performance Assessment for Yucca Mountain," SAND91-2795, Sandia National Laboratories, Albuquerque, New Mexico.

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CHAPTER 4

THREE-BUCKET APPROACH

CHAPTER 4

THREE-BUCKET APPROACH

4.1 STATEMENT OF THE PROBLEM

Agencies affected by 40 CFR Part 191 have experimented with the complementary cumulative distribution function (CCDF) that the 1985 rule suggests for demonstrating compliance with the containment requirements. The NRC came forward with an alternative approach in 1991 and offered it up for discussion in informal forums. The approach came to be known as the "three-bucket approach" because it attempts to divide into three categories the phenomena that might affect waste isolation. The EPA has informally circulated a somewhat modified version in the draft Federal Register notice (2/3/92). The DOE examined both the NRC and the EPA statements of the approach and some further statements by the NRC staff: material in a letter, dated July 1, 1992, from B. J. Youngblood (Director of the NRC Division of High-Level Waste Management) to J. W. Gunter (Director of the EPA Criteria and Standards Division) and in an informally circulated draft, dated October 10, 1991, giving examples of compliance demonstration. The DOE has also benefited from an informal technical exchange with the NRC staff (July 22, 1992) at which the three-bucket approach was discussed in some detail. A preliminary series of calculations done under contract to the DOE has suggested that the approach is not completely compatible with the DOE understanding of what will be needed for determining compliance and is not necessarily simpler to implement than the original standard. The problem, then, is to state the difficulties that the DOE sees in the "three-bucket approach."

4.2 RECOMMENDED APPROACH

The DOE finds that the "three-bucket approach," as it has been stated up to now, contains some difficulties that keep it from being a completely acceptable way to demonstrate compliance. The DOE would prefer to leave the approach as an option in a draft Federal Register notice intended to solicit comment on the revision of 40 CFR 191. Additional comment could help to clarify the difficulties that the DOE finds in the approach and might help to produce an acceptably simplified form of the original standard.

4.3 SUPPLEMENTARY INFORMATION

This material is not "supplementary information" in the sense that it is normally used in the rulemaking process. Instead, it simply explains, in brief summary form, the reasoning behind the DOE statement that the three-bucket approach may not be a useful alternative to the original EPA standard. The material in this section may be useful to the EPA if its next proposal for 40 CFR Part 191 is accompanied by supplementary information that explains the EPA position on the "three-bucket approach."

The analysis reported in the technical support documentation reveals some features of the three-bucket approach that appear to make it unacceptable, at least in its present form:

1. It is possible to construct some scenario classes for which the three-bucket approach and the original standard disagree about compliance. The analysis began by applying the three-bucket approach to the results of a recent total-system performance assessment of the potential Yucca Mountain site; the original assessment had already compared its results to the original standard. The comparison suggested that the two methods agree about compliance for those particular results. Nevertheless, when those results were modified slightly, the analysis showed that the two methods can easily disagree. Sometimes, depending on the particular modifications, the original standard is stricter; sometimes the three-bucket approach is stricter. This conclusion suggests that three-bucket approach is probably not completely compatible with the original standard: i.e., it does not yield the same conclusions about compliance. Whether it would nevertheless be acceptable to the regulatory community can probably be determined only after the community has examined the approach more thoroughly and has debated the acceptability of the apparent inconsistencies.

2. The three-bucket approach is sensitive to the way in which "scenarios" are defined as part of the compliance examination. The approach introduces the term "scenario" into the regulation and therefore requires that scenarios be used in the examination of compliance. The technical community does not appear to be in complete agreement about the role of scenarios in constructing complementary cumulative distribution functions, and that lack of agreement would be an obstacle to the implementation of the approach. More important, the analysis shows that compliance may, in at least some examples, be demonstrable when a disposal system is described by one set of scenarios, but not demonstrable when it is described by another set. It would be preferable for the standard to yield the same results about compliance regardless of the details of the definition of scenarios. Studies of how to implement the current standard have suggested that it is not necessarily sensitive to the details of the definitions. Because of the way the three-bucket approach treats scenarios, however, it probably cannot be made insensitive without fairly drastic revision.

3. The three-bucket approach adopts a bounding value for sequences of events and processes that have low likelihoods. This bounding value, 0.01, is applied in the analyses of all sequences whose likelihood is (a) great enough for the sequences to warrant regulatory consideration and (b) smaller than 0.01. The adoption of this bounding value can lead to an overemphasis of some low-probability sequences--an overemphasis that contributes to the disagreements discussed in item 1 above.

4. The examples that the NRC staff has used in its explanations of the three-bucket approach have assumed that it is possible to identify for each scenario a single associated value of radionuclide release. Although this assumption was made for didactic purposes and not because the staff felt that it will be appropriate in actual licensing, the simplification achieved by it appears to mask some of the difficulties with implementing the standard. The approach, as now stated, does not explain how to reduce the distribution of releases associated with realistic scenarios to simple, if not single, values of release.

Many of these difficulties may be avoided by further definitions within the three-bucket approach and by detailed guidance about how to apply the approach in licensing. The additional details that would be needed, however, appear to require efforts that would be approximately as complex as the effort needed to show compliance with the original standard. For example, to overcome the difficulty with associating a single value of release with each scenario class would probably require something like deriving a CCDF for each class--an effort that would not be a reduction below the efforts required by the current standard.

These points are derived in much greater detail in the accompanying technical support documentation. Although there may be solutions to the problems that the documentation raises, the three-bucket approach does not appear to necessarily offer less difficulty in implementation than the current standard. And it does pose potential problems of its own. Until these possibilities are sorted out, it would not be wise to adopt the three-bucket approach <u>in place of</u> the original EPA standard. At most, the three-bucket approach should be provided as an option for compliance demonstration.

4.4 TECHNICAL SUPPORT DOCUMENTATION

The technical analysis of the three-bucket approach in Appendix A is offered as support for the DOE recommendation that the approach not be taken as a replacement for the original EPA containment standard. It should be offered as an option in the draft Federal Register notice in hopes that additional review and analysis will provide answers to the Department's concerns.

CHAPTER 5

MULTIMODE RELEASE LIMITS

CHAPTER 5

MULTIMODE RELEASE LIMITS

5.1 STATEMENT OF THE PROBLEM

In some instances, the release requirements of Table 1 in 40 CFR Part 191 may result in an inappropriate or overly conservative evaluation of repository sites because they do not adequately account for significant features of a site. The 1985 version of 40 CFR Part 191 contained only one release limit table (Table 1) for all release modes. The table was based on simultaneous releases to all the world's rivers and oceans. The three other basic release modes--atmospheric, land surface, and withdrawal-well, which are the only expected release modes for sites presently under consideration--were not taken into account. Because a single release limit table cannot represent all release modes and release locations, cumulative releases would have been evaluated at the boundary of the repository instead of at locations of release.

5.2 RECOMMENDED APPROACH

A multimode release limit option is proposed in addition to the existing Table 1 limit in Appendix A of the standard. This additional option would include limits for all release modes to be considered in the containment requirements (land, well, river, and ocean). The atmospheric release mode is addressed in the individual protection requirements (as explained in Chapter 9, which discusses Carbon-14), and the human intrusion component is addressed in Chapter 3. In incorporating the proposed new table, a number of corresponding changes to the wording of the rule are needed. These changes are described below.

A number of new terms have been introduced. As used here, these terms are defined as follows:

Point of compliance - the location, for a given release mode, where radionuclides enter the biosphere. At this location, cumulative releases over 10,000 years are calculated for comparison to the multimode release limits table.

In calculating cumulative releases over 10,000 years, the points of compliance are as follows:

Release Mode	Point of Compliance
Land	Location where radioactive material released from the repository is brought directly to the land surface.
Well	Any wellhead outside the controlled area from which groundwater containing radionuclides released from the repository is withdrawn for irrigation or supplying drinking water.
River	Location(s) of existing discharge of groundwater containing radionuclides released from the repository to a river.
Ocean	Location where river-water or groundwater containing radionuclides released from the repository discharges to an ocean.

Release mode - one of four potential ways in which radionuclides are transported from the lithosphere to the biosphere, resulting in exposure to humans. The release modes are: land (contaminated solids deposited on the land surface, such as volcanic materials); well (contaminated groundwater pumped to the land surface); river (all fresh surface waters); and ocean.

Biosphere - the zone of the Earth extending from (and including) the surface into the surrounding atmosphere.

Subsection 191.13(a) needs to be changed to accommodate the option of multimode release limits. The proposed wording is as follows:

a) Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation, based upon performance assessments, that the cumulative releases of radionuclides to the accessible environment (for Table 1 in Appendix A), or the cumulative releases of radionuclides, considering all applicable release modes, to the biosphere (for Tables 2 and 3 in Appendix B) for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall:

   1. Have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix A) or Tables 2 and 3 (Appendix B); and

   2. Have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 (Appendix A) or Tables 2 and 3 (Appendix B).

The Department shall select the release limits method to be used in evaluating compliance.

Appendix A remains the same as in the 1985 version of 40 CFR Part 191.

A new Appendix B would be created. It would be the same as Appendix A except for these changes: replacement of Table 1 with Tables 2 and 3, the addition of two notes, and minor changes to the original Note 6 from Table 1. (The creation of a new Appendix C will be discussed in Chapter 6.)

Tables 2 and 3 provide release limits for the four potential release modes to be considered in the containment requirements expressed in curies and terabequerels, respectively. The proposed tables are included at the end of this section.

New information would have to be added as Note 6 of Tables 2 and 3 of Appendix B. The wording for the new Note 6 would be:

> The Agency assumed, in deriving the release limits for the river and well releases in Tables 2 and 3, that the entire drainage system of all rivers (for river releases) and all aquifers (for well releases) is contaminated by the released radionuclides. Site Adjustment Factors (SAFs) may be used with Tables 2 and 3 to account for specific site locations. The following are examples of how SAFs might be developed for the surface flow system and other geologic and hydrologic components of a geologic disposal system.
>
> Example 1--River Releases: For the river column, the release limits are calculated assuming that the entire drainage of all rivers is contaminated. For an actual site, only the downstream section of the tributary that is fed by groundwater passing through the repository is contaminated. To correct for this, a Site Adjustment Factor for the river release mode (SAF_R) is used as a multiplier to adjust the risk factors. The Reciprocal Site

Adjustment Factor ($RSAF_R$), with which the release limits are multiplied, is calculated as follows:

$$RSAF_R = \frac{\sum_{i=1}^{n} (L_{C(i)} * F_{c(i)}) + \sum_{j=1}^{n} (L_{U(j)} * F_{U(j)})}{\sum_{i=1}^{n} (L_{C(i)} * F_{C(i)})}$$

This approximation represents the sums of the products of all tributary lengths and flow rates divided by the equivalent sums of contaminated tributaries. "L" is the length of the river segments and "F" is the volumetric flow rate of that segment. The subscripts "C" and "U" refer to contaminated and uncontaminated segments, respectively. The release limits in Tables 2 and 3 are then multiplied by this ratio to provide a site-specific release limit for the river release mode.

Example 2--Well Releases: The derivation of the release limits for the well release mode using world average parameters assumes all groundwater from the recharge area to the locations where it enters surface waters is contaminated. For an actual site, wells up-gradient of the repository do not produce contaminated water. In addition, during the 10,000-year regulation period, the contaminated plume may not reach the discharge location, thus some uncontaminated water may also be withdrawn down-gradient from the repository.

A method for approximating the ratio of contaminated water to total available water can be determined by dating the water at the repository (A_1), at the point it is expected that the radionuclides will reach in 10,000 years (A_2), and at the location where groundwater discharges to a river (A_3). With these ages, the Site Adjustment Factor for the well release mode (SAF_W) may then be calculated and used as a multiplier to adjust the risk factors. Calculation of the Reciprocal Site Adjustment Factor ($RSAF_W$) is done by dividing the age of the water at the river by the difference in the ages of the water at the repository and at the farthest point of migration in 10,000 years, or:

$$RSAF_W = \frac{A_3}{A_2 - A_1}$$

However, if it is found that the contaminated plume will reach a river within 10,000 years, the formula becomes:

$$RSAF_W = \frac{A_3}{A_3 - A_1}$$

Release limits in Tables 2 and 3 are then multiplied by one of these ratios (the $RSAF_W$s) to provide a site-specific release limit for the well release mode.

The use of SAFs and the parameters to be considered in calculating SAFs shall be determined by the Department.

A second new note, describing the concept of points of compliance for the multimode release limits in the containment requirements will also need to be added to Tables 2 and 3 of the new Appendix B. The note would read as follows:

In calculating cumulative releases over 10,000 years, the points of compliance are as follows:

Release Mode	Point of Compliance
Land	Location where radioactive material released from the repository is brought directly to the land surface.
Well	Any wellhead outside the controlled area from which groundwater containing radionuclides released from the repository is withdrawn for irrigation or supplying drinking water.
River	Location(s) of existing discharge of groundwater containing radionuclides released from the repository to a river.
Ocean	Location where river-water or groundwater containing radionuclides released from the repository discharges to an ocean.

The existing Note 6 from Appendix A, Table 1 should be revised and used as Note 8 for Tables 2 and 3 of the new Appendix B. Two changes will be necessary.

- The third and fourth sentences should be rephrased as follows:

  For each radionuclide in the mixture, determine the ratio between the cumulative release quantity projected over 10,000 years and the limit for that radionuclide for each applicable release mode as determined from Tables 2 or 3 and Notes 1 through 7.

- The last paragraph, the example, should be reworded as follows:

  For example, if all release modes (L,W,R, and O referring to land, well, river, and ocean release modes) are used in the example, if radionuclides a and b are projected to be released in amounts Q_a and Q_b, and if the applicable release limits are RL_a and RL_b, then the cumulative releases over 10,000 years shall be limited so that the following relationship exists:

$$Q_{L,a}/RL_{L,a} + Q_{L,b}/RL_{L,b} + \ldots + Q_{W,a}/RL_{W,a} + Q_{W,b}/RL_{W,b} + \ldots +$$

$$Q_{R,a}/RL_{R,a} + Q_{R,b}/RL_{R,b} + \ldots + Q_{O,a}/RL_{O,a} + Q_{O,b}/RL_{O,b} + \ldots +$$

$$Q_{O,n}/RL_{O,n} < 1.$$

The existing Appendix B from the 1985 standard would be renamed Appendix D. The introductory paragraph of this Appendix discusses evaluating long-term predictions of compliance, focusing on compliance with 191.13. Because of the other proposed changes outlined above, this introductory paragraph should acknowledge two additional steps in 191.13 compliance. The following sentences should be inserted between sentences 2 and 3:

> Quantitative evaluations for these predictions compare predicted releases with either Table 1 of Appendix A or Tables 2 and 3 of Appendix B. If the multimode release limits in Tables 2 and 3 of Appendix B are used, the presence or absence of the four possible release modes (land, well, river, and ocean) to be considered in the containment requirements must be determined. The fifth release mode, for atmospheric releases, is considered under the individual protection requirements. Site Adjustment Factors for the well and river release modes, to be determined by the Department, may be calculated to account for differences between the actual site-specific availability of water and the original assumption that the entire drainage system is available and contaminated.

Table 2. Cumulative Release Limits for 10,000 years (curies per 100,000 MTHM) for Multiple Release Modes

Release Limit (curies per 100,000 MTHM)				
Nuclide	River	Well	Ocean	Land
C-14	TBD[a]	TBD	TBD	TBD
Ni-59	2E+07	9E+06	TBD	1E+09
Sr-90	4E	2E+04	4E+07	3E+07
Zr-93	7E	3E+06	3E+07	4E+07
Tc-99	3E+06	1E+06	6E+08	2E+10
Sn-126	1E+04	4E+03	9E+03	7E+05
I-129	1E+04	5E+03	4E+06	3E+05
Cs-135	1E+05	6E+04	2E+07	2E+06
Cs-137	9E+04	8E+04	2E+06	5E+07
Sm-151	1E+08	4E+07	TBD	1E+10
Pb-210	8E+03	4E+03	TBD	7E+06
Ra-226	6E+03	3E+03	TBD	2E+05
Ra-228	4E+04	2E+04	TBD	6E+07
Ac-227	1E+04	6E+03	7E+03	8E+06
Th-229	3E+04	1E+04	6E+03	5E+04
Th-230	2E+03	8E+02	TBD	3E+03
Th-232	3E+03	1E+03	TBD	3E+03
Pa-231	7E+03	3E+03	2E+04	4E+04
U-233	5E+04	2E+04	1E+06	1E+06
U-234	5E+04	2E+04	TBD	2E+06
U-235	5E+04	2E+04	1E+06	1E+06
U-236	5E+04	2E+04	TBD	2E+06
U-238	5E+04	2E+04	TBD	1E+06
Np-237	1E+04	8E+03	7E+04	8E+06
Pu-238	2E+04	1E+04	TBD	3E+06
Pu-239	2E+04	8E+03	2E+04	2E+05
Pu-240	2E+04	8E+03	2E+04	2E+05
Pu-241	5E+05	2E+05	TBD	4E+08
Pu-242	2E+04	8E+03	TBD	2E+05
Am-241	2E+04	8E+03	5E+03	1E+06
Am-243	2E+04	8E+03	5E+03	4E+05
Cm-245	1E+04	4E+03	3E+03	1E+05
Cm-246	2E+04	8E+03	TBD	3E+05

[a] To be determined

Table 3. Cumulative Release Limits for 10,000 years (TBq per 100,000 MTHM) for Multiple Release Modes

Release Limit (TBq per 100,000 MTHM)				
Nuclide	River	Well	Ocean	Land
C-14	TBD[a]	TBD	TBD	TBD
Ni-59	8E+05	3E+05	TBD	5E+07
Sr-90	2E+03	7E+02	2E+06	1E+06
Zr-93	2E+05	1E+05	9E+05	2E+06
Tc-99	1E+05	4E+04	2E+07	7E+08
Sn-126	4E+02	1E+02	3E+02	3E+04
I-129	5E+02	2E+02	1E+05	9E+03
Cs-135	5E+03	2E+03	6E+05	6E+04
Cs-137	3E+03	3E+03	8E+04	2E+06
Sm-151	4E+06	2E+06	TBD	6E+08
Pb-210	3E+02	1E+02	TBD	2E+05
Ra-226	2E+02	1E+02	TBD	7E+03
Ra-228	2E+03	7E+02	TBD	2E+06
Ac-227	6E+02	2E+02	2E+02	3E+05
Th-229	1E+03	4E+02	2E+02	2E+03
Th-230	7E+01	3E+01	TBD	1E+02
Th-232	1E+02	4E+01	TBD	1E+02
Pa-231	3E+02	1E+02	6E+02	2E+03
U-233	2E+03	7E+02	4E+04	5E+04
U-234	2E+03	8E+02	TBD	6E+04
U-235	2E+03	7E+02	4E+04	4E+04
U-236	2E+03	8E+02	TBD	6E+04
U-238	2E+03	7E+02	TBD	5E+04
Np-237	5E+02	3E+02	3E+03	3E+05
Pu-238	9E+02	4E+02	TBD	1E+05
Pu-239	7E+02	3E+02	6E+02	6E+03
Pu-240	8E+02	3E+02	6E+02	7E+03
Pu-241	2E+04	7E+03	TBD	1E+07
Pu-242	8E+02	3E+02	TBD	6E+03
Am-241	7E+02	3E+02	2E+02	4E+04
Am-243	6E+02	3E+02	2E+02	2E+04
Cm-245	4E+02	2E+02	1E+02	5E+03
Cm-246	7E+02	3E+02	TBD	1E+04

[a] To be determined

5.3 SUPPLEMENTARY INFORMATION

The following material explains why the rule is reasonable when written in the form on the preceding pages. This material could be used by the EPA as supplementary information for the proposed rule.

The 1985 release limits contained in 40 CFR Part 191, Section 191.13, which were stated in terms of the allowable release from a repository containing 1,000 metric tons of heavy metal, were developed by estimating how many curies of each radionuclide would cause 10 premature deaths over 10,000 years if released to the environment. For these calculations, the Agency used very general models of environmental transport, based upon a simultaneous release to all the world's rivers and oceans. The resulting release limits table (Appendix A, Table 1 of the 1985 version), provided a single cumulative release limit per radionuclide that was to be evaluated at the boundary of the controlled area.

Several commenters have suggested that release limits based solely upon a simultaneous release to the world's rivers and oceans may not be appropriate for all releases at all sites. As a result, the Agency has further evaluated the appropriateness of the single generic derived version of the release limits. While the Agency continues to believe that cumulative release limits per radionuclide are an appropriate way in which to regulate the disposal of radioactive waste, several changes have been implemented in order to accommodate any site-specific circumstances which may differ from the assumed circumstances underlying the Table 1 release limits. The Agency further feels that today's proposal gives the Department greater flexibility in complying with the standard, while at the same time it provides at least the same level of protection to human health and the environment as did the 1985 standard.

Given below is a brief description of the relevant changes in the present version from the 1985 version, with a more detailed explanation to follow:

- Table 1 in Appendix A is retained as an option for determining the releases to the accessible environment.

- New multimode release tables (Tables 2 and 3 in Appendix B) for the containment requirements are included as an option for determining releases to the biosphere. Each table consists of four release modes (land, wells, rivers and oceans), each with specific release limits, that can be used to account for site-specific features. Atmospheric releases are considered in the Individual Protection Requirements.

- The multimode release limits (Tables 2 and 3 in Appendix B) are based upon a repository containing 10^5 (100,000) MTHM rather than 10^3 (1,000) MTHM.

- Compliance with the release limits from the multimode tables is evaluated at the point of release to the biosphere for the particular release mode rather than at the boundary of the controlled area.

- Site Adjustment Factors (SAFs) are provided for use with the multimode release limits. The Department may use SAFs for the river and well release modes. The department would determine the parameters to be used in accounting for specific site locations.

10^5 (100,000) MTHM v. 10^3 (1,000) MTHM

The multimode release limits contained in today's version of Appendix A (Table 1) and Appendix B (Tables 2 and 3) are based upon a 10^5 (100,000) MTHM repository rather than a 10^3 (1,000) MTHM repository. This modification reflects no quantitative change in the level of protection. It simply presents the information in a manner more clearly related to the fundamental criterion (1,000 deaths per 10,000 years per reference repository, whether HLW or TRU waste) and the individual protection dose standards which are based upon a 10^5 (100,000) MTHM repository. For consistency and scaling efficiency, 10^5 (100,000) MTHM for HLW and 20 MCi for TRU will now be used as the reference repositories for the multimode release method.

Four Column Release Limits Tables

After receiving comments that a single generic derived release limit based upon a simultaneous release to all of the world's rivers and oceans as a radionuclide escapes the controlled area may not be appropriate for all repositories, the Agency has reevaluated the basis of the rule. The Agency feels that more is known now about release modes and pathways than when the 1985 version of the standard was promulgated. Advances in the understanding of geologic disposal systems should be incorporated into the present version of the rule. As a result, the Agency has retained the single generic derived release limit table and added an option of multimode release limit tables consisting of four columns addressing land, well, river (including all fresh surface water), and ocean release modes. A fifth release mode, for atmospheric releases, is considered in the individual protection requirements.

The Agency feels that today's version of the multimode release limit tables applies uniformly to all repositories and pathways while allowing all major components of a disposal system to be included in a risk assessment. In setting the multimode release limits for today's rule, the Agency has used the same methodology described in the Background Information Document (BID) for the 1985 version. That is, for each radionuclide, the maximum number of fatalities allowed by the fundamental criterion (1000) was divided by the fatal cancers per curie for each release mode. The summed normalized release limit for each scenario or event would include the release fractions for each radionuclide for each release mode.

The derivations from the 1985 version of the standard have been reexamined. The derivation for the land and river release modes in the 1985 version were basically complete. The well release mode limits consist of a minor modification to the river release mode, and the ocean release mode limits have been completely recalculated. For a thorough treatment on exactly how the release limits were derived, the BID should be consulted.

Implementation of Multimode Release Limits

While both the BID and the standard address the implementation of the multimode release limits approach, the Agency feels that it should be addressed here also. It should be stressed that the level of protection provided to human health and the environment, for both present and future populations, has remained the same for today's version of the standard as that contained in the 1985 version. The only significant change in the containment requirements is the optional method that the Agency is allowing the Department to use in determining compliance with the containment requirements. The Agency believes that in some instances this option may more realistically reflect the actual processes and events that will take place between the repository and the potential release points and therefore may more realistically reflect the potential risks posed by any such repository.

Multimode Well Release Limits Not Applicable within the Controlled Area

The Agency feels that it is necessary to make one point particularly clear with regard to the implementation of the multimode well release limits. That is, these release limits do not apply within the controlled area. This view was upheld by the First Circuit Court (Natural Resources Defense Council v. U.S.E.P.A., 824 F.2d 1258 (1st Cir. 1987)). As the Court stated in upholding the Agency's decision not to apply the groundwater protection standards within the controlled area:

> "... the EPA's choice to sacrifice the purity of water at repository sites as part of the control strategy was impliedly sanctioned by Congress when, subsequent to passage of the SDWA [Safe Drinking Water Act], it enacted the Nuclear Waste Policy Act."

Thus, the concept that a certain amount of area directly surrounding the repository is devoted to the disposal of radioactive waste is clearly accepted. Application of the multimode release limits for wells will therefore begin at the boundary of the controlled area.

The multimode release limits method, in addition to expanding the release limits to a four column table, also allows the Department to evaluate potential releases at the points of release to the biosphere for each release mode rather than at the boundary of the controlled area for all potential releases. This approach is consistent with the 1985 approach in that the Agency has modeled the effects of a release of each radionuclide via each of the four release modes for the containment requirements and based the release limits upon this modeling.

In setting the current multimode release limits, the Agency has assessed the impacts upon human health and the environment once a radionuclide escapes through one of the four release modes for the containment requirements. This modeling from the release points to humans ensures uniformity of the biosphere for all applications of multimode release limits in the containment requirements. In contrast, the Agency has decided in providing multimode release limits that it would be more appropriate for the Department to assess

the movement of radionuclides from the repository to the points of release. This decision is a result of comments received and further evaluation of potential repository locations.

While the Agency believes that the use of generic models to assess the impacts of radionuclides once they are released into the environment via one of the four release modes is an appropriate method to regulate the release of radionuclides, it is also the Agency's belief that the Department may most appropriately assess the movement of radionuclides from the repository to the points of release. This belief is based upon the fact that the Department will be in a better position to evaluate the site-specific attenuation factors and their impact upon the movement of radionuclides through the lithosphere to the points of release. Attenuation factors depend on: groundwater velocity, retardation factor, dispersivity, distance of the actual release from the repository in the direction of groundwater flow, duration of regulation, radionuclide half life, time of release from the repository, and rate of release. All components of the disposal system should be evaluated when determining compliance with the multimode release limits unless it can be shown that their effects are negligible.

Site Adjustment Factors

In determining compliance with the multimode river and well release limits, the Agency allows the Department to use site adjustment factors (SAFs). This is necessary because, in deriving the release limits for the river and well release modes, the Agency assumed the entire drainage system of all rivers (for the river release mode) and all aquifers (for the well release mode) would be contaminated by the released radionuclides. Thus, in order to obtain a more realistic depiction of the potential releases from specific sites, the Agency allows SAFs to be used when determining the release limits for actual sites.

As stated earlier, there is no need for adjustment factors in computing compliance with the release limits for the land and ocean release modes. The Department determines the factors to be used in determining SAFs for a specific repository. In applying the multimode release limits to specific sites, the Department should recognize that it will be necessary to allocate radionuclides that reach an aquifer to either the well or river release modes. Surface (river) and groundwater (well) usages vary for different regions in the United States. Thus, the Department will be responsible for determining the appropriate allocations for the specific region in which the site is located.

The effect of multimode release tables on the release CCDF is to change the magnitude of the normalized release (R) for each scenario or event relative to the single release method in the 1985 version of 40 CFR Part 191. The probabilities of the individual scenarios or events that make up the CCDF are unchanged.

The Agency believes that today's rule satisfies comments received concerning the appropriateness of using only a single generic derived release limit applied at the boundary of the controlled area. The option of multimode release limits refines the release limit approach used in the 1985 version of 40 CFR Part 191, Section 191.13. The use of multimode release limits accounts for all release modes to be considered in the containment requirements in assessing the performance of a disposal system. The

Department is responsible for determining release modes and release locations for all pathways for each repository. Because the Agency has computed all transport and biological effects from the release location to humans for all four release modes, the biosphere and effects are uniform for all applications of the containment requirements. Multimode release limits are not site specific and can therefore be applied to future repositories.

5.4 TECHNICAL SUPPORT DOCUMENTATION

The following material is supporting information that could be cited as reasons for the suggestions in the proposed revision. It could be part of a technical support document for the rule.

Background

The 1985 version of 40 CFR Part 191 (Reference 5-1) contained a single derived release limit for all release modes that was based on simultaneous release to all the world's rivers and oceans. Cumulative releases would have been evaluated at the boundary of a repository. The EPA based the decision to use this approach on their determinations that releases to surface water through groundwater are usually the most important release mode for mined repositories and that the health effects per curie released are usually the highest for this release mode (Reference 5-2).

In reexamining 40 CFR Part 191, the EPA has received substantial comments addressing release limits based on a single release mode. Characterization of disposal sites currently under consideration indicates that release modes for these proposed repositories are gaseous, land surface, and withdrawal wells. Therefore, it is appropriate to add the option of multimode release limits that, except for gaseous releases, may be used to evaluate these additional release modes in compliance evaluations for the containment requirements. Gaseous releases, although included in this discussion for completeness, are considered in the individual protection requirements of the regulation. The option of multimode release limits satisfies any deficiencies that may have existed in the 1985 version by providing the ability to account for all applicable release modes in assessing the performance of a disposal system. The use of multimode release limits applies the standard at actual release locations (Figure 5-1), so risk attenuation between the boundary and the release locations is considered in the risk assessment. In addition, the methodology for multimode release limits allows corrections for repository locations.

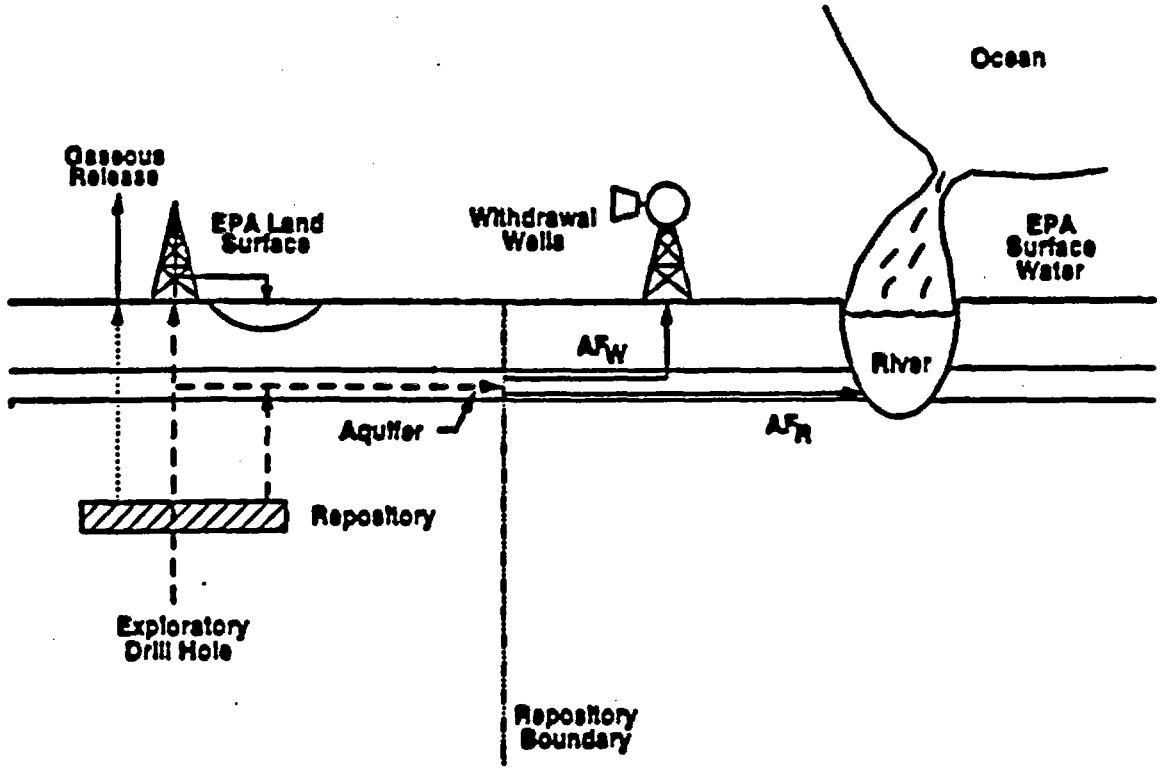

Figure 5-1. Schematic of a Radioactive Waste Disposal System Showing Possible Release Modes and Risk Attenuation Factors Outside the Repository.

(Gaseous releases are considered in the individual protection requirements. In some instances, human intrusion may not be considered in evaluations of the land release mode, as explained in Chapter 3.)

Description of Multimode Generic Release Limits

Tables 2 and 3 are included in Appendix B of 40 CFR Part 191 to supply generic release limits that are set at the locations of release to the biosphere for each applicable release mode, which is just one step in the derivation prior to where they were set in the 1985 version of 40 CFR Part 191. The following sections describe multimode release limits, methods used in developing the four-column table of release limits, methods for combining releases from all applicable modes into a single summed normalized release limit, corrections for repository locations and geologic risk attenuation, and suggestions for performance assessments. These multimode release limits contain some generalizations that may not apply to specific repositories, but the generalizations are limited to the processes between the release locations and humans. Multimode standards apply uniformly to all repositories and all release modes considered in the containment requirements. All major components in the disposal system are included in risk assessments.

EPA generic analyses from the release locations to humans ensure uniform modeling of the biosphere for all applications (dashed lines in Figure 5-2). The four-column release table proposed for 40 CFR Part 191 covers all applicable release modes for repositories. The appropriate release mode is selected for each pathway, and all disposal system components are included in the performance assessment. This is similar to the approach used for the 1985

version of 40 CFR Part 191, and most of the derivations of risk factors were completed for that version of the standard (References 5-2 and 5-3). Differences are that risk factors for well releases have been calculated, and risk factors for ocean releases have been recalculated.[a] Release limits are still calculated by dividing the fundamental criterion (1,000 deaths per 10,000 years per reference repository) by the risk factor for each radionuclide.

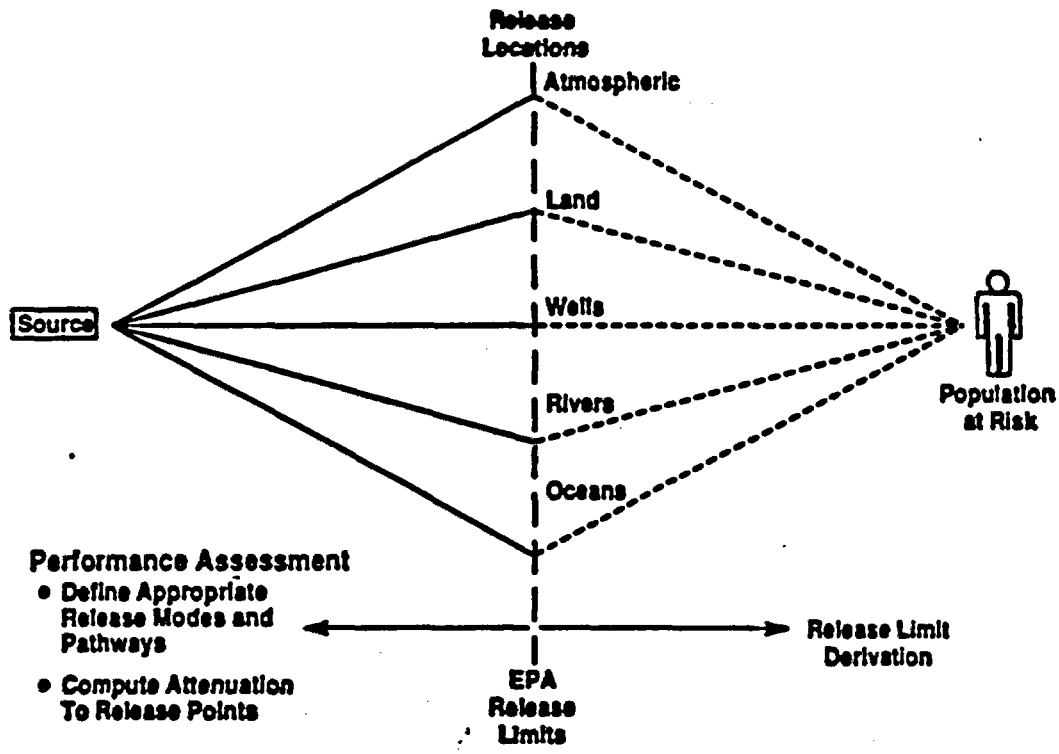

Figure 5-2. Multimode Release Limits in the Risk Assessment Process. (Atmospheric releases are considered in the individual protection requirements.)

Derivation and Implementation of Multimode Release Limits

The following sections summarize the factors considered in the derivation of the four-column tables of release limits in the present version of 40 CFR Part 191. Factors considered in analyses for the river and land release modes are from the Background Information Document (BID) for the 1985 version of 40 CFR Part 191. Factors considered in analyses for the ocean release mode are from a recent study. Data for the well release mode are new and are presented in this chapter.

[a]This technical support document assumes that analyses will be completed using a program such as MARINRAD (Reference 5-4) and a detailed model with a shelf compartment. Other references in this document to ocean releases make the same assumption. If this study is completed, values obtained from the evaluation should be substituted in Tables 5-3 through 5-6 of this Technical Support Document and in Tables 2 and 3 in Appendix B of 40 CFR Part 191.

The derivation of the single generic table for release limits in the 1985 version of 40 CFR Part 191 assumed that all the fresh water that is used comes from the world's rivers. The new multimode release tables separate fresh water into surface water and groundwater. Surface water comes from lakes and rivers, but these sources are combined into a river release mode to be consistent with earlier notation. The USGS publishes estimates of water sources and uses at 5-year intervals. Table 5-1 gives the 1985 percentages of water used for irrigation, livestock, and human drinking water that came from groundwater and surface water. Values are given for the United States and for regions with disposal sites currently under consideration. This table (or an updated version of it) is used to allocate water use to the well and river release modes. The values in Table 5-1 represent the percentages of each radionuclide that reach an aquifer by any means that would be available for well withdrawal or discharge to a river. It does not mean that all or any of these radionuclides will reach any points of release before they decay or during the 10,000 years of regulation. The DOE selects the percentages appropriate for each repository region.

Table 5-1. Fresh Water Sources in 1985 (Reference 5-5)

Region	Percentage	
	Groundwater	Surface Water
Rio Grande Region	28	72
Great Basin	19	81
United States	36	64

Adjustments of Generic Release Limits

Generic or world average parameter values are used to compute multimode release limits, just as they were in the derivation of the present standards. Therefore they may not represent the actual radionuclide pathways or risk of specific repository locations. There are many site adjustment factors (SAFs) that could be applied to release limits for specific repositories to compensate for these generalizations. Alternatively, generic SAFs could be defined in the standards that would apply to all sites, or the selection of site specific adjustment factors could be left to the implementing agency for each repository. Generic SAFs have the advantage of consistent use for all repositories, and an equitable selection of SAFs that increase and decrease the release limits would be predefined. The disadvantages of generic adjustment factors are that they may overcorrect or undercorrect at any given site. The advantage of developing SAFs for each repository is that local conditions such as repository location relative to rivers, oceans, agriculture, and populations at the time of assessment can be defined more precisely. The disadvantage is the potential for nonuniformity in the selection of SAFs and demands for an unreasonable number of SAFs.

Either option should produce more accurate predictions of actual risk than generic analyses with no site adjustments. The magnitude of the net adjustment would depend on site characteristics and may be insignificant for some repositories. Generic SAFs for two of the most obvious cases

are suggested for the river and well release modes in their respective sections. The alternative to SAFs for repositories that cannot be adequately assessed with generic release limits is the use of collective dose limits, which do not require adjustments, but require additional site characterization and PA.

River Release Mode

World-average parameters were used to compute risk factors included in the 1985 version of the standards (Reference 5-3). This approach is compatible with fundamental criteria for collective risk and can be used with multimode derivations. The pathways to humans for the river release mode include ingestion of drinking water, freshwater fish, food crops, milk, and beef; inhalation of resuspended material; and external exposure to ground contamination and air submersion. "River" includes all sources of fresh surface water. Derivations for the river mode have not been updated with more recent data. Ocean releases, which were included in the 1985 version of the table, have been removed from the river release mode and are now considered separately.

The derivation of the risk factors for the river release mode, using world-average parameters, assumes that the entire drainage system of all rivers is contaminated with the released radionuclides regardless of the repository location (Reference 5-2). Site Adjustment Factors (SAF_R) may be used to correct for actual repository locations and may be selected by the DOE.

As an example, Figure 5-3 shows that, in reality, only the downstream section of the tributary that is fed by groundwater passing the repository is contaminated. The ratio of the actual available contaminated water to the total available water in the drainage system is approximated by dividing the sum of the products of contaminated tributary lengths and flow rates by equivalent sums of all tributaries:

$$SAF_R = \frac{\sum_{j=1}^{n} (L_{C(j)} * F_{C(j)})}{\sum_{i=1}^{n} (L_{C(i)} * F_{c(i)}) + \sum_{j=1}^{n} (L_{U(j)} * F_{U(j)})} \qquad (5\text{-}1)$$

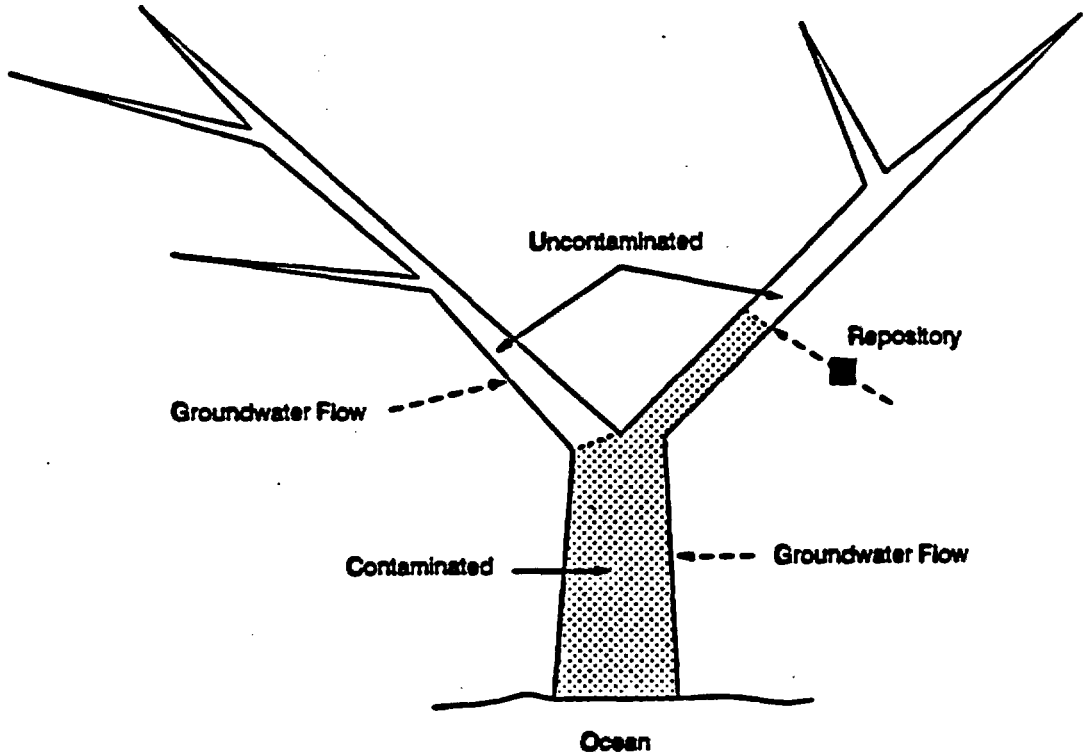

Figure 5-3. Generic River Basin for the River Release Mode

SAF_R is the site adjustment factor used to correct the risk factors for the river release mode. "L" is the length of the river segments and "F" is the volumetric flow rate of that segment. The subscripts "C" and "U" refer to contaminated and uncontaminated segments, respectively. The risk factors for the river release mode are adjusted by multiplying by the SAF_R. If the adjustment is applied to the release limits rather than to the risk factors, the Reciprocal Site Adjustment Factor ($RSAF_R$) is used as the multiplier to adjust the release limits. This definition of water availability is compatible with the derivation in the 1985 version of 40 CFR Part 191.

Attenuation factors (AFs) for radionuclide transport in aquifers depend on flow rates, diffusion, dispersion, retardation, decay rates of the nuclides, the duration of regulation, and the performance of all preceding repository components (Reference 5-6). Determining AFs for the river release mode would extend the present assessments beyond the controlled area.

Well Release Mode

Pathways for the well release mode are the same as those for the river mode except for fish consumption. The radionuclide concentrations in groundwater used to compute risk factors for the well mode are based on world averages, the same as the river mode, so that the standards are consistent. The total volumetric flow rates for both modes are computed by dividing the volumes of each part of the hydrosphere by their exchange activities. This information is available in a UNESCO report for all the major hydrosphere divisions (Reference 5-7) and is summarized in Table 5-2.

Table 5-2. World Hydrosphere Activities (Reference 5-7)

Part of Hydrosphere	Volume (km^3)	Exchange Activity (yrs)	Volumetric Flow (km^3/yr)
Rivers	1.2×10^3	.032	3.8×10^4
Lakes	2.3×10^5	10	2.3×10^4
Active Groundwater	4.0×10^6	330	1.2×10^4
Total Groundwater	6.0×10^7	5000	1.2×10^4
World Oceans	1.4×10^9	3000	4.6×10^5

The derivation of the river risk factors in the 1985 version of 40 CFR Part 191 used a volumetric flow rate of 3×10^4 km^3/yr. This flow rate is a good average of the lake and river divisions, which comprise surface water sources. The flow rates for groundwater are a factor of 2.5 lower, or the radionuclide concentrations in groundwater are a factor of 2.5 higher. Because the risk factors in the EPA derivations (Reference 5-3) are linear functions of concentration, the risk factors for the two modes scale with concentration. The ratio of release limits for the well release mode to those for the river mode range from 0.400 for Zr-93 to 0.803 for Cs-137. This variation is caused by fish consumption in the river mode.

The derivation of the limits for the well release mode using world average parameters assumes all groundwater from the recharge area to the locations where it enters surface waters is contaminated. Site Adjustment Factors (SAF_W) may be used in the same manner as for the river release mode. As an example, Figure 5-4 shows that, in reality, wells upgradient of the repository do not produce contaminated water. In addition, during the 10,000-year regulatory period, the contaminated plume may not reach the discharge location, and some uncontaminated water also would be withdrawn downgradient from the repository. The ratio of contaminated to total available water can be approximated by dating the water at the repository (A_1), at the point that the radionuclides are expected to reach in 10,000 years (A_2), and at the location where groundwater is discharged to a river (A_3), as shown in Figure 5-4. The site adjustment factor (SAF_W) can then be approximated by dividing the difference in the ages of the water at the farthest point of projected radionuclide migration in 10,000 years (A_2) and at the repository (A_1) by the age of the water at the point of discharge to the river (A_3):

$$SAF_W = \frac{A_2 - A_1}{A_3} \qquad (5\text{-}2)$$

However, if the contaminated plume is projected to reach a river within 10,000 years, the SAF_W is approximated by the following formula:

$$SAF_W = \frac{A_3 - A_1}{A_3} \tag{5-3}$$

The risk factors are multiplied by these ratios. If the correction is applied directly to the release limits rather than to the risk factors, the release limits are multiplied by the Reciprocal Site Adjustment Factor ($RSAF_W$).

Computations of attenuation factors are similar to those for the river release mode. Over a 10,000-year period, withdrawal wells could be located anywhere in the contaminated plume outside the controlled area. Therefore, to assume uniform withdrawal in the plume for the entire time is reasonable. The well AFs are then based on a statistical sampling of distances to wells instead of being based on a single distance, as the river mode AFs are.

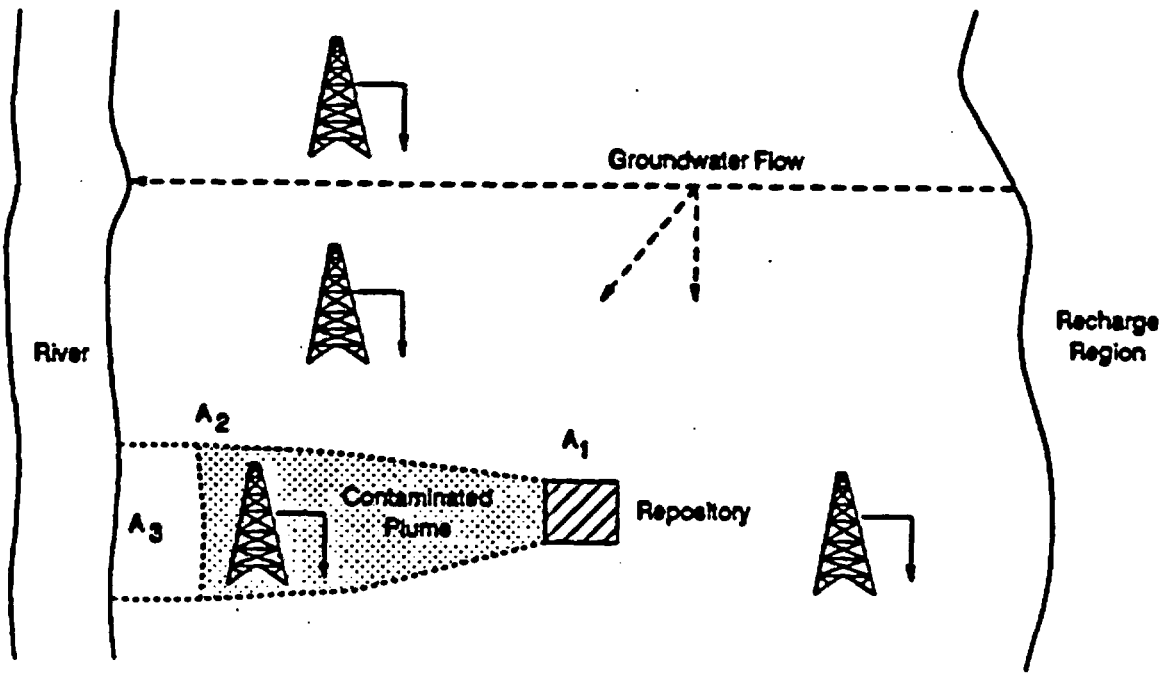

Figure 5-4. Generic Groundwater Diagram for the Well Release Mode

Ocean Release Mode

Ocean risk factors in References 5-2 and 5-3 were compared with those computed with the MARINRAD (Reference 5-4) computer program and deep ocean and shelf models for the Subseabed Disposal Project (References 5-8 and 5-9). The comparison showed that the ocean risk factors used to derive the release limits in the 1985 version of 40 CFR Part 191 were up to a factor of 100 too low (Reference 5-10). This difference was confirmed by a preliminary study of ocean risk factors that were defined in a letter from R.D. Klett (SNL) to D. Ensminger (TASC) concerning the "Ocean Model for Release Limit Derivation," dated October 22, 1991. The preliminary study was conducted by TASC and explained in a letter from S. Oston (TASC)

to R. Williams (EPRI) about "Ocean Pathway Modeling," dated December 10, 1991. [Note: A thorough study of the ocean mode should be conducted with MARINRAD.]

No correction factors for repository location are required for the ocean mode. With the conservative assumptions of no risk attenuation in the rivers and the return of all irrigation water to the rivers, the same geologic AFs are used for the river and ocean release modes for each repository.

Land Release Mode

Changing the method of computing risk factors for the land mode is not necessary, and the risk factors have not been updated with more recent data. No corrections for repository location and no computations of risk attenuation are required for the land release mode.

Atmospheric Release Mode

For the multimode release approach, no corrections for repository location and no additional computations of attenuation are required. The method for computing C-14 risk factors in EPA 520/5-85-026 (Reference 5-3) used a good global circulation model with release to the atmosphere. Updating the analysis with a later version of the global circulation model would only increase the release limit by a factor of 1.4. For completeness, a value for I-129 (Reference 5-11) has been added to the atmospheric column.

Risks from releases to the atmosphere are proportional to the amount of radioactivity in the atmosphere during the period of regulation, not the total amount of activity released. Because the release limits are based on total released activity, the C-14 limits are accurate for early releases but very conservative for later releases. One alternative would be to regulate atmospheric releases under the Individual Protection Requirements of 40 CFR Part 191. This would result in an evaluation of releases in a manner that is consistent with the National Emissions Standards for Hazardous Air Pollutants (NESHAPS, 40 CFR Part 61). A dose limit of 10 mrem/yr for atmospheric releases would be added to the individual Protection Requirements in addition to the existing 25 mrem/yr limit for individual exposure from all pathways.

For completeness, limits for atmospheric releases have been provided in the Tables. However, as discussed earlier, atmospheric releases will be regulated under the Individual Protection Requirements.

Risk Factors

This section presents the derivation results in terms of risk factors, the premature fatal cancers induced over 10,000 years for each curie of the various radionuclides that may be released to the biosphere. These risk factors were used to develop the radionuclide release limits proposed for Tables 2 and 3 of Appendix B of 40 CFR Part 191. Risk factors in cancers per TBq are shown here in Table 5-3, and risk factors in cancers per curie are shown in Table 5-4.

Development of Release Limits for 40 CFR Part 191

The analyses described in this chapter were used to develop radionuclide release limits for the multimode method that correspond to the level of protection chosen for the containment requirements of the final rule (Section 191.13). The 1985 BID describes the procedure used to determine release limits from the risk factors. The maximum number of fatalities allowed by the fundamental criterion were divided by the fatal cancers per curie for each release mode and each radionuclide. The release limits in SI units are shown here in Table 5-5, and the release limits in curies and associated units are shown in Table 5-6.

Summed Normalized Releases

Note 8 for Tables 2 and 3 included in Appendix B of 40 CFR Part 191 indicates how release limits are used in determining compliance with the containment requirements (Section 191.13). In most instances, a mixture of radionuclides is projected to be released to the biosphere. The summed normalized release limit for each scenario or event includes the release fractions for each nuclide for each release mode:

$$Q_{L,a}/RL_{L,a} + Q_{L,b}/RL_{L,b} + \ldots + Q_{W,a}/RL_{W,a} + Q_{W,b}/RL_{W,b} + \ldots +$$

$$Q_{R,a}/RL_{R,a} + Q_{R,b}/RL_{R,b} + \ldots + Q_{O,a}/RL_{O,a} + Q_{O,b}/RL_{O,b} + \ldots + \qquad (5\text{-}4)$$

$$Q_{O,n}/RL_{O,n} < 1. \quad (5\text{-}4)$$

Q is the computed 10^4 year release of a radionuclide for each release mode at the release location, and RL is the release limit for that nuclide and release mode. The subscripts L, W, R, and O refer to the land, well, river, and ocean release modes, respectively, and the subscripts a, b, . . ., n refer to the individual radionuclides listed in the tables. The effect of multimode release tables on the release CCDF is to change the magnitude of the normalized release (R) for each scenario or event relative to the single release method in the 1985 version of 40 CFR Part 191, as illustrated in Figure 5-5. The probabilities of the individual scenarios or events that make up the CCDF are unchanged.

Table 5-3. Fatal Cancers per TBq Released to the Biosphere over 10,000 Years for Multiple Release Modes

Cancers per TBq					
Nuclide	River[a]	Well[b]	Ocean[c]	Land[a]	Atmosphere
C-14	TBD[d]	TBD	TBD	TBD	1.57E+00[e]
Ni-59	1.24E-03	3.03E-03	TBD	1.83E-05	NA[e]
Sr-90	6.08E-01	1.51E+00	6.62E-04	1.02E-03	NA
Zr-93	4.08E-03	1.02E-02	1.06E-03	6.10E-04	NA
Tc-99	9.86E-03	2.41E-02	4.29E-05	1.53E-06	NA
Sn-126	2.84E+00	6.95E+00	2.89E+00	3.73E-02	NA
I-129	2.18E+00	5.43E+00	7.32E-03	1.07E-01	6.72E+00[f]
Cs-135	2.09E-01	4.69E-01	1.73E-03	1.55E-02	NA
Cs-137	2.89E-01	3.60E-01	1.33E-02	5.91E-04	NA
Sm-151	2.53E-04	6.14E-04	TBD	1.81E-06	NA
Pb-210	3.19E+00	7.03E+00	TBD	4.10E-03	NA
Ra-226	4.40E+00	1.05E+01	TBD	1.52E-01	NA
Ra-228	6.51E-01	1.52E+00	TBD	4.24E-04	NA
Ac-227	1.80E+00	4.34E+00	4.13E+00	3.35E-03	NA
Th-229	9.42E-01	2.30E+00	4.64E+00	5.13E-01	NA
Th-230	1.45E+01	3.60E+01	TBD	1.04E+01	NA
Th-232	9.18E+00	2.29E+01	TBD	1.02E+01	NA
Pa-231	4.00E+00	9.87E+00	1.60E+00	6.37E-01	NA
U-233	5.81E-01	1.44E+00	2.50E-02	2.03E-02	NA
U-234	5.29E-01	1.31E+00	TBD	1.77E-02	NA
U-235	5.86E-01	1.45E+00	2.26E-02	2.27E-02	NA
U-236	5.00E-01	1.24E+00	TBD	1.67E-02	NA
U-238	5.56E-01	1.38E+00	TBD	1.86E-02	NA
Np-237	2.15E+00	3.27E+00	3.89E-01	3.27E-03	NA
Pu-238	1.14E+00	2.82E+00	TBD	8.37E-03	NA
Pu-239	1.34E+00	3.32E+00	1.55E+00	1.68E-01	NA
Pu-240	1.31E+00	3.23E+00	1.55E+00	1.41E-01	NA
Pu-241	5.86E-02	1.45E-01	0.00E+00	6.75E-05	NA
Pu-242	1.29E+00	3.20E+00	TBD	1.71E-01	NA
Am-241	1.46E+00	3.28E+00	5.48E+00	2.84E-02	NA
Am-243	1.54E+00	3.49E+00	5.37E+00	6.62E-02	NA
Cm-245	2.73E+00	6.58E+00	8.07E+00	2.18E-01	NA
Cm-246	1.35E+00	3.25E+00	TBD	9.56E-02	NA

Sources:
[a]Reference 5-2
[b] This report
[c]Preliminary incomplete analysis by TASC using MARINRAD
[d]To be determined
[e]Not Applicable
[f]Reference 5-1 using 0.04 cancers per Sv

Table 5-4. Fatal Cancers per Curie Released to the Biosphere Over 10,000 years for Multiple Release Modes

	Cancers per curie				
Nuclide	River[a]	Well[b]	Ocean[c]	Land[c]	Atmosphere
C-14	TBD[d]	TBD	TBD	TBD	5.83E-02[a]
Ni-59	4.61E-05	1.12E-04	TBD	6.79E-07	NA[e]
Sr-90	2.25E-02	5.60E-02	2.45E-05	3.76E-05	NA
Zr-93	1.51E-04	3.77E-04	3.94E-05	2.26E-05	NA
Tc-99	3.65E-04	8.93E-04	1.59E-06	5.65E-08	NA
Sn-126	1.05E-01	2.57E-01	1.07E-01	1.38E-03	NA
I-129	8.07E-02	2.01E-01	2.71E-04	3.96E-03	2.49E-01[f]
Cs-135	7.73E-03	1.74E-02	6.39E-05	5.75E-04	NA
Cs-137	1.07E-02	1.33E-02	4.92E-04	2.19E-05	NA
Sm-151	9.38E-06	2.27E-05	TBD	6.71E-08	NA
Pb-210	1.18E-01	2.61E-01	TBD	1.52E-04	NA
Ra-226	1.63E-01	3.87E-01	TBD	5.62E-03	NA
Ra-228	2.41E-02	5.62E-02	TBD	1.57E-05	NA
Ac-227	6.67E-02	1.61E-01	1.53E-01	1.24E-04	NA
Th-229	3.49E-02	8.51E-02	1.72E-01	1.90E-02	NA
Th-230	5.38E-01	1.33E+00	TBD	3.86E-01	NA
Th-232	3.40E-01	8.47E-01	TBD	3.76E-01	NA
Pa-231	1.48E-01	3.66E-01	5.94E-02	2.36E-02	NA
U-233	2.15E-02	5.33E-02	9.25E-04	7.51E-04	NA
U-234	1.96E-02	4.86E-02	TBD	6.54E-04	NA
U-235	2.17E-02	5.38E-02	8.36E-04	8.42E-04	NA
U-236	1.85E-02	4.59E-02	TBD	6.18E-04	NA
U-238	2.06E-02	5.11E-02	TBD	6.90E-04	NA
Np-237	7.95E-02	1.21E-01	1.44E-02	1.21E-04	NA
Pu-238	4.23E-02	1.05E-01	TBD	3.10E-04	NA
Pu-239	4.97E-02	1.23E-01	5.73E-02	6.23E-03	NA
Pu-240	4.84E-02	1.20E-01	5.73E-02	5.22E-03	NA
Pu-241	2.17E-03	5.36E-03	TBD	2.50E-06	NA
Pu-242	4.79E-02	1.18E-01	TBD	6.34E-03	NA
Am-241	5.42E-02	1.22E-01	2.03E-01	1.05E-03	NA
Am-243	5.72E-02	1.29E-01	1.99E-01	2.45E-03	NA
Cm-245	1.10E-01	2.44E-01	2.99E-01	8.08E-03	NA
Cm-246	4.99E-02	1.20E-01	TBD	3.54E-03	NA

Sources:
[a] Reference 5-2 [b] Preliminary incomplete analysis by TASC using MARINRAD [c] Not applicable
[d] This report [e] To be determined [f] Reference 5-11 using 0.04 cancers per Sv

Table 5-5. Cumulative Release Limits for 10,000 years (TBq per 100,000 MTHM) for Multiple Release Modes
(This table should be used only with RSAFs)

	Release Limit (TBq per 100,000 MTHM)				
Nuclide	River[a]	Well[b]	Ocean[c]	Land[d]	Atmosphere
C-14	TBD[e]	TBD	TBD	TBD	6E+02[f]
Ni-59	8E+05	3E+05	TBD	5E+07	NA[g]
Sr-90	2E+03	7E+02	2E+06	1E+06	NA
Zr-93	2E+05	1E+05	9E+05	2E+06	NA
Tc-99	1E+05	4E+04	2E+07	7E+08	NA
Sn-126	4E+02	1E+02	3E+02	3E+04	NA
I-129	5E+02	2E+02	1E+05	9E+03	1E+02[f]
Cs-135	5E+03	2E+03	6E+05	6E+04	NA
Cs-137	3E+03	3E+03	8E+04	2E+06	NA
Sm-151	4E+06	2E+06	TBD	6E+08	NA
Pb-210	3E+02	1E+02	TBD	2E+05	NA
Ra-226	2E+02	1E+02	TBD	7E+03	NA
Ra-228	2E+03	7E+02	TBD	2E+06	NA
Ac-227	6E+02	2E+02	2E+02	3E+05	NA
Th-229	1E+03	4E+02	2E+02	2E+03	NA
Th-230	7E+01	3E+01	TBD	1E+02	NA
Th-232	1E+02	4E+01	TBD	1E+02	NA
Pa-231	3E+02	1E+02	6E+02	2E+03	NA
U-233	2E+03	7E+02	4E+04	5E+04	NA
U-234	2E+03	8E+02	TBD	6E+04	NA
U-235	2E+03	7E+02	4E+04	4E+04	NA
U-236	2E+03	8E+02	TBD	6E+04	NA
U-238	2E+03	7E+02	TBD	5E+04	NA
Np-237	5E+02	3E+02	3E+03	3E+05	NA
Pu-238	9E+02	4E+02	TBD	1E+05	NA
Pu-239	7E+02	3E+02	6E+02	6E+03	NA
Pu-240	8E+02	3E+02	6E+02	7E+03	NA
Pu-241	2E+04	7E+03	TBD	1E+07	NA
Pu-242	8E+02	3E+02	TBD	6E+03	NA
Am-241	7E+02	3E+02	2E+02	4E+04	NA
Am-243	6E+02	3E+02	2E+02	2E+04	NA
Cm-245	4E+02	2E+02	1E+02	5E+03	NA
Cm-246	7E+02	3E+02	TBD	1E+04	NA

[a]Reference 5-2 [b]This Report [c]Preliminary incomplete analysis by TASC using MARINRAD
[d]To be determined [e]Not applicable [f]Reference 5-11 using 0.04 cancers per sv

Table 5-6. Cumulative Release Limits for 10,000 years (curies per 100,000 MTHM) for Multiple Release Modes
(This table should be used only with RSAFs)

	Release Limit (curies per 100,000 MTHM)				
Nuclide	River[a]	Well[b]	Ocean[c]	Land[d]	Atmosphere
C-14	TBD[d]	TBD	TBD	TBD	2E+04[e]
Ni-59	2E+07	9E+06	TBD	1E+09	NA[e]
Sr-90	4E	2E+04	4E+07	3E+07	NA
Zr-93	7E	3E+06	3E+07	4E+07	NA
Tc-99	3E+06	1E+06	6E+08	2E+10	NA
Sn-126	1E+04	4E+03	9E+03	7E+05	NA
I-129	1E+04	5E+03	4E+06	3E+05	4E+03[f]
Cs-135	1E+05	6E+04	2E+07	2E+06	NA
Cs-137	9E+04	8E+04	2E+06	5E+07	NA
Sm-151	1E+08	4E+07	TBD	1E+10	NA
Pb-210	8E+03	4E+03	TBD	7E+06	NA
Ra-226	6E+03	3E+03	TBD	2E+05	NA
Ra-228	4E+04	2E+04	TBD	6E+07	NA
Ac-227	1E+04	6E+03	7E+03	8E+06	NA
Th-229	3E+04	1E+04	6E+03	5E+04	NA
Th-230	2E+03	8E+02	TBD	3E+03	NA
Th-232	3E+03	1E+03	TBD	3E+03	NA
Pa-231	7E+03	3E+03	2E+04	4E+04	NA
U-233	5E+04	2E+04	1E+06	1E+06	NA
U-234	5E+04	2E+04	TBD	2E+06	NA
U-235	5E+04	2E+04	1E+06	1E+06	NA
U-236	5E+04	2E+04	TBD	2E+06	NA
U-238	5E+04	2E+04	TBD	1E+06	NA
Np-237	1E+04	8E+03	7E+04	8E+06	NA
Pu-238	2E+04	1E+04	TBD	3E+06	NA
Pu-239	2E+04	8E+03	2E+04	2E+05	NA
Pu-240	2E+04	8E+03	2E+04	2E+05	NA
Pu-241	5E+05	2E+05	TBD	4E+08	NA
Pu-242	2E+04	8E+03	TBD	2E+05	NA
Am-241	2E+04	8E+03	5E+03	1E+06	NA
Am-243	2E+04	8E+03	5E+03	4E+05	NA
Cm-245	1E+04	4E+03	3E+03	1E+05	NA
Cm-246	2E+04	8E+03	TBD	3E+05	NA

[a]Reference 5-2 [b]This Report [c]Preliminary incomplete analysis by TASC using MARINRAD
[d]To be determined [e]Not applicable [f]Reference 5-11 using 0.04 cancers per sv

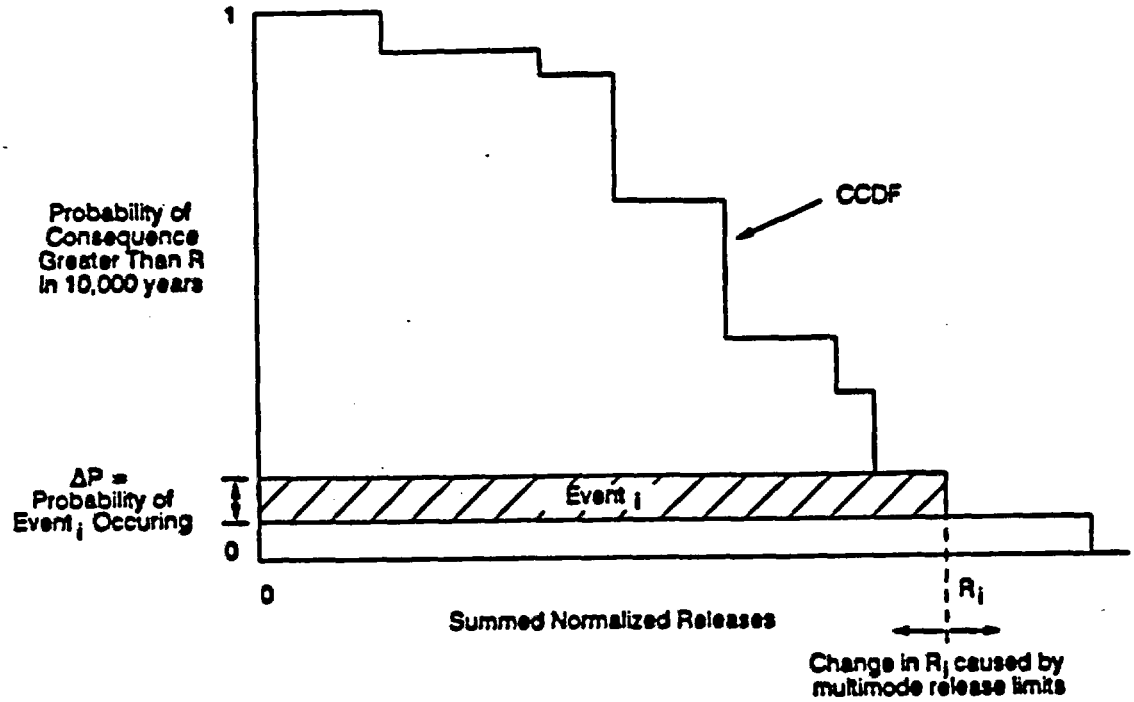

Figure 5-5. Effects of Multimode Release Limits on the Release CCDF

Performance Assessments with Multimode Release Limits

Figure 5-2 illustrates the function of performance assessments (PA) using multimode release limits. Some releases from disruptive geologic events (e.g. volcanos) would be through the upper surface of the controlled volume as shown in Figure 5-1. For these pathways, the PA segment of the risk assessment evaluates releases against land release limits.

For radionuclide transport through an aquifer, the groundwater that is not withdrawn by wells would eventually reach rivers, lakes, and oceans. Computations of releases to wells, rivers, and oceans may require additional attenuation factor analyses (Reference 5-6) by PA, and some site characterization past the controlled volume may be required. Site characterization and analyses only have to extend far enough to show compliance. The remainder of the disposal system could be considered an additional, but unquantified, margin of safety. Because the standards do not specify average fractions of fresh water usage obtained from ground and surface water, regional values are defined by the DOE and incorporated into assessments. The river and well release limits are adjusted by PA to account for the location of each repository relative to the recharge location and closest river or ocean.

Summary

The inclusion of multimode release limits as an option in the containment requirements refines the release limit approach used in the 1985 version of 40 CFR Part 191. The use of multimode release limits accounts for the applicable release modes in assessing the performance of a disposal system for the containment requirements. The DOE would be able to select release modes and release locations for all pathways for each repository. PA will include all pre-release disposal system components in the assessments, from the repository to the release locations.

Because all transport and biological effects from the release location to humans for all four release modes have been calculated, the biosphere and effects are uniform for all applications. These derivations were conducted with generic models and data, so the multimode release limits still contain some generalizations that may affect risk assessments. Multimode release limits are not site-specific and can therefore be applied to future repositories. This approach is compatible with the 40 CFR Part 191 format. The derivations for the river and land release modes were performed for the 1985 version of 40 CFR Part 191 and are complete. The limits for the ocean release mode should be recalculated, and the derivation for the well release mode is a modification of the limits for the river release mode. The roles of the DOE in PA have been expanded to include release mode selection, corrections to account for repository locations, and possible analyses of attenuation factors outside the controlled area. Site characterization and analyses only have to extend far enough to show compliance.

REFERENCES

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5-3. J. M. Smith, T. W. Fowler, and A. S. Goldin, "Environmental Pathway Models for Estimating Population Health Effects from Disposal of High-Level Radioactive Waste in Geologic Repositories; Final Report," EPA 520/5-85-026, August 1985.

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5-9. G. de Marsilly, et al., "Feasibility of Disposal of High-Level Radioactive Waste into the Seabed: Volume 2, Radiological Assessment," Nuclear Energy Agency, Paris, 1988.

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5-11. G. M. Smith and I. F. White, "A Revised Global-Circulation Model for Iodine-129," National Radiological Protection Board, Chilton, Didcot, Oxon, June 1983.

5-12. Clean Air Act, 42 U.S.C.A., Section 7401 et seq. (1983 & Supp. 1991).

5-13. United States Environmental Protection Agency, "National Emission Standards for Hazardous Air Pollutants," 40 CFR Part 61, Subpart I.

CHAPTER 6

COLLECTIVE DOSE

CHAPTER 6

COLLECTIVE DOSE

6.1 STATEMENT OF THE PROBLEM

In some instances the release limits of Table 1 in 40 CFR Part 191 may result in an inappropriate or overly conservative evaluation of repository sites because they do not adequately account for significant features of a site. Release limits are derived standards used only to facilitate regulation. A more fundamental criterion of dose limits could be used without jeopardizing safety. A dose option similar to that provided in the draft Federal Register notice of 40 CFR Part 191 (2/3/92) would allow the Department to show compliance with collective dose limits that are equivalent to the fundamental criterion, i.e., equivalent to 1,000 health effects over 10,000 years per 100,000 metric tons of heavy metal.

6.2 RECOMMENDED APPROACH

Incorporation of the collective dose option requires only minor wording changes to language developed in EPA's draft Federal Register notice (2/3/92). Issues to be considered in using this option are discussed in Chapter 2. Atmospheric releases are considered in the Individual Protection Requirements, as discussed in Chapter 9. Human intrusion is discussed in Chapter 3. A standard biosphere, as described in the "Future States" section to be added to Appendix D (Guidance for Implementation of Subpart B), should be specified.

The following material suggests a way that the standard might be rewritten to incorporate the collective dose option. Most of the text for subsection (b) is taken from the draft Federal Register notice (2/3/92) but is provided here for clarity. Section 191.13 would be rewritten as follows:

191.13 Containment Requirements

The Department may invoke either subsection (a) or (b) of this section.

(a) Disposal systems for spent fuel ....; or

(b) Disposal systems for radioactive waste shall be designed to provide a reasonable expectation, based upon performance assessments, that the collective (population) effective dose, calculated using the weighing factors in Appendix C, caused by releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall:

(1) Have a likelihood of less than one chance in 10 of exceeding 2.5 million person-rem (25,000 person-sieverts); and

(2) Have a likelihood of less than one chance in 1,000 of exceeding 25 million person-rem (250,000 person-sieverts).

Dose limits are based upon an HLW/SF repository of 10^5 MTHM and 20 MCi for a TRU repository.

Appendix C should contain the information that was in Appendix B of the draft Federal Register notice (2/3/92). However, the information in that Appendix has yet to be fully accepted in the United States. Consideration should be given to returning to the information contained in Appendix A of Working Draft 3 (4/25/91) until acceptance of the ICRP 60 methods used in the draft Federal Register notice (2/3/92) has been achieved.

Appendix D would contain the information found in Appendix B of the 1985 version of the standard. Guidance on "future states" would provide the Department with a means of addressing some of the uncertainties that could result from predicting conditions 10,000 years into the future.

The following wording should be added to Appendix D, Guidance for Implementation of Subpart B:

> *Future States.* Uncertainties about the future involving conditions that are unknowable can only be dealt with by making assumptions and recognizing that these may, or may not, correspond to a future reality. The Agency believes that speculation concerning future conditions should not be the focus of the compliance-determination process. Therefore, it would be appropriate for assessments made for Part 191 to proceed under the assumption that many future conditions related to humans or to interactions between humans and their environment will remain the same as those of today's world. Factors in this category include human physiology and nutritional needs, level of knowledge and technical capability, the state of medical knowledge, societal structure and behavior, patterns of water use, and pathways through the accessible environment that are affected by or result from human interactions with the accessible environment. In some instances, consideration of these factors may be specific to the region in which a disposal site is located (e.g., population distributions or patterns of water and land use). In contrast, the Agency would not find it appropriate to include in this category the future states of geologic, hydrologic, and climatic conditions that may be estimated by examining the geologic record. Additionally, the Agency would find inappropriate the assumption that national or world populations will remain unchanged; however, assuming future world populations that cannot reasonably be sustained by current abilities to produce, distribute, and consume food would also be inappropriate. For this reason, future world populations in excess of 10 billion people need not be assumed for evaluations under 191.13. For standardization, a "reference person" is assumed to ventilate (breathe) at a rate of _____ m^3/sec and to ingest _____ liters/day of drinking water; _____ kg/day of fish; _____ kg/day of mollusks; _____ kg/day of aquatic invertebrates; _____ kg/day of water plants; _____ kg/day of leafy vegetables; _____ kg/day of root vegetables; _____ kg/day of grains; _____ kg/day of fruit; _____ kg/day of meat; _____ kg/day of poultry; _____ kg/day of eggs; and _____ liters/day of milk.

Some standardization of current conditions unrelated to particular sites can be attained by providing parameters for a "reference person." A physiological model of "reference man" is available from the International Commission on Radiological Protection (see the example table at the end of this chapter). Values for other parameters need to be determined. In addition, the Nuclear Energy Agency initiated a BIOsphere MOdel Validation Study (BIOMOVS) in 1985. The first phase of the study examined environmental assessment models for selected contaminants and exposure scenarios. The second phase of the study, which began in 1991, has as one of its objectives the development of a reference biosphere model that could be used in performance assessments of radioactive waste repositories. Although this phase is not complete, preliminary results of the study may provide an additional means for standardizing current conditions that could be used as guidance for future states. The provisional reference biosphere(s) should be formulated by October 1992, but the guidance for using the reference biosphere(s) is not expected until 1996.

The following wording should be added between the 2nd and 3rd sentences of the paragraph entitled, *"Compliance with Section 191.13"*:

> Section 191.13 contains options for comparing results of performance assessments with release limits and dose limits. The complementary cumulative distribution function may represent both summed release fractions and summed dose fractions. It is appropriate to apply dose standards to specific events or processes for which the release limits are inappropriate. The predicted doses for each event may then be normalized relative to the dose limits set by the Agency in the same manner as predicted releases. The dose fraction then replaces the summed release fraction for that event in the complementary cumulative distribution function.

6.3 SUPPLEMENTARY INFORMATION

The following information explains the basis for incorporating a collective dose option in the rule. This material could be used by the EPA as supplementary information for the proposed rule.

The fundamental criterion, which is the basis for the containment requirements in 40 CFR Part 191, is that in disposing of radioactive waste there must be a reasonable expectation that releases from a reference repository will cause no more than 1,000 premature cancer deaths over the entire 10,000-year regulatory period. This criterion was based primarily upon technical achievability and the premise that the overall risks to future generations be comparable to the risks that those generations would have faced from the uranium ore used to create the wastes. The Agency intends that the fundamental criterion shall be met in either of two ways: (1) through the use of derived release limits or (2) through the use of a collective dose standard.

The Agency has provided a collective dose alternative in the present version of the standard as a result of comments received. Some commenters have expressed the view that, in some instances, the use of a dose standard may be more appropriate than the use of generic derived release limits. According to the commenters, generic release limits do not fully account for site-specific attenuation factors that indicate variability in the lithosphere and biosphere surrounding repositories. It is the Agency's belief that derived release limits, either single generic or multimode, are appropriate for application to repositories. However, the Agency does realize that there may exist instances where comparisons to a dose standard more clearly reflects the performance of a repository. In applying the dose alternative, the Department would assess the movement of radionuclides from the repository to contact with humans. When applying the release limits, the Department assesses the movement of radionuclides from the repository to the accessible environment (for Table 1 in Appendix A) or to a point of compliance or the biosphere (for Tables 2 and 3 in Appendix B), with the Agency generically assessing the impacts beyond this point.

The performance of dose-based risk assessments may require extensive site characterization for repositories that may not have attenuation processes adequately represented by comparison with release limits. Any extensive site characterization activity may prove to be prohibitively expensive. Uncertainties arise as more parameters are included in compliance demonstration analyses. The larger the number and extent of uncertainties, the greater the cost of the site characterization activity required to reduce them. To reduce somewhat the scope of such site characterizations, the Agency has added a section in Appendix D of this rule that provides guidance concerning projections of occurrences in the future.

It would be appropriate to apply the dose standards only to specific events or processes for which comparisons to the release limits do not adequately reflect repository performance. Predicted dose for each analyzed event may be normalized relative to the dose limits set by the Agency in the same manner as predicted releases. The dose fraction then replaces the summed release fraction for that event in the CCDF. The probability remains the same, so the only effect is to change the consequence level for that event in the CCDF.

A preliminary performance evaluation may be needed to select the most appropriate standard for a particular repository. Repository evaluations using release limits are less expensive and can be completed in less time because they require less site characterization and a less complex performance assessment. However, the approximate release limits may not adequately represent the attenuating processes of some repositories, and the less approximate dose standards may be used.

The Agency believes that the collective dose alternative and the release limits alternative are both viable means of providing protection to human health and the environment. In fact, the fundamental criterion, which is expressed in terms of health effects per unit waste over time, remains the same regardless of which alternative is used. The containment requirements are simply a method of showing compliance with the fundamental criterion. Providing both release limits and dose limits does not mean that proposed repositories are expected to comply with both standards. An unsafe repository could not comply with either dose or release limits, so evaluating compliance against both standards is neither expected nor required.

Thus, the Agency is providing the Department with the option of using the alternative it determines is the most appropriate for a given site. The key in determining the appropriateness of one alternative over the other should be based upon the ability of the particular alternative to reflect more clearly the capability of a disposal system to meet the fundamental criterion.

6.4 TECHNICAL SUPPORT DOCUMENTATION

The following material is supporting information that could be cited as justification for the proposed revision. It could be part of a technical support document for the rule.

The 1985 version of 40 CFR Part 191 (Reference 6-1) contained derived release limits as the standard for evaluating protection of future populations for at least 10,000 years from disposal of radioactive wastes. These release limits, which were derived from a dose standard, used predictive assumptions, generalizations, and simplifications in order to provide a generic standard. The EPA believes that, in most instances, exceptionally good protection can be achieved with release limits. However, in reexamining 40 CFR Part 191, the EPA has received substantial comments addressing the use of derived release limits. One aspect that has been commented on in depth is that, for some repositories, the conservative approximate release limits may not adequately account for attenuating processes and that evaluation against a dose standard, which would be more comprehensive, may be required. Dose limits provide a more precise measure of actual risk but may require more extensive site characterizations and performance assessments. In order to allow for possible circumstances that may require a more comprehensive analysis, the Agency has provided dose limits as an alternative to using the release limits in the present rule. Performance assessments now have the option of constructing the CCDF by using all normalized releases, all normalized doses, or a combination of the two. Providing both release limits and dose limits does not mean that proposed repositories are expected to comply with both standards. An unsafe repository could not comply with either dose or release limits, so evaluating compliance against both standards is neither expected nor required.

Description of the Dose Limit Alternative

The information used to develop the dose limit was used in the development of release limits. The implementation of dose and release limits have many similarities.

The dose limits are based on the fundamental criterion of 1,000 premature deaths during the 10,000 year regulatory period for the reference repository. The premature cancer deaths in the fundamental criterion were converted to allowable effective doses using a conversion factor supplied by the ICRP (Reference 6-2) to produce the dose limits. This procedure is explained in the next section.

Consequences using dose limits are normalized for an event or process similar to the way they are normalized using release limits. The normalized dose consequence is the computed dose divided by the dose limit. Performance assessments using dose limit standards produce the same type of normalized CCDF that is produced using release limits. Therefore, consequence CCDFs based on the dose standard and release limits are regulated by the same containment requirements. The probabilities of events or processes in the CCDF are the same with either limit. Only the values of individual normalized consequences (R for summed normalized release and D for normalized dose) are different, as illustrated in Figure 6-1. The CCDF may be constructed using all normalized releases, all normalized doses, or a combination of the two. The latter option is particularly advantageous for repositories that are expensive to characterize and analyze and have only a few events or processes that cannot be represented properly by generic release limits.

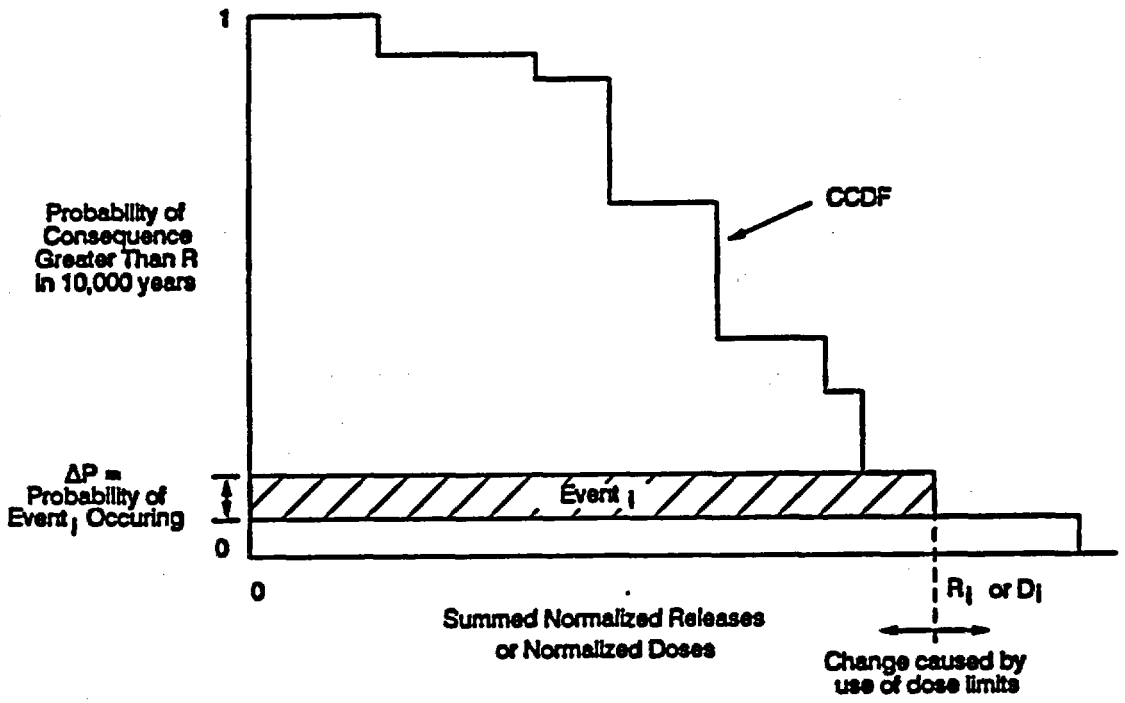

Figure 6-1. CCDF Made Up of Normalized Doses or Normalized Releases

Dose Criteria and Reference Future States

The consequences of radiation exposure that were used to develop the dose standard in the draft Federal Register notice of 40 CFR Part 191 (2/3/92) (Reference 6-3) are the same as the latest ICRP recommendations (Reference 6-2), which have not yet been accepted in the United States. The nominal probability coefficient for stochastic effects used to set the effective dose limits is 0.04 premature cancer deaths per Sv. Applying this coefficient to the fundamental criterion of 1,000 premature deaths in 10,000 years for the reference HLW repository containing 100,000 MTHM gives an effective dose limit of 25,000 person-sieverts per 100,000 MTHM (0.25 person-sieverts/MTHM). For the reference TRU repository containing 20 MCi, the effective dose limit is 25,000 person-sieverts per 20 MCi of radioactive waste (0.00125 person-sieverts/Ci).

Two basic procedures can be used to compute collective effective doses. The procedures in Appendix B of the Draft Federal Register Notice of 40 CFR Part 191 (2/3/92) (Reference 6-3) for computing the effective dose are identical to those in Annex A of ICRP 60 (Reference 6-2). The effective dose (E) is the sum of weighted absorbed doses from all radiation types and energies, in all tissues and organs of the body. It is given by the expression:

$$E = \sum_R w_R \sum_T w_T \cdot D_{T,R} = \sum_T w_T \sum_R w_R \cdot D_{T,R} \qquad (6\text{-}1)$$

where $D_{T,R}$ is the mean absorbed dose to organ T delivered by radiation R. The radiation is that incident on the body or emitted by a source within the body. Values for the radiation weighing

factors (w_R) are given in Table 6-1, and values of the tissue weighing factors (w_T) are given in Table 6-2.

Table 6-1. Radiation Weighing Factors, w_R[1]

Radiation Type and Energy Range[2]		w_R value
Photons, all energies		1
Electrons and muons, all energies		1
Neutrons, energy	<10 keV	5
	10 keV to 100 keV	10
	>100 keV to 2 MeV	20
	>2 MeV to 20 MeV	10
	>20 MeV	5
Protons, other than recoil protons, >2 MeV		5
Alpha particles, fission fragments, heavy nuclei		20

1 All values relate to the radiation incident on the body or, for internal sources, emitted from the source.
2 The choice of values for other radiation types and energies not in the table, see paragraph A14 in ICRP Publication 60 (Reference 6-2)

Table 6-2. Tissue Weighing Factors, w_T[1]

Organ or Tissue	w_T Value
Gonads	0.20
Red bone marrow	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Oesophagus	0.05
Thyroid	0.05
Skin	0.01
Bone surfaces	0.01
Remainder	0.05[2,3]

[1] The values have been developed from a reference population of equal numbers of both sexes and a wide range of ages. In the definition of effective dose, they apply to individuals and populations and to both sexes.

[2] For purposes of calculation, the remainder is comprised of the following additional tissues and organs: adrenals, brain, upper large intestine, small intestine, kidney, muscle, pancreas, spleen, thymus, and uterus. The list includes organs which are likely to be selectively irradiated. Some organs in the list are known to be susceptible to cancer induction. If other tissues and organs subsequently become identified as having a significant risk of induced cancer, they will be included either with a specific w_T or in this additional list constituting the remainder. The latter may also include other tissues or organs selectively irradiated.

[3] In those exceptional cases in which a single one of the remainder tissues or organs receives an equivalent dose in excess of the highest dose in any of the twelve organs for which a weighing factor is specified, a weighing factor of 0.025 should be applied to that tissue or organ and a weighing factor of 0.0225 to the average dose in the rest of the remainder as defined above.

An additional method for calculating doses is provided here because it was considered as an alternative to the approach in Appendix C of the proposed final rule. The NEA used a modification of the ICRP procedures in the dose analyses for the Subseabed Disposal Program (Reference 6-4). The average effective dose per unit intake of activity for the ingestion and inhalation pathways was computed for each radionuclide. Similar dose conversion factors were computed for external exposure. Most of the radioactive doses per unit intake for all the major radionuclides were taken from ICRP Publication 30 (Reference 6-5). The exceptions are the doses per unit intake values for isotopes of plutonium and neptunium; these were calculated using values appropriate to the forms of these radionuclides found in environmental materials (Reference 6-6). Tables 6-3 and 6-4 list the dose conversion factors for both systems of units. These tables simplify the dose calculations and assure uniform application. The values used in the averaging of tissue and organ exposure are reasonable approximations considering the accuracy of the dose model and the weighing factors. Tables 6-3 and 6-4 were computed using

Table 6-3. Dose Equivalent Factors for Humans (Curies and Related Units)

Nuclide	Ingestion (Rem/Ci)	Inhalation (Rem/Ci)	Immersion (Rem/Hr-Ci-M**3)	Exposure to Soil (REM/Hr-Ci-M**3)
C-14	2.07E+03	2.07E+03	0.00E+00	0.00E+00
Ni-59	2.00E+02	1.33E+03	2.30E-03	0.00E+00
Sr-90	1.44E+05	1.26E+06	5.40E-04	0.00E+00
Zr-93	1.55E+03	3.18E+05	0.00E+00	0.00E+00
Tc-99	1.26E+03	7.40E+03	1.30E-04	0.00E+00
Sn-126	1.89E+04	7.40E+04	1.80E-02	9.00E+00
I-129	2.74E+05	1.74E+05	1.70E-02	4.50E-01
Cs-135	7.03E+03	4.44E+04	6.60E-05	0.00E+00
Cs-137	5.18E+04	3.22E+04	1.00E+00	4.20E+00
Sm-151	3.37E+02	2.81E+04	2.60E-04	4.80E-02
Pb-210	5.18E+06	1.30E+07	3.00E-03	1.30E-02
Ra-226	1.15E+06	7.77E+06	1.80E+00	6.40E+00
Ra-228	1.22E+06	4.44E+06	6.75E+00	2.60E+01
Ac-227	1.41E+07	6.66E+09	1.69E+00	8.21E+00
Th-229	3.70E+06	2.11E+07	5.80E-01	2.20E+00
Th-230	5.55E+05	3.18E+08	1.80E+00	6.50E+00
Th-232	2.74E+06	1.63E+09	4.00E+00	1.56E+01
Pa-231	1.07E+07	1.26E+09	5.00E-01	2.20E+00
U-233	2.66E+05	1.33E+08	5.90E-01	2.30E+00
U-234	2.63E+05	1.33E+08	1.18E-03	7.32E-03
U-235	2.52E+05	1.22E+08	2.96E-01	1.31E+00
U-236	2.48E+05	1.26E+08	2.97E-06	2.06E-04
U-238	2.33E	1.18E+08	7.36E-02	3.52E-01
Np-237	4.07E+06	4.81E+08	3.60E-01	1.40E+00
Pu-238	1.85E+06	4.44E+08	1.50E-04	1.30E-03
Pu-239	2.22E+06	5.18E+08	1.20E-04	7.90E-04
Pu-240	2.22E+06	5.18E+08	1.40E-04	1.30E-03
Pu-241	4.44E+04	1.04E+07	6.10E-05	4.60E-03
Pu-242	2.04E+06	4.81E+08	1.10E-04	1.10E-03
Am-241	2.22E+06	5.18E+08	3.90E-02	1.80E-01
Am-243	2.18E+06	5.18E+08	3.10E-01	1.30E+00
Cm-245	6.66E+04	1.74E+07	3.40E-04	5.50E-03
Cm-246	1.11E+06	2.74E+08	2.60E-04	2.90E-03

Table 6-4. Dose Equivalent Factors for Humans (TBq and Related Units)

Nuclide	Ingestion (Sv/TBq)	Inhalation (Sv/TBq)	Immersion (Sv/Hr-TBq-M**3)	Exposure to Soil (Sv/Hr-TBq-M**3)
C-14		5.60E+02	0.00E+00	0.00E+00
Ni-59	5.40E+01	3.60E+02	6.21E-04	0.00E+00
Sr-90	3.90E+04	3.40E+05	1.46E-04	0.00E+00
Zr-93	4.20E+02	8.60E+04	0.00E+00	0.00E+00
Tc-99	3.40E+02	2.00E+03	3.51E-05	0.00E+00
Sn-126	5.10E+03	2.00E+04	4.86E-03	2.43E+00
I-129	7.40E+04	4.70E+04	4.59E-03	1.22E-01
Cs-135	1.90E+03	1.20E+04	1.78E-05	0.00E+00
Cs-137	1.40E+04	8.70E+03	2.70E-01	1.13E+00
Sm-151	9.10E+01	7.60E+03	7.02E-05	1.30E-02
Pb-210	1.40E+06	3.50E+06	8.10E-04	3.51E-03
Ra-226	3.10E+05	2.10E+06	4.86E-01	1.73E+00
Ra-228	3.30E+05	1.20E+06	1.82E+00	7.02E+00
Ac-227	3.80E+06	1.80E+09	4.56E-01	2.22E+00
Th-229	1.00E+06	5.70E+06	1.57E-01	5.94E-01
Th-230	1.50E+05	8.60E+07	4.86E-01	1.76E+00
Th-232	7.40E+05	4.40E+08	1.08E+00	4.21E+00
Pa-231	2.90E+06	3.40E+08	1.35E-01	5.94E-01
U-233	7.20E+04	3.60E+07	1.59E-01	6.21E-01
U-234	7.10E+04	3.60E+07	3.19E-04	1.98E-03
U-235	6.80E+04	3.30E+07	7.99E-02	3.54E-01
U-236	6.70E+04	3.40E+07	8.02E-07	5.56E-05
U-238	6.30E+04	3.20E+07	1.99E-02	9.50E-02
Np-237	1.10E+06	1.30E+08	9.72E-02	3.78E-01
Pu-238	5.00E+05	1.20E+08	4.05E-05	3.51E-04
Pu-239	6.00E+05	1.40E+08	3.24E-05	2.13E-04
Pu-240	6.00E+05	1.40E+08	3.78E-05	3.51E-04
Pu-241	1.20E+04	2.80E+06	1.65E-05	1.24E-03
Pu-242	5.50E+05	1.30E+08	2.97E-05	2.97E-04
Am-241	6.00E+05	1.40E+08	1.05E-02	4.86E-02
Am-243	5.90E+05	1.40E+08	8.37E-02	3.51E-01
Cm-242	1.80E+04	4.70E+06	9.18E-05	1.49E-03
Cm-244	3.00E+05	7.40E+07	7.02E-05	7.83E-04

1975 to 1985 models and data. In defining reference future states, demography, and human characteristics, uncertainties involving things that are unknowable about the future can only be dealt with by making assumptions and recognizing that these may, or may not, correspond to a future reality. Speculation concerning future conditions should not be the focus of the compliance determination process. Therefore, it is appropriate for assessments to contain the assumption that many conditions remain the same as today's. Conditions included in this category are population distributions (i.e., current population distributions should be assumed), level of knowledge and technical capability, human physiology and nutritional needs, the state of medical knowledge, societal structure and behavior, patterns of water use, and pathways through the accessible environment. However, including in this category the geologic, hydrologic, and climatic conditions whose future states may be estimated by examining the geologic record would not be appropriate. Although assuming that national or world populations will remain unchanged is not appropriate, assuming future world populations that cannot reasonably be sustained by current abilities to produce, distribute, and consume food would likewise be inappropriate. For this reason, future world populations in excess of 10 billion people need not be assumed in evaluations for the containment requirements.

Changes covering varying climatic, geologic, and hydrologic conditions may be assessed with sensitivity studies and stochastic analyses.

Performance Assessment

Dose based risk assessments, for repositories that may not have attenuation processes adequately represented by comparison with release limits, could result in extensive site characterization and analyses. If release limits are inappropriate for evaluation of only a few events or processes that are responsible for the significant releases, these events or processes may be analyzed using dose criteria. The predicted doses for each event are normalized relative to the dose limits set by the EPA in the same manner as predicted releases. The dose fraction then replaces the summed release fraction for that event in the CCDF. The probability remains the same, so the only effect is to change the consequence level for that event in the CCDF.

Summary and Conclusions

It is appropriate to add a collective dose option to 40 CFR Part 191. In addition, a method for selectively substituting dose limits for events or processes that cannot be represented accurately with generic derived release limits is also an appropriate alternative. Dose analyses are possible on only selected events and processes, and doses can be normalized to the EPA supplied dose limits. These normalized doses would replace the corresponding normalized releases in the CCDF.

REFERENCES

6-1. Federal Register, Environmental Protection Agency, 40 CFR Part 191, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes; Final Rule," Vol. 50, No. 182, September 19, 1985.

6-2. "ICRP Publication 60 - 1990 Recommendations of the International Commission on Radiological Protection," Annals of the ICRP, Volume 21, Numbers 1-3, 1990.

6-3. U. S. Environmental Protection Agency, "Draft Federal Register Notice of 40 CFR Part 191," February 3, 1992.

6-4. G. de Marsilly, et al., "Feasibility of Disposal of High-Level Radioactive Waste into the Seabed: Volume 2, Radiological Assessment," Nuclear Energy Agency, Paris, 1988.

6-5. "ICRP Publication 30 - Limits for Intakes of Radionuclides by Workers, Parts 1-3," Annals of the ICRP, Volumes 2-8, 1979.

6-6. "Metabolic and Dosimetric Models for Application to Members of the Public: Recommended Models for the Metabolism of Plutonium, Americium, Curium, and Neptunium," NRPB-GS3, National Radiation Protection Board, Chilton, 1984.

EXAMPLE TABLE

(From ICRP 23, 1975)

REFERENCE MAN: SUMMARY OF PHYSIOLOGICAL DATA

		Page
Carbon dioxide exhaled	1000 g/day	340
Dietary intake (nutrients)		
Protein	95 g/day	351
Carbohydrates	390 g/day	351
Fat	120 g/day	351
Dietary intake (major elements)		
Carbon	300 g/day	352
Hydrogen	350 g/day	352
Nitrogen	16 g/day	352
Oxygen	2600 g/day	352
Sulfur	1 g/day	352
Elements (summary of model values for daily balance)	See section O	
Energy expenditure	3000 kcal/day	338
Feces, weight of	135 g/day	353
Feces, components of		
Water	105 g/day	353
Solids	30 g/day	353
Ash	17 g/day	353
Fats	5 g/day	353
Nitrogen	1.5 g/day	353
Other substances	6.5 g/day	353
Feces, major elements in		
Carbon	7 g/day	353
Hydrogen	13 g/day	353
Nitrogen	1.5 g/day	353
Oxygen	100 g/day	353
Human milk, composition of	See Table 128	361
Intake of milk	300 ml/day	357
Lung capacities		
Total capacity	5.6 l	345
Functional residual capacity	2.2 l	345
Vital capacity	4.3 l	345
Dead space	160 ml	345
Lung volume and respiration		
Minute volume, resting	7.5 l/min	346
Minute volume, light activity	20 l/min	346
Air breathed, 8 h light work activity	9600 l	346
Air breathed, 8 h nonoccupational activity	9600 l	346
Air breathed, 8 h resting	3600 l	346
Metabolic rate	17 cal/min-kg W	341
Nasal secretion, composition of (major elements)		
Water	95-97 g/100 ml	365
Calcium	11 g/100 ml	365
Chlorine	495 g/100 ml	365
Potassium	69 g/100 ml	365

EXAMPLE TABLE

(From ICRP 23, 1975)

REFERENCE MAN: SUMMARY OF PHYSIOLOGICAL DATA (Continued)

		Page
Sodium	295 g/100 ml	365
Oxygen Inhaled	920 g/day	340
Saliva, composition of	See Table 130	364
Sweat, composition of	See Table 129	362
Urine values		
Volume	1400 ml/day	354
Specific gravity	1.02	354
pH	6.2	354
Solids	60 g/day	354
Urea	22 g/day	354
"Sugars"	1 g/day	354
Bicarbonates	0.14 g/day	354
Urinary loss of major elements		
Nitrogen	15 g/day	354
Hydrogen	160 g/day	354
Oxygen	1300 g/day	354
Carbon	5 g/day	354
Water balance (gains)		
Total fluid intake	1950 ml/day	360
Milk	300 ml/day	360
Tap water	150 ml/day	360
Other	1500 ml/day	360
In food	700 ml/day	360
By oxidation of food	350 ml/day	360
Total	3000 ml/day	360
Water balance (losses)		
In urine	1400 ml/day	360
In feces	100 ml/day	360
Insensible loss	850 ml/day	360
In sweat	650 ml/day	360
Total	3000 ml/day	360

[1]All sections reference ICRP 23, 1975

CHAPTER 7

TRU WASTE EQUIVALENCE UNIT

CHAPTER 7
TRU WASTE EQUIVALENCE UNIT

7.1 STATEMENT OF PROBLEM

The use of values in the 1985 version of 40 CFR Part 191 that equate transuranic (TRU) waste with high-level waste (HLW) and spent nuclear fuel (SF) is not technically sound because military TRU waste is not associated with commercial reactor fuel, does not have a unit comparable to a metric ton of heavy metal (MTHM) of fuel, and does not have a comparable risk/benefit relationship. None of the proposed quasi-equivalent units equate the risks of a TRU repository to those of a HLW/SF repository. It would only be possible to equate HLW and TRU repository risks for a specific pair of inventories and a specific pair of repositories. One option is to develop a fundamental criteria for TRU waste based on acceptable risk to the populace.

7.2 RECOMMENDED APPROACH

The following material proposes a fundamental criterion for transuranic (TRU) waste disposal, based upon the same risk allowed for high level waste/spent fuel (HLW/SF) repositories and upon the same concept of a reference-size repository. To incorporate this proposed approach, only minor changes in the wording of the rule are needed. The definition of the TRU unit of waste (in the notes accompanying Table 1) must be modified in two ways. First, the definition must include the activity from all radionuclides (not just long-lived alpha-emitters) contained in the waste. Second, the definition must reflect the adoption of a reference size for a TRU repository. Table 1 should be adjusted to the reference release limits (based on 10^5 MTHM for HLW/SF). The activity associated with the TRU unit of waste would be changed to 20 MCi.

Appendix A: Table for Subpart B

TABLE 1 - RELEASE LIMITS FOR CONTAINMENT REQUIREMENTS
{See Table 1 at end of section}

Application of Table 1

Note 1: *Units of Waste.* The Release Limits in Table 1 apply to the amount of wastes in any one of the following:
(a) An amount of spent nuclear fuel containing 100,000 metric tons of heavy metal (MTHM) exposed to a burnup between 25,000 megawatt-days per metric ton of heavy metal (MWd/MTHM) and 40,000 MWd/MTHM;
(b) The high-level radioactive wastes generated from reprocessing each 100,000 MTHM exposed to a burnup between 25,000 MWd/MTHM and 40,000 MWd/MTHM;
(c) Each 10,000,000,000 curies of gamma or beta-emitting radionuclides with half-lives greater than 20 years but less than 100 years (for use as discussed in Note 5 or with materials that are identified by the Commission as high-level radioactive waste in accordance with part B of the definition of high-level waste in the NWPA);
(d) Each 100,000,000 curies of other radionuclides (i.e., gamma or beta-emitters with half-lives greater than 100 years or any alpha-emitters with half-lives greater than 20 years) (for use as discussed in Note 5 or with materials that are identified by the Commission as high-level radioactive waste in accordance with part B of the high-level waste in the NWPA); or
(e) An amount of transuranic (TRU) wastes containing twenty million curies of radionuclides.

Note 2: *Release Limits for Specific Disposal Systems.* To develop Release Limits for a particular disposal system, the quantities in Table 1 shall be adjusted for the amount of waste included in the disposal system compared to the various units of waste defined in Note 1. For example:
(a) If a particular disposal system contained the high-level wastes from 50,000 MTHM, the Release Limits for that system would be the quantities in Table 1 multiplied by .5 (50,000 MTHM divided by 100,000 MTHM).

(b) If a particular disposal system contained two million curies of alpha-emitting transuranic wastes, the Release Limits for that system would be the quantities in Table 1 multiplied by .1 (two million curies divided by twenty million curies).

(c) If a particular disposal system contained both the high-level wastes from 50,000 MTHM *and* 2 million curies of alpha-emitting transuranic wastes, the Release Limits for that system would be the quantities in Table 1 multiplied by .6:

$$\frac{50{,}000 \text{ MTHM}}{100{,}000 \text{ MTHM}} + \frac{2{,}000{,}000 \text{ curies TRU}}{20{,}000{,}000 \text{ curies TRU}} = .6$$

Note 3: {same as 1985 standard}
Note 4: {same as 1985 standard}
Note 5: {same as 1985 standard}
Note 6: {same as 1985 standard}

TABLE 1 - RELEASE LIMITS FOR CONTAINMENT REQUIREMENTS
[Cumulative releases to the accessible environment for 10,000 years after disposal]

Radionuclide	Release limit per 100,000 MTHM or other unit of waste (see notes) (curies)
Americium-241 or -243	10,000
Carbon-14	10,000
Cesium-135 or -137	100,000
Iodine-129	10,000
Neptunium-237	10,000
Plutonium-238, -239, -240, or -242	10,000
Radium-226	10,000
Strontium-90	100,000
Technetium-99	1,000,000
Thorium-230 or -232	1,000
Tin-126	100,000
Uranium-233, -234, -235, -236, or -238	10,000
Any other alpha-emitting radionuclide with a half-life greater than 20 years	10,000
Any other radionuclide with a half-life greater than 20 years that does not emit alpha particles	100,000

7.3 SUPPLEMENTARY INFORMATION

The following material could be used as supplementary information in explaining why the rule is reasonable when written as suggested in the preceding section.

The Agency based the cumulative release limits of 40 CFR Part 191 on the fundamental criterion of no more than 1,000 premature cancer deaths over 10,000 years from the disposal of the wastes from 100,000 metric tons of reactor fuel. This fundamental criterion is expressed in terms of allowable health effects per quantity of waste over a specific time.

The Agency reasoned that this fundamental criterion satisfied two objectives. First, it provided a level of protection that appeared to be reasonably achievable by the then-considered geologic disposal options. The Agency reached this conclusion after assessing the performance of a number of model repositories, using very general transport models. The second objective satisfied by the selected fundamental criterion was the limitation of risks to future populations to acceptably small levels. This conclusion was made after comparing the estimated risks posed by a HLW/SF repository to those that would result if the uranium ore used to create the waste had never been mined. In meeting these two objectives, the Agency established a rational fundamental criterion for the disposal of HLW and spent fuel.

The fundamental criterion formed the basis for the derived release limits, expressed as radionuclide release per "unit of waste." For HLW and spent fuel, the unit of waste selected was 1,000 metric tons of heavy metal. The Agency provided a scaling rule for different sized repositories. The Agency selected one million curies as the TRU equivalent to the HLW/SF unit, which was intended to provide the same degree of control for the long-lived alpha-emitting radionuclides.

For a number of reasons, the Agency is reconsidering the appropriateness of defining the TRU unit of waste in terms of equivalence with the HLW/SF unit of waste. The 1,000,000 Ci/1,000 MTHM relationship was derived based on specific initial inventories of transuranics in TRU wastes and HLW. The method of application of the release limits table, by limiting the summed normalized release fractions of both transuranics and fission products, may result in inconsistent controls of HLW and TRU repositories. In addition, the original approach, comparing initial inventories, fails to take into account the 10,000-year timeframe, the presence of radionuclides other than long-lived alpha emitters, and the risk attenuation which depends on pathways and release modes.

A number of options in redefining the TRU waste unit were examined. The possibility of using an alternative basis for comparison (i.e., other than initial inventories) of TRU and HLW was dismissed due to the lack of a defensible relationship between the two. Another option, a comparable risk/benefit analysis, is not possible due to the remote (and difficult to quantify) nature of benefits associated with TRU wastes. For these reasons, the Agency is proposing a TRU unit of waste, independent of the HLW/SF unit of waste. The option of developing a new absolute collective risk limit was considered, but was deemed impractical because of the extensive analysis required (comparable to the original

analyses supporting the development of 40 CFR Part 191) and the difficulty in eliminating (or justifying) inconsistencies between the approaches for HLW and TRU wastes.

The approach proposed for defining the TRU unit of waste is to derive a fundamental criterion for TRU waste disposal, developed in the same manner as the HLW fundamental criterion. The fundamental criterion for both HLW and TRU waste disposal would then be based on collective risk limit. This approach is consistent with the radiation protection objectives and methods recommended by the Nuclear Energy Agency and the International Commission on Radiological Protection. An advantage to the proposed risk limit approach is the ability to frame the approach in a manner that is both consistent with the technical basis for the HLW release limits and compatible with other provisions of 40 CFR Part 191. Finally, the collective risk limit can be applied uniformly, to all release modes, all repositories, and all inventories. A collective risk limit of 1,000 deaths over 10,000 years for a reference repository is proposed, adopting the same basis used for the HLW standards.

The remaining element in the implementation of the collective risk limit is the total quantity of TRU waste to which the limit applies. For HLW, the fundamental criterion was based upon an inventory (100,000 MTHM) expected to accumulate by the year 2000, encompassing all existing HLW and most future waste from all then-operating reactors. This inventory constituted the HLW "reference repository," used in the original risk analyses supporting the containment requirements. For TRU wastes, a "reference repository" must be defined, consistent with the approach used for HLW.

Using the same timeframe as that for the HLW reference repository, TRU wastes quantities are expected to reach 9.8 MCi by the year 2,000. Extending the timeframe to 2013, the latest dated cited in the Integrated Data Base, provides an estimate of 14.3 MCi.

Based on the projections, a reference TRU repository size of 20 MCi is proposed. Because this reference size was developed consistent with that established for HLW, the conservatism inherent in the HLW criteria is maintained for TRU. In addition, this size is believed to be conservative in light of the projected inventories and is also thought to represent, in practical terms, the largest geologic repository size due to the limited lateral extent of suitable, homogenous formations.

As the final step in incorporating the fundamental criterion for disposal of TRU wastes into the rule, modification of the cumulative release limit table (now based upon 10^3 MTHM) is proposed. This modification reflects no quantitative change to the level of protection required, but merely presents the information in a manner more clearly related to the fundamental criteria (1,000 deaths per 10,000 years per reference repository, whether HLW or TRU waste). For consistency and scaling efficiency, 10^5 MTHM of HLW/SF and 20 MCi of TRU will now be used throughout the standards. To apply the release limits, the standards would be scaled, the proportion determined by rationing the size of the actual repository to the reference repository. For example, for a TRU repository with an inventory of 5 MCi and a reference repository of 20 MCi, the limits applicable to the repository would be 5/20, or 0.25, of the reference release limits. This

approach is consistent with that used for the HLW standards and does not require any new derivations.

7.4 TECHNICAL SUPPORT DOCUMENTATION

The following material is supporting information that could be cited as reasons for the DOE suggestions for the above revision. It could be part of a technical support document for the rule.

The 1985 version of 40 CFR Part 191 contained a fundamental criterion for high-level waste and spent fuel that allowed no more than 1,000 premature deaths over the first 10,000 years from disposal of the wastes from 100,000 metric tons of reactor fuel. In developing the disposal criterion for transuranic waste, the Agency felt that it was possible to equate a TRU waste unit to the HLW/SF waste unit. Thus, the standards did not contain a fundamental criterion specifically developed for TRU waste disposal.

Fundamental criteria (Level 1) are the only standards that explicitly define the radiological safety requirements of the repositories. Level 1 criteria control risks to the populace, have a significant effect on the cost of repositories, and are the basis for other levels of radiological criteria. To have any radiological risk significance, all other levels of criteria must be traceable to an appropriate fundamental criterion (Reference 7-1).

In reexamining 40 CFR Part 191, the EPA has received substantial comments addressing the TRU waste unit. One aspect in particular, that has been commented on in depth, is the need to more fundamentally define the TRU waste unit, rather than trying to equate it to the HLW/SF unit. The Agency believes that the fundamental TRU waste criterion promulgated in the present version of the rule satisfies any deficiencies that may have existed in the 1985 version that resulted from equating TRU waste and HLW/SF. The present version of the TRU criterion is based on established principles for fundamental criteria, and all steps in the development are parallel to those used in the HLW fundamental criterion development. The regulatory philosophy for this fundamental criteria is consistent with that for the HLW/SF fundamental criterion and is also compatible with the existing release limits approach.

HLW Fundamental and Derived Criteria in 40 CFR Part 191

The present fundamental criterion for HLW and SF allows no more than 1,000 premature cancer deaths over the first 10,000 years from disposal of the wastes from 100,000 metric tons of reactor fuel (average of 10^{-6} HE/MTHM-yr). This is a risk/benefit criterion that allows the risk from waste disposal to be proportional to the amount of power generated. For convenience of analyses power is equated to the amount of fuel used to generate the power (MTHM). It is also based on collective world population risk over the 10,000 year period of regulation. The HLW/SF release limits were derived by computing the risk factors (fatal cancers per curie released) for each radionuclide for several release modes (Reference 7-2). The fundamental criterion was divided by each of these risk factors to produce a table containing release limits for each radionuclide (Reference 7-3), which is compatible with the risk/benefit, collective population risk fundamental criterion.

The allowable risk level for HLW/SF disposal was based on predicted capabilities of a reference HLW/SF repository in several geologic media. This results in a high level of stringency relative to standards for other carcinogens. The 100,000 MTHM size of the reference repository was selected because it was the estimated cumulative inventory by the year 2000 (Reference 7-4).

This is also about the largest geologic HLW/SF repository that would be built because of the limited horizontal extent of homogeneous formations with characteristics acceptable for HLW/SF repositories.

Fundamental and Derived Standards for TRU Waste Disposal

The 1985 version of 40 CFR Part 191, while providing a fundamental criterion for HLW and SF, did not contain a fundamental criterion for TRU waste disposal. In developing the TRU waste unit, one million curies was selected as equivalent to the HLW/SF unit. The TRU waste unit was intended to provide comparable levels of protection for the long-lived alpha-emitting radionuclides in TRU waste to that provided for all radionuclides in the HLW/SF unit. The 1,000,000 Ci/1,000 MTHM relationship was derived based on the initial inventories of transuranics in a TRU inventory compared to an inventory of HLW/SF. However, due to the method of application of the release limits table (limiting the summed normalized release fractions of both transuranics and fission products), the specified limits do not provide the same levels of protection for HLW/SF and TRU repositories. Thus, after a review of the numerous comments submitted addressing the need for a more accurately defined fundamental TRU waste disposal criterion, the Agency has concluded that military TRU waste cannot be appropriately associated with commercial reactor fuel to facilitate the equation of the TRU unit to the HLW/SF unit. As a result, the Agency has decided to promulgate a separate fundamental criterion for TRU waste in 40 CFR Part 191.

Several alternatives have been suggested to the Agency in addressing the fundamental criterion for TRU waste. One approach was to develop a fundamental criterion based on acceptable risk to the populace and the expected quantity of TRU waste (Reference 7-5, 7-6). However, since collective population risks are the basis for the fundamental criteria and derived release limits used to show compliance in 40 CFR Part 191, neither the ICRP standards nor the EPA standards for chemical carcinogens could be used for TRU waste. The ICRP fundamental standards are based on a *peak individual* risk *rate*, which is not compatible with *collective* risks or release limits. The standards for chemical carcinogens are based on individual risks as a function of the number of people at risk. This method is also incompatible. Since the benefits associated with military TRU waste are not readily quantifiable, it has been suggested that the EPA develop a new *absolute collective* risk limit. This TRU fundamental criterion would be completely independent of the HLW/SF fundamental criterion and based solely on expected quantities of TRU waste and acceptable levels of risk. One of the difficulties with an absolute TRU criterion is the uncertainty in predicting the total quantity of TRU waste that will be generated so that a risk allocation can be made for each repository. There are also inconsistencies in regulatory philosophy between the risk/benefit HLW/SF criteria and an absolute TRU criteria, and a new release limit table would have to be derived. Thus, it would not be practical to develop this form of fundamental criteria for TRU waste disposal at this time.

Instead, the Agency has developed a TRU fundamental risk criterion parallel to that for HLW/SF repositories. Development of this fundamental criterion used the same rationale and type of analyses as the development of the HLW/SF standards (Reference 7-2, 7-3). No new release limits are needed, and the fundamental criterion for TRU waste is compatible with the HLW/SF criteria and all other requirements in 40 CFR Part 191. Although this TRU fundamental criterion

is not a true risk/benefit criterion, the allowable risk can be scaled relative to repository size, making the allowable risk units for TRU waste comparable to those of HLW/SF.

The fundamental risk criterion for HLW/SF is intended to assure adequate protection for a HLW/SF reference repository of 100,000 MTHM. This reference repository was then used, in the 1985 version of 40 CFR Part 191 (Reference 7-3), for derivations and in comparison studies of waste disposal systems and undisturbed ore bodies (Reference 7-7, 7-8). A TRU reference repository has now been established by equating the allowable risk to that of the HLW/SF reference repository.

The size of the HLW/SF reference repository was based on the expected inventory in the year 2000 including all existing HLW/SF and the projected waste from existing reactors. A consistent size for the TRU reference repository has been defined using the same guidelines. The Integrated Data Base for 1991 (Reference 7-9) lists the current inventory and projected accumulation of known remote-handled (RH) and contact-handled (CH) TRU waste in the year 2000 as 9.8 MCi (Table 7-1). This value is 14.3 MCi in 2013, which is the last year listed. Following the same rationale used to select the size of the HLW reference repository, a conservative size for the TRU reference repository, including RH and CH waste, is 20 MCi. Given the conservatism built into the HLW/SF criteria, this size gives the TRU reference repository a very conservative allowable risk. The allowable risk for smaller TRU repositories is scaled proportionally to their size relative to the reference repository.

New release limits were not required for the TRU fundamental criterion. The risk factors used to derive the release limits were computed for individual radionuclides and apply to any inventory or waste category. The fundamental HLW/SF risk criterion and dose limits in the 1985 version of 40 CFR Part 191 were based on 100,000 MTHM, while the release limits in Table 1 of Appendix A were based on 1,000 MTHM. For consistency and scaling efficiency, 100,000 MTHM for HLW/SF and 20 MCi for TRU waste are now used throughout the standards.

Scaling the release limits for different sizes of repositories uses the method defined in Note 2 to Table 1 in Appendix A of 40 CFR Part 191 (Reference 7-7). Calculating release limits for both HLW/SF and TRU-waste repositories uses the values in the release limit table for the applicable reference repository (100,000 MTHM or 20 MCi) multiplied by the ratio of actual repository size to the reference repository size.

Summary

A TRU fundamental criterion has been developed that is related to the allowable risk for HLW/SF repositories. Development of this criterion used the same rationale and type of analyses as development of the HLW standards, as shown in Table 7-2. The approach using a TRU fundamental criterion is based on repository safety and applies equally to all release modes, all repositories, all inventories, and all times. It uses the same format and regulatory philosophy as the HLW standards, and it is completely compatible with other aspects of the standards. It eliminates the need for a TRU waste unit that is "equivalent" to the HLW/SF waste unit, and the repository risks may be more accurately computed because the release limits are traceable to a fundamental criterion.

Table 7-1. Total System Inventories, Projections, and Characteristics of all Stored DOE TRU Waste in 5-year Increments[1]

End of calendar year	Volume (m^3)		Mass[2] (kg)		Radioactivity[3] (10^3 Ci)		Thermal power[3] (10^3 W)	
	Annual rate[4]	Accumu-lation	Annual rate	Accumu-lation	Annual rate	Accumu-lation	Annual rate	Accum u-lation
Stored, contact-handled[5]								
1990	1,478.2	59,022.1	37.5	2,114.5	166.70	2,534.45	5.26	68.42
1991	2,108.9	61,131.0	120.5	2,235.0	535.65	3,020.68	16.90	84.41
1995	2,108.9	69,566.6	120.5	2,717.0	535.65	4,896.56	16.90	146.15
2000	2,108.9	80.111.1	120.5	3,319.5	535.65	7,098.98	16.90	218.71
2005	2,108.9	90,655.6	120.5	3,922.5	535.65	9,160.70	16.90	286.71
2010	2,108.9	101,200.1	120.5	4,524.5	535.65	11,097.70	16.90	350.64
2013	2,108.9	170,526.8	120.5	4,886.0	535.65	12,205.72	16.90	387.23
2015[6]								

[1] Assembled from data provided in Tables 3.2, 3.5, 3.8, and 3.9, Reference 7-9.

[2] Mass of TRU nuclides.

[3] Values were calculated using the estimated isotopic compositions for TRU waste at the several sites given in Table 3.8. See Section 3.3 for details, Reference 7-9.

[4] Annual rate is an average.

[5] Excludes waste managed as LLW. See Table 3.5, Reference 7-9.

[6] The destination of TRU waste after 2013 is not defined.

Table 7-1. Total System Inventories, Projections, and Characteristics of all Stored DOE TRU Waste in 5-year Increments (Continued)

End of calendar year	Volume (m^3)		Mass (kg)		Radioactivity (10^3 Ci)		Thermal power (10^3 W)	
	Annual rate	Accumulation	Annual rate	Accumulation	Annual rate	Accumulation	Annual rate	Accumulation
Stored, remote-handled[5,7]								
1990	14.4	1,585.4	0.03	118.2	30.38	2,244.76	0.18	6.15
1991	191.1	1,776.5	0.1	118.3	101.26	3,345.43	.60	12.31
1995	191.1	2,540.9	0.1	118.7	101.26	2,966.12	.60	10.87
2000	191.1	3,495.4	0.1	119.2	101.26	2,675.07	.60	9.91
2005	191.1	4,451.9	0.1	119.7	101.26	2,420.13	.60	9.07
2010	191.1	5,407.4	0.1	120.2	101.26	2,193.49	.60	8.32
2013	191.1	5,980.7	0.1	120.5	101.26	2,069.82	.60	7.91
2015[6]								
Total stored[7]								
1990	1,492.6	60,607.5	37.5	2,232.7	197.08	4,779.21	5.44	74.57
1991	2,300.0	62,907.5	120.6	2,353.3	636.91	6,366.11	17.50	96.72
1995	2,300.0	72,107.5	120.6	2,835.7	636.91	7,862.68	17.50	157.01
2000	2,300.0	83.607.5	120.6	3,438.7	636.91	9,774.05	17.50	228.62
2005	2,300.0	95,107.5	120.6	4,041.7	636.91	11,580.83	17.50	295.78
2010	2,300.0	106,607.5	120.6	4,644.7	636.91	13,291.19	17.50	358.96
2013	2,300.0	113,507.5	120.6	5,006.5	636.91	14,275.53	17.50	395.14
2015[6]								

[5] Excludes waste managed as LLW. See Table 3.5, Reference 7-9.

[6] The destination of TRU waste after 2013 is not defined.

[7] The total radioactivity and thermal power columns do not include values for Hanford's projected stored, remote-handled waste. The isotopic composition of this waste is unknown.

Table 7-2. Parallelism in the HLW/SF and TRU Fundamental Criteria

	Waste Type	
Feature	HLW/SF	TRU Waste
Maximum deaths from reference repository in 10,000 years	1000	1000
Basis for reference repository site	Cumulative inventory by year 2000. Waste from currently operating reactors ≈ 100,000 MTHM	Cumulative inventory by year 2013. Wastes from existing facilities rounded up to 20 MCi
Fundamental Criterion	Deaths per 10,000 years/ Reference repository size	Deaths per 10,000 years/ Reference repository size
Release limit values	40 CFR 191, Table 1	40 CFR 191, Table 1
Scaling factor for release limits	Actual repository size/ Reference repository size	Actual repository size/ Reference repository size

REFERENCES

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7-2. J. M. Smith, T. W. Fowler and A. S. Goldin, "Environmental Pathway Models for Estimating Population Health Effects from Disposal of High-Level Radioactive Waste in Geologic Repositories; Final Report," EPA 520/1-82-023, August 1985.

7-3. "Background Information Document - Final Rule for High-Level and Transuranic Radioactive Wastes," EPA 520/1/85/023, August 1985.

7-4. "Draft Environmental Impact Statement for 40 CFR 191: Environmental Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," EPOA 520/1-82-025, December 1982.

7-5. "Long-Term Radiation Protection Objectives for Radioactive Waste Disposal," Nuclear Energy Agency, OEDC/NEA, Paris, 1984.

7-6. "ICRP Publication 46 - Radiation Protection Principles for the Disposal of Solid Radioactive Waste," Annals of the International Commission on Radiological Protection, Volume 15, November 14, 1985.

7-7. Federal Register, Environmental Protection Agency, 40 CFR Part 191, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes; Final Rule," Vol. 50, No. 182, September 19, 1985.

7-8. W. A. Williams, "Population Risks from Uranium Ore Bodies," EPA 520/3-80-009, October 1980.

7-9. "Integrated Data Base for 1991: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characterizations," DOE/RW-0006, Rev. 7, 1991.

CHAPTER 8

UNCERTAINTY PROPAGATION

CHAPTER 8

UNCERTAINTY PROPAGATION

8.1 STATEMENT OF THE PROBLEM

In 1985, the U.S. Environmental Protection Agency (EPA) promulgated standards for disposal of spent fuel, high-level and transuranic radioactive wastes in the United States. These standards included an individual protection requirement of maximum individual dose rate that was applicable for 1,000 years and a containment requirement of cumulative radionuclide releases to the accessible environment applicable for 10,000 years. In 1986, the Natural Resources Defense Council and others challenged EPA's decision to limit the individual protection requirement to 1,000 years as arbitrary and capricious. The First Circuit Court of Appeals ruled on this matter and others on July 17, 1987. The court held that the Agency's choice of a 1,000-year individual protection criterion was arbitrary and capricious and remanded that portion of the regulations to the Agency for reconsideration or a more thorough explanation of the reasons underlying the choice of 1,000 years.

In addition, the Draft Federal Register Notice of 40 CFR Part 191 (2/3/92) includes proposed requirements for calculation of dose and radionuclide release projections for undisturbed conditions up to 100,000 years.

The problem is that there are significant uncertainties associated with calculation of individual doses for 10,000 years, or with projections of doses and radionuclide releases out to 100,000 years.

This task consists of calculating uncertainty propagation from 1,000 to 10,000 years to select an appropriate time period for individual protection and for groundwater protection requirements, and from 10,000 to 100,000 years to evaluate the usefulness of requiring performance assessment calculations beyond 10,000 years.

8.2 RECOMMENDED APPROACH

The time period for assessments of individual and groundwater protection should be no more than 1,000 years after disposal (as in sections 191.15 and 191.16 of the 1985 standard), rather than 10,000 years (as proposed in sections 191.14 and 191.23 of the Draft Federal Register Notice of 40 CFR Part 191 (2/3/92)).

In addition, the new standard should not include requirements for projection of potential releases or doses out to 100,000 years after disposal, as proposed in subsections 191.12(c) and 191.14(b) of the Draft Federal Register Notice of 40 CFR Part 191 (2/3/92).

8.3 SUPPLEMENTARY INFORMATION

The following material provides an explanation of why the rule should be retained as originally suggested in the 1985 standard. This material could be used by the EPA as supplementary information to accompany the proposed rule.

The containment requirements in 40 CFR Part 191 limit cumulative releases to the accessible environment for 10,000 years after disposal. These requirements were based on a world-wide population risk criterion. The Science Advisory Board (SAB) Subcommittee recommendation at the time the 1985 standard was being promulgated (50 FR 38073, September 19, 1985), included the following statements: "We support the use of a population risk criteria. We believe it is impractical to provide absolute protection to every individual for all postulated events for very long periods. On the other hand, in our view it is important that, for the first several hundred years, residents of the region immediately outside the accessible environment have very great assurance that they will suffer no, or negligible, ill effects from the repository." Therefore, the Agency felt that this additional assurance (individual protection requirements) was needed to provide protection for the individual since the primary containment standard was for cumulative releases over 10,000 years, with no limits placed on the rate of such releases.

The individual protection requirements in the final rule issued in 1985 limited annual exposures to individuals from a disposal system during the first 1,000 years after disposal. The Agency examined the effects of different time periods and selected 1,000 years for the individual protection requirement because the Agency's assessments indicated that 1,000 years was long enough to ensure that good engineered barriers would be used.

Demonstrating compliance with individual exposure limits over time frames much longer than 1,000 years appeared to be difficult because of the uncertainties involved. The performance assessments that must be conducted to demonstrate compliance with regulatory requirements include evaluation of parameters and processes that are uncertain. Regardless of how extensive a site characterization program is, these uncertainties will be present. In addition to the initial uncertainty inherent in these parameters and processes, the uncertainty will increase with time. The extent to which these uncertainties change depends, in part, on the extent to which projected site conditions are expected to change. All these uncertainties result in uncertainties in calculation of the performance measures. Demonstrating compliance, therefore, requires an understanding of all the uncertainties, including those inherent in the estimates of future site conditions.

If the present hydrologic conditions at a waste disposal site are expected to persist over time, the uncertainties in calculation of individual dose arise primarily from uncertainties in the description of hydrologic parameters, geochemical parameters, and radionuclide release rates from the repository (canister failure times and leach rates). The uncertainties in calculation of the individual dose rates will increase with time for time periods significantly longer than the radionuclide travel times. These uncertainties will increase significantly over the time period of 1,000 to 10,000 years.

If the present hydrologic conditions at the site are expected to change over time, additional uncertainties are introduced. For example, a change in climate, and thus in infiltration, could affect the hydrologic system at the disposal site. In addition to changing the parameters discussed in the paragraph above, it could change the hydrologic boundary conditions affecting both the radionuclide release rates from the repository (through changed leach rate) and groundwater flow rates. Since uncertainties in the climate change are larger over longer time periods, the uncertainties would further increase over the time period of 1,000 to 10,000 years.

The Agency believes that a 1,000-year time period is more than adequate to protect individuals from the potential risks associated with geologic disposal. The containment and individual protection requirements are complementary to each other and are not inconsistent with each other. They apply to different site conditions (undisturbed versus disturbed performance). Therefore, there is no need for them to cover similar time periods. The containment requirements in Section 191.13, which cover releases over 10,000 years after disposal, are the primary standard for waste isolation. This standard covers all significant processes and events that may affect the disposal system, thus ensuring that the site has natural characteristics that will adequately protect the environment. The individual protection requirement governs only the undisturbed performance of the disposal system. It is designed to ensure that engineered barriers provide adequate protection to individuals living in the vicinity of the repository.

The groundwater protection requirements contained in Section 191.16 of this proposed rule are similar to the individual protection requirements. Their primary purpose is to ensure that engineered barriers perform in such a way as to prevent significant degradation of the groundwater in the vicinity of the disposal facility, and thereby protect the individuals in the area. These requirements only apply to the undisturbed performance of the disposal system and are deterministic in nature, just like the individual protection requirements. Consequently, the Agency has decided to also retain the 1,000-year time period for groundwater protection.

As discussed above, the regulations being proposed by the Agency for individual and groundwater protection cover a time period of 1,000 years after disposal. The containment requirements cover a time period of 10,000 years. Questions have been raised regarding the extent to which periods past 10,000 years should be evaluated. As indicated in the supplementary information accompanying the 1985 standard, the Agency believes that 10,000 years is an adequate time period for demonstration of compliance with the containment requirements, and 1,000 years for individual and groundwater protection. Nevertheless, the Agency asked for comments on whether 100,000-year assessments are likely to provide useful information in selecting preferred disposal sites. Comments received from various groups, including the Nuclear Regulatory Commission, the Advisory Committee on Nuclear Waste, and the Department of Energy, agree that such assessments would not be meaningful as a measure of disposal system performance.

The discussions in the paragraphs above were limited to change in performance of the disposal system for undisturbed conditions over the time period of 1,000 to 10,000 years. If the time period for dose or release projections is increased to 100,000 years, then the

uncertainties may become so large as to render the calculations not meaningful. If disturbances were included, then the uncertainties in calculation of the performance measures would increase further, depending on the uncertainties in the disturbed conditions. Estimating the effects of disturbances to 100,000 years requires the inclusion of relatively low-probability geologic events in the modeling of repository behavior. Hydrologic and geochemical properties of the site may change significantly as well. Merely extrapolating the present conditions is not a defensible way to extend performance assessment calculations over long periods of time.

The Agency continues to believe that a disposal system capable of meeting the containment requirements for 10,000 years would continue to protect people and the environment well beyond 10,000 years and, therefore, assessments for time periods past 10,000 years should not be required. This is supported by the views of other groups. When the 1985 standard was being promulgated, the SAB Subcommittee reviewed and supported the technical arguments for limiting the containment requirements to a 10,000-year period. In addition, NRC requirements in 10 CFR Part 60 already contain siting criteria and performance objectives that reduce the potential for significant release after the 10,000-year period has elapsed.

Consequently, the Agency has decided to not require projections of releases or doses out to 100,000 years after disposal.

8.4 TECHNICAL SUPPORT DOCUMENTATION

The following material is supporting information that could be cited as reasons for the suggestions in the proposed revision. It could be part of a technical support document for the rule.

Background

The performance assessment of radioactive waste repositories involves the comparison of potential radionuclide releases from the repository and the resultant dose to man with regulatory standards. The U.S. Environmental Protection Agency (EPA) standards for such releases and doses are contained in 40 CFR Part 191. The 1985 EPA standards included (i) individual protection in terms of maximum individual dose rate applicable for 1,000 years, (ii) groundwater protection standard applicable for 1,000 years, and (iii) a containment requirement of cumulative radionuclide releases for 10,000 years. The individual and groundwater protection standards were deterministic and the containment standards were probabilistic. In 1986, a lawsuit was filed against the EPA questioning the choice of 1,000 years as the time limit for the individual protection. In 1987, the First Circuit Court of Appeals ruled that the EPA needed to reconsider the 1,000 year time limit or provide more thorough explanation of the 1,000 year time limit.

The EPA has considered extending the individual and groundwater protection time limit to 10,000 years. In addition, the EPA has also considered requiring performance assessments using undisturbed conditions for up to 100,000 years without any quantitative standards. Both of these considerations have resulted in a number of comments to the EPA in opposition to these time limits on the grounds of uncertainties in the performance assessments. The analysis presented below shows how uncertainties propagate with time, and can be used to support the selection of an appropriate time period for the individual and groundwater protection standards.

Measure of Uncertainty

Uncertainty is defined as the level of confidence or degree of accuracy in prediction or calculation of results. Uncertainty is quantitatively defined as a probability density function. Generally accepted quantitative measures of uncertainty are variance and standard deviation. Standard deviation of the total release is used as a quantitative measure of uncertainty in this discussion.

There is both initial uncertainty and uncertainty propagation with time. Initial uncertainty is due to uncertainty in site description. The initial uncertainty may be due to several factors, including: data/parameter uncertainty and model uncertainty. Data collection methods are imprecise and spacially incomplete. Understanding of the natural processes occurring at any site is also incomplete in that it is impossible to know exactly which processes are controlling under all conditions. These factors will lead to uncertainty in the data and model of the repository. Uncertainty propagation with time is primarily due to uncertainty in the future state of the system that results from changes in boundary conditions, such as climatic conditions. Assuming hydrologic conditions persist with time, and assuming the model is truly representative of the natural processes, then that same model can be used to calculate uncertainty propagation.

However, hydrologic conditions may not persist with time. A change in rainfall and thus in infiltration could affect the hydrology of a disposal site situated in the unsaturated zone. A pluvial climate cycle could increase the volume of fluid contacting the repository, which may lead to accelerated canister corrosion and may allow an increased mass of leached radionuclides to dissolve. In addition, the groundwater travel time from the repository to the water table could be shortened. The effect on discharge rates could be approximated using the same steady state flow model by broadening the range of uncertainties inherent in the source term and groundwater flow rate. Discharge rate uncertainty would not only increase, but also the period of growing uncertainty would be extended. Thus, it becomes even more likely uncertainty will increase during the time periods of regulatory interest. For a repository in the saturated zone, the impact of increasing the infiltration would affect primarily the travel time from the repository to the accessible environment.

Analyses to show compliance with the standards generally include simulations of the repository for the specified time period. The simulations incorporate as much information about the repository and surrounding site as possible. Given the information which is known about a site, the simulations may provide results in terms of release to the accessible environment and dose to man.

The individual and groundwater protection standards have been defined in terms of individual dose rates. The dose rates depend upon biosphere transport and dose-to-man pathways. Uncertainties in projection of these transport processes, pathways, and biospheric parameters and variables are substantial. An accepted practice is to assume the present biosphere for analysis purposes. While some limited dose analysis is presented below, the primary focus of the uncertainty analysis conducted was on cumulative radionuclide release. Consequently, the cumulative release analysis results have grossly underestimated uncertainty because dose calculations involve more parameters (e.g., dose pathways, human behavior) than cumulative release analysis. Inherent uncertainty in models and these additional parameters substantially increase uncertainty in dose calculations.

Simulations were conducted to analyze the propagation of uncertainty with time, considering total radionuclide discharge at selected times. Release of several specific radionuclides was analyzed. The results include a common evaluation of uncertainty, the standard deviation of total release as a measure of how uncertainty propagated, and an evaluation of the significance of the uncertainty.

Repository Inventory

For the uncertainty propagation analyses, projected spent fuel inventory for the high-level waste/spent fuel (HLW/SF) repository was used. Table 8-1 lists the expected total curies for 70,000 metric tons of heavy metal (MTHM) for 13 of the significant radionuclides in a potential HLW/SF repository (Reference 8-1). These thirteen radionuclides contain virtually all of the radioactivity contained in the repository. Table 8-1 also shows cumulative release limits contained in Table 1 of EPA's 1985 standard.

Table 8-1. Curies in Proposed Repository

Radionuclide	Ci/MTHM		Total Curies for 70,000 MTHM using EPRI Values	Cumulative Release Limit per 1000 MTHM (Ci)[3] for 10,000 Years
	Wilson [1]	EPRI [2]		
C-14	1.54	1.38	9.66E+04	100
Se-79	3.81E-01	5E-01	3.50E+04	--
Tc-99	1.23E+01	1.3E+01	9.10E+05	10,000
I-129	2.95E-02	3.15E-02	2.21E+03	100
Cs-135	3.51E-01	3.45E-01	2.42E+04	1,000
Ra-226	3.67E-07	3.12E-03	2.18E+02	100
U-234	1.13	2.03	1.42E+05	100
U-235	1.68E-02	1E-02	7E+02	100
U-238	3.18E-01	1E-02	7E+02	100
Np-237	2.87E-01	9.99E-01	6.99E+04	100
Pu-239	3.08E+02	3.05E+02	2.14E+07	100
Pu-240	5.07E+02	4.78E+02	3.35E+07	100
Pu-242	1.60	1.72	1.20E+05	100

1) Reference 8-2
2) Reference 8-1
3) Reference 8-3

Radionuclide Selection

To evaluate uncertainty propagation, it is not necessary to simulate all the radionuclides present in the repository. It is not even necessary to simulate all thirteen of the major radionuclides. However, it is important that the radionuclides selected for simulation be representative and generally cover the range of possibilities.

Three radionuclides were evaluated in the analyses: Technetium-99, Iodine-129, and Neptunium-237. The first two were chosen primarily because of their low retardation characteristics (1-10).

Neptunium-237 was chosen because of its higher retardation characteristics (10-200). The half-lives for these three radionuclides vary from 100,000 to 10,000,000 years.

Generic Site Description

For regulatory analysis, a generic site description must be used as the basis for evaluation. Since the standards are to be applicable to all types of geologic sites, the site description should be representative of potential sites under consideration as well as those which may be considered in the future. Any site description will include a near surface unsaturated zone, a combination of hydrologic layers with varying permeabilities, and source (recharge) and discharge for the hydrologic system. For geosphere transport of radionuclides to take place, the radionuclides must dissolve in water and be transported vertically to an aquifer for subsequent transport to the biosphere. The accessible biosphere is assumed to be at 5 km distance from the repository.

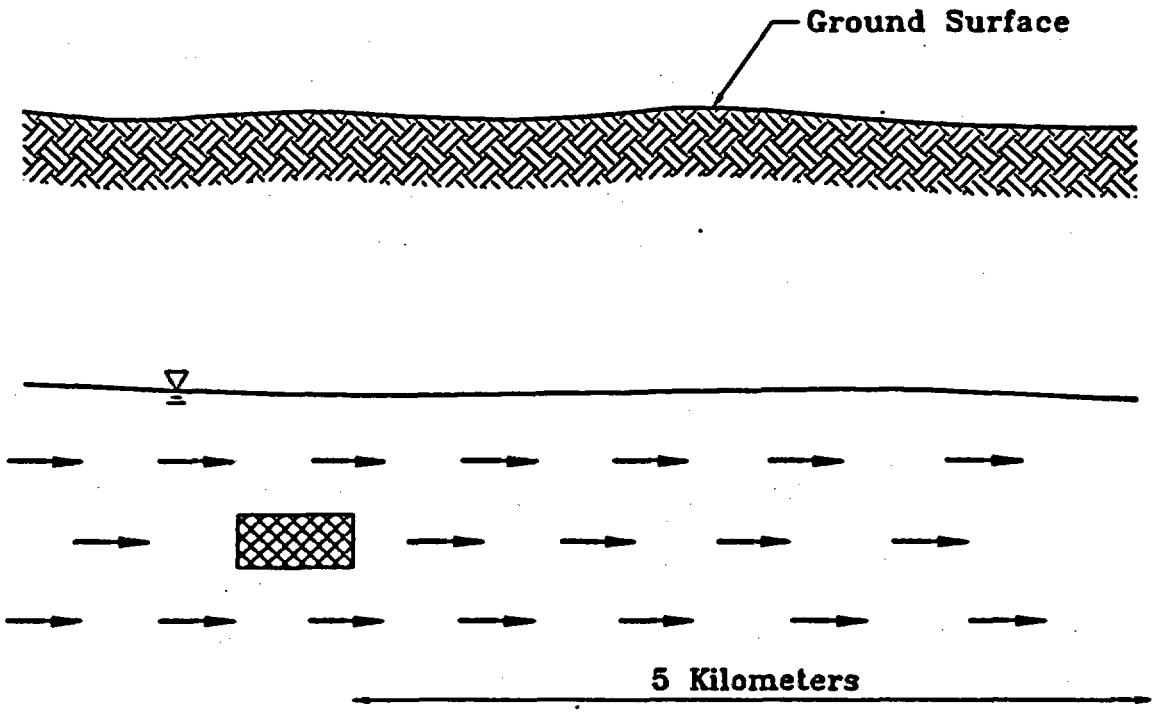
The generic repository is defined as a simplified one-dimensional system. For generic description, convenience, and simplicity, a constant permeability, homogeneous, one-dimensional flow system is assumed. This description does not represent a specific potential site, but can be assumed to represent virtually any site in a simplistic manner. Site specificity comes from differences in hydrologic properties. The one-dimensional site description used here assumes constant thickness and width of the aquifer.

The generic repository was assumed to be in a steady-state, saturated hydrogeologic environment (Figure 8-1). The repository, or source term, was composed of one radionuclide per simulation. The radionuclide was allowed to decay, but daughter products were not accounted for in the analyses. The accessible environment where radionuclide release was accumulated was defined to be 5 kilometers from the repository. Simulations of the repository and transport of the radionuclide to the accessible environment covered 100,000 years. Conditions, such as climate change or human intrusion, were not included in the generic site.

Parameter Description

The major parameters in the analyses included groundwater velocity, retardation of transported radionuclides, alteration rate of radionuclides, and access fraction of radionuclides. The groundwater velocity was specified to provide groundwater travel times within the range of 500 to 50,000 years. Base case retardation values were selected based on generic geologic environments, and were varied about that range. The alteration rate or leach rate of radionuclides specifies the fraction of the radionuclide inventory in the repository which leaches per unit time. The access fraction of radionuclides indicates the fraction of the inventory which is available for transport. For completeness, a dispersivity term was included in the analyses, though it had little effect. Also, the source of the radionuclide (curies), and its half-life were included in the analyses.

The analyses were based on several assumptions. The parameters of groundwater velocity, retardation, dispersivity, alteration rate, and access fraction were assumed to be uncertain with specified probability distributions. The parameter values were considered representative of generic sites. Disturbed conditions, such as human intrusion were not considered. The simulations assumed one isotope was transported in a saturated flow system 5 kilometers to the



LEGEND

- Repository
- Groundwater Flow Direction
- Groundwater Table

Figure 8-1. Schematic of Generic Repository

accessible environment. The simulations were conducted to 100,000 years. Source rate was assumed to be essentially uniform for the transport simulation time. This could be caused by a constant rate of leaching for a long-lived isotope. As noted, only one radionuclide was transported at a time and the decay products of Np-237 were ignored.

Groundwater velocity was assigned a loguniform distribution with the endpoints defined in the following tables on each of the radionuclides (Tables 8-2, 8-3, and 8-4). Over a travel path of 5 km (16,000 ft), the base case range of 0.33 ft/yr to 30 ft/yr gives a range of groundwater travel times from less than 500 years to greater than 50,000 years. For a specific site, uncertainty in groundwater velocity is usually characterized by assigning a lognormal distribution. However, to include all sites in a generic simulation, a loguniform distribution is more appropriate. Such a distribution gives equal weight to each log decade. Thus, there are as many realizations having groundwater travel times from 500 years to 5,000 years as there are from 5,000 years to 50,000 years. Retardation, dispersivity, and alteration rate were varied loguniformly. Access fraction was varied from .01 to 1, also in a loguniform distribution.

The most realistic simulations are those that include probability distribution functions for each parameter in the simulation. Such simulations represent the overall uncertainty in the parameters and thus in the repository system. These simulations indicate whether or not uncertainty grows with time and thus, whether or not the level of confidence in the simulation results changes with increasing time of simulation. For each radionuclide, one such base case simulation (analysis 1) was conducted. Note that in the analyses, the base case is not a single simulation but rather a compilation of a significant number of realizations, so that uncertainty is included in the base case results. Uncertainty propagation simulations for selected parameters were also conducted. In particular, groundwater velocity, retardation, and access fraction probability distribution functions were varied from analysis to analysis in order to determine the effect on total release and thus on the uncertainty with time. Four simulations were conducted for Technetium-99 and Neptunium-237. Only three simulations were conducted for Iodine-129, because its retardation factor of 1 is considered relatively certain.

A summary of the parameter distributions for the Technetium-99 analyses is presented in Table 8-2. The base case simulations are followed by the groundwater velocity, retardation, and access fraction variations.

Table 8-2. Summary of Parameter Distributions for Technetium-99

Analysis	GW Velocity (ft/yr)		Retardation[1]		Dispersivity (ft)		Alteration Rate (1/yr)		Access Fraction	
	Lo	Hi	Lo	Hi	Lo	Hi	Lo	Hi	Lo	Hi
1	.33	30	1	10	100	630	1E-6	1E-4	.1	1
2	.1	100	1	10	100	630	1E-6	1E-4	.1	1
3	.33	30	1	100	100	630	1E-6	1E-4	.1	1
4	.33	30	1	10	100	630	1E-6	1E-4	.01	1

[1] Values from Reference 8-4, p. 65. Range was 1 to 100 for Tuff.

The distributions used in analysis of Iodine-129 are presented in Table 8-3. The base case probability distribution functions are followed by the groundwater velocity and access fraction variations.

Table 8-3. Summary of Parameter Distributions for Iodine-129

Analysis	GW Velocity (ft/yr)		Retardation[1]		Dispersivity (ft)		Alteration Rate (1/yr)		Access Fraction	
	Lo	Hi	Lo	Hi	Lo	Hi	Lo	Hi	Lo	Hi
1	.33	30	1	--	100	630	1E-6	1E-4	.1	1
2	.1	100	1	--	100	630	1E-6	1E-4	.1	1
3	.33	30	1	--	100	630	1E-6	1E-4	.01	1

[1] Reference 8-4

The distributions used in analysis of Neptunium-237 are shown in Table 8-4. As with the Technetium-99, the base case probability distribution functions are followed by the groundwater velocity, retardation, and access fraction variations.

Table 8-4. Summary of Parameter Distributions for Neptunium-237

Analysis	GW Velocity (ft/yr)		Retardation[1)]		Dispersivity (ft)		Alteration Rate (1/yr)		Access Fraction	
	Lo	Hi	Lo	Hi	Lo	Hi	Lo	Hi	Lo	Hi
1	.33	30	10	200	100	630	1E-6	1E-4	.1	1
2	.1	100	10	200	100	630	1E-6	1E-4	.1	1
3	.33	30	5	500	100	630	1E-6	1E-4	.1	1
4	.33	30	10	200	100	630	1E-6	1E-4	.01	1

[1)] Reference 8-4

Codes

NEFTRAN (Reference 8-5), a code developed by INTERA and Sandia National Laboratories for the Nuclear Regulatory Commission, was used for simulating release of radionuclides from a repository. NEFTRAN is a pseudo 3-dimensional flow and transport code. However, the 1-dimensional flow and analytical transport options of NEFTRAN were utilized. The model was linked to a statistical sampling routine to evaluate the uncertainty propagation with time. The values for each of these parameters were selected by the sampling routine, and then 200 realizations of the flow and transport model were evaluated.

Response Variables

The primary response variable for the analyses was total radionuclide release (Ci). The uncertainty of total radionuclide release at a point in time was measured by the sample standard deviation (cumulative). Since the distribution of releases at a point in time is not, in general, normally distributed, the standard deviation cannot be used to calculate percentiles and confidence limits as though the distribution were normal. However, the standard deviation is a valid indicator of the spread (or uncertainty) of the response variable. Furthermore, a second measure of uncertainty (the difference between the 95th and the 5th percentile) exhibits the same behavior as the standard deviation does.

Another response variable for the analyses is annual effective dose equivalent. The standard deviation of such doses were calculated for selected times. By using the NEFTRAN code to calculate the release, concentrations and total discharge as a function of time were obtained. To compare to the groundwater standards, concentration was converted to dose. This was done in three steps:

Step1 - Assume that the primary dose pathway is ingestion. According to ICRP-77 the effective dose commitment by ingestion is:

Tc-99	1.02×10^{-3} rem/*u*Ci = 1.02×10^{6} mrem/Ci
I-129	2.34×10^{-1} rem/*u*Ci = 2.34×10^{8} mrem/Ci
Np-237	4.69 rem/*u*Ci= 4.69×10^{9} mrem/Ci

Step 2 - Assume ingestion of 2 liters of water per day and ignore accumulation. For each radionuclide the effective dose equivalent is found by multiplying the effective dose commitment by the consumption rate of 0.73 m^3/year giving:

Tc-99	7.4×10^{5} mrem-m^3/Ci-yr
I-129	1.7×10^{8} mrem-m^3/Ci-yr
Np-237	3.4×10^{9} mrem-m^3/Ci-yr

Step 3 - The annual effective dose equivalent for year T is the product of the release concentration at time T and the effective dose equivalent. The dose units are mrem/yr.

Results

The results presented show the uncertainty propagation with time for Technetium-99, Iodine-129, and Neptunium-237. The analyses show without exception an increase in uncertainty from 1,000 to 100,000 years.

Each of the three radionuclides analyzed here has a long half-life, compared to the simulation time, and a long source pulse in time. Consequently, for a given radionuclide, each realization has a discharge curve that exhibits three phases. There is no discharge during the first phase, particles have not yet reached the release point. During the second phase, discharge increases from zero to its maximum. The third phase shows a flat (constant) discharge rate equal to the source rate. Uncertainty is measured at a point in time by measuring the spread of discharge from all realizations at that time. During the time that none of the realizations show positive discharge (all realizations are in the first phase), the uncertainty is zero. Since each realization has unique values for groundwater velocity and radionuclide retardation, each will exhibit a different breakthrough time, passing from the first to the second phase. As soon as one realization breaks through, uncertainty begins its increase. The uncertainty continues to grow until most of the realizations have reached their maximum discharge rate. At that point each curve maintains its separation from the other curves and any measure of the spread amongst the curves is constant. Thus, uncertainty starts at zero, increases while most curves are in the second phase, and levels off after most curves have reached the third phase. In the results that follow, the time of increasing uncertainty begins earlier than 1,000 years and begins to level off between 10,000 years and 100,000 years. For a candidate disposal site, the time at which uncertainty begins to increase and the duration of the increase will depend on the physical properties of the disposal site and the chemical properties of the migrating radionuclides.

It is important to remember that the results presented are for individual radionuclides. Any uncertainty in the results would be increased if the complete suite of radionuclides present in the repository were included in the analyses. In addition, if daughter products were included in the

analyses, the uncertainty in the results would increase. Also, if disturbed conditions were included in the analyses, the uncertainty in the results would increase.

Base Case: The base case (analysis 1) for each of the three radionuclides is presented in Figure 8-2. Remember that the base case represents 200 realizations of the given parameter distributions. The magnitude of uncertainty is summarized in Table 8-5. An indication of the increase in uncertainty with length of simulation time is explicitly shown in the base cases. For Technetium-99, the standard deviation of total release in curies increases by nearly 3 orders of magnitude from 1,000 years to 10,000 years. The increase from 10,000 years to 100,000 years is approximately an order of magnitude, and the increase in the value (from 50,000 Ci to nearly 200,000 Ci) is significant. This pattern is repeated often in the results: initial increase in uncertainty by orders of magnitude, followed by continued increase in uncertainty by a significant amount. For Iodine-129, the standard deviation of total release in curies increases by 2 orders of magnitude from 1,000 years to 10,000 years, and nearly an order of magnitude from 10,000 to 100,000 years. The value of the standard deviation of total release is smaller than that of Technetium-99, owing primarily to Iodine-129's significantly smaller source term. For Neptunium-237, the standard deviation of total release in curies increases by 12 orders of magnitude from 2,000 years to 10,000 years, and by over 2 orders of magnitude from 10,000 years to 100,000 years. Note that even though it appears that the increase in uncertainty is slowing at later times, the actual value is significantly higher than early times.

Table 8-5. Summary of Uncertainty in Base Case Analyses

Analysis	Increase in Uncertainty (Orders of Magnitude)	
	1,000 - 10,000 yrs.	10,000 - 100,000 yrs.
Technetium - 99	3	1
Iodine - 129	2	1
Neptunium - 237	12[1]	2

[1]Increase from 2,000 years to 10,000 years

The standard deviations of the annual effective doses at 1,000 years and 10,000 years for the assumed base case (analysis 1) and groundwater velocity variation (analysis 2) are presented in Table 8-6. For the base case analyses, the dose uncertainty increases from 1,000 to 10,000 years for all radionuclides reviewed.

Table 8-6. Annual Effective Dose Uncertainty

Radionuclide	Standard Deviation of Annual Effective Dose (mrem/yr)	
	1,000 years	10,000 years
Technetium-99		
Analysis 1	0.55	7.94
Analysis 2	5.37	9.51
Iodine-129		
Analysis 1	2.46	5.80
Analysis 2	4.07	5.46
Neptunium-237		
Analysis 1	~0	118
Analysis 2	0.0075	1498

For the individual radionuclides, additional results are presented below to show the effect on uncertainty propagation caused by varying a parameter. Groundwater velocity, retardation, and access fraction were varied and the results are presented.

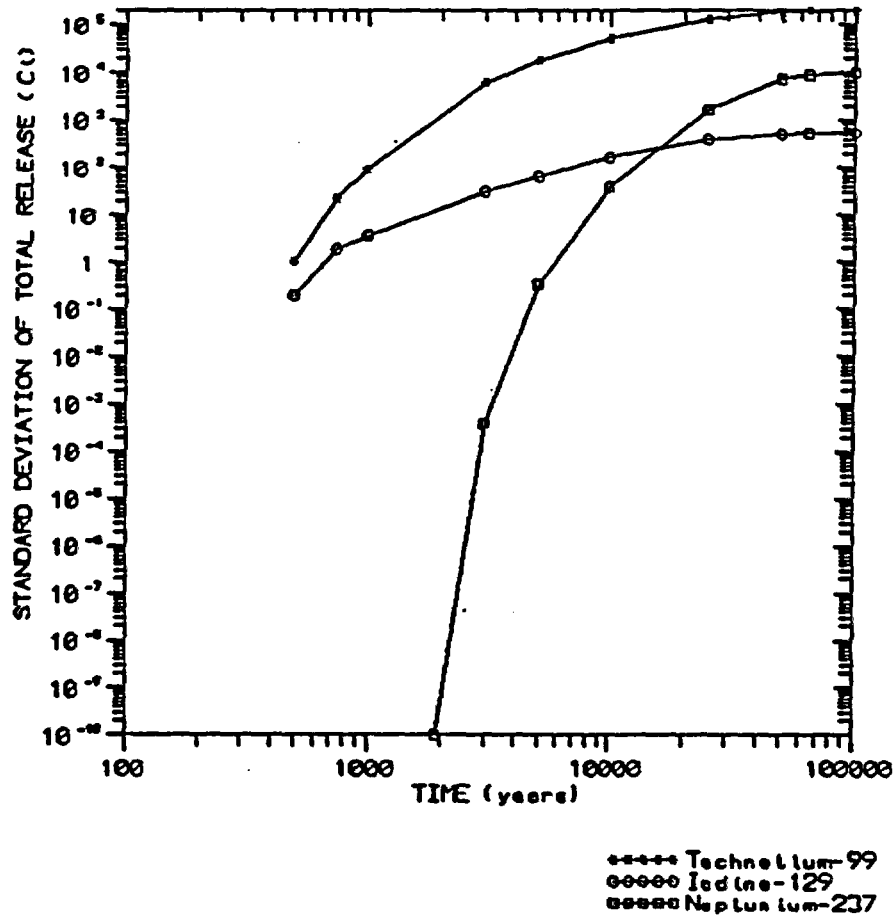


Figure 8-2. Uncertainty Propagation for Base Case Simulations

Technetium-99 (Figures 8-3, 8-4, and 8-5)

The base case simulation is presented on each plot.

Groundwater velocity: Increasing the uncertainty in the groundwater velocity distribution by an order of magnitude increases the standard deviation of the total release by over an order of magnitude at 1,000 years and a smaller amount at 10,000 years (Figure 8-3). By 100,000 years there is not a significant difference between the curves. Also, the dose uncertainty increases when the groundwater uncertainty is increased (Table 8-6).

Retardation: Increasing the uncertainty in the retardation parameter distribution by increasing the highest retardation value an order of magnitude reduces the standard deviation of the total release slightly at 1,000 years as well as at 10,000 years (Figure 8-4). However, the overall uncertainty increases several orders of magnitude from 1,000 to 10,000 years. The simulated response of total release to a larger uncertainty in the retardation parameter value indicates the importance of the endpoints of the range of uncertainty as well as the total order of magnitude of uncertainty.

Access Fraction: The sensitivity of the standard deviation of the total release to a decrease of one order of magnitude in accessible fraction shows a small decrease in the standard deviation of the total release (Figure 8-5). This change may be explained similar to the retardation variation. While the access fraction was more uncertain, only the lower end of the distribution was modified so that the reduction in uncertainty may be explained by the reduction in the availability of the radionuclide for transport. Again, the endpoints of the uncertainty are significant as well as the overall range of uncertainty.

The multiple order of magnitude change observed for Technetium-99 becomes more significant when one considers that the total release of each radionuclide inventory in the repository inventory may have a similar amount of uncertainty. Thus, the uncertainty effects are additive.

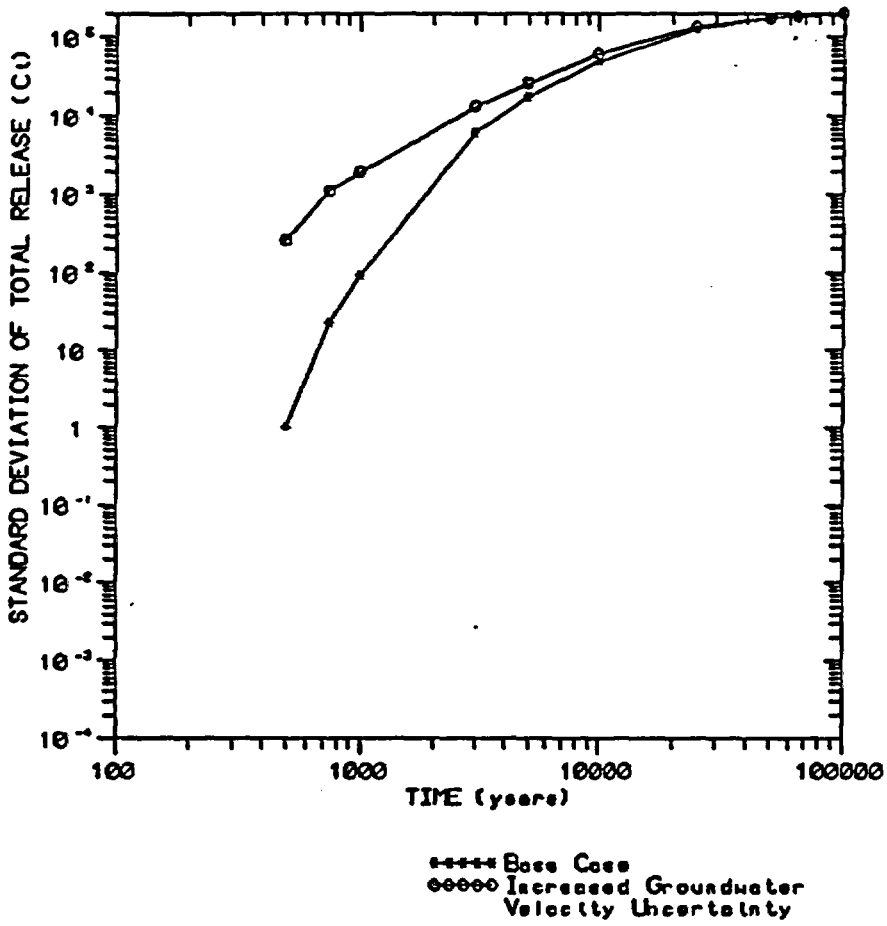


Figure 8-3. Groundwater Velocity Uncertainty Propagation for Technetium--99

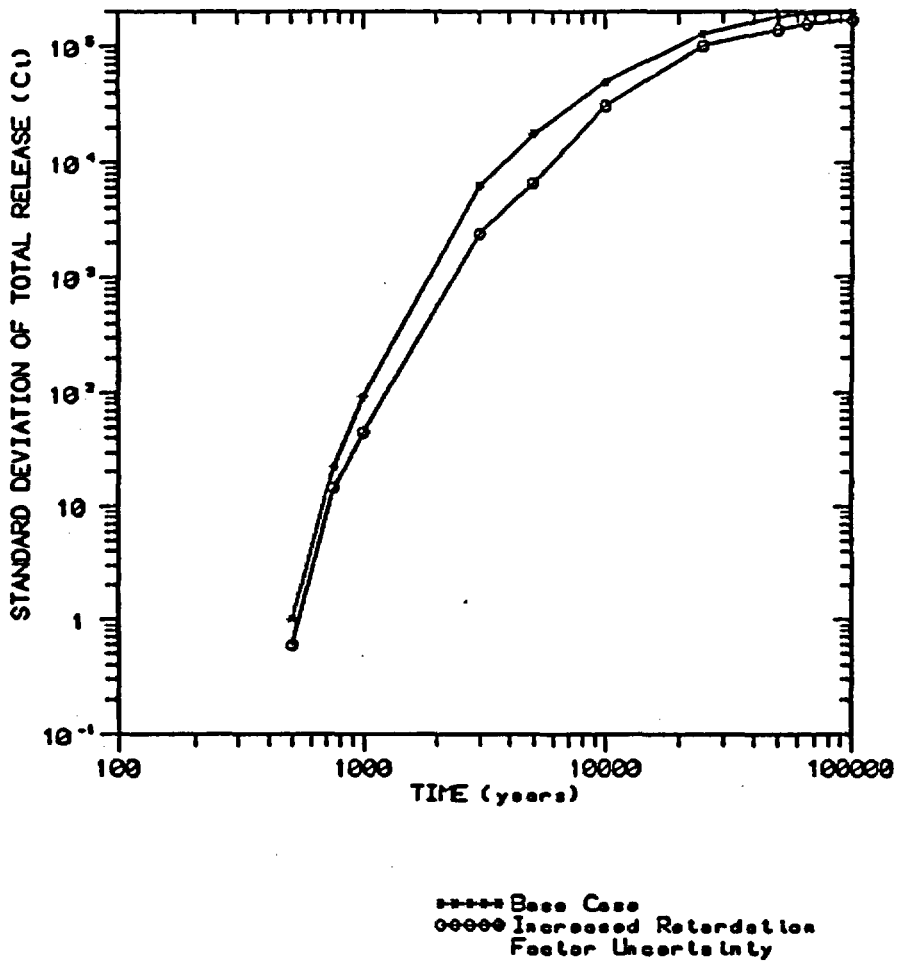


Figure 8-4. Retardation Uncertainty Propagation for Technetium--99

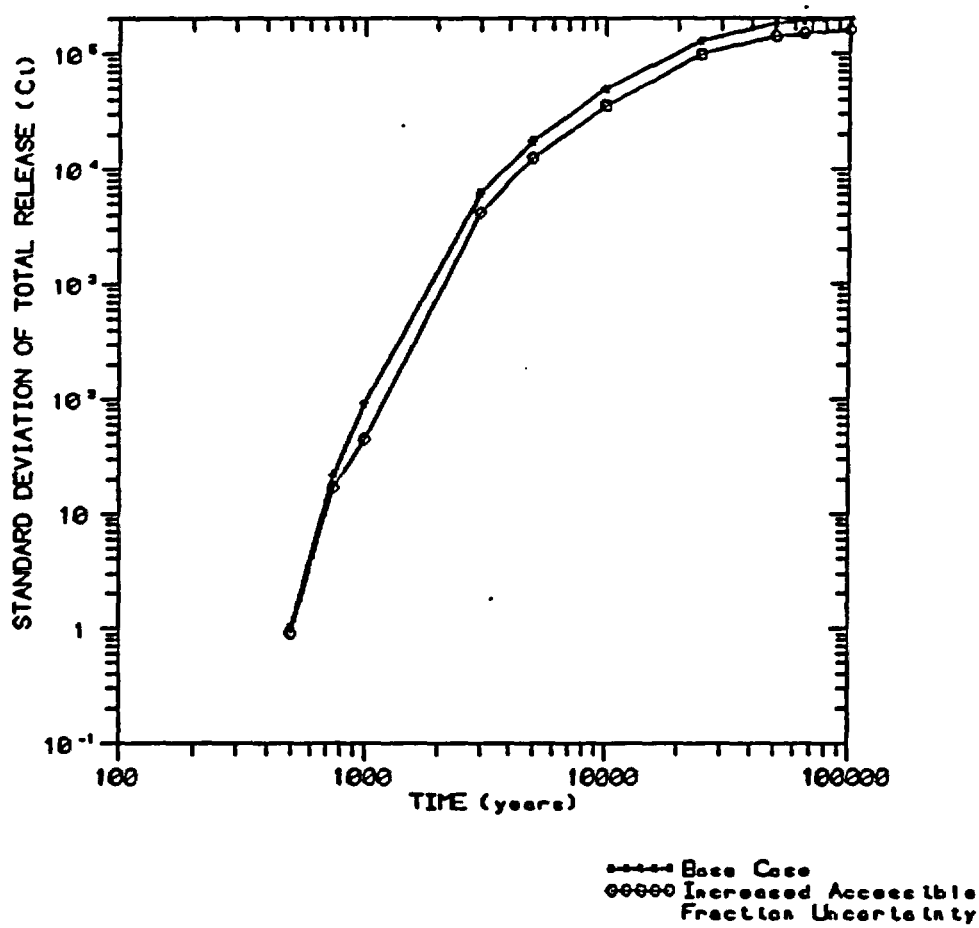


Figure 8-5. Accessible Fraction Uncertainty Propagation for Technetium-99

Iodine-129 (Figures 8-6, 8-7)

The base case simulation is presented on each plot.

Groundwater Velocity: Increasing the uncertainty in the groundwater velocity distribution by an order of magnitude increases the standard deviation of the total release by nearly an order of magnitude at 1,000 years and a smaller amount at 10,000 years (Figure 8-6). By 100,000 years, the curves have converged similar to the Technetium-99 curves. Again, the dose uncertainty increases when the groundwater velocity uncertainty is increased (Table 8-6).

Retardation: The retardation of Iodine-129 is 1 so this parameter was not varied.

Access Fraction: Increasing the uncertainty range by decreasing the lower limit of the distribution of the access fraction results in a small decrease in the standard deviation of the total release (Figure 8-7). The simulation with a low end value of .01 shows a slightly smaller uncertainty than the simulation with a low end value of .1. However, the standard deviation grows nearly 2 orders of magnitude from 1,000 years to 10,000 years.

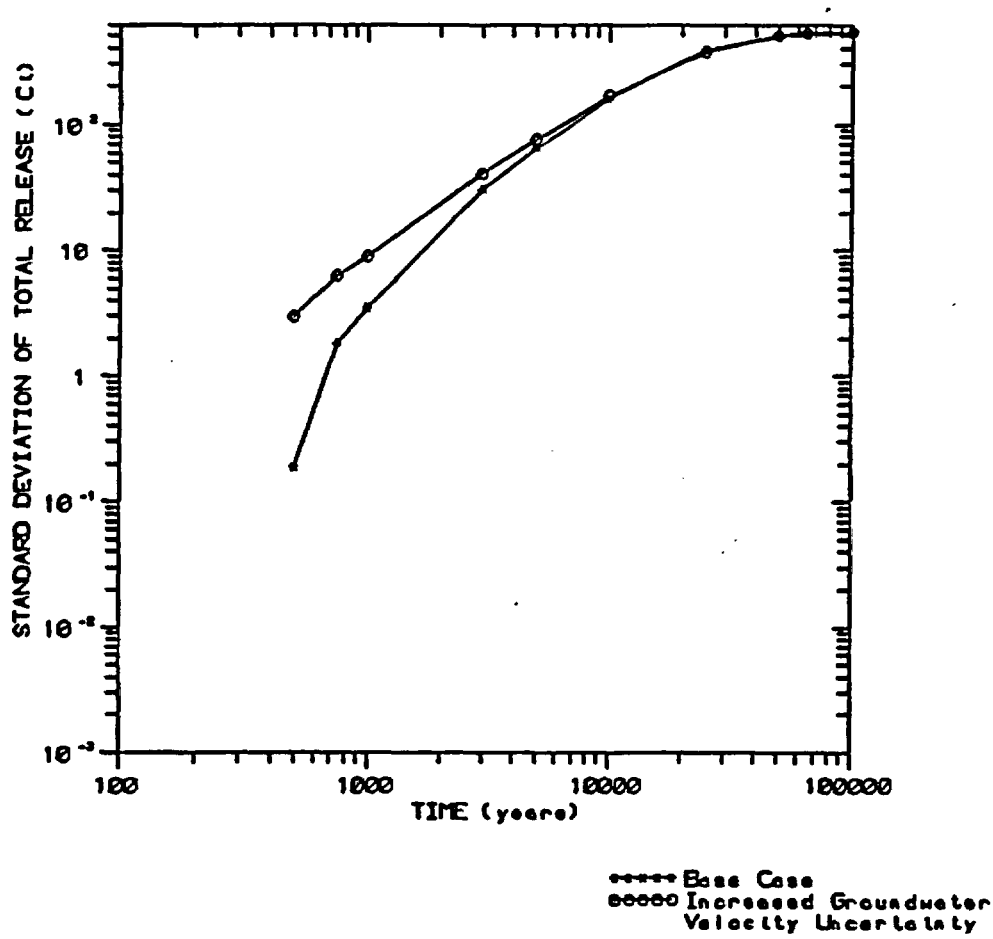


Figure 8-6. Groundwater Velocity Uncertainty Propagation for Iodine-129

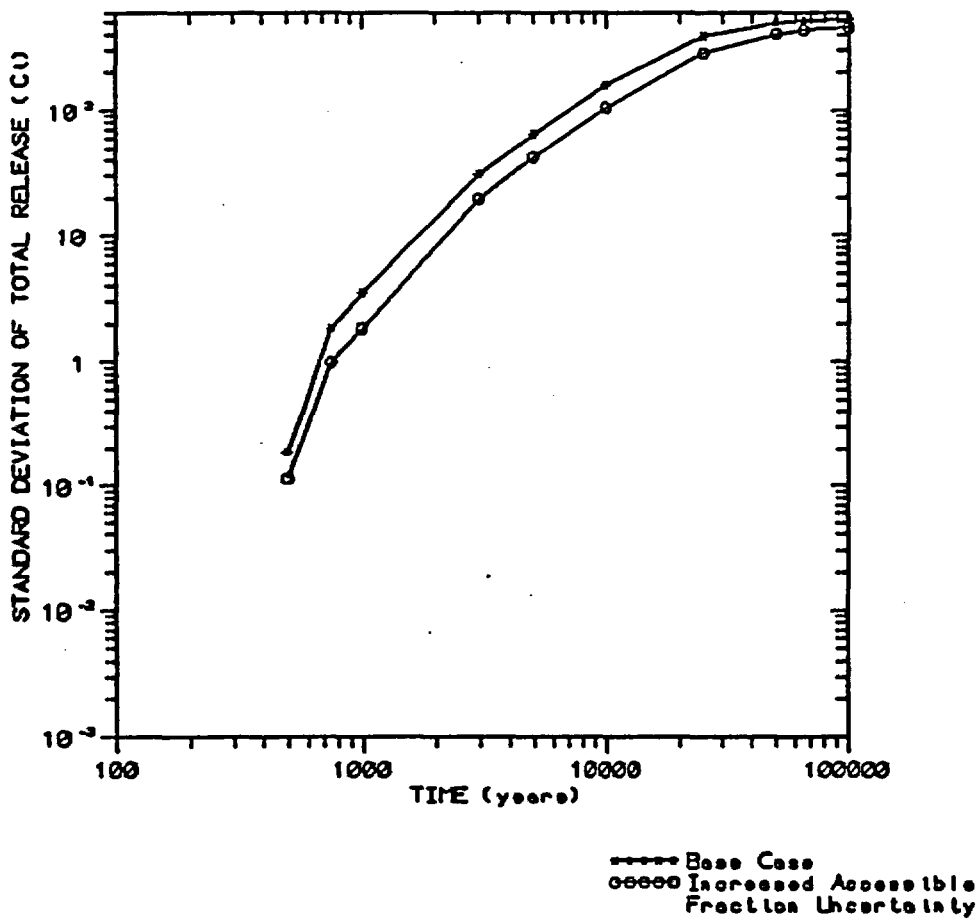


Figure 8-7. Accessible Fraction Uncertainty Propagation for Iodine-129

Neptunium-237 (Figures 8-8, 8-9, 8-10)

The base case simulation is presented on each plot.

Groundwater Velocity: Increasing the uncertainty in the groundwater velocity distribution by an order of magnitude increases the standard deviation of the total release by 9 orders of magnitude at 2,000 years and by over 1 order of magnitude at 10,000 years (Figure 8-8). By 100,000 years, the total release is not sensitive to groundwater velocity as indicated by the convergence of the curves. However, there is considerable uncertainty in the total release as shown by the large values of the standard deviation. Likewise, the dose uncertainty increases significantly from 1,000 to 10,000 years (Table 8-6).

Retardation: Increasing the uncertainty in the retardation of Neptunium-237 by one order of magnitude produces an increase in the total release standard deviation at 2,000 years of 6 orders of magnitude and nearly an order of magnitude difference at 10,000 years (Figure 8-9). Both ends of the distribution of the retardation parameter were modified, unlike the Technetium-99 and Iodine-129 analyses. Thus, much of the increase in uncertainty may be due to the reduction of retardation caused by lowering the endpoint of the distribution from 10 to 5.

Access Fraction: Increasing the uncertainty in the access fraction distribution does not affect the total release at early times and only slightly affects the results at times after 10,000 years (Figure 8-10). However, the standard deviation grows 11 orders of magnitude from 2,000 to 10,000 years, and over 2 orders of magnitude from 10,000 to 100,000 years.

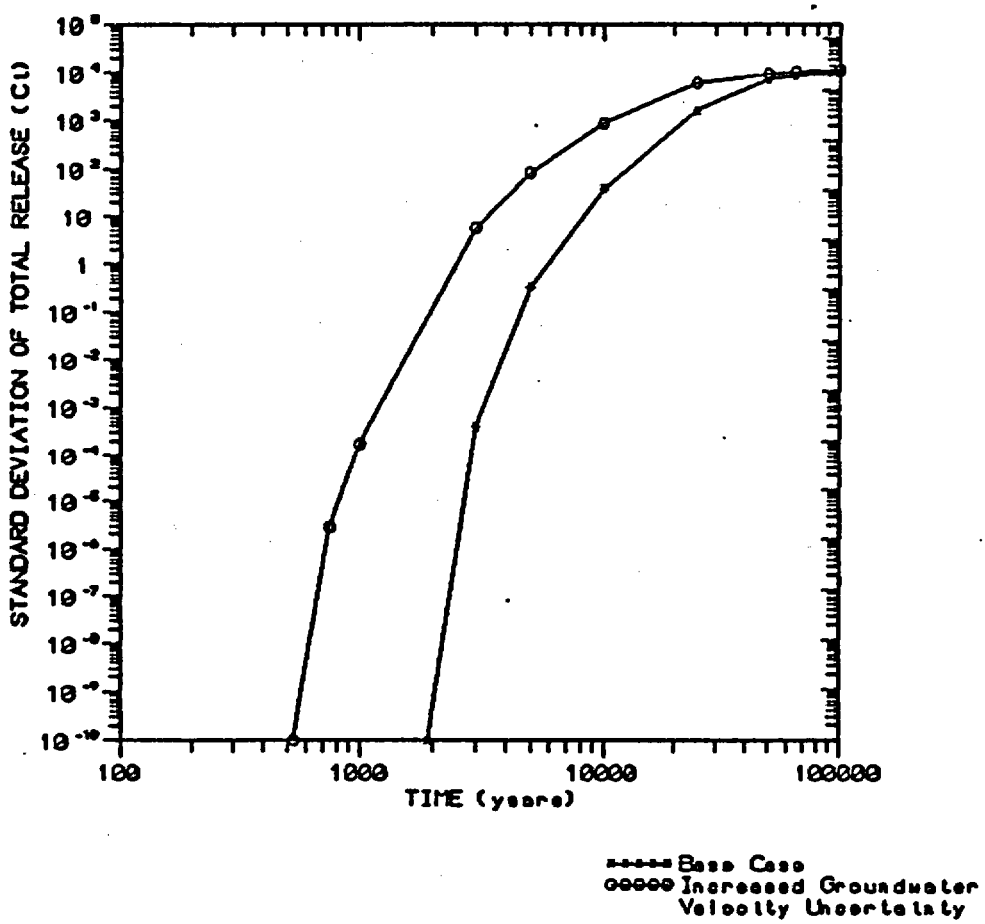


Figure 8-8. Groundwater Velocity Uncertainty Propagation for Neptunium-237

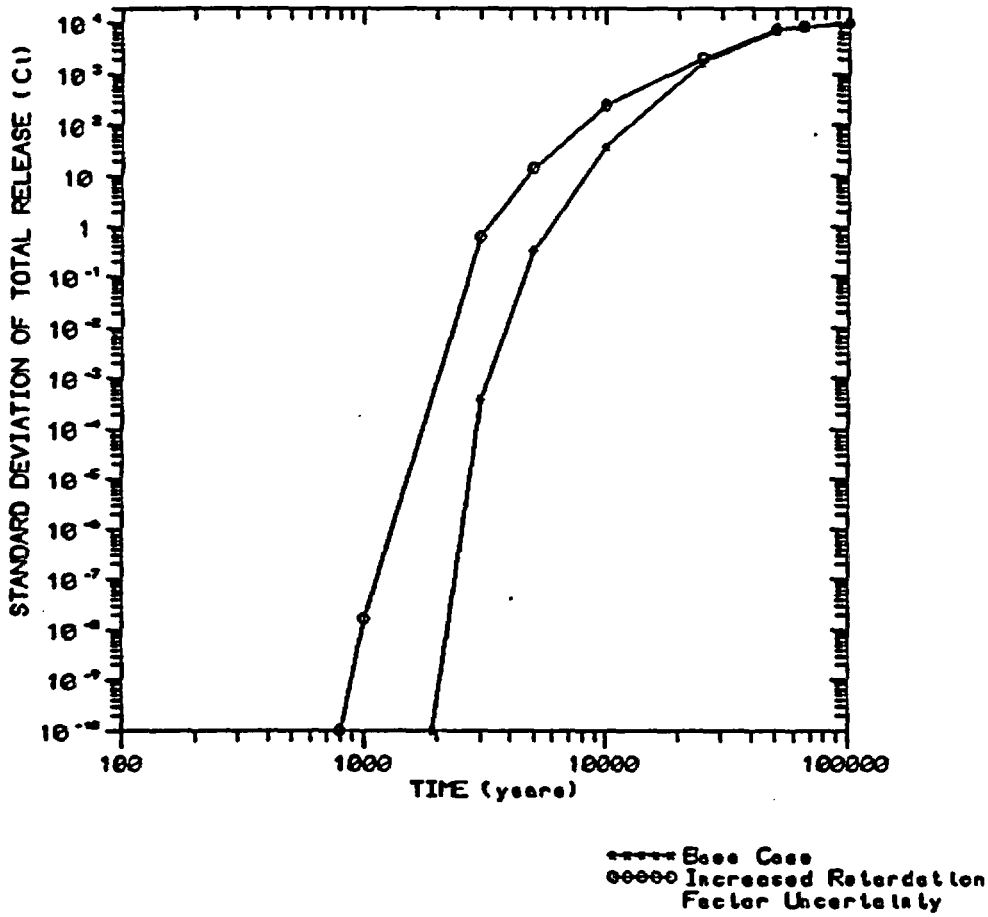


Figure 8-9. Retardation Uncertainty Propagation for Neptunium-237

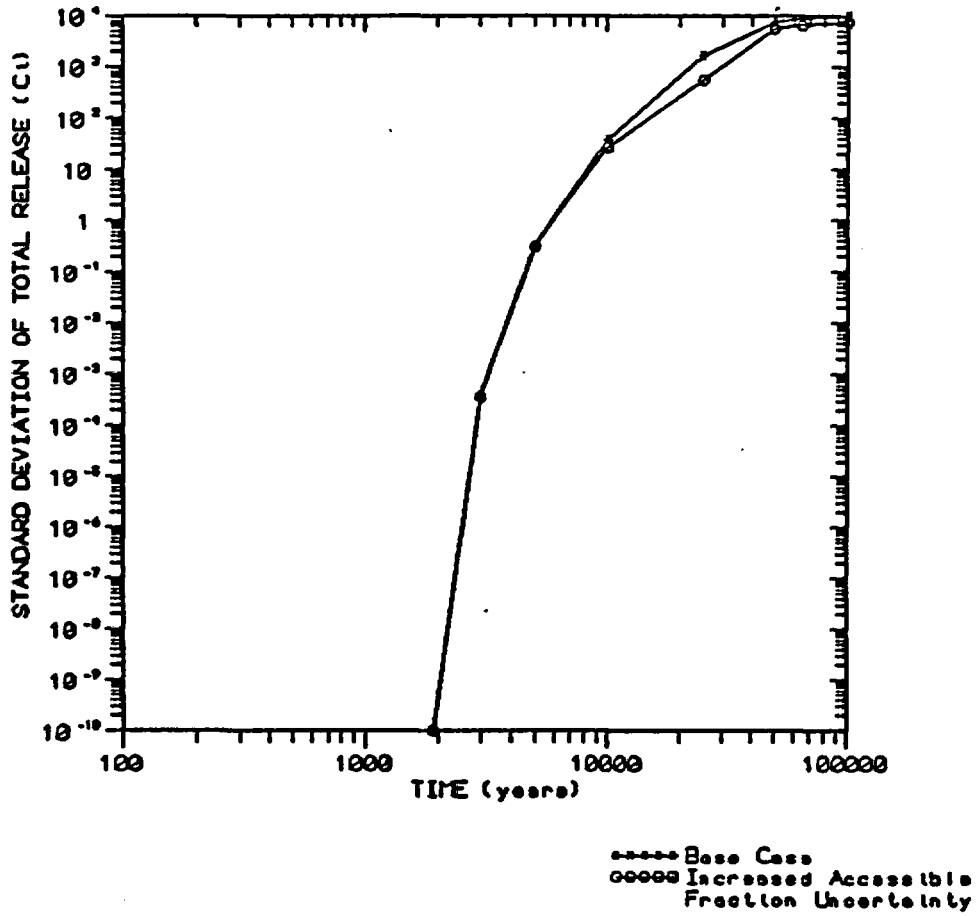


Figure 8-10. Accessible Fraction Uncertainty Propagation for Neptunium-237

Demonstration of Compliance

The individual and groundwater protection limits in the 1985 standard and in the draft revised standards are presented in terms of individual dose rates. As stated previously, only a portion of the radionuclide inventory was used in these analyses and all the results are interpreted in terms of cumulative release. However, some results are also presented in terms of annual effective dose equivalent.

Table 8-6 summarizes the propagation of uncertainty with time as it relates to the calculation of doses that were generated using the assumptions discussed earlier in this chapter. The calculated values are an extremely conservative (i.e., low) estimate of uncertainty in dose as critical parameters were not modeled. Areas of that would contribute to uncertainty in any actual compliance demonstration that were ignored in these analyses include transport mechanisms, climate changes, gradual changes in geologic parameters, and human behavior. One key assumption that greatly reduced the uncertainty in these calculations is the assumption that the model is representative of actual site conditions. Even if the uncertainties appear low in absolute terms, their real significance lies in the percentage of the allowable dose rate that these uncertainties represent. Under the individual protection requirements in the 1985 standard, the annual individual dose equivalent is limited to 25 mrem. Under the groundwater protection requirements of the same standard, the limit is 4 mrem/yr. As shown in Table 8-6, the standard deviation at 1,000 years for the base case (analysis 1) for Technetium-99 and Neptunium-237 is a small percentage of the EPA limits. On the other hand, the standard deviation for Iodine-129 at 1,000 years is more significant when compared to these limits. Uncertainty in dose increases from 1,000 to 10,000 years, particularly for Neptunium-237. The dose uncertainty at 10,000 years is quite large in comparison to the standard for all three radionuclides. Furthermore, the other sources of uncertainty that were not considered in the analysis would increase the uncertainty even more.

Summary/Conclusions

This work was done to analyze uncertainty propagation in order to examine various time periods for the individual and groundwater protection standards. Unquestionably, uncertainties exist through the waste isolation system and grow with time. For the three radionuclides evaluated, simulations of a simple, generic repository show that the uncertainty propagation with time is significant. In particular, the analyses show considerable uncertainty in the total curies released, and resulting doses. The total uncertainty continues to grow in all circumstances to the end of the simulation period at 100,000 years. They grow so large at 10,000 years that demonstrating compliance with the standard is meaningless. The assumptions used in these analyses made the modeling exercises extremely simple when compared to those that would be required in any actual compliance demonstration. The uncertainties resulting from these simple modeling simulations grossly underestimate the uncertainties that would result had the full range of model parameters and values been utilized. Any actual compliance demonstration for a dose limit at 10,000 years would be totally speculative . The uncertainty band would be so large that it would be impossible to determine any defensible endpoint. Therefore, based on uncertainty propagation analysis, the time period for regulatory concern for the individual and groundwater protection limits should be set at no more than 1,000 years.

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CHAPTER 9

CARBON-14

9.1 STATEMENT OF THE PROBLEM

The purpose of 40 CFR Part 191 is to protect public health and safety. The 1985 rule was developed on the basis of the assumption that the repository would be located in a geologic formation that lies below the water table. It was therefore assumed that the principal mechanism of pollutant migration would be via dissolution of radionuclides in groundwater and transport by aqueous means.

We now find the nation examining the suitability of unsaturated sites, specifically Yucca Mountain, a site that is located above the water table. At this site, and other unsaturated sites, it is appropriate to examine gaseous release and transport of pollutants in order to determine site adequacy. When the provisions of the 1985 standard are applied to Yucca Mountain, specifically the limits for Carbon-14, we can release in 10,000 years no more than 7,000 curies of Carbon-14 in the form of carbon dioxide. Meanwhile, calculations indicate that the repository may release about 8,000 curies of Carbon-14 dioxide, an amount that exceeds the standard by 10 to 20 percent.

For the first 1,000 to 2,000 years after the repository is closed, it is expected that the host rock will contain the Carbon-14 dioxide. For containment for longer periods of time, we must rely on a durable waste package, one utilizing a multiple-layer design. Such an approach could be very costly. Estimates indicate the repository program cost would increase by approximately $3.2 billion if the multiple-layer waste package is required.

The basis of the 1985 standard was that, in a site below the water table, the limit for Carbon-14 was technically achievable. It was not a standard based on a release level that would prevent a danger to public health. If we examine the danger to public health of the release of 8,000 curies of Carbon-14 dioxide during an 8,000-year period, this release would not pose a significant threat to public health. Industry and natural sources release many times this amount of Carbon-14 dioxide each year. The question therefore becomes: is it appropriate to spend an additional $3 billion on waste packages when this will not provide an improvement in public health?

A situation exists in which the 1985 rule has an unintended result. It appears that a potential repository at Yucca Mountain can release its inventory of Carbon-14 dioxide without endangering public health, yet the site may not be able to satisfy a standard that has as its ultimate purpose the protection of public health. Thus, an alternative approach is needed. The EPA should regulate Carbon-14 dioxide under a more equitable standard, similar to those in the clean air regulations, or not regulate it at all.

9.2 RECOMMENDED APPROACH

The following material suggests an alternative method of regulating gaseous releases from the repository. The containment requirements, expressed as curies/1,000 MTHM, would apply only to solid and liquid releases to the land, a well, a river, and the ocean (see Chapter 5). The individual protection requirements, expressed as millirems/year, would continue to apply to all releases through all pathways. However, exposures from radioactive gases cannot exceed 10 millirems/year.

The following is a possible revision of subsection 191.13(a) of the 1985 standard:

> 191.13 Containment requirements.
>
> (a) Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation, based upon performance assessments, that the cumulative releases of radionuclides in the solid and liquid phases to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall:
>
> (1) Have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix A); and
>
> (2) Have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 (Appendix A).

The following is a possible revision of Section 191.15 of the 1985 standard:

> 191.15 Individual protection requirements.
>
> (a) Disposal systems for radioactive wastes shall be designed to provide a reasonable expectation that, for 1,000 years after disposal, undisturbed performance of the disposal system shall not cause the annual committed effective dose received through all potential pathways from the disposal system to any member of the public in the accessible environment to exceed 25 millirems (250 microsevents). The annual committed effective dose for gases released through the atmospheric pathway shall not exceed 10 millirems.

9.3 SUPPLEMENTARY INFORMATION

The following material explains the basis for the revisions suggested in the preceding pages. This material could be used by the EPA as part of the supplementary information for the proposed rule.

Besides the remand from the First District Court of Appeals, much has transpired since the Agency issued its standards in September, 1985, that has led us to reconsider our containment and individual protection requirements. Congress amended the Nuclear Waste Policy Act (Act); the Agency proposed and issued new clean air regulations (40 CFR Part 61); and the U.S. Department of Energy (DOE) has begun to characterize an unsaturated site.

The Act directed the Agency to issue generally applicable standards, and the amended Act directed the DOE to characterize only Yucca Mountain, an unsaturated site. We issued our standards after the Act was passed but before the Act was amended. At that time, saturated sites were the leading contenders for a repository. Consequently, our containment requirements were not intended to control gases that would be released through fractures in unsaturated rock.

Information developed by the DOE and others indicates that, when applied to gases, namely Carbon-14 dioxide, the containment requirements become overly stringent - millions of times more stringent than the clean air regulations. The stringency would not affect a saturated repository, but would discourage the development of any unsaturated repository. Thus, to keep our standards generic and consistent with other regulations, the Agency proposes these changes.

After considering these developments, we propose to change the requirements. The containment requirements would apply only to solid and liquid releases to the land, a well, a river, and the ocean. The individual protection requirements would continue to apply to all releases from an undisturbed repository through all pathways, but now exposures from radioactive gases cannot exceed 10 mrem/year. Without these changes, the standards would not be generic, they would not be consistent with the clean air regulations, and the standards could force the DOE to needlessly spend billions of dollars.

The Agency proposes to regulate solid and liquid releases under the containment requirement and regulate gases in a manner that is consistent with our National Emissions Standards for Hazardous Air Pollutants (NESHAP) (40 CFR Part 61). In developing NESHAP, we found that a maximum individual dose of 10 millirems per year (mrem/yr) provides an ample margin of safety. We now propose this same dose limit for a repository. The dose would appear in our individual protection requirements along with the current 25 mrem/yr limit that an individual could receive through all pathways.

Even though these changes could potentially allow approximately 8,000 curies of Carbon-14 dioxide to be released over a 10,000-year period, such a release does not pose a significant threat to public safety. If the 8,000 curies were released in just one year, an individual would be exposed to less than 0.5 mrem. During the same year, this individual would

receive 300 mrem from natural background radiation and 1.3 mrem from the Carbon-14 within his own body.

Without the above revisions, DOE would be forced to design and fabricate an overly expensive waste package to completely contain the 8,000 curies of Carbon-14 dioxide. Complete containment does not make sense when Carbon-14 dioxide is routinely released throughout the world. A typical nuclear power plant releases, without any restriction, about 24 curies of Carbon-14 each year and a typical reprocessing plant about 860 curies. But under the 1985 standard, a repository filled with 70,000 MTHM can average a release of no more than 0.7 curies/year. If just 3 waste packages fail in 1 year, about 1 curie of Carbon-14 dioxide will be released.

The more durable Carbon-14 package could cost $213,000 each, or $5.3 billion for the 25,000 packages that will be needed. The DOE is considering several designs, such as thick-walled packages and multi-layered packages with either metallic or ceramic inserts. The fabrication of these more conservative packages will need development, particularly those made of ceramic materials. The DOE believes that ceramics are feasible but development will be difficult. For example, a hot isostatic press must be designed and constructed to remotely fuse the ceramic around the spent fuel assemblies. With an additional $100 million for research and development, the Carbon-14 packages cost a total of $5.4 billion.

The DOE's present reference waste package could cost $88,000 each or $2.2 billion for 25,000. Fabricated from a corrosion-resistant alloy, these packages may provide substantially complete containment for 1,000 years, but the DOE cannot guarantee that they will contain the radioactive gases for 10,000 years.

The difference between these two types of waste packages, $3.2 billion, constitutes the cost of meeting the current (1985) limits for Carbon-14 dioxide. Stated another way, the DOE must spend $400 million to contain 1 curie of Carbon-14 dioxide, while the world's industries release thousands of curies each year. The Agency finds that the negligible benefits to public safety do not justify the high cost. We therefore propose to exclude gases from our containment requirements and regulate them under the more equitable individual dose limits of 10 mrem/yr. A cost-benefit analyses follows.

The NRC requires applicants to employ "reasonably demonstrated technology" that can reduce, in a cost-effective manner, a population's exposure to radiation. A population's exposure to radiation, called collective dose, is expressed in person-rems. The NRC values a "favorable cost-benefit ratio" at $1,000 per person-rem (Reference 9-1). Most utilities use a higher ratio; the DOE uses $10,000 to $15,000 per person-rem.

A cost-benefit analysis of the reduction of world population exposure for 10,000 years would appear to be consistent with the objectives of our HLW standards. Here the collective dose is taken to mean the world's exposure to Carbon-14 dioxide. The cost, $3.2 billion, would reduce 10 billion persons' exposure to Carbon-14 dioxide by 8,000 curies. Over a 10,000-year period, each curie of Carbon-14 dioxide would expose the world to 400

person-rems (Reference 9-2). Thus the cost-benefit ratio is $3.2 billion / 8,000 curies x 400 person-rem per curie or $1,000/person-rem.

Despite the Agency's objective to protect 10 billion people for 10,000 years, a cost-benefit ratio with the same objective is meaningless. The cost is diluted by a high collective dose. Composed of tiny doses over thousands of years to billions of people, this collective dose grossly overstates the risk and thereby makes expensive but trivial benefits appear cost-effective. As stated by the NRC in the Below Regulatory Concern Policy Statement, "As a practical matter, consideration of dose rates in the microrem per year range and large numbers of hypothetical individuals potentially exposed ... may unduly complicate the dose calculations.... The Commission believes that inclusion of individual doses below 0.1 mrem per year (0.001 mSv per year) introduces unnecessary complexity into collective dose assessments and could impute an unrealistic sense of the significance and certainty of such dose levels." (Reference 9-3) The National Council for Radiation Protection sets the collective dose lower cut-off limit at $\leq$ 1 mrem/yr. (Reference 9-4).

More traditional analyses confine the collective dose to a local population. Often called ALARA or as low as reasonably achievable, these analyses must be completed by most NRC applicants and licensees (Reference 9-5). Here the collective dose is taken to mean a "population reasonably expected to be within 50 miles of the [repository]" (10 CFR 50, Appendix I). Approximately 12,000 people live within 50 miles of Yucca Mountain (Reference 9-6). We conservatively assume that the 8,000 Curies of Carbon-14 dioxide exposes each of the 12,000 people to the same radiation dose that the maximally exposed individual would receive (0.5 mrem or 0.0005 rems). Thus the cost/benefit ratio is $3.2 billion / 12,000 persons x 0.0005 rems or $533 million/person-rem.

We prefer the more traditional ALARA-type analysis. While not totally accurate or equitable, this analysis at least gives a cost-benefit ratio that can be compared to an industrial baseline.

No nuclear industry has ever been compelled to spend $533 million to reduce a collective dose by one person-rem. Moreover, the collective dose is caused by a radionuclide that the world's industries freely and routinely release. The Agency therefore finds that it is not cost-effective to contain Carbon-14 dioxide for 10,000 years.

9.4 TECHNICAL SUPPORT DOCUMENTATION

The following material is information that could be cited in support of the above revisions. It could be part of a technical support document for the rule.

Carbon-14 Inventory

By law, the repository can hold no more than 70,000 metric tons of heavy metal (MTHM). Approximately 7,000 MTHM comprises defense waste which contains little or no Carbon-14. The remaining 63,000 MTHM comprises spent fuel from nuclear power plants; approximately 60% from pressurized water reactors and 40% from boiling water reactors. With the above assumptions, Carbon-14 inventories can be estimated.

Based on nitrogen impurities and experimental data, Van Konynenburg estimates the total Carbon-14 inventory at 71,000 curies (Reference 9-7). Park adjusted this estimate to account for a higher spent nuclear fuel burn-up and reports 78,000 curies (Reference 9-8).

The literature reports that one to ten percent of the Carbon-14 inventory can be rapidly released as Carbon-14 dioxide. The one percent value (Reference 9-9) is probably too low and the ten percent value (Reference 9-10) may not be the upper bound. The term "rapid release" means that the Carbon-14 dioxide escapes immediately after the waste container fails. The rate at which the gas escapes has been investigated (Reference 9-11), but not determined. The rapid release fraction is assumed to be 8000 Ci as a maximum value.

The remaining Carbon-14 will gradually oxidize and reach the accessible environment. Some or all may escape as a gas; some or all may dissolve and escape in the ground water; and some or all of the Carbon-14 dioxide may partition between the gaseous and aqueous phases (Reference 9-12). Given these uncertainties, performance assessments completed by the NRC staff (Reference 9-13) and the DOE (Reference 9-14) have not attempted to model the gradual release fraction of Carbon-14. However, even if these 70,000 curies of carbon-14 are ignored, the other 8,000 curies (i.e., the rapid release fraction) dominates all other releases combined (Ibid).

Cost to Contain Carbon-14 Dioxide

Containment of Carbon-14 dioxide, or any other radioactive gas, requires a multi-barrier waste package concept with, at least, one of the barriers utilizing a material that has very low corrosion characteristics. The DOE is currently considering robust waste packages to increase design margins, but DOE is not specifically addressing Carbon-14 containment. This evaluation attempts to quantify the additional costs of developing and manufacturing such a containment without a determination of its technical feasibility, which can come only after considerable research and development.

Using a statistical model to calculate the cumulative failure distribution for high-level radioactive waste containers, Bullet (Reference 9-15) shows that multiple-barrier systems have the potential to delay the failure of waste packages depending on the choice of each barrier material. A multi-barrier approach was assumed for the Carbon-14 containment cost evaluation, with one

barrier utilizing a ceramic material known to have very low corrosion rates. Other barriers would be similar to the reference design described in the Site Characterization Plan (SCP) allowing the cost evaluation to focus on added costs to contain Carbon-14 within a ceramic barrier.

The selection of ceramics infers a requirement for considerable research and development (R&D) to develop the data, processes, and equipment necessary to produce this material and predict its performance. The consensus of the Engineered Barrier System Concepts Workshop (Reference 9-16) regarding use of ceramics was that their feasibility was undetermined because of the current lack of appropriate data on these materials. An R&D program for ceramics costing $10-15 million per year out to license application in the year 2001, totaling $80-100 million, would be necessary to generate the performance data and develop the manufacturing processes (see Table 9-1). These costs would be in addition to the currently estimated costs of developing the reference waste package. Currently, no facility in the U.S. can fabricate a ceramic large enough to hold the spent fuel. Moreover, the DOE would have to build a facility to remotely encapsulate the spent fuel within the ceramic.

For this cost evaluation, it was assumed that the Carbon-14 package would contain the same amount of waste as the reference design, so that direct comparisons can be made. This design contains three PWR and four BWR spent-fuel assemblies. Approximately 25,000 waste packages would be required to accommodate the first repository inventory. Larger concepts are currently being evaluated that could reduce the number of packages, but this effort has not proceeded far enough to provide a basis for comparison.

The Carbon-14 package, defined for this evaluation, is based on an external metallic barrier and an inner second barrier of alumina or titania ceramic to contain Carbon-14. Inside the ceramic, a steel handling canister would hold the spent fuel. Alloy 825 is assumed for the outer container because cost data are available for it (Reference 9-17). The diameter of this external container must be increased over the reference design to accommodate the ceramic barrier. The ceramic barrier would be approximately 3 inches thick, and the steel canister would be 0.39 inch thick.

Cost estimates for the ceramic barrier in the size needed are not readily available, because these sizes are larger than what is currently manufactured. However, it is the opinion of ceramic researchers and manufacturers that a ceramic container of the size needed would have costs comparable to the corrosion-resistant high-nickel alloy container being considered for the metallic barrier. The cost of 25,000 ceramic packages plus R&D totals $5.4 billion. The cost of 25,000 reference packages plus R&D totals $2.2 billion. The difference, $3.2 billion, constitutes the cost of containing Carbon-14 dioxide (see Table 9-2).

Other Information

Many technical analyses and evaluations regarding Carbon-14 have been done by the DOE, its contractors, national laboratories, and others. These have included analyses of the source term, transport mechanisms, health effects, uncertainties, as well as evaluation of the regulatory implications concerning releases of Carbon-14. Appendix A of this document contains a paper written by Dr. U-Sun Park, of Science Applications International Corporation, that discusses these various aspects. This paper was prepared in support of the workshop on 40 CFR Part 191 sponsored by the Electric Power Research Institute in February 1992.

Table 9-1. Ceramic Research and Development Costs Leading to License Application, $1,000

ACTIVITY	FY 93	FY 94	FY 95	FY 96	FY 97	FY 98	FY 99	FY 00	FY 01	TOTAL
Design	500	1000	1000	1000	1000	1000	1000	1000	1000	8500
Process Development	200	500	1500	2000	2000	2000	2000	2000	2000	14200
Fabrication Equipment Development	50	100	500	2000	2000	2000	2000	2000	2000	12650
Nondestructive Exam (NDE)	50	100	200	200	200	200	200	200	200	1550
Material Characterization & Test	200	1000	2000	4000	4000	4000	4000	4000	4000	27200
Prototype Fabrication & Test	50	100	500	1000	2000	2000	2000	5000	5000	17650
TOTAL	1050	2800	5700	10200	11200	11200	11200	14200	14200	81750

Table 9-2. Carbon-14 Containment Costs, $

CONCEPT	PACKAGE QUANTITY	BARRIER NUMBER	BARRIER MATERIAL	UNIT COST	ASSEMBLY COST	TOTAL COST
SCP	25000	1	ALLOY 825	83000	5000	88000
C-14 Package	25000	1	ALLOY 825	95000	2000	97000
		2	CERAMIC	75000	5000	80000
		3	STEEL	31000	5000	36000
		TOTAL C-14		201000	12000	213000
Added Costs Over SCP Per Package				118000	7000	125000
Added Cost for Research and Development						1.E+08
Total Added Cost to Program						3.2E+09

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APPENDIX A

A COMPARISON OF THE 1985 EPA STANDARD WITH THE "THREE BUCKET" APPROACH

This information is supplied as reference material only.

A Comparison of the 1985 EPA Standard with the "Three Bucket" Approach

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July 27, 1992

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1 Introduction

The U. S. Environmental Protection Agency (EPA) is currently reconsidering its 1985 radioactive-waste standard, 40 CFR Part 191, because it was partially remanded by a U. S. district court. Although the part of the standard regulating releases of radioactivity to the environment (the "containment requirements") was not part of the reason that the court ordered the remand, the entire standard is being reconsidered. The staff of the U. S. Nuclear Regulatory Commission (NRC) have proposed to the EPA a different way of combining releases of radioactivity estimated from different scenarios and comparing them to the release limits set by the EPA. The NRC staff's proposal has come to be known as the "three bucket" approach because release scenarios are divided into three groups, depending on their probabilities of occurrence. The EPA, in its most recent working draft of 40 CFR Part 191, included a version of the three-bucket approach (somewhat different from the NRC staff's proposal) in a section of the draft that is for comment only. The purpose of this paper is to compare the three approaches: EPA's original standard, NRC's version of the three-bucket approach, and EPA's version of the three-bucket approach.

Let us begin with a description of the three approaches.

1.1 The original 40 CFR Part 191

The original statement of the standard's containment requirements, in 40 CFR 191.13(a) (EPA, 1985), is as follows:

> (a) Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation, based upon performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall:
>
> (1) Have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix A); and
>
> (2) Have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 (Appendix A).

Furthermore, Appendix B of 40 CFR Part 191 offers the following guidance about how to apply the standard:

> *Scope of Performance Assessments.* Section 191.13 requires the implementing agencies to evaluate compliance through performance assessments as defined in §191.12(q). The Agency assumes that such performance assessments need not consider categories of events or processes that are estimated to have less than one chance in 10,000 of occurring over 10,000 years. Furthermore, the performance assessments need not evaluate in detail the releases from all events and processes estimated to have a greater likelihood of occurrence. Some of these events and processes may be omitted from the performance assessments if there is a reasonable expectation that the remaining probability distribution of cumulative releases would not be significantly changed by such omissions.
>
> *Compliance with §191.13.* The Agency assumes that, whenever practicable, the implementing agency will assemble all of the results of the performance assessments

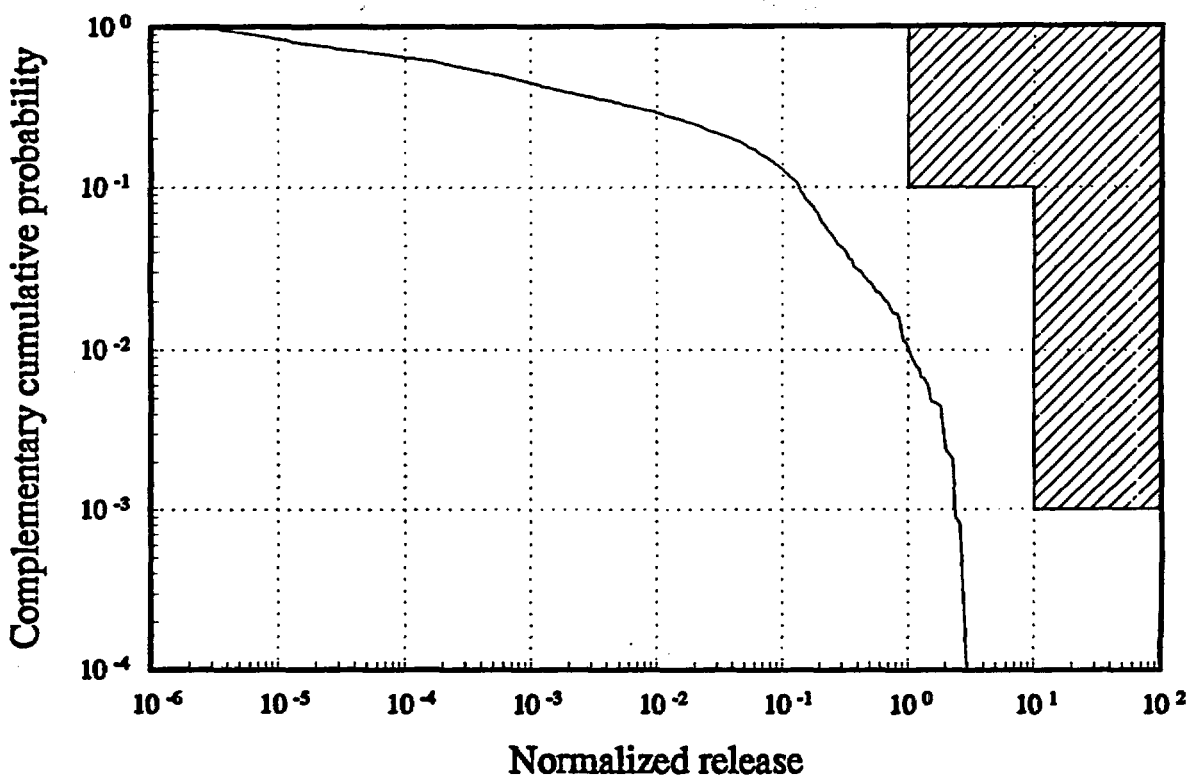

Figure 1: An example of a complementary cumulative distribution function, or CCDF.

> to determine compliance with §191.13 into a "complementary cumulative distribution function" that indicates the probability of exceeding various levels of cumulative release. When the uncertainties in parameters are considered in a performance assessment, the effects of the uncertainties considered can be incorporated into a single such distribution function for each disposal system considered. The Agency assumes that a disposal system can be considered to be in compliance with §191.13 if this single distribution function meets the requirements of §191.13(a).

It is not necessary to discuss here all details of the above requirements. For our purposes, it is not necessary to know anything about Table 1 of Appendix A, for example. What is important to know is that the standard defines a performance measure—the cumulative release of radioactivity to the accessible environment over 10,000 years, normalized in a particular way. This performance measure is assumed to have some uncertainty because of geologic variability, uncertainty about the future, etc., so that a probability distribution for the performance measure is to be presented rather than a single estimate or a simple range of possible values. An example of such a probability distribution, or CCDF, is given in Figure 1.

Each point on the curve in Figure 1 gives the probability of exceeding a particular value of the performance measure, which is referred to as "normalized release." Mathematically, if the normalized release in general is denoted by M and a particular value of the normalized release is denoted by m, then the CCDF at that point, $G(m)$, is defined by

$$G(m) = Pr(M > m) \ .$$

With this notation, the EPA limits in 40 CFR 191.13(a) may be restated as

$$G(1) < 0.1 \ ,$$

$$G(10) < 10^{-3} \ .$$

These limits are represented in Figure 1 by the cross-hatching. Though the EPA limits are only stated as upper bounds on the CCDF curve at two points, the fact that a CCDF, by definition, must never increase (it is a monotonically nonincreasing curve) implies the restriction of the curve from the entire cross-hatched region.

Let us conclude the discussion of the original EPA standard with three important points. (1) Only one CCDF is called for, and it is expected to include all significant sources of uncertainty; (2) there is a cutoff probability of 10^{-4}, below which "categories of events or processes" need not be considered; (3) the implementing agency for the Yucca Mountain site is the NRC, so interpretation of ambiguities (such as "reasonable expectation") would be up to the NRC.

1.2 The NRC staff's three-bucket approach

The following discussion of the NRC staff's proposed alternative to the containment requirements in the EPA standard is based on NRC (1991). Their recommendation is to change the wording of 40 CFR 191.13(a) to read

> Disposal systems...shall be designed to provide a reasonable expectation that, for 10,000 years after disposal:
>
> (1) anticipated performance will not cause cumulative releases of radionuclides to the accessible environment to have a likelihood greater than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix B); and
>
> (2) the release resulting from any process, event, or sequence of processes and events that is sufficiently credible to warrant consideration will not exceed ten times the quantities calculated according to Table 1 (Appendix B).

(The Table 1 of Appendix B referred to here is the same as Table 1 of Appendix A referred to previously.) The key concepts here are "anticipated performance" and "sufficiently credible to warrant consideration." These terms are deliberately vague to allow the implementing agency flexibility in evaluating a proposed site. From NRC (1991) and from discussions with NRC staff members, it appears that "anticipated performance" is meant to encompass events, processes, and sequences of events and processes with probabilities greater than about 0.01 to 0.1; "sufficiently credible to warrant consideration" is meant to encompass events, processes, and sequences of events and processes with probabilities greater than about 10^{-3} to 10^{-4}. In their examples in NRC (1991), the NRC staff use a probability of 0.01 to determine which processes and events are "anticipated." For the examples in NRC (1991), a probability of 10^{-3} to determine which processes and events are "sufficiently credible to warrant consideration" was thought to make the three-bucket approach equally stringent to the original EPA standard (that is, the same scenarios survived the scenario-screening process). Because the NRC staff seem to prefer the values 10^{-2} and 10^{-3} for the two cutoffs, those values will be used in this report when this alternative is applied.

We can now see the origin of the phrase "three-bucket approach." The first "bucket" contains processes and events with probabilities greater than 0.01, the second "bucket" contains processes and events with probabilities between 0.001 and 0.01, and the third "bucket" contains processes and events with probabilities below 0.001. In the three-bucket approach, the comparison with regulatory limits is done differently for the different buckets.

The first bucket contains the anticipated processes and events, and they are handled as in the EPA standard, but with only one limit rather than two. A CCDF of normalized releases caused by those processes and events is constructed and its value at $m = 1$ must be less than or equal to 0.1 ($G_a(1) \leq 0.1$). Note that the change in wording from the EPA standard results in a $\leq$ rather than a $<$; this difference is not important. Note also that G_a is being used rather than G to denote this CCDF, to indicate that only anticipated processes and events are included, rather than all significant processes and events. Rather than the limit on the CCDF at $m = 10$ that is in the EPA standard, each sequence of processes and events is to be evaluated and the normalized release is supposed to be no more than ten for each one: $m \leq 10$.

The second bucket contains unlikely processes and events, but ones judged sufficiently credible to consider. It is the NRC staff's intention that calculations for these processes and events be done deterministically rather than probabilistically, with a single normalized release calculated for each sequence of processes and events and compared with a limit of 10, as above. For each one, we must have $m \leq 10$.

The third bucket contains processes and events that are judged too unlikely to consider further, so there is no limit on releases from these processes and events and therefore no need to perform any calculations.

The NRC staff's concern is the difficulty in quantifying the probabilities for unlikely events. The EPA's cutoff probability of 10^{-4} for a 10,000-year period is a rate of only 10^{-8} per year. Probabilities this low are very difficult to estimate accurately. The NRC staff's three-bucket concept is an attempt to change the standard so as to require less precision in the estimates of low probabilities and to require simpler modeling of the low-probability events. The three-bucket method also has the advantage of not requiring calculation of the extreme tail of the CCDF, down to a probabilty of 10^{-3}, as the EPA standard requires.

One final comment on the above statement of the three-bucket-approach requirements. In the second part, "release" is referred to without the qualifiers "cumulative" and "to the accessible environment," thereby creating some confusion. It will be assumed throughout this report that releases are calculated in the same way as in the first part and in the original EPA containment requirements—by calculating cumulative releases to the accessible environment.

1.3 The EPA's version of the three-bucket approach

The most recent working draft of the EPA's modified 40 CFR Part 191 (EPA, 1992a) contains a number of changes from the 1985 version, including adding radiation dose as an optional alternative to cumulative releases of radioactivity as the performance measure. This report will not be concerned with the issue of radiation dose *vs.* radioactivity as performance measures. In all the following discussion, cumulative release of radioactivity to the accessible environment will be used as the performance measure to make comparisons with the 1985 standard easier.

The part of the revised (draft) standard of interest here is essentially unchanged from the 1985 standard. In addition, however, the EPA included a section for comment in which it states that they are considering inclusion of the following as an option for the implementing agency:

> Disposal systems for radioactive waste shall be designed to provide a reasonable expectation that releases resulting from all significant processes and events (including both natural and human-initiated processes and events) that may affect the disposal system for 10,000 years after disposal shall:
>
> (1) Have a likelihood of less than one chance in 10 that cumulative releases to the accessible environment will exceed the quantities calculated according to Table 1 (Appendix A); and
>
> (2) Not exceed ten times the quantities calculated according to Table 1 (Appendix A) based upon the projected release resulting from any process, event, or sequence of processes and events which has a likelihood of occurrence between one chance in 10 and one chance in 10,000.

This statement of the three-bucket approach is different from the NRC staff's recommendation in two ways. First, there is less ambiguity. Rather than using terms such as "anticipated performance" and "sufficiently credible to warrant consideration," concrete probability values are specified. Second, the CCDF calculated for part (1) is to include releases resulting from all significant processes and events rather than only those anticipated, as in the NRC version.

As stated here, the CCDF is the same as the one that would have been calculated for use with the original EPA standard, but only the first limit is applied: $G(1) < 0.1$. The second limit on the CCDF is replaced by the NRC staff's idea of requiring $m \leq 10$ for any sequence of processes and events with probability between 10^{-4} and 0.1 (but remember that the NRC staff used probabilities between 10^{-3} and 1 for this limit).

Finally, note that the EPA statement of the three-bucket approach has the same problem the NRC staff's statement had in not specifying *cumulative* releases *to the accessible environment* in the second part. As stated before, it will be assumed throughout this report that releases always refer to cumulative releases to the accessible environment.

1.4 Summary of the three methods

In the rest of this report, the three methods will be referred to as EPA1 (the containment requirements of the 1985 EPA standard, unchanged in EPA's most recent revision), NRC1 (the NRC staff's three-bucket approach), and EPA2 (the EPA's version of the three-bucket approach, included for comment only in the recent revision of the EPA standard).

- All three methods allow events or processes with very low probability to be neglected. The guidance section in both the original standard and the recent draft revision states that categories of events or processes with probabilities less than 10^{-4} need not be considered. Presumably this cutoff also applies to the NRC staff's three-bucket approach since they did not suggest any revision to it.
- Methods EPA1 and EPA2 both require construction of a CCDF $G(m)$ that includes releases resulting from all significant processes and events. EPA1 and EPA2 both place the

following restriction on the CCDF:

$$G(1) < 0.1 \ .$$

EPA1 places the additional restriction

$$G(10) < 10^{-3} \ .$$

- Method NRC1 requires construction of a CCDF G_a that includes only anticipated processes and events, and places the following restriction on it:

$$G_a(1) \leq 0.1 \ .$$

In NRC (1991), the NRC staff include scenario classes with probabilities greater than 0.01 in the CCDF calculation. This quantification is not necessarily easy, especially since there is considerable difference of opinion over just what is meant by a "scenario." This report conforms to the Yucca Mountain Site Characterization Plan (DOE, 1988) in the following terminology: An individual scenario is a possible future history of the repository system and has a probability of zero, just as a single line has a volume of zero. It takes a bundle of lines with nonzero cross-sectional area to have a nonzero volume. Similarly, a "scenario class" is a "bundle" of scenarios. The scenario classes must be defined in such a way that they include all the modeled scenarios and are mutually exclusive.

- Methods NRC1 and EPA2 place a limit on the normalized release resulting from individual scenario classes: $m \leq 10$. This limit only applies to scenario classes within a range of probabilities. EPA2 specifies that this limit applies to scenario classes with probabilities between 0.1 and 10^{-4}, while for NRC1 the limit applies to scenario classes with probabilities greater than about 10^{-3} (and less than or equal to 1, of course).

NRC1 and EPA2 are more different than this description would indicate, as can be seen by looking at the examples in the NRC staff's paper (NRC, 1991). When comparing a scenario-class probability against the 10^{-3} cutoff, they do not use an actual estimate of the scenario-class probability, but rather a number that is more of an upper bound. In their Examples 1 and 2, they consider a possible sequence of events as follows: faulting; drilling, but with no hits on waste packages; and volcanism (not necessarily in that order). Their estimates of the probabilities of these events are 0.55, 0.975, and 3×10^{-4}, respectively, when calculating a CCDF for comparison with the EPA standard, leading to a probability of 1.6×10^{-4} for the sequence. But, when making a comparison with their three-bucket approach (NRC1), they replace the 3×10^{-4} probability for volcanism with "$< 10^{-2}$," on the grounds that the probability of 3×10^{-4} is not well known but the probability is fairly certain to be below 10^{-2}. Following this replacement, the probability for the sequence comes out to be $< 5.4 \times 10^{-3}$, which is above the 10^{-3} cutoff even though the initial estimate of 1.6×10^{-4} would have been below the cutoff.

For this report it was decided not to apply this bounding-probability procedure to method EPA2 for two reasons. (1) It doesn't seem appropriate to use this procedure with the very low probability cutoff (10^{-4}) specified by the EPA; and (2) by using different interpretations for methods NRC1 and EPA2, we can see the advantages and disadvantages of the two interpretations.

A problematic issue for this limit ($m \leq 10$) is that it is stated deterministically. How is it to be applied when there is uncertainty about the normalized release for a scenario class

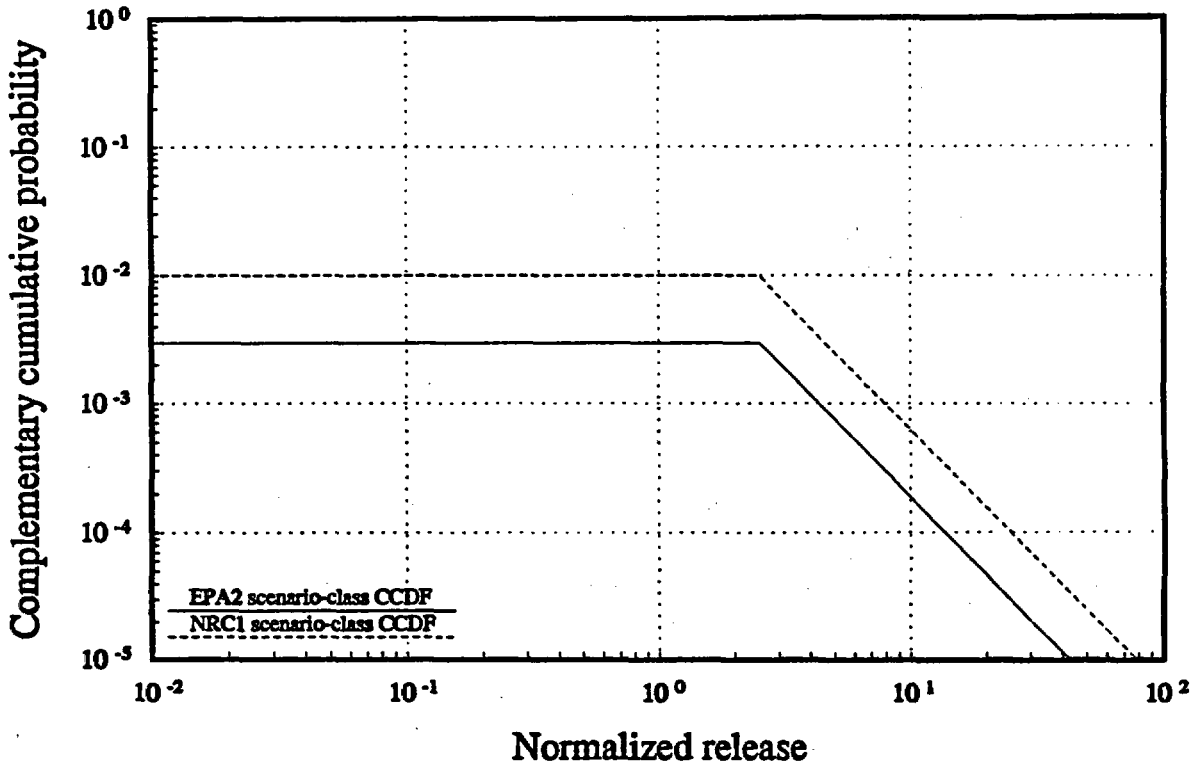

Figure 2: CCDFs for a single scenario class, with scenario-class probability of 3×10^{-3} and mean normalized release of 5 (1.5×10^{-2} if adjusted for probability of occurrence).

(as would normally be the case)? Suppose, for illustration, that there is a scenario class that has probability 3×10^{-3} and a normalized release of 5 but, if probability distributions for the input parameters are included to take into account their uncertainty, has a distribution of normalized releases extending up beyond the limit of 10 (see Figure 2). There are calculated m's greater than 10, but they occur at low probability levels—well below the 10^{-3} probability that would cause the 1985 EPA standard to be exceeded. It is not clear how the NRC or the EPA intend for this situation to be resolved; it is very difficult to assign a single release number to each scenario class in a consistent manner. For this report it will be assumed that the release limit is exceeded only if $m > 10$ at the cutoff probability level or higher, where the cutoff probability is 10^{-3} for NRC1 and 10^{-4} for EPA2. To put it another way, the release value that will be assigned to each scenario class will be the value at the cutoff probability level. The hypothetical case illustrated in Figure 2 would exceed the EPA2 limit but would not exceed the NRC1 limit.

The choice was made for this paper to use the release value at the probability cutoff so that the same level of stringency in probability is applied to all scenario classes. Certainly, there are simpler choices that could be made. For example the mean of the release distribution for each scenario class could be used, or some percentile of the release distribution (e.g., the median or the 90th percentile) could be used. Such choices have the problem that the $m \leq 10$ limit is then much more stringent for low-probability scenario classes than for high-probability scenario classes. This lack of consistency is what leads to the problem of

sensitivity to the choice of scenario classes that will be discussed in Section 3.3.

The problem with using percentiles of the release distribution may be illustrated as follows. Suppose we have a set of scenario classes, $\{S_i\}$. The limit $m \leq 10$ is required for each one (ignoring for now the fact that the EPA's version of the three-bucket approach does not place this restriction on scenario classes with probabilities greater than 0.1), and for this example we choose the median of the release distribution as the value to use for m. Thus,

$$Pr(m > 10 \,|\, S_i) < 0.5 \quad \text{for all } i.$$

The notation $Pr(m > 10 \,|\, S_i)$ means the probability that m is greater than 10, assuming that scenario class S_i occurs. The exceedence probabilities for the individual scenario classes can be summed to obtain the exceedence probability for the system as a whole if they are weighted by the probabilites of occurrence for the scenario classes:

$$Pr(m > 10) = \sum_i p_i Pr(m > 10 \,|\, S_i) \,,$$

where p_i is the probability of occurrence for scenario class S_i. Using the above inequality for the individual exceedence probabilities, the following result is obtained:

$$\begin{aligned} Pr(m > 10) &< \sum_i p_i \cdot 0.5 \\ &< 0.5 \,. \end{aligned}$$

The final result follows from the fact that all the scenario-class probabilities have to add up to one:

$$\sum_i p_i = 1 \,.$$

In the notation used previously, $Pr(m > 10) = G(10)$, where G is the CCDF. So, if medians are used to define the scenario-class releases, the $m \leq 10$ part of the three-bucket approach only requires that $G(10) < 0.5$. This is not the whole story, because the three-bucket approach also requires that $G(1) < 0.1$ (ignoring for the moment the fact that the NRC's version of the three-bucket approach only places restrictions on the CCDF for anticipated events, G_a). Because the CCDF is a nonincreasing function, the restriction at $m = 1$ is also a restriction at $m = 10$: $G(10) \leq G(1) < 0.1$. Thus, interpreting the three-bucket approach in this manner would be much less restrictive than the 1985 EPA requirement that $G(10) < 10^{-3}$. To ensure that the $m \leq 10$ limit was at least as stringent as the original EPA standard would require taking m to be the value at the 99.9th percentile. Such a definition would place a much more stringent restriction on low-probability scenario classes than required by the original EPA standard.

There are similar problems with other methods of assigning scenario-class releases. For example, requiring the mean normalized release for a high-probability scenario class to be less than or equal to 10 does not ensure that its CCDF probability is less than 10^{-3} at $m = 10$. Another way (and, in fact, an easier way) to assign a single release number to a scenario class is to use some statistical measure of the input parameters and calculate the release only once (for example, calculate the release using the mean values of all the state variables). This procedure can still have some of the problems just discussed, and it has

the additional drawback of being sensitive to the choice of model parameters. There is no unique way to specify what the model parameters should be, and in a nonlinear system the mean of some combination of parameters is not necessarily the same as the combination of the means (and similarly for any other statistical measure). It should also be kept in mind that a calculation with typical (mean) values for the input parameters does not necessarily produce typical (mean) output. See, for example Barnard *et al.* (1992, Section 4.8).

2 Sandia Laboratories' TSPA-1991

Sandia National Laboratories recently completed a preliminary total-system performance assessment (TSPA) of the Yucca Mountain site (Barnard *et al.*, 1992). The calculations for that report were made using the EPA performance measure from the 1985 standard. The results are presented in terms of CCDFs and are compared to the EPA limits from the standard. A natural starting point for the comparison of the three regulatory methods described in the previous section is to recast the TSPA results in the form prescribed by each method. This will be done in the following subsections.

The results from TSPA-1991 are summarized in Figure 3. The calculations were made for different processes separately and combined at the end into overall CCDFs for comparison with the EPA standard. Figure 3 shows the component CCDFs before the final combination. It should be kept in mind that these results are preliminary and do not constitute a final performance assessment for Yucca Mountain. They are used here only to illustrate the methods described in the previous section. Note that the curves stop abruptly rather than going all the way down to zero probability. This is a result of the Monte Carlo method that was used to generate the curves. In a Monte Carlo simulation, there is always a minimum probability that can be observed because of the finite number of realizations. The curves could be extended vertically downward, but the curves were left as is because that way the reader has some additional information about the statistical significance of the results.

The curves included in Figure 3 are as follows:

1. Gaseous (composite). Releases resulting from gaseous transport of ^{14}C under nominal conditions (no disruptive events). The composite-porosity model of water flow was used in defining the releases of ^{14}C from the waste packages.

2. Gaseous (weep). Releases resulting from gaseous transport of ^{14}C under nominal conditions. The "weeps" model of water flow was used in defining the releases of ^{14}C from the waste packages.

3. Aqueous (composite). Releases resulting from aqueous transport of radionuclides under nominal conditions. The composite-porosity model of water flow was used in defining the releases from the waste packages and in calculating groundwater flow and transport.

4. Aqueous (weep). Releases resulting from aqueous transport of radionuclides under nominal conditions. The weeps model of water flow was used in defining the releases from the waste packages and in calculating unsaturated-zone flow and transport (saturated-zone flow and transport were calculated using the composite-porosity model.)

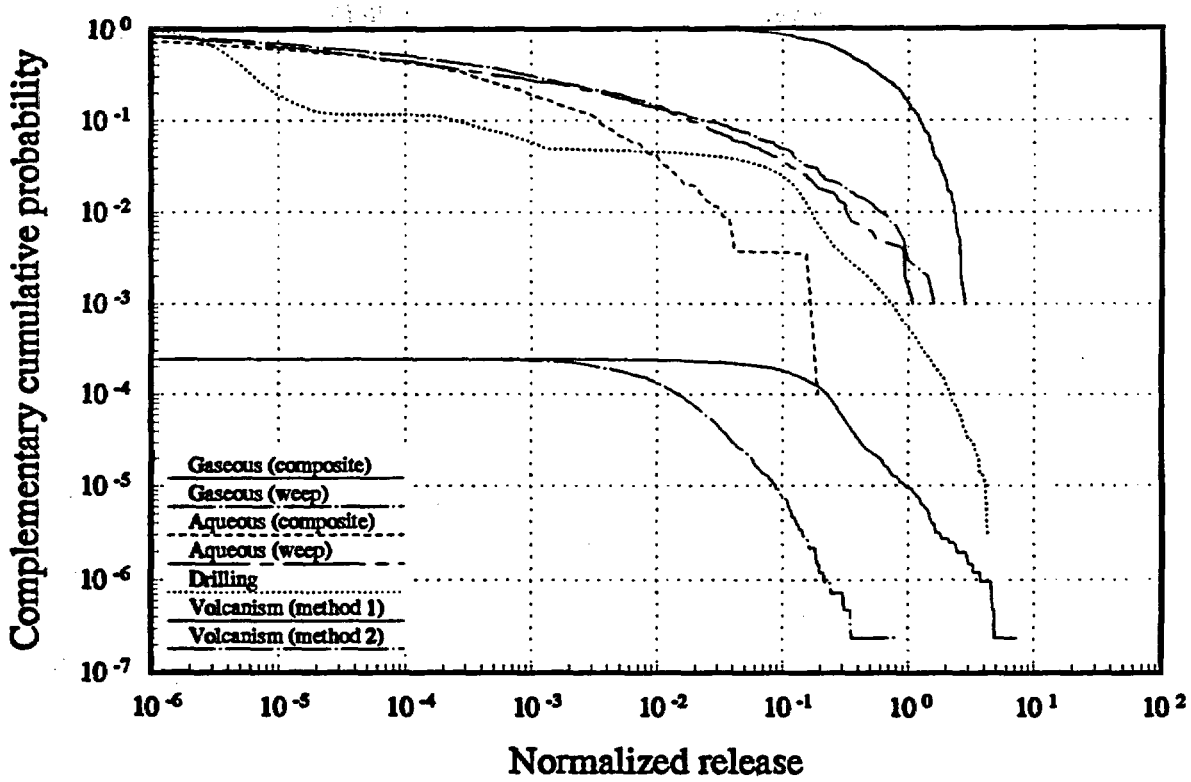

Figure 3: Component conditional CCDFs from TSPA-1991.

5. Drilling. Releases resulting from exploratory drilling. Four components have been combined in this curve: releases directly to the surface due to direct hits of waste packages, releases directly to the surface due to "near misses" that bring up contaminated rock, releases due to direct hits of waste packages that cause some waste material to fall to the tuff aquifer and be transported to the accessible environment by groundwater, and releases due to direct hits of waste packages that cause some waste material to fall to the carbonate aquifer and be transported to the accessible environment by groundwater.

6. Volcanism (method 1). Releases directly to the surface resulting from a basaltic igneous intrusion through the repository.

7. Volcanism (method 2). Releases directly to the surface resulting from a basaltic igneous intrusion through the repository.

It is not necessary for the present discussion to know what the composite-porosity and weeps models are, except to know that they are treated as alternative conceptual models of unsaturated-zone flow and transport. For the TSPA-1991 report, they were taken to be mutually exclusive models, i.e., either one is correct or the other is correct, with no mixtures allowed. In the following, results will be presented separately for the two conceptual models: when nominal releases are combined with drilling releases and volcanic releases to get the total release, it will be done twice, first using the composite-porosity model to obtain the nominal releases and then using the weeps model to obtain the nominal releases.

The two volcanism curves represent modeling of the same processes, but different formulas were used to calculate the amount of waste entrained by the intruding dike. The two volcanism release models will also be treated as alternative conceptual models, including only one of them in any given composite CCDF. However, the treatment of the two volcanism models has little or no effect on the overall CCDF because of the low probability of direct volcanic releases (2.4×10^{-4}).

2.1 The original 40 CFR Part 191 (EPA1)

For method EPA1, all of the model results should be combined into a single CCDF, and the CCDF compared to the EPA limits at $m = 1$ and $m = 10$. To combine releases from the different processes (that is, to combine the component CCDFs shown in Figure 3) the simplifying assumption is made that the processes operate independently. With that assumption, the CCDFs can be combined by a simple Monte Carlo simulation in which a release is drawn at random from each of the three release categories (nominal flow and transport, exploratory drilling, and igneous intrusion) and the releases are added together to obtain the total release. Those operations are repeated many (10,000) times to obtain the probability distribution of the total release. The results are shown in Figure 4. As already noted, the composite-porosity and weeps models are not combined, but instead results are reported for each separately. The volcanism releases do not contribute significantly to the combined CCDFs, no matter which model is used, so it is not necessary to present alternative curves for the two volcanism models.

It can be seen in Figure 4 that the releases calculated using the composite-porosity model exceed the EPA limits by having a probability greater than 0.1 for a normalized release exceeding 1 ($G(1) \simeq 0.2$). Neither CCDF exceeds the limit at $m = 10$. At this point, it should be emphasized once again that these results are preliminary, and it is likely that the EPA-limit violation is a result of overly conservative assumptions made in the gaseous-release calculations.

2.2 The EPA's version of the three-bucket approach (EPA2)

Method EPA2 uses the same CCDF as method EPA1, including all significant processes and events. Only the limit at $m = 1$ is applied, however, so it would not be necessary to calculate the CCDF as precisely (i.e., fewer realizations could be used in the Monte Carlo simulations). Since we already have the CCDFs in Figure 4, we will use them. The result is the same as for EPA1: the CCDF for the composite-porosity model exceeds the limit, but the CCDF for the weeps model does not.

Determination of compliance with the other EPA2 criterion, that $m \leq 10$ for any sequence of processes and events with probability between 10^{-4} and 0.1, depends on resolution of some ambiguous terms. The largest calculated normalized release for nominal conditions (using the composite-porosity model) is 3.0, the largest calculated normalized release for exploratory drilling is 4.3, and the largest calculated normalized release for volcanism (using method 1) is 7.2. Thus, it is possible to get a normalized release as high as 14.5 from the calculations that we are using. But, the high releases from volcanism occur at very low probability levels in Figure 3, so one would not expect them to cause violation of the regulatory limits.

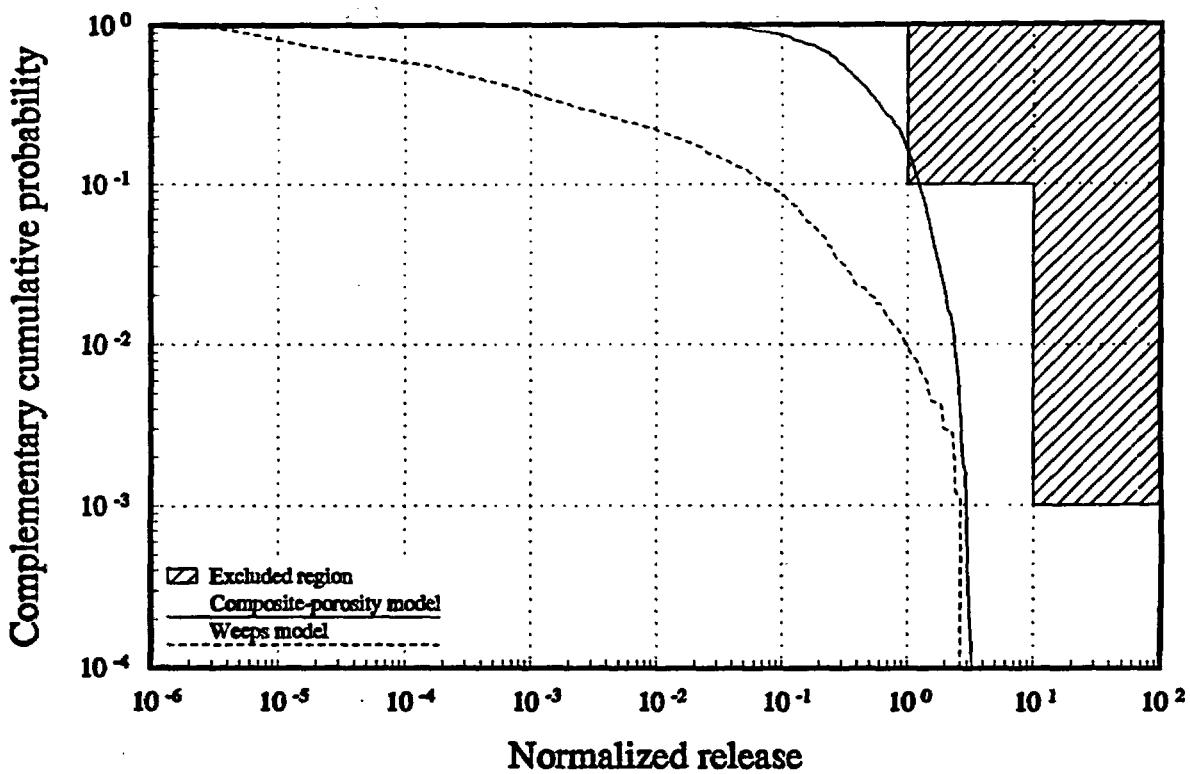

Figure 4: Conditional CCDFs including all modeled processes and events.

As discussed in Section 1.4, in this report it is assumed that the $m \leq 10$ limit is applied to scenario classes, and is applied to the scenario-class CCDF at the 10^{-4} probability level (for EPA2). So, the next step must be to define the scenario classes for the system, as modeled in TSPA-1991. The assumption in TSPA-1991 is that nominal gaseous and aqueous transport and exploratory drilling always occur, so the only one of the modeled processes and events that may or may not occur is volcanism. The occurrence or non-occurrence of an igneous intrusion into the repository defines two scenario classes, which will be referred to as V and $\bar{V}$, respectively. V is assigned probability 2.4×10^{-4}, so $\bar{V}$ has probability 0.99976. Conditional CCDFs for scenario class V, normalized by probability of occurence, are given in Figure 5. There are four curves because of the two alternative models of flow and the two alternative models of volcanism; all combinations are shown in Figure 5. The cross-hatched region in the figure shows the EPA2 limitation on scenario-class CCDFs at $m = 10$, as it is interpreted in this report. It can be seen that the $m \leq 10$ limit is not exceeded for the conditions modeled. The conditional CCDFs for scenario class $\bar{V}$ are the same as those shown in Figure 4, because releases due to volcanism had negligible effect on those curves. Curves for scenario class $\bar{V}$ are not shown in Figure 5 because in EPA2 the limit at $m = 10$ does not apply to scenario classes with probabilities above 0.1.

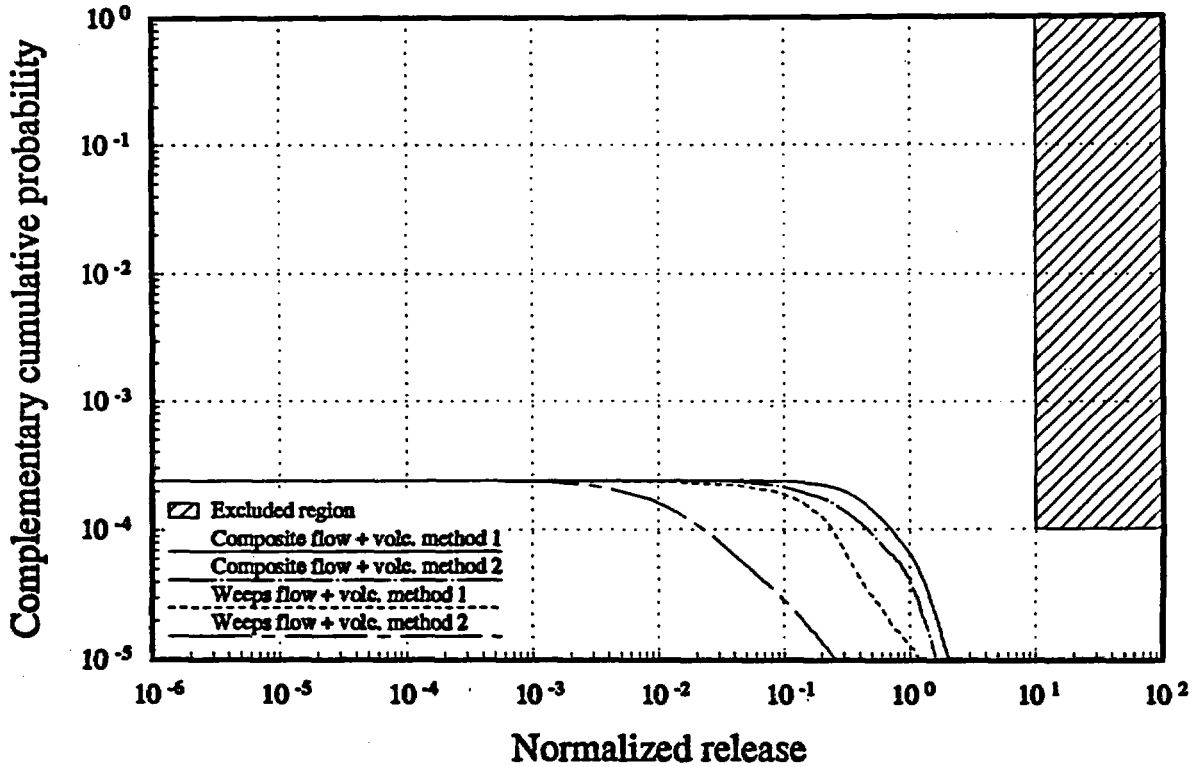

Figure 5: Conditional CCDFs for scenario class V (method EPA2).

2.3 The NRC staff's three-bucket approach (NRC1)

Compliance with method NRC1 will be determined by following the steps outlined in NRC (1991). As described there, the method is closely linked to a particular methodology requiring the specification of "scenarios," which are referred to here as scenario classes. This is a significant change from the 1985 EPA standard; note that compliance with EPA1 was determined without ever having to specify a set of mutually exclusive scenario classes.

For TSPA-1991, the scenario classes are defined as in the previous subsection. There are only two scenario classes, V and $\bar{V}$. What is different from the previous subsection is the way the NRC staff handles low-probability events such as volcanism. In their method, rather than using 2.4×10^{-4} as the probability for an igneous intrusion, the bounding estimate "< 0.01" is used instead. After the scenario classes are defined and their releases are estimated, the next step is to check them against the $m \leq 10$ criterion. This comparison is presented in Figure 6. There are three differences between Figure 6 and Figure 5. (1) Scenario class V (with four possible alternatives) is shown at probability 0.01 rather than 2.4×10^{-4}. (2) All scenario classes with probabilities greater than 10^{-3} are included, rather than scenario classes with probabilities between 10^{-4} and 0.1. Scenario class V is included because its probability was adjusted upward to 0.01. (3) The limit at $m = 10$ is set to probability 10^{-3} rather than 10^{-4}. The last point is a result of the interpretation being followed in this report, and is open to other interpretations. From Figure 6, it can be seen that all alternatives of the scenario classes are below the $m \leq 10$ limit.

The last step in determining compliance with NRC1 is to construct a CCDF from only

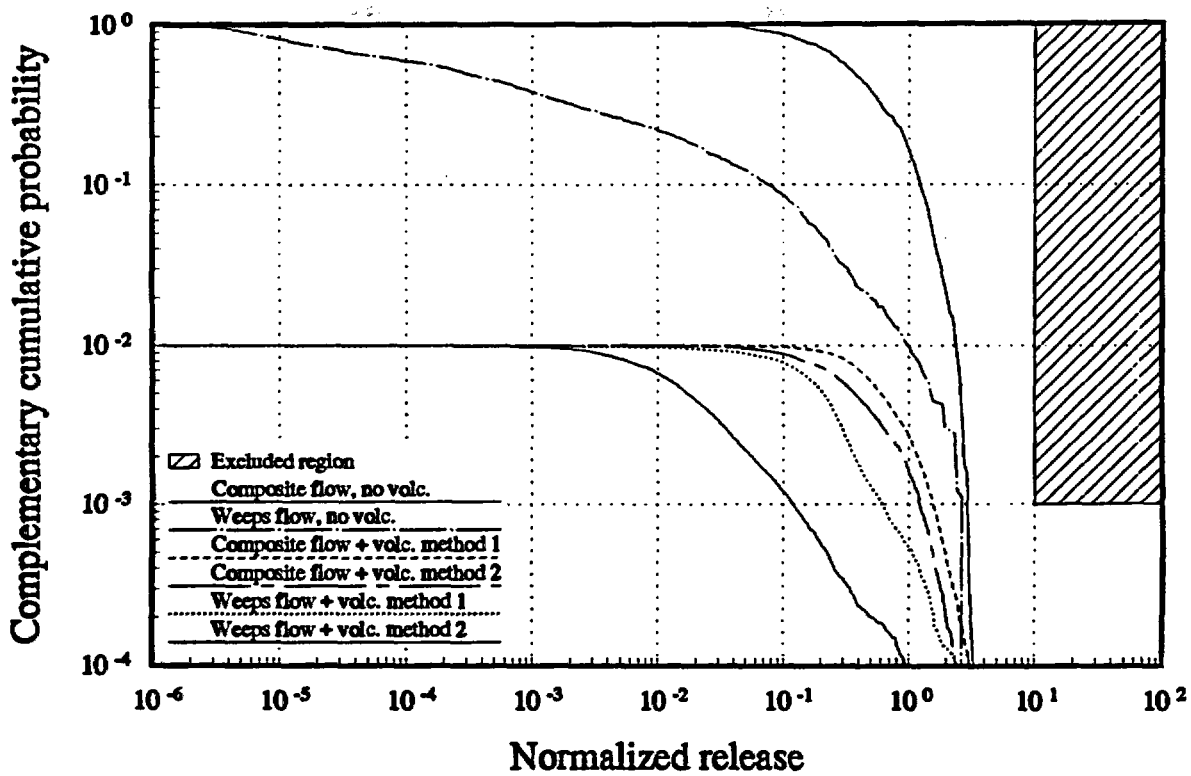

Figure 6: Conditional CCDFs for scenario classes (method NRC1).

the scenario classes having probabilities greater than 0.01. The CCDFs so constructed are indistinguishable from the CCDFs shown in Figure 4 because volcanism had no significant effect on those curves. The CCDFs are then checked to determine whether they have probabilities greater than 0.1 at $m = 1$. As before, the CCDF for the weeps model passes this test and the CCDF for the composite-porosity model fails.

2.4 Summary

To sum up, there is no difference in compliance when comparing the TSPA-1991 results with the three regulatory methods. The details of the three comparisons are different, but the answers are the same: the weeps model passes and the composite-porosity model fails.

3 Variations on TSPA-1991

In this section, some variations on the TSPA-1991 results will be used to illustrate various peculiarities and differences in the three regulatory methods. Of particular interest are cases in which the three methods disagree about compliance.

3.1 A high-consequence, low-probability scenario class

The NRC1 and EPA2 methods place more severe restrictions on low-probability processes and events than did the old EPA standard, EPA1. To illustrate this fact, let us consider what would happen if the consequences (i.e., the normalized releases) were higher for an igneous intrusion. There is no need to carry along all the alternative models to make the point, so we will just consider the case where nominal flow and transport are represented by the weeps model. Releases from volcanism will be represented by the "method 1" releases increased by a factor of 100. The resulting total-system CCDF is presented in Figure 7. The CCDF is the same as the weeps-model curve in Figure 4 except for the low-probability tail, where releases are increased because of the higher volcanic releases. The higher releases in the tail do not affect compliance with the EPA1 standard because they occur at probability below 10^{-3}.

The CCDF for the volcanism scenario class, V, is shown in Figure 8. This figure corresponds to Figure 5, but with higher volcanic releases. As shown in the figure, this scenario class no longer satisfies the $m \leq 10$ criterion for method EPA2. Similarly, it can be shown that this scenario class fails to satisfy the criteria of method NRC1. Thus, for high-consequence, low-probability scenario classes, methods NRC1 and EPA2 are stricter than EPA1.

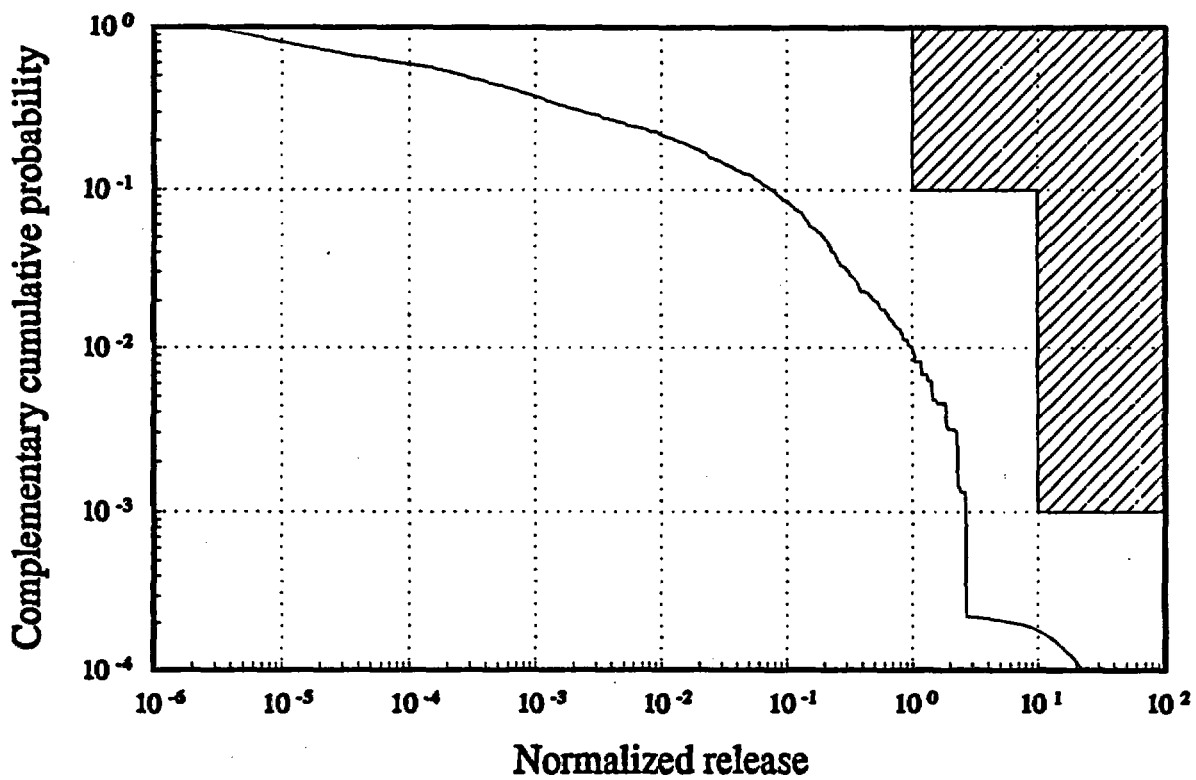

Figure 7: Total-system CCDF with volcanic releases multiplied by 100.

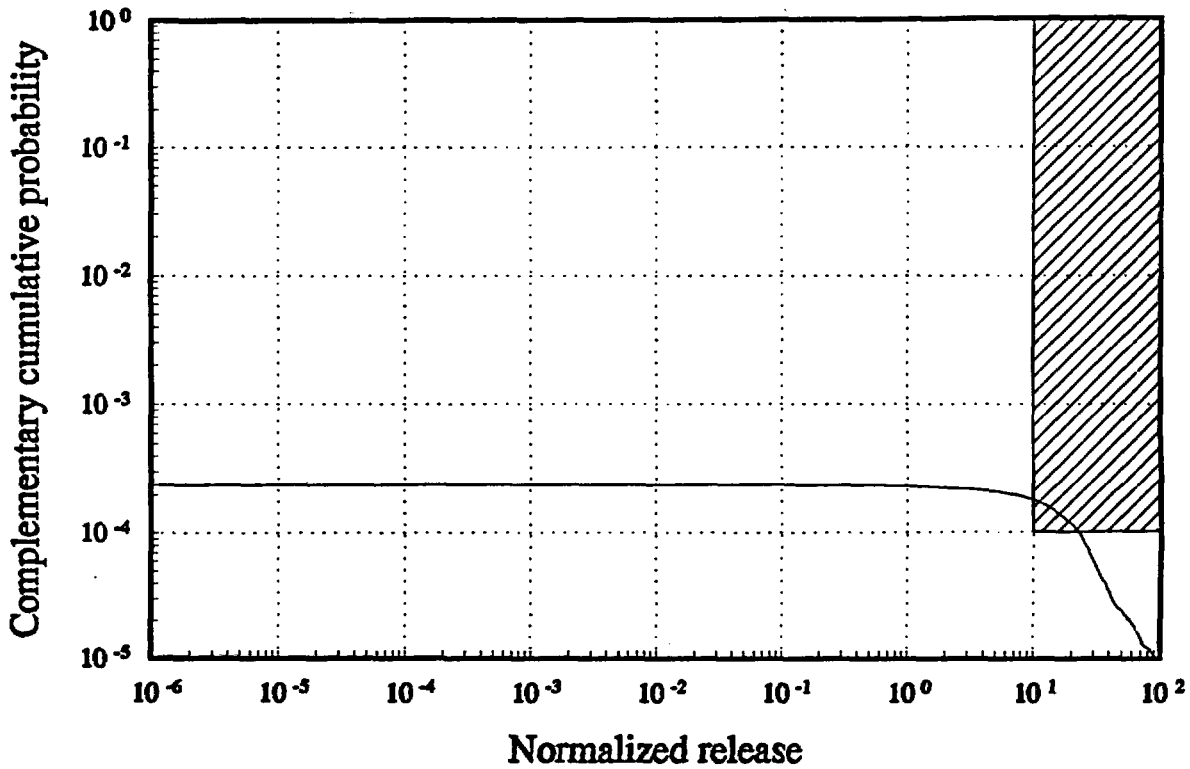

Figure 8: CCDF for scenario class V with volcanic releases multiplied by 100 (method EPA2).

3.2 A high-probability scenario class with some high releases

A peculiarity of method EPA2 is that the $m \leq 10$ limit does not apply to scenario classes with probabilites greater than 0.1. Because of that, it is possible to have a situation in which the EPA2 criteria will be satisfied while the EPA1 and NRC1 limits are exceeded. To illustrate this occurrence, let us consider the situation in which the direct releases from exploratory drilling are increased by a factor of 100. The weeps model will be used for nominal flow and transport, method 1 will be used for volcanic releases, and the near-miss and aqueous-release components of the drilling releases will be kept the same as previously. To make the point, the probability of getting a direct drill-hit on a waste package will be decreased from 12% to 9%.

The total-system CCDF for the system just described is shown in Figure 9. It can be seen that the CCDF is below the EPA1 limit at $m = 1$ but exceeds the limit at $m = 10$. The same CCDF is used for method EPA2, but only the $m = 1$ limit is applicable. Thus, this system passes the CCDF part of the EPA2 requirements. Similarly, it passes the CCDF part of the NRC1 requirements.

Figure 10 shows the comparison with the $m \leq 10$ criterion for EPA2. The limit only applies to scenario class V, as was the case previously (Figure 5), and scenario class V passes the test. Figure 11 shows the comparison with the $m \leq 10$ criterion for NRC1. In method NRC1, the limit applies to both scenario classes, V and $\bar{V}$ (see also Figure 6), and scenario class $\bar{V}$ exceeds the limit. Thus, this system is in violation of methods EPA1 and NRC1, but is in compliance with method EPA2.

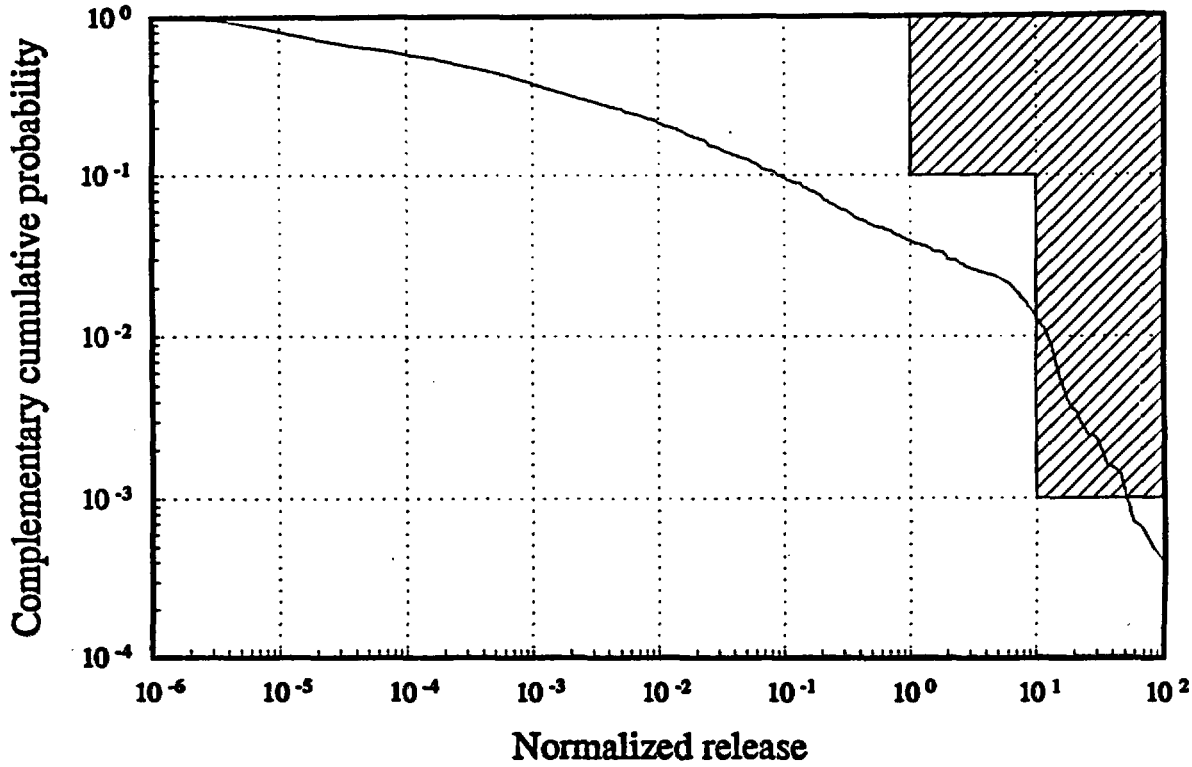

Figure 9: Total-system CCDF with direct drilling releases multiplied by 100.

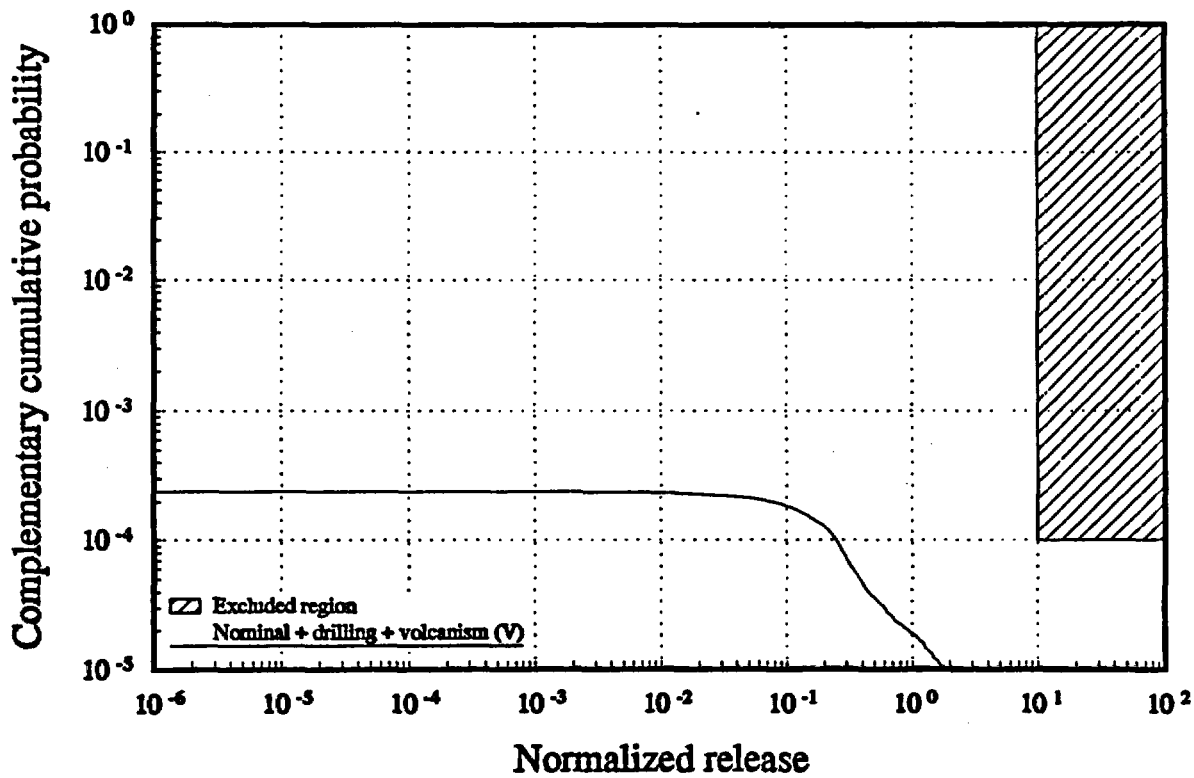

Figure 10: Scenario-class CCDF with direct drilling releases multiplied by 100 (EPA2).

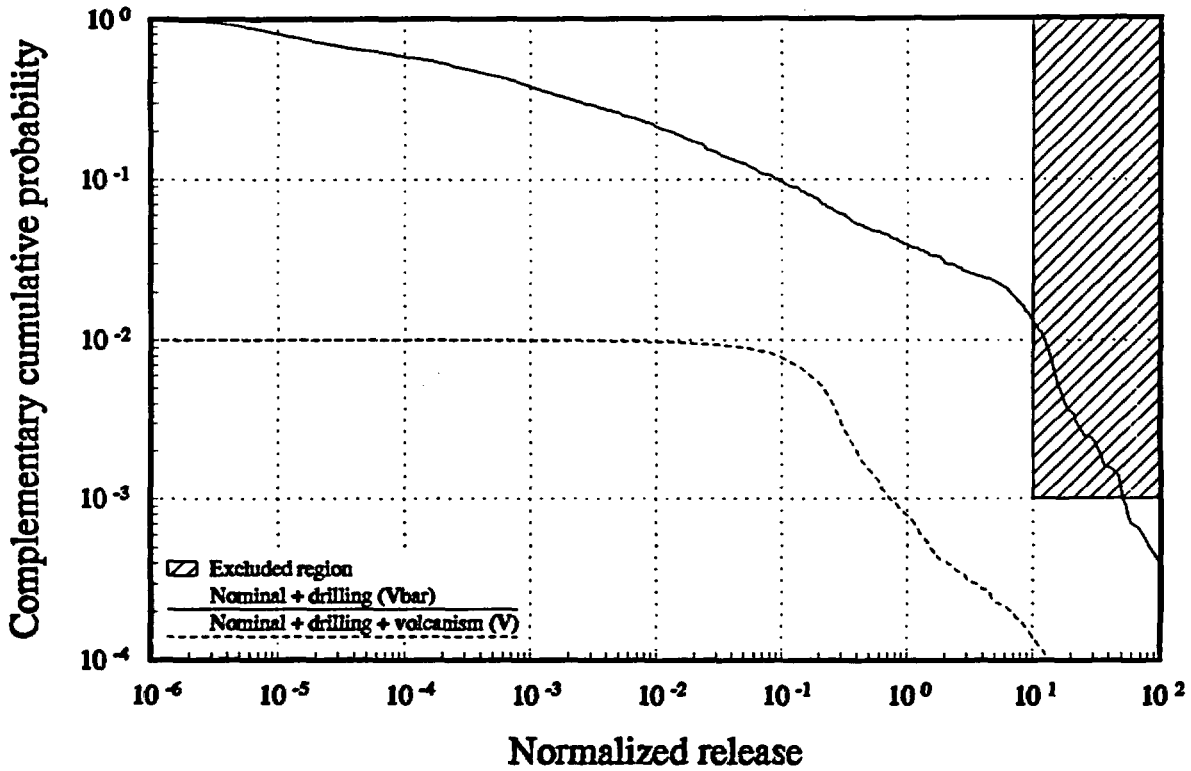

Figure 11: Scenario-class CCDFs with direct drilling releases multiplied by 100 (NRC1).

3.3 Sensitivity to the choice of scenario classes

For the methods (NRC1 and EPA2) that place restrictions directly on scenario classes rather than on the system as a whole, compliance or noncompliance depends on how the system is split up into scenario classes. This is potentially a serious problem because there is not a single, unique way to define the scenario classes. The possible problems will be illustrated with two examples.

For the first example, consider the system described in the previous subsection, in which direct drilling releases were increased and their probability decreased from the values used in TSPA-1991. We could have separated drilling with a direct hit on a waste package and drilling with no direct hits into separate scenario classes (the NRC staff make such a separation in the examples in NRC, 1991). Such a split results in four scenario classes: S_1 (nominal flow and transport, no direct hits, no volcanism), S_2 (nominal flow and transport, no direct hits, releases from an igneous intrusion), S_3 (nominal flow and transport, at least one direct hit, no volcanism), and S_4 (nominal flow and transport, at least one direct hit, releases from an igneous intrusion). This separation into scenario classes is illustrated in Figure 12. The figure also shows the scenario-class probabilities.

Changing the way the system is separated into scenario classes does not affect the total-system CCDF (Figure 9), but the component scenario-class CCDFs are different and so the comparison with the $m \leq 10$ criterion must be revisited. The comparison of the new scenario classes with the $m \leq 10$ criterion in EPA2 is shown in Figure 13. Note that S_1 is not included because its probability is too high and S_4 is not included because its probability is too low.

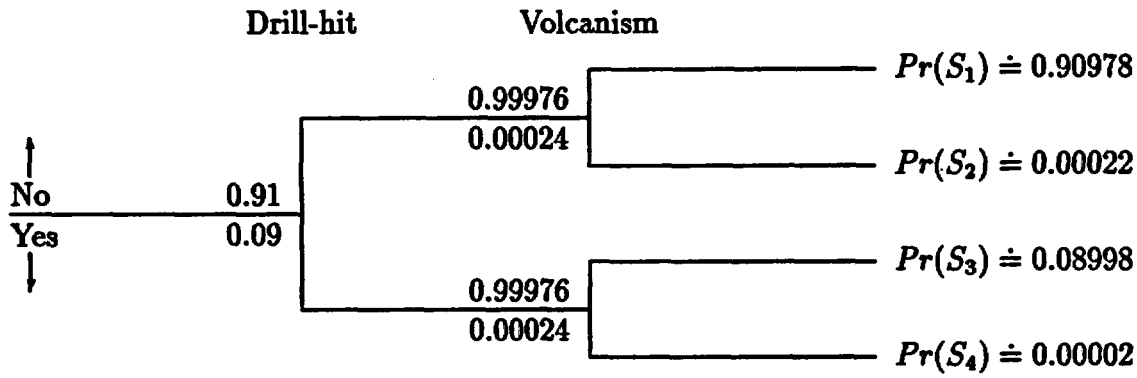

Figure 12: Logic tree for separation of the model system into four scenario classes.

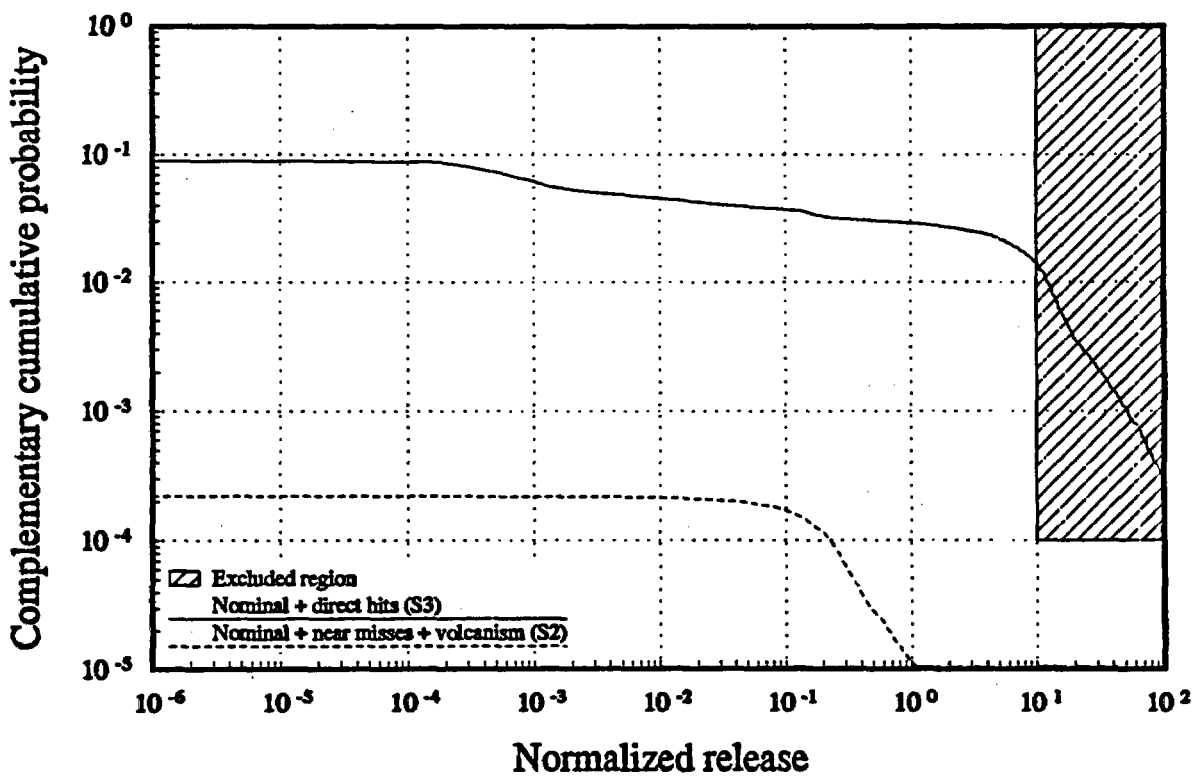

Figure 13: Scenario-class CCDFs with direct drilling releases multiplied by 100 (EPA2).

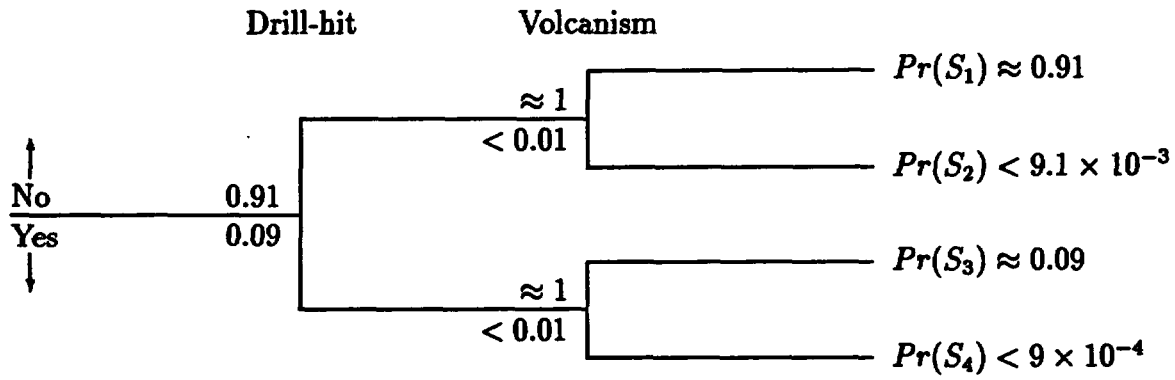

Figure 14: NRC1 version of the logic tree.

Scenario class S_3 exceeds the limit.

Because of the different way method NRC1 handles scenario-class probabilities, it is worth showing the NRC1 comparison to the $m \leq 10$ criterion as well. Figure 14 shows the logic tree in Figure 12 with the probabilities amended as was done by the NRC staff in NRC (1991). Scenario class S_4 has a probability low enough that the $m \leq 10$ limit does not apply to it, but the limit must be applied to the other three scenario classes. The comparison with the limit is shown in Figure 15. It can be seen that scenario class S_3 exceeds the limit. Thus, using this set of scenario classes all three regulatory methods agree on failure but, using the previous set of scenario classes, method EPA2 showed compliance while the other two methods showed failure.

To show that the problem of sensitivity to the choice of scenario classes is not unique to method EPA2 but also occurs in method NRC1, let us go back to the system as modeled in TSPA-1991 and put an additional hypothetical effect into the exploratory-drilling calculation. Suppose that, for some reason, a small portion of the direct hits have unusually large releases, at about a probability level of 5×10^{-4} (0.4% of the direct hits, with direct hits occuring about 12% of the time). The occurrence of these "special" direct hits need not be a result of some event occuring 0.05% of the time, but could result from some combination of undesirable parameter values, out in the tail of the joint probability distribution. The total-system CCDF for this hypothetical system is shown in Figure 16. It is quite similar to the CCDF in Figure 7, but with the high releases in the tail resulting from a different process. Because of the low proabability of the special direct hits, the EPA1 criteria are satisfied.

A comparison of the V and $\bar{V}$ scenario-class CCDFs with the NRC1 $m \leq 10$ criterion is shown in Figure 17. The limit is not exceeded. A comparison of the S_1, S_2, S_3, and S_4 scenario-class CCDFs with the same criterion is shown in Figure 18. Once again, the limit is not exceeded.

Suppose now that we break out the special, high-release drilling events into separate scenario classes, as shown in Figure 19. Because of the way the NRC staff handle low-probability events, the probability level for the special drilling events is changed from 5×10^{-4} to "< 0.01," and treated as if the probability were 0.01. Because of the increase of over an order of magnitude in the probability, the $m \leq 10$ criterion is now exceeded (Figure 20;

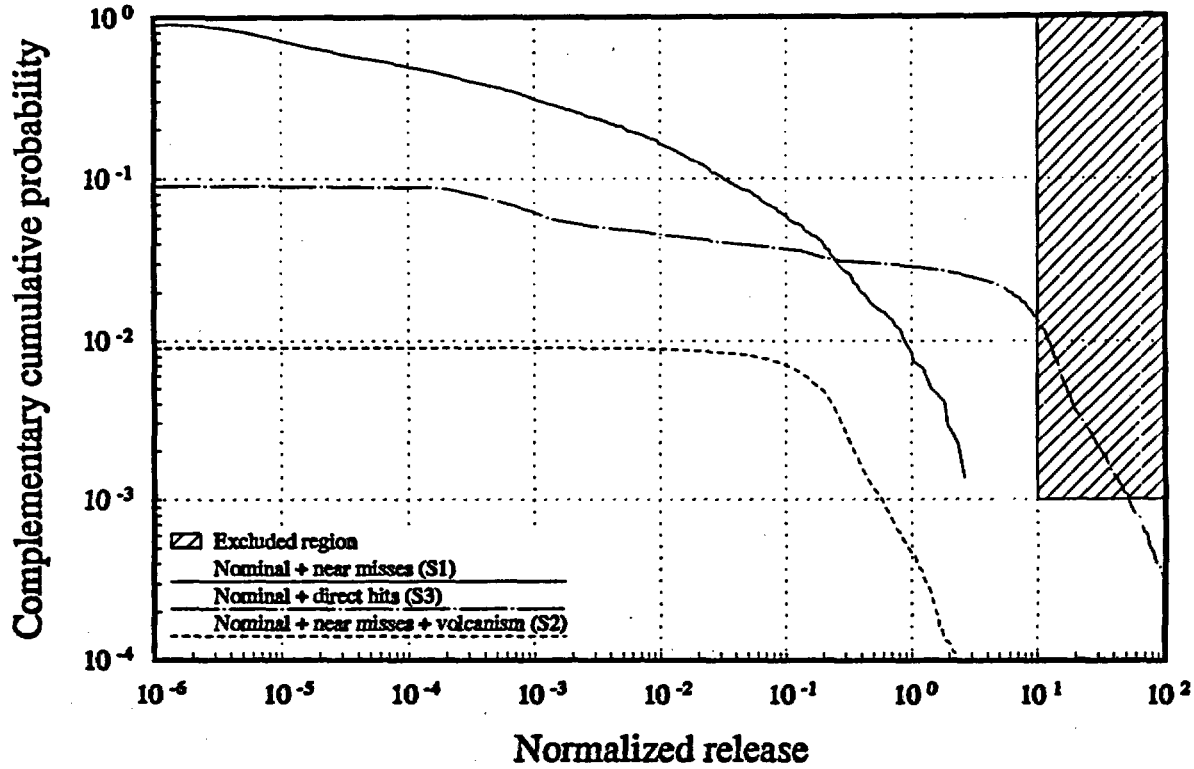

Figure 15: Scenario-class CCDFs with direct drilling releases multiplied by 100 (NRC1).

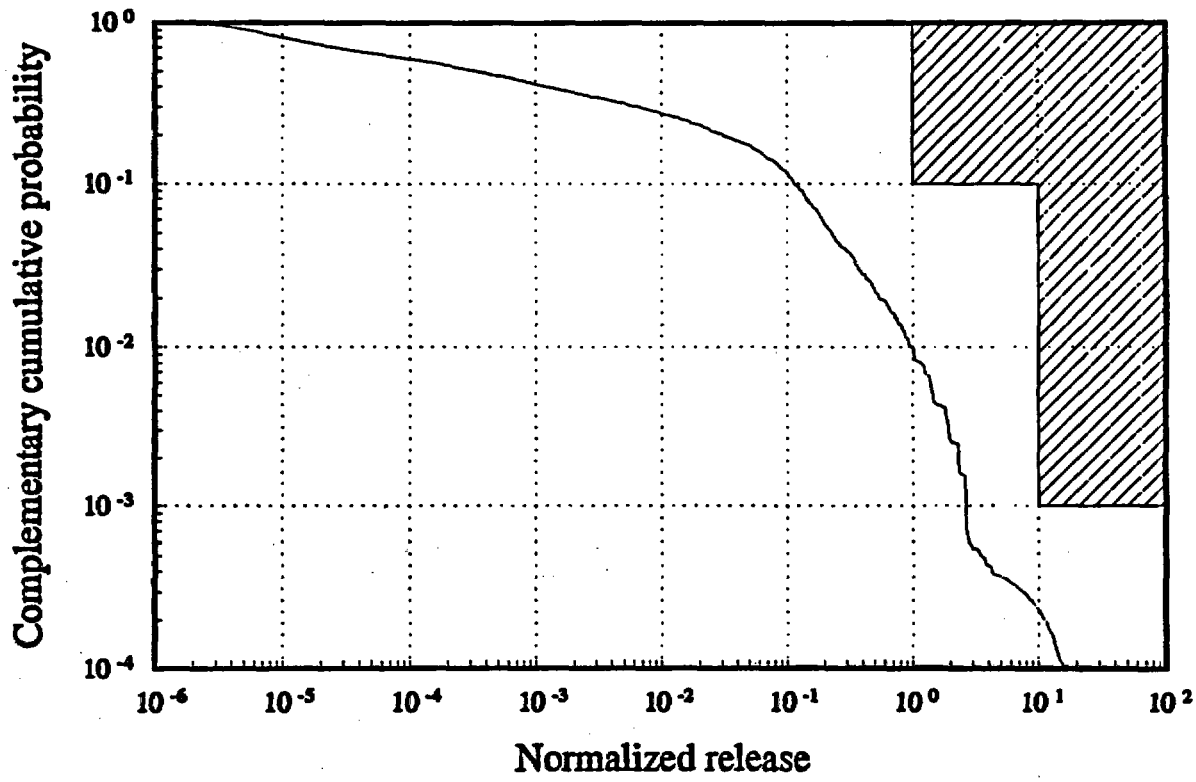

Figure 16: Total-system CCDF for the system with "special" drilling events.

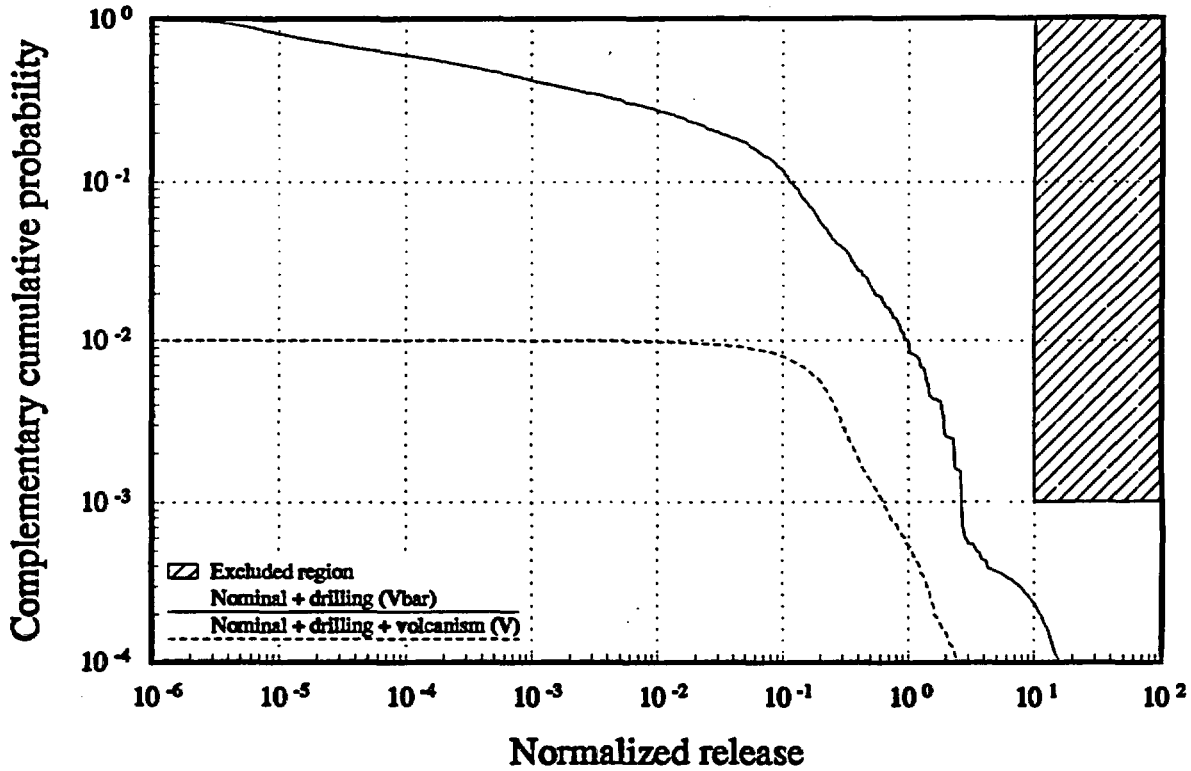

Figure 17: CCDFs for two scenario classes, with special drilling events (NRC1).

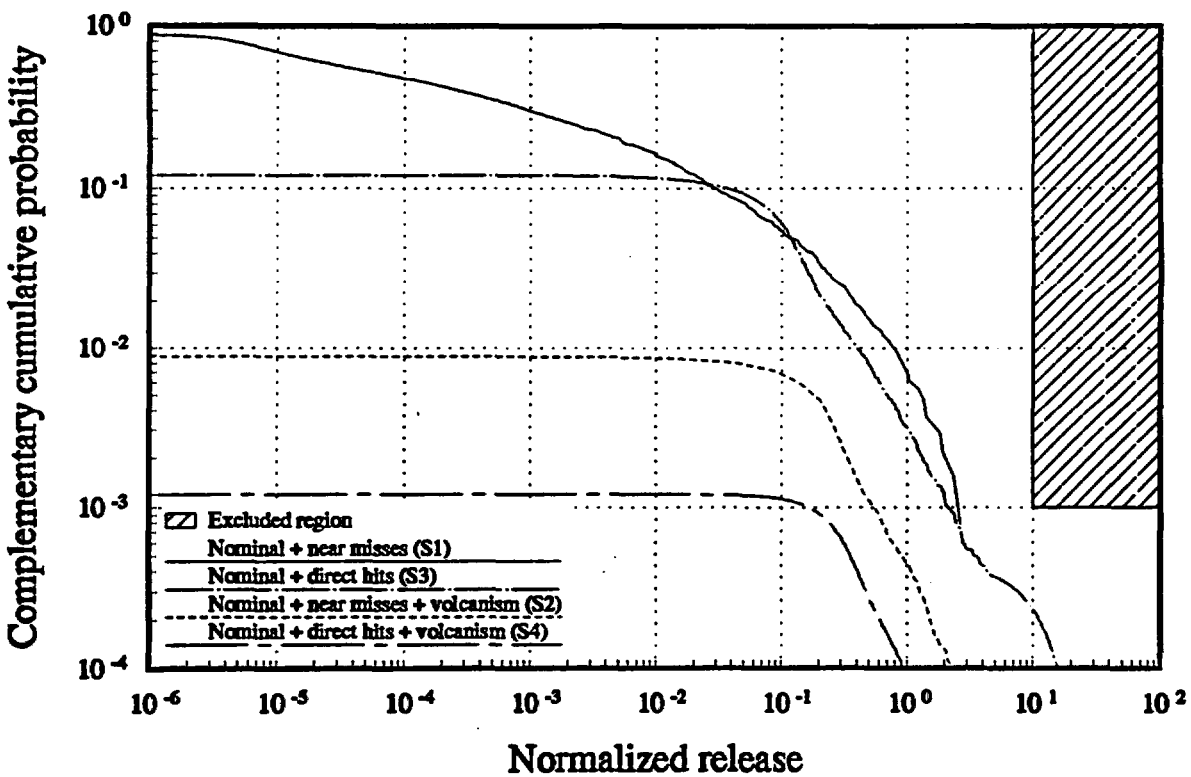

Figure 18: CCDFs for four scenario classes, with special drilling events (NRC1).

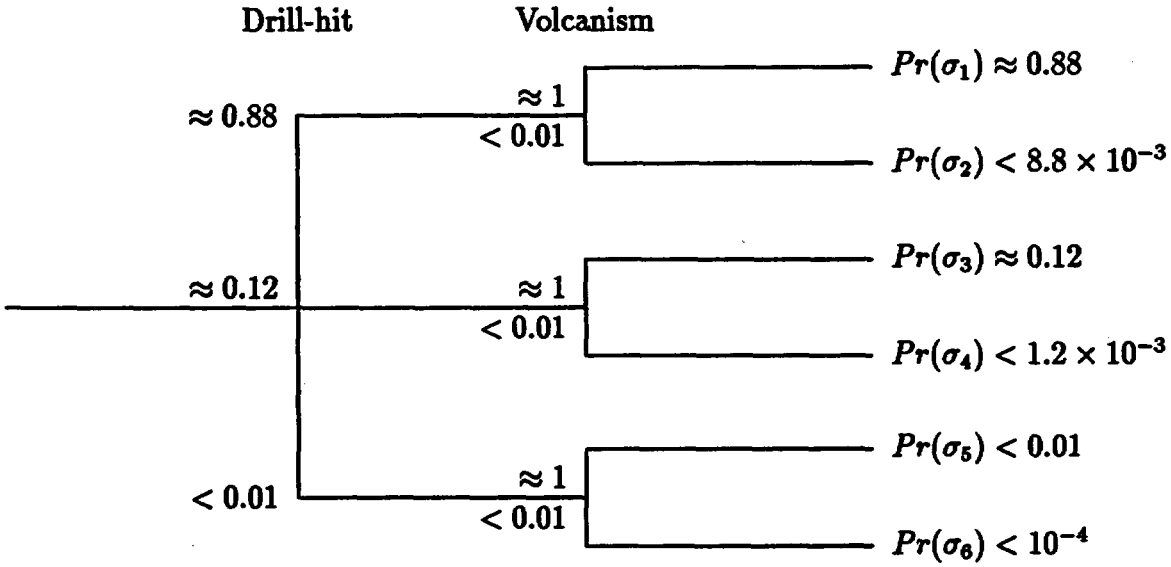

Figure 19: Logic tree for six scenario classes (including special drilling events).

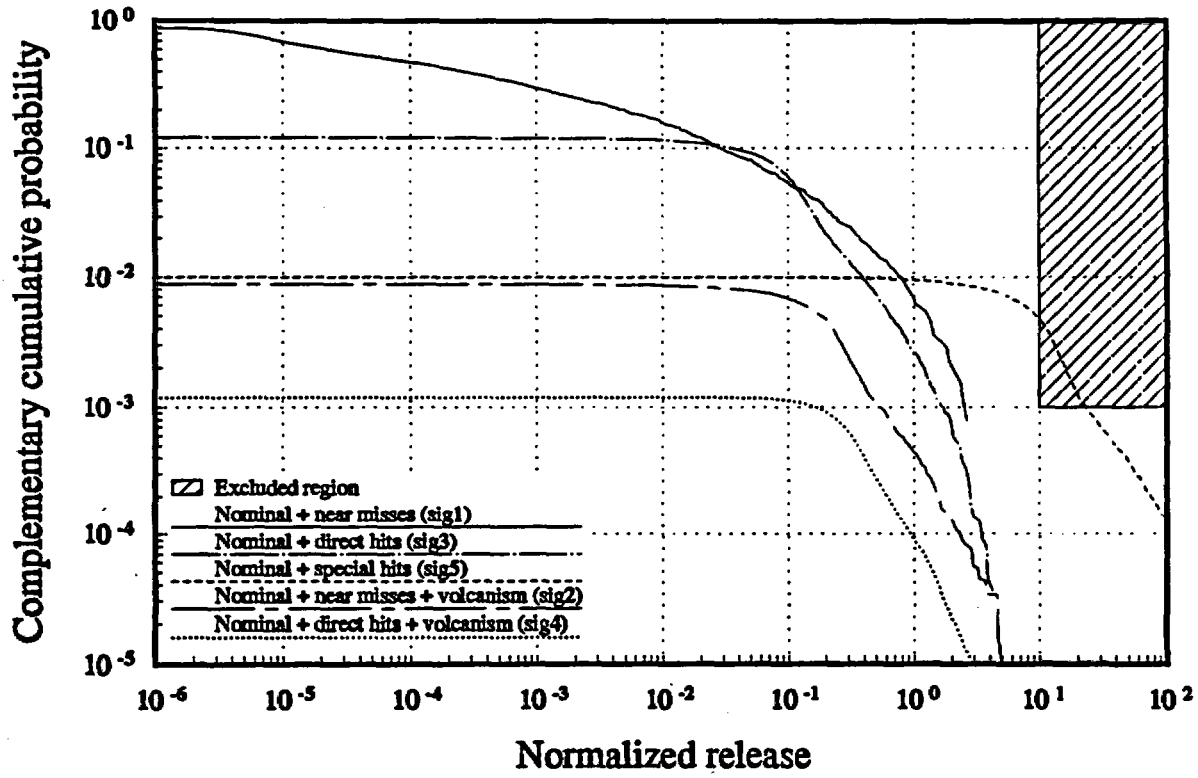

Figure 20: CCDFs for six scenario classes, with special drilling events (NRC1).

note that the CCDF for scenario class σ_6 is not shown in the figure because its probability is below the 10^{-3} cutoff). This result shows that the NRC1 procedure is sensitive to the choice of scenario classes; the NRC1 requirements are more restrictive when the system is split up into several low-probability scenario classes rather than a few higher-probability ones. The sensitivity results from the modification of scenario-class probabilities without corresponding modification of probabilities for calculations within a scenario class. With additional guidance, this ambiguity could possibly be removed. The NRC staff's concern was with estimating probabilities for infrequent events. What complicates resolution of the ambiguity is that the event probabilities may not always be direct inputs to the calculations, but may be calculated from other input parameters in a way that is not easily amenable to the procedure outlined by the NRC staff.

Comparison of this system with the EPA2 version of the $m \leq 10$ criterion will not be illustrated here, but it is easy to see that the limit is exceeded for all three choices of scenario classes. The compliance of this model system with the three regulatory methods is summarized as follows. With method EPA1 it passes, with method EPA2 it fails, and with method NRC1 it depends on how the system is divided up into scenario classes.

Table 1 summarizes some of the results of this subsection. To produce Table 1, normalized releases were calculated for all of the scenario classes in the four different scenario-class structures for this model system. Releases were calculated using four different statistical definitions of the release for each scenario class: (1) the mean of the distribution of releases; (2) release at the 90th percentile; (3) release at a fixed probability level of 10^{-4}; and (4) release at a fixed probability level of 10^{-3}, but with probabilities handled in the way suggested by the NRC staff in NRC (1991). Number 3 is the same as the method called EPA2 in this report, but scenario classes with probabilities above 0.1 were not excepted. Number 4 is the method called NRC1 in this report. In each case, Table 1 lists the highest release for the scenario classes with probabilities greater than 10^{-4} (greater than 10^{-3} for method NRC1).

It can be seen from the table that the only one of these methods for which releases do not depend on how the system is divided into scenario classes is the third one, that uses the release at a fixed probability level. This result should not be surprising, because choosing the release value at a fixed probability level is the only one of the methods that treats the probability cutoff consistently. If the mean release or a percentile of the release distribution is used, then low-probability scenario classes are restricted at lower absolute probability levels than are high-probability scenario classes. Using the value at a fixed probability level, as was done for the EPA2 examples in this report, is consistent and avoids the problem of sensitivity to the choice of scenario classes, but it requires calculating the scenario-class CCDFs down to that probability level and so is no savings over the original EPA standard, which required calculating CCDFs down to the 10^{-3} probability level. (The sensitivity to choice of scenario classes that was demonstrated in Section 3.2 for method EPA2 resulted from the unfortunate stipulation that the $m \leq 10$ limit does not apply to scenario classes with probabilities over 0.1. With this aspect of method EPA2 removed, sensitivity to choice of scenario classes is no longer a problem.) For the NRC1 examples in this report the value at a fixed probability level was used, but the scenario-class probabilities were replaced by bounding values. This procedure is effectively a cross between using the value at a fixed probability level and using the value at some percentile of the release distribution, and it was shown above that such a procedure does not give a unique answer, but depends on how the system is split up into

Table 1: Normalized releases for four scenario-class structures, calculated in four ways.

	mean	90 percent.	10^{-4} cutoff	10^{-3} cutoff (NRC method)
1 scenario class	0.057	0.11	16	2.6
2 scenario classes	0.33	0.62	16	2.6
4 scenario classes	0.31	0.60	16	2.5
6 scenario classes	14	23	16	23

scenario classes.

To conclude this section, let us consider what determines the choice of scenario classes. Is one choice more natural or preferable to the others for some reason? For regulatory method EPA1 (the old EPA standard), the choice of scenario classes is primarily driven by the modeling tools available. If a single computer program were available that contained models for all the significant processes and events, there would be no real reason to separate the system into scenario classes at all: a single all-encompassing Monte Carlo simulation would suffice. This method of using a "total-system simulator" is described in DOE (1988). It is often the case that computer models exist that can represent only part of the parameter space, so that to represent the whole parameter space requires using several models. The big advantage of breaking the system up into subsystems appears in this situation, if the scenario classes can be chosen so that each scenario class corresponds to one (or a particular combination of more than one) of the models. For example, for the system just considered, there could be a good reason to split the system into six scenario classes, as shown in Figure 19, if different models are used for calculating drilling releases resulting from near misses, drilling releases resulting from direct hits, drilling releases for the special circumstances that lead to the high releases, and releases resulting from igneous intrusions. On the other hand, if one model is used to calculate all drilling releases and one model is used to calculate volcanic releases, the simple division into the two scenario classes V and $\bar{V}$ is more natural.

For the other methods, EPA2 and NRC1, with their more explicit reliance on the division of the system into scenario classes, some criteria for how that division should be made are needed. Without such criteria, as has been shown in this subsection, there may be too much ambiguity to be able to determine whether the system meets the regulations or not.

4 Advantages and disadvantages

From the preceeding discussion and the examples presented, some of the strengths and weaknesses of the three regulatory methods should be apparent. In this section, advantages and disadvantages of each method will be summarized.

4.1 The original 40 CFR Part 191 (EPA1)

Advantages:

- History. The EPA standard has been around for a number of years and considerable effort has gone into developing methods for determining compliance with it. That effort will not necessarily be wasted if a version of the three-bucket approach is implemented instead, but changing to different containment requirements will certainly require re-examination of what has been done and development of some new methods.
- Clarity. The EPA standard is relatively clear. Specific numbers are given for release limits and probability cutoffs.
- Aesthetics. The EPA standard treats the disposal system as a whole, rather than requiring treatment of susbsytems in any particular way. The system can be broken into scenario classes or not, according to the judgment of researchers as to the best way to do it.

Disadvantages:

- CCDF tail. The EPA standard puts a limit on the probability distribution of normalized releases—the CCDF—at a probability level of one part in 1000. Thousands of realizations may be required in a Monte Carlo simulation to reach statistical significance at such a low probability level. The three-bucket approaches only put limits on the CCDF at a probability level of one part in 10, which would only require tens of realizations to reach statistical significance. Thus, the three-bucket approaches are potentially much cheaper in computer time and analyst time.
- Low-probability events. The NRC staff regard the necessity of estimating probabilities for low-probability events as a serious disadvantage of the EPA standard. However, there is nothing in the standard that requires an *accurate* estimate of the probabilities. If there is uncertainty about a probability, a defensible conservative estimate would normally be used. And if the results (i.e., compliance or noncompliance with the standard) are sensitive to the value used for the probability, within its plausible range, then it would be difficult to argue that the standard's requirements were met with "reasonable expectation."

4.2 The EPA's version of the three-bucket approach (EPA2)

Advantages:

- Low-probability events. Rather than putting a limit on the total-system CCDF at a low probability level, the three-bucket approaches put release limits on low-probability events and processes directly. This avoids the necessity of determining the probabilities as precisely, to some extent, and also avoids having to calculate the tails of the CCDFs for high-probability events and processes down to the probability level of one part in 1000, as discussed above under "CCDF tail." Unfortunately, with the $m \leq 10$ limit interpreted as in this report, it is still necessary to know the probabilities of the events so that the conditional CCDFs for the scenario classes can be compared with $m \leq 10$ down to the 10^{-4} probability level, as in Figures 5, 8, 10, and 13. Furthermore, the conditional CCDFs must be calculated down to the 10^{-4} probability level, so there is no savings from not having to calculate the tail of the CCDF. Choosing a relative probability level rather than an absolute probability level (for example, only examining the scenario-class CCDF down one order of magnitude below the scenario-class probability) would alleviate these problems but causes additional problems. It was shown previously that using a relative probability rather than an absolute probability as the criterion for determining the scenario-class release (1) causes the determination of compliance to depend on how the system is split into scenario classes, and (2) can make the approach less stringent than the original EPA standard for high-probability scenario classes while at the same time making it more stringent than the original standard for low-probability scenario classes.

- Clarity. Like the original EPA standard, the EPA's version of the three-bucket approach gives specific numbers for release limits and probability cutoffs.

Disadvantages:

- More stringent. The three-bucket approaches place greater restrictions on releases from low-probability events than does the original EPA standard. The standard basically places no restrictions on an event with probability below 10^{-3}, but the three-bucket approaches place limits on releases down to a probability of 10^{-4}.

- Requirement of a particular methodology. The three-bucket approaches apparently require dividing the system into mutually exclusive scenario classes to determine compliance. The total-system-simulator method proposed for the Yucca Mountain Site Characterization Project (DOE, 1988) may not be allowed since it treats the system globally rather than breaking it into parts.

- Sensitivity to choice of scenario classes. Because the $m \leq 10$ limit only applies to scenario classes with probabilities less than 0.1, applicability of the limit depends on how the system is divided up into scenario classes. This problem could be eliminated by applying the $m \leq 10$ limit to all scenario classes with probabilities above the cutoff probability level. With the restriction of this limit to scenario classes with probabilities less than 0.1, as the draft of the EPA's three-bucket approach is written, it is necessary

to provide guidance about how the system should be divided into scenario classes in order to avoid ambiguity.

4.3 The NRC staff's three-bucket approach (NRC1)

Advantages:

- Low-probability events. Much of the discussion of the EPA's three-bucket approach applies here. The difference is that the NRC staff in NRC (1991) explicitly suggested a mechanism to make sure that precise probability values are not needed for low-probability events in their method: they do not use the actual probability estimate, but only the bounding estimate of "< 0.01." To some extent, this approach amounts to using a relative probability cutoff rather than an absolute probability cutoff, as discussed above. And the consequence is a significant sensitivity to the choice of scenario classes (see below).

Disadvantages:

- Vagueness. The statement of the NRC staff's three-bucket approach is extremely vague, giving no explicit values for probability cutoffs. Such vagueness makes a regulation very difficult to work to—like trying to hit a moving target. Furthermore, one would never guess from the statement of the requirement that one is supposed to follow the procedure that the NRC staff go through in their examples (using the bounding probability estimate of 0.01, etc.). Any such procedure that is expected should be clearly stated in the guidance section of the regulation.

- More stringent. As discussed above, both versions of the three-bucket approach place release restrictions down to the 10^{-4} probability level rather than down to the 10^{-3} level, as in the old EPA standard.

- Requirement of a particular methodology. Same as above.

- Sensitivity to choice of scenario classes. Because of the probability "renormalization" feature of the NRC staff's procedure, the results are sensitive to how the system is divided into scenario classes. Such sensitivity would be very undesirable, and to avoid it some explicit guidance would have to be given on just how to define the scenario classes.

5 Conclusions

The original EPA standard is wonderfully concise. It is simple, logical, and consistent. The biggest problem with it is that it requires calculation of the system probability distribution all the way down to a probability level of one part in 1000. As a result, Monte Carlo simulations to calculate the probability distribution may require thousands of realizations to obtain the desired statistical significance. The three-bucket approach seems like a way around having to make such large numbers of calculations. However, as has been shown in

this report, the simple statements of the three-bucket approach given in recent NRC and EPA draft documents have problems with logic and consistency. To make the three-bucket approach work, it is necessary to provide far more guidance about how it is to be applied.

The original concept of the three-bucket approach was to evaluate the anticipated, or high-probability, part of the system with a CCDF as in the original EPA standard, but with the CCDF calculated only down to a probability level of one part in 10 rather than one part in 1000. Then the low-probability part of the system was to be evaluated by simpler means—deterministic calculations rather than probabilistic ones, so that it was not necessary to know the precise probabilities.

This sounds like a reasonable concept, but it is not easy to apply. Some estimate of the probabilities is necessary in order to avoid disqualifying every site because of extremely unlikely combinations of events that lead to high releases. The process of assigning probabilities is not as straightforward as it might seem because any given "process, event, or sequence of processes and events" has a probability of zero—to get nonzero probabilities it is necessary to look at collections (or, as it is stated in Appendix B of 40 CFR Part 191, categories) of processes and events. That is why the concept of "scenario class" was introduced in this report. A scenario class is a collection of future histories of the system, and so a scenario class can have a nonzero probability and the probability cutoffs can be applied to it (see DOE, 1988). But the problem is that there is no unique way to define the scenario classes, and if the system is divided into scenario classes in different ways, the comparison with the probability cutoffs and the release limits can be different.

An additional difficulty with the three-bucket approach is that, because a scenario class is not a single event or process but rather a collection of them, there is no single answer for its release. Several methods are possible for assigning a single release value to a scenario class, including using the mean of the release distribution, using some percentile of the distribution (e.g., using release at the 90th percentile of the distribution), and using the release at some fixed probability value. Using release at a fixed probability value is consistent, but eliminates the simplicity of the three-bucket approach and makes it as difficult, or even more difficult, to work with as the original EPA standard because CCDFs have to be calculated down to some low probability level for every scenario class. With any of the other methods, compliance or noncompliance with the three-bucket-approach release limit (normalized release less than or equal to 10 for each scenario class) depends on how the system is split up into scenario classes. This phenomenon is referred to in this report as sensitivity to the choice of scenario classes.

The considerations just presented lead to the conclusion that a simple, concise statement of the three-bucket approach is probably not possible. Additional guidance is needed to remove the ambiguities in order to make the three-bucket approach usable

6 Addendum

After this report was written, the EPA and the NRC staff both came out with suggested revisions to the wording of the three-bucket approach. The revised versions will be discussed in this section. The revisions make no material difference in the conclusions stated above, though there are some minor changes in the advantages and disadvantages.

6.1 Revised NRC three-bucket approach

In NRC (1992), the NRC Staff suggest the following new wording for the three-bucket approach:

> 191.01 *Definitions*
>
> "Scenario" means a hypothetical future set of repository environmental conditions including any sequence of potentially disruptive processes and events that is sufficiently credible to warrant consideration.
>
> 191.12a *Consequence limit*
>
> Disposal systems for radioactive waste shall be designed to provide a reasonable expectation that, for 10,000 years after disposal, the release of radionuclides caused by any scenario will not exceed ten times the quantity calculated according to Table 1 (Appendix A).
>
> 191.12b *Containment requirement*
>
> Disposal systems for radioactive waste shall be designed to provide a reasonable expectation that, for 10,000 years after disposal, there will be at least a 90 percent likelihood that the cumulative release of radionuclides to the accessible environment will not exceed the quantity calculated according to Table 1 (Appendix A).

The first part (191.12a) is essentially the same as the corresponding part of the previous NRC-staff statement of the three-bucket approach, though different words are used. For any "sufficiently credible" scenario class, the normalized release should be less than or equal to 10: $m \leq 10$. Note that the NRC staff is using the word "scenario" to mean essentially the same thing as this report's term "scenario class." However, they have the idea of credibility built into their definition of scenario, so presumably a very-low-probability scenario class is not a scenario, in their terminology, but remains nameless.

The second part (191.12b) is worded quite differently from its original wording. If we put it in terms of a restriction on a CCDF to make comparison with previous methods clearer, we have $G(1) \leq 0.1$. As before, the NRC staff's version of the statement is slightly different from the EPA statements in having a $\leq$ rather than a $<$. The statement also is different from the previous NRC-staff statement in not mentioning "anticipated" processes and events, so the restriction has been stated here in terms of the total-system CCDF, G, rather than the CCDF including only anticipated processes and events, G_a.

6.2 Revised EPA three-bucket approach

The following revised statement of the EPA's three-bucket approach is taken from EPA (1992b):

> (1) Disposal systems for radioactive waste shall be designed to provide a reasonable expectation, based upon performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events (including both natural and human-initiated processes

and events) that may affect the disposal system shall have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix A).

(2) Disposal systems for radioactive waste shall be designed to provide a reasonable expectation that the release of radionuclides to the accessible environment for 10,000 years after disposal resulting from any one of the set of mutually exclusive scenarios that may affect the disposal system and is sufficiently credible to warrant consideration shall not exceed ten times the quantities calculated according to Table 1 (Appendix A).

The following paragraph has been added to the guidance section (now Appendix C rather than Appendix B):

Evaluation of compliance with §§191.12(b)(1) and (d)(1) need not consider categories of processes or events that are estimated to have less than one chance in 100 of occurring over 10,000 years. Sections 191.12(b)(2) and (d)(2) require the implementing agency to evaluate mutually exclusive scenarios which are sufficiently credible to warrant consideration. Such evaluations will be warranted at a likelihood greater than one chance in 1,000 over 10,000 years if the potential for release is dominated by a single release scenario. Consideration will be warranted at a likelihood greater than one chance in 10,000 over 10,000 years if there is the potential for releases from more than one scenario at probabilities near this value.

It can be seen that these statements continue the EPA's policy of stating probabilities explicitly rather than using only vague qualitative terms. However, in most other respects this statement of the three-bucket approach is closer to the NRC staff's original statement. The CCDF that is required in part (1) is to include only scenario classes with probabilities greater than 0.01, as was the case of the original NRC-staff three-bucket approach. The limit placed on the CCDF can be stated as $G_a(1) < 0.1$.

In part (2), the EPA took out the restriction of the $m \leq 10$ release limit to scenario classes with probabilities less than 0.1 that was in their previous three-bucket-approach statement. In addition, rather than restricting the releases of all scenario classes with probabilities greater than 10^{-4}, the lower probability limit now depends on whether there is one or more than one scenario class at the 10^{-4} to 10^{-3} probability level.

6.3 Analysis

- The most important change in the new three-bucket approaches may be that both versions now explicitly refer to "scenarios." The concept of mutually exclusive scenarios, or scenario classes in this paper's terminology, was present implicitly in the previous versions of the three-bucket approach, but with these new versions the concept is elevated to a higher status.
- In their statement of the $m \leq 10$ limit, the NRC staff still refer to release without specifying that it is *cumulative* release *to the accessible environment.* The EPA included "to the accessible environment" to their statement, but not "cumulative." The whole phrase should be included to avoid confusion.
- Though the EPA seems to have realized that their old three-bucket approach placed release limits on processes and events at lower probabilities than the original EPA standard, the new wording is not really any better. Under the new wording, the $m \leq 10$ limit would

not apply to a single scenario class with a probability of 2.4×10^{-4} (such as the volcanism scenario class V in the examples in Sections 2 and 3 of this report), but if there were two such scenario classes the limit would apply. It would take five such scenario classes to reach the probability of 10^{-3} where the CCDF is restricted in the original standard (at low probability, the probabilities are approximately additive). It is unclear whether the NRC staff's three-bucket approach is stricter than the 1985 containment requirements, just as it was unclear with their previous wording. If the procedure from NRC (1991) is followed, limits are placed on processes and events down to a probability of 10^{-4} because event probabilities between 10^{-2} and 10^{-4} are called "< 0.01" and treated as though they have a probability of 0.01.

- The EPA's new wording places the $m \leq 10$ limit on all scenario classes with probabilities greater than a probability cutoff rather than excepting probability classes with probabilities greater than 0.1. This change eliminates one source of sensitivity to the choice of scenario classes, but making the lower probability cutoff depend on the number of scenario classes introduces a new source of sensitivity to the choice of scenario classes—whether there is one low-probability scenario class or several just depends on how the system is split into scenario classes and not on any intrinsic property of the system.
- It was not mentioned before in this report, but the procedure of using only the scenario classes with probabilities greater than 0.01 to construct the CCDF that is to be compared to a probability limit at $m = 1$ could potentially lead to problems with sensitivity to the choice of scenario classes. This possibility seems like a minor problem because it would require the system to be divided into a large number of low-probability (below 0.01) scenario classes before there would be a significant problem. Nonetheless, it is preferable to avoid this possibility by simply specifying that all significant processes and events be used in constructing the CCDF.
- The problem of how to define a single release for a scenario class that has a distribution of releases is not addressed by either of the new wordings of the three-bucket approach. The problem of how the system is to be divided into scenario classes is also not addressed. Because there is no unique way to define the scenario classes or the release from a scenario class, compliance or noncompliance of a given system will be ambiguous unless the standard includes guidance for the division into scenario classes and for the assignment of a single release value to each scenario class.

Acknowledgment

I am indebted to Suresh Pahwa of INTERA for pointing out to me the fact that assigning scenario-class releases for comparison with the $m \leq 10$ limit on the basis of some percentile of the release distribution is potentially much less restrictive than the 10^{-3} probability limit at $m = 10$ in the original EPA containment requirements.

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APPENDIX B

JUSTIFICATION FOR DEVELOPING ALTERNATIVE RELEASE LIMITS

This information is supplied as reference material only.

Excerpt from SAND92-0556 (To be published)

EXPANDED PROPOSED EXTENSIONS OF STANDARDS FOR HIGH-LEVEL AND TRANSURANIC RADIOACTIVE WASTE DISPOSAL

Robert D. Klett and Marilyn M. Gruebel

Task 3
MULTIMODE RELEASE LIMITS

Justification for Developing Alternative Release Limits

Bases for the Release Limits

The EPA standards for radioactive waste disposal are unique in several ways [1], and this uniqueness must be taken into account when changes or extensions are considered. 40 CFR 191 [2] is different in philosophy, method of regulation, and level of protection than the recommendations made by the International Commission on Radiological Protection (ICRP) [3], high-level waste (HLW) standards being considered by other countries [1], and standards for environmental carcinogens promulgated by the EPA and other U.S. regulatory agencies [4,5]. The subject of this chapter is the generic derived release limits (Table 1 in Subpart B of Reference 2), briefly described below.

A single derived standard that limits time integrated radionuclide releases from repository boundaries applies to all HLW, SF, and TRU repositories and all release modes. The variability in lithosphere and biosphere surrounding the repositories, the site locations, and repository designs were not considered in the derivation. The dependence of the detriment on the release mode was also not considered. Because there is a large difference in dose attenuation by each disposal system and for each pathway, the single release limit forces the level of protection to be different for every site and every mode of release for each site. The derivation of the release limits omitted one of the most important components in the disposal system, omitted the three most likely release modes for the Yucca Mountain Project (YMP) and the Waste Isolation Pilot Plant (WIPP), and was based on two release modes that are highly improbable for these repositories.

Differences of opinion exist on this and other features of 40 CFR 191. Review panels [6], advisory boards [7,8], and individual investigators have recommended numerous modifications to all versions of the EPA radioactive

waste standards [2,9,10]. Most of the reviewers recommended substantial changes in regulation philosophy, format, and stringency, but there was not complete agreement on what to change or how to change it. Minor changes have been made that involve models and data, but there have been no changes to the promulgated standards or to any of the many drafts that involve the philosophy, methodology, or format of the standards. Although major changes of the type recommended probably would produce the most appropriate standards, they may not be practical at this stage of standards development and could result in challenges, unacceptable delays, and loss of public confidence. The second obstacle to change is obtaining agreement on all aspects of the standards. There is no single solution that is best for all situations and meets everyone's values. Whichever approach is selected, the development of the standards should be consistent and logical.

This chapter, which is based on one of the features of 40 CFR 191 discussed in Reference 11, suggests two extensions of the existing release limit standard without changing the form or the way it is developed. The suggested modifications pertain to development and application of the standards, rather than to the level of regulation.

The designs of radioactive waste repositories, performance assessments to evaluate them, and licensing are all driven by radiological criteria as much or more than they are by scientific and engineering principles. Therefore, accuracy and appropriateness of the regulations are essential. Apparent stringency of a standard alone does not assure safety if the standards are inaccurate or inappropriate for the application. Inappropriate standards can greatly increase the cost of a repository while offering inadequate protection to the populace.

The first step in the critique of the release limits is a review of development procedures, functions, and characteristics of derived standards. The development and resulting release limits in 40 CFR 191 are then reviewed and analyzed. Differences between the requirements and the standards and their possible causes are discussed. Problems that may be caused by the present standards are also covered. Suggestions are then made for modifying or extending the present standards but no specific recommendations are made. Methods of analyses are suggested for the proposed extensions.

Derivation of the Release Limits

The background documents [12,13,14,15] for 40 CFR 191 were reviewed to trace the development of the 10^4 year, time integrated release limits that apply to any surface of the repository controlled volume. The generic models

used by the EPA [13,14] considered four general modes of radionuclide releases to the environment: direct releases to a river, to an ocean, to a land surface, and from a volcanic or meteorite interaction with a waste repository. The purpose of the EPA analyses was to compute the number of premature cancer deaths per curie of each radionuclide released to the biosphere via the various release modes. The geologic formations and resulting dose attenuation between the repository and the release points were not included in any of the computations. It was assumed that all radionuclides leaving any surface of any repository boundary instantaneously enter the river, ocean, etc. This is an extremely conservative simplification because a large part of the risk attenuation takes place outside the repository.

All consequences were assumed to be independent of release rates and times of release. The models were based on world average values and contained many predictive assumptions. Except for the world population, which was assumed to be a constant 10^{10}, the values of all parameters were the current world average. The following is a list of some parameters used in the analyses with values that are very likely to change over time or are not presently well defined.

- World population
- Total flow rates of all rivers
- The amount of river water drunk by humans
- Fresh water fish consumption
- Fraction of river water used for irrigation
- Land area irrigated
- Consumption of irrigated crops, milk, and meat
- Number of people fed per unit area of irrigated land
- Salt water fish and shellfish consumption
- Resuspension factors
- Household shielding and occupancy factor
- Uptake factors
- Whole body effective dose equivalents
- Health effect conversion factors

The river release mode biosphere model included ingestion of drinking water, freshwater fish, food crop, milk, and beef; inhalation of resuspended material; and external exposure to ground contamination and air submersions. No other pathways and no sorption or sedimentation in the rivers were included in the river model. The derivation only accounted for the approximately 60% of water use that comes from surface water. Contaminated well water was omitted. Considering the uncertainties in the data, the model simplifications, and the variability with site location, the model for the river release mode could be either conservative or non-conservative.

The same source term was used for both the river and ocean models. That is, all nuclides leaving the controlled volume of the repository were assumed to enter both the river and the ocean instantaneously. There are several serious problems with the ocean model. The model represents all ocean waters with only two layered compartments (elements). It assumes that all releases instantaneously mix in the top compartment. The model contains no ocean circulation and only vertical nuclide transport is allowed between the two compartments. The coastal shelves, where rivers enter the ocean and aquatic food is concentrated, are not represented in the model. The Subseabed Disposal Project [16,17] showed that radioactive waste released to a shelf region would result in 10^4 year population doses that are about 100 times higher than those predicted by a model without shelf compartments. There are much better ocean models available than the one used in the EPA analysis [17,18]. Ingestion of fish and shellfish was the only pathway to humans that was considered in the EPA model. References 16 and 17 showed that these are not the only significant pathways when radionuclides enter the shelf as they would from a river. Harvest limits of the ocean fisheries would affect maximum population dose rates but were not included in the analyses.

Dose rates are proportional to release rates only when the radionuclides have a short residence time, such as in rivers or on ocean shelves [16]. In ocean waters beyond the shelves, dose rates are proportional to the accumulated inventory in the oceans (the time integrated release rate minus decay, scavenging, and removal) or concentration. The peak accumulation occurs long after the time of peak release rates. Therefore, doses from the deep ocean are very sensitive to when the radionuclides enter the ocean. The only reliable derived metric to represent doses from deep oceans was found to be the accumulated inventory, not the time integrated release. The present EPA model assumes exponentially decreasing release rates to the oceans, whereas any releases to the ocean during the period of regulation would start late and gradually increase. Considering all the omissions, simplifications, and predictive assumptions, the ocean model is probably non-conservative.

The land surface biosphere model represents waste brought to the surface by inadvertent human intrusion. These releases were assumed to be small and of short duration. Pathways to humans include food crop, milk, and beef ingestion, inhalation of resuspended material, and external exposure from ground and air contamination. This is a realistic model because instantaneous release is appropriate, fewer assumptions are required, and it is less dependent on predicted data values.

Carbon-14 was treated as a separate case and a single risk factor was used for all release modes. It was assumed that all C-14 is released as carbon dioxide. Although Reference 14 describes the release mode used in the

derivation as global atmospheric, the EPA stated they did not consider the possibility of pure gaseous release of C-14 when developing the standards [19].

The risk factors (cancers per curie) for each release mode apply uniformly to all repositories and all release modes. If release limits based on each of these risk factors had been applied at the points of release, PA could have selected the appropriate release mode for each pathway and included the entire lithospheric pathway in the analyses. However, the derivation was carried one step further, which caused non-uniform levels of protection, several inconsistencies, and the omission of an important component of the disposal system.

In the derivation of the release limits the EPA chose to base the values on only surface water releases, which is the combination of the river and ocean release modes. Risk factors for the other modes that had been computed [13] were not used. The release limits were derived by dividing the number of acceptable premature deaths from 10^3 MTHM of waste for 10^4 years (10) by the risk factors (deaths per curie) for each radionuclide. The limits are applied to releases from all surfaces of the CV rather than to the locations for which they were computed. They are also used for all repository locations, all applications, and all pathways.

The variabilities and uncertainties found in risk assessments also apply to derived standards. The single valued release limits in Table 1 of 40 CFR 191 are actually distributions that span from five to over nine orders of magnitude, depending on the radionuclide [12,14,20,21]. The Envirosphere Company performed a combined variability and uncertainty analysis on the river release model that was used to derive the release limits [20,21]. Probability distributions were assigned to 12 of the input parameters and stochastic analyses were conducted for each radionuclide. Uncertainties in process assumptions and varying expert opinion on probability distributions were not included in this study. Figure 1 is a typical predicted probability distribution of population risk per curie released to the generic river. Also shown is the risk factor that corresponds to the EPA release limit. Most release limit equivalents for the 13 radionuclides that were analyzed are above the medians of the risk factor distributions. In the Envirosphere analysis, 90 percent confidence intervals for release limits span an average of about four orders of magnitude. This illustrates that the use of derived standards does not reduce the total uncertainty in risk assessments. Reference 20 concluded that identifying specific repositories would considerably reduce many key uncertainties, which is another reason for not using a single generic release table.

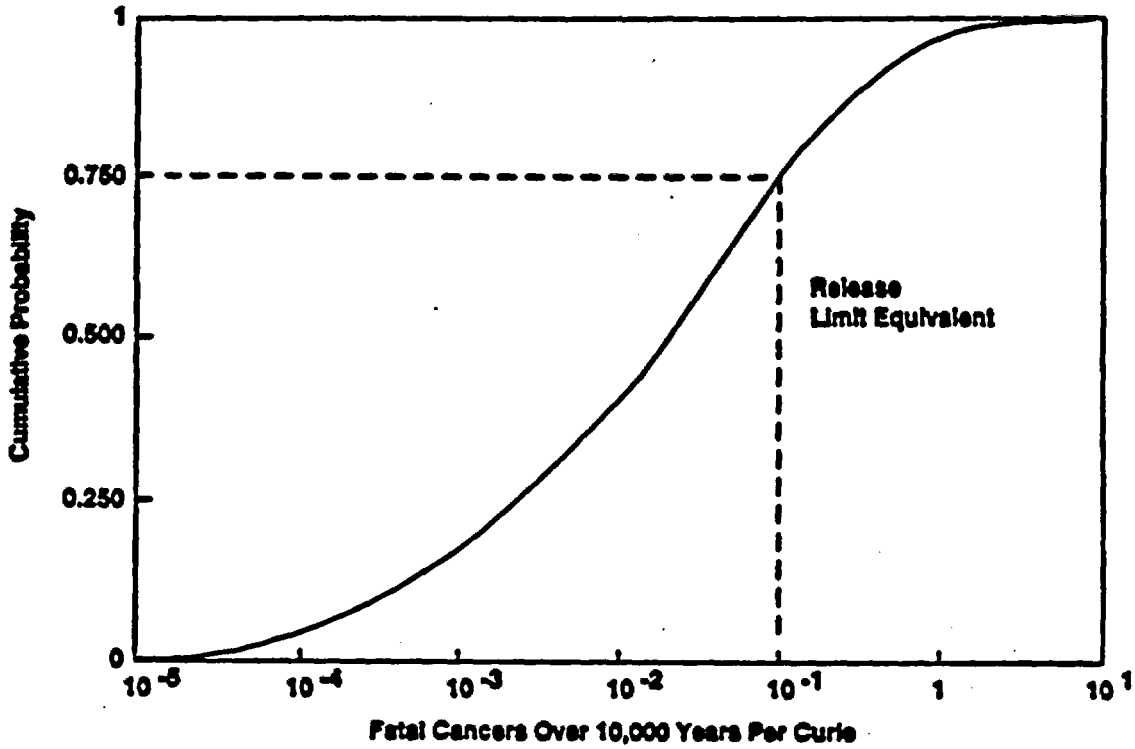

Figure 1. Probability distribution of the population risks per curie of Am-243 released to a generic river.

The derived release limits in Table 1 of Reference 2 specify the amount of each radionuclide that can be released from the controlled volume (CV) during the regulation period, if that is the only radionuclide in the inventory. For the actual waste inventories a summed normalized release (R) is used. The ratios of total release of each radionuclide (Q_i), to the release limit for that nuclide (RL_i), summed over all radionuclides, must be less than one (Equation 1).

$$Q_a/RL_a + Q_b/RL_b + \ldots + Q_n/RL_n < 1. \quad (1)$$

Problems with the Present Release Limits

There are several inconsistencies and discrepancies between the derivation and application of the 40 CFR 191 release limits, and the reasoning for some decisions is obscure. Most of the assumptions and limitations of the risk factor computations were clearly stated in Reference

14, but the results were not used accordingly. The following are the major features of the present release limits and their derivation that do not meet the requirements for derived standards, or are internally inconsistent.

1. One of the most important parts of the disposal system was left out of the release limit derivation. The assumption that all radionuclides, except C-14, that exit through any surface of the controlled volume of any repository, instantaneously enter both a generic river and the ocean is not realistic. This assumption ignores all forms of risk attenuation outside the repository boundary.

2. The release limits were not applied to the same circumstance for which they were derived. The generic model used in the derivation cannot represent specific disposal systems. The two systems currently being considered, the WIPP and YMP, have very dissimilar lithologies, hydrologies, inventories, distances to release points, pathways to humans, and system attenuation factors. The only release modes considered in the release limit derivation were rivers and oceans. It is unlikely that any waste from either repository would reach any surface water in 10^4 years. The only plausible non-gaseous release modes are withdrawal wells, which were not included in the derivation, and release to the land surface, which was computed but not used. It is unclear which release mode and which assumptions were used in the C-14 release limit derivation. The inappropriateness of the release modes used in the generic release limit derivation is illustrated in Figure 2.

3. Reference 22 requires that the regulatory process must "consist of establishing generally applicable <u>environmental</u> standards for the protection of the general environment." The EPA interprets this to mean that requirements may not be site-specific [23]. Environmental standards are level 1 criteria, which means that the fundamental criterion for HLW/SF of no more than 10 premature deaths from 10^3 MTHM of waste in 10^4 years, cannot be site-specific. Presently a single <u>derived</u> standard, that only limits radionuclide releases from repository boundaries, applies to all repositories. Since there is a large difference in the dose attenuation of each repository system and each mode of release, the single generic release limit forces the fundamental criterion (population safety) to be different for every site and for every mode of release at each site.

4. The degree of conservatism in the derivation is unknown. Some simplifications, predictions, and assumptions were conservative and some non-conservative. Even when the assumptions and omissions are definitely conservative, the level of conservatism is far from uniform for all repositories and all release modes. The standard is probably unrealistically conservative for all applications, but the confidence level of this conclusion is low.

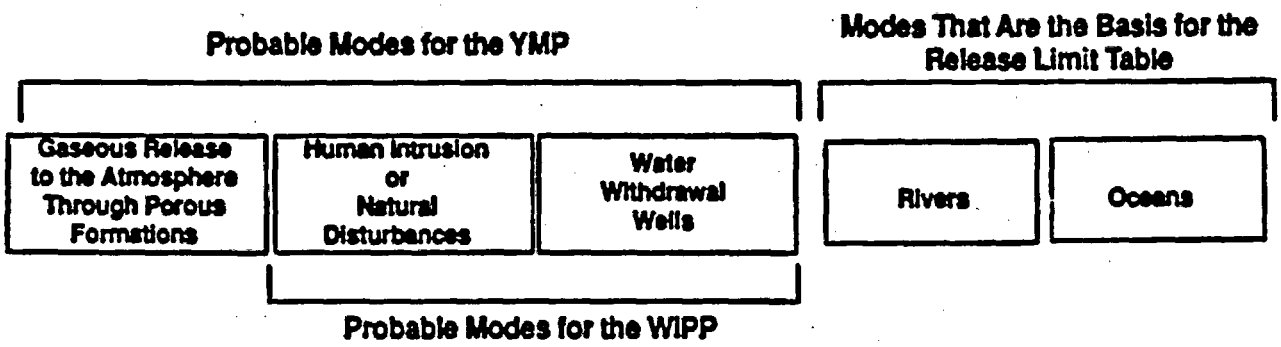

Figure 2. Possible release modes from radioactive waste repositories.

It may be that the release limits in 40 CFR 191 were promulgated before there was sufficient information available on repository designs, waste forms, site locations, and site properties. Deming [24] stated that the quality of a process suffers when standards and regulations are set before they have operational meaning. A workshop on radiological protection standards for the Subseabed Project [25] recommended that procedures for implementing dose standards should not be specified until the feasibility phase of the project is complete and that derived and prescriptive standards should not be set until a site is selected and the basic repository design is defined. Fundamental safety criteria should remain fixed but derived standards should reflect the characteristics of the waste disposal system. If derived standards are set before this can be accomplished, provisions should be made to update them if there are significant changes in repository configurations, data, or process definitions, or if new information shows that they are inappropriate for a specific application.

Alternatives to the Present Release Limits

Fundamental and dose standards are the only criteria levels that accurately limit risk for all repositories and all pathways [11,26]. Derived standards are requirements placed on the performance of components or processes in the disposal system, or on the flux or concentration of radionuclides at locations along pathways from the waste site to humans. Because derived standards require modeling of specific pathways and processes, using specific data, the ICRP cautions that they must only be used for the circumstances for which they were derived [27]. Therefore, generic derived standards that apply to all repositories and all pathways, such as the present release limits in 40 CFR 191, are inherently inaccurate and the more generic they are, the more inaccurate they are. All assumptions and simplifications in generic derivations must be conservative to assure adequate protection for all applications. The compounding of conservatism resulting from generic derivations can lead to excessive repository costs or exclude some repositories that have acceptable risk levels. In all cases the degree of conservatism is unknown and is not uniform for all applications. In addition, generic derived standards applied close to the repository force the fundamental criterion to change for each repository and each pathway. The generic derived release limits in 40 CFR 191 are briefly described below.

The present release limit table in 40 CFR 191 [2] is an example of a single generic derived standard. Only one release limit table is used for all release modes, and it is based on simultaneous release to all the world's rivers and oceans. Cumulative releases are evaluated at the boundary of the repository instead of at actual locations of release (Figure 3). Because the limits are based on releases to surface waters and the only release modes expected for the WIPP and the YMP are gaseous, land surface, and withdrawal wells, this single generic derived standard is not being applied to the circumstances for which it was derived. This distinction is illustrated in Figure 2. By applying the standard at the repository boundary instead of at the actual release locations, one of the most important components of the disposal system, risk attenuation between the boundary and the release locations, is left out of the risk assessment. Examples of the magnitude of geologic risk attenuation outside the repository boundary are given later in this chapter. In addition, the risk limit derivation was based on world-average parameters, which could cause inaccurate risk evaluations unless corrections for local conditions are made during performance assessments. These difficulties are to be expected with a single generic derived standard.

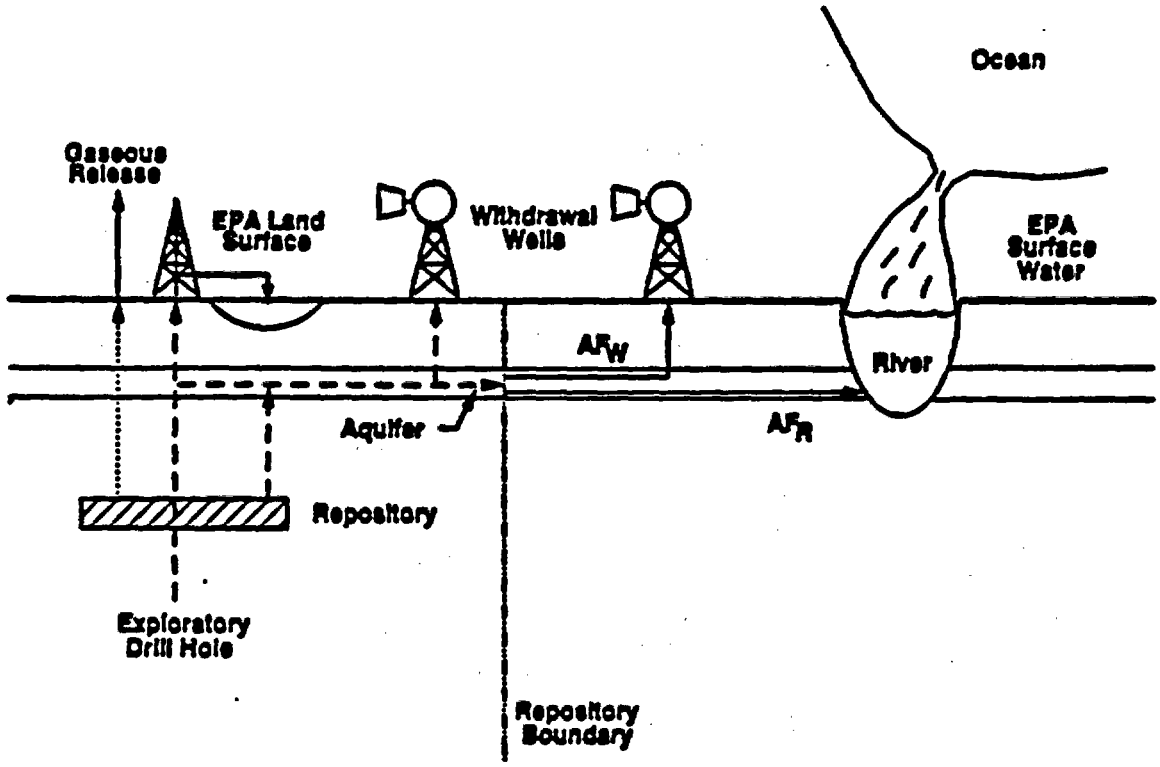

Figure 3. Schematic of a radioactive waste disposal system showing possible release modes and risk attenuation factors outside the repository.

At least four approaches can be taken to obtain more realistic risk assessments and regulation of HLW and TRU repositories than with the present release limits. The two approaches discussed in this chapter would retain the derived release limit format. The other two, which are not described here, would use the higher level, more precise dose limits as either the primary criterion or an option to release limits.

Site-and-Pathway-Specific Release Limits

One alternate approach that uses release limits is an extension of the present derivations. More is known about release modes and pathways than when the present release limits were computed, and better data and models are available. Two very different candidate sites have been selected, and site-specific definitions of the disposal systems between the repository and the release points can now be made. Although site-specific parameters probably

will change more with time than world-average parameters, site-specific analyses do not have the initial bias that world-average analyses do. More complete and appropriate release limits applied at the surfaces of the CV could be computed for each site using the same basic methodology that the EPA used in the original derivation.

The first step for each site would be to define the possible pathways and release modes. The generic illustration in Figure 2 applies to both sites. For the YMP, C-14 gas could escape through the unsaturated tuff. Other radionuclides could be brought to the land surface by human intrusion or abnormal natural events. Unsaturated flow could eventually transport radionuclides to the underlying aquifer, which would carry them off site. The most likely release mode for this pathway would be withdrawal wells. For the WIPP, human intrusion could bring radionuclides to the land surface, and drilling into the repository could enhance transport to the overlying aquifer. Diffusion and advection would eventually transport radionuclides to the aquifer, where they would be carried off site. The most likely release mode from the aquifer would be withdrawal wells. It is very unlikely that either repository would release any radionuclides to surface water in 10^4 years. Gaseous, land surface, and withdrawal wells are the only probable release modes for these two sites, but the river and ocean modes should be included unless they can be conclusively shown to be insignificant. Methods of updating the risk factors, computing the attenuation in the formations outside the CV, and allocating releases for each mode are suggested in Reference 11.

Release tables would be computed using the EPA method described in Reference 12. The maximum number of fatalities allowed by the fundamental criterion would be divided by the fatal cancers per curie for each site and each significantly different pathway. Upward movement of gas, radionuclide movement to the land surface caused by human intrusion, and radionuclide transport through an aquifer and subsequent withdrawal by a well would be the only pathways for release from the controlled volume, resulting in a maximum of three release tables for each repository. The summed normalized release limit would include the release fractions for each nuclide for each release pathway (up to three release fractions for each nuclide).

$$Q_{G,a}/RL_{G,a} + Q_{G,b}/RL_{G,b} + \ldots + Q_{L,a}/RL_{L,a} + Q_{L,b}/RL_{L,b} + \ldots + Q_{A,a}/RL_{A,a} + Q_{A,b}/RL_{A,b} + \ldots + Q_{A,n}/RL_{A,n} < 1. \quad (2)$$

Q is the computed 10^4 year release of a radionuclide from the controlled volume and RL is the release limit for that nuclide. The subscripts G, L, and A refer to the gas, land surface, and aquifer pathways, respectively, and

the subscripts a, b, . . ., n refer to the individual radionuclides listed in the tables.

This approach would use the appropriate release mode for each pathway and include all pathways and all components for the repositories covered in the derivation. It also would assure uniform safety requirements. The negative aspects of this approach are that the requirements would be site specific, would not cover future repositories, and would require additional site characterization and considerable time and effort to develop.

Description of Multimode Generic Release Limits

The other alternate approach that uses generic release limits would set release limits at the points of release to the biosphere for each release mode, which is just one step in the derivation prior to where they are presently set. The following sections describe multimode release limits, methods used in developing the five-column table of release limits as well as methods for combining releases from all modes into a single summed normalized release limit, corrections for repository locations and geologic risk attenuation, and suggestions for performance assessments. These multimode release limits would still be generic derived standards and consequently would contain some generalizations that may not apply to specific repositories, but the generalizations would be limited to the processes between the release points and humans. Multimode standards would apply uniformly to all repositories and all pathways. All major components in the disposal system would be included in risk assessments.

EPA generic analyses from the release points to humans would ensure uniform modeling of the biosphere for all applications (dashed lines in Figure 4). A five-column release table would be required to cover all possible release modes for generic repositories. As previously described, only three of the modes—gaseous, land surface, and withdrawal wells—are probable for the YMP and the WIPP. PA would be able to select the appropriate release mode for each pathway and include all disposal system components in the assessment. This is similar to the present approach, and most of the derivations of risk factors have already been completed [12,14]. The only difference is that the final step in the present derivation is eliminated. Release limits would still be computed by dividing the fundamental criterion by the risk factor for each radionuclide. Besides eliminating inconsistencies and omissions, this approach would not be site specific and would allow the fundamental standard to remain constant for all repositories and all pathways.

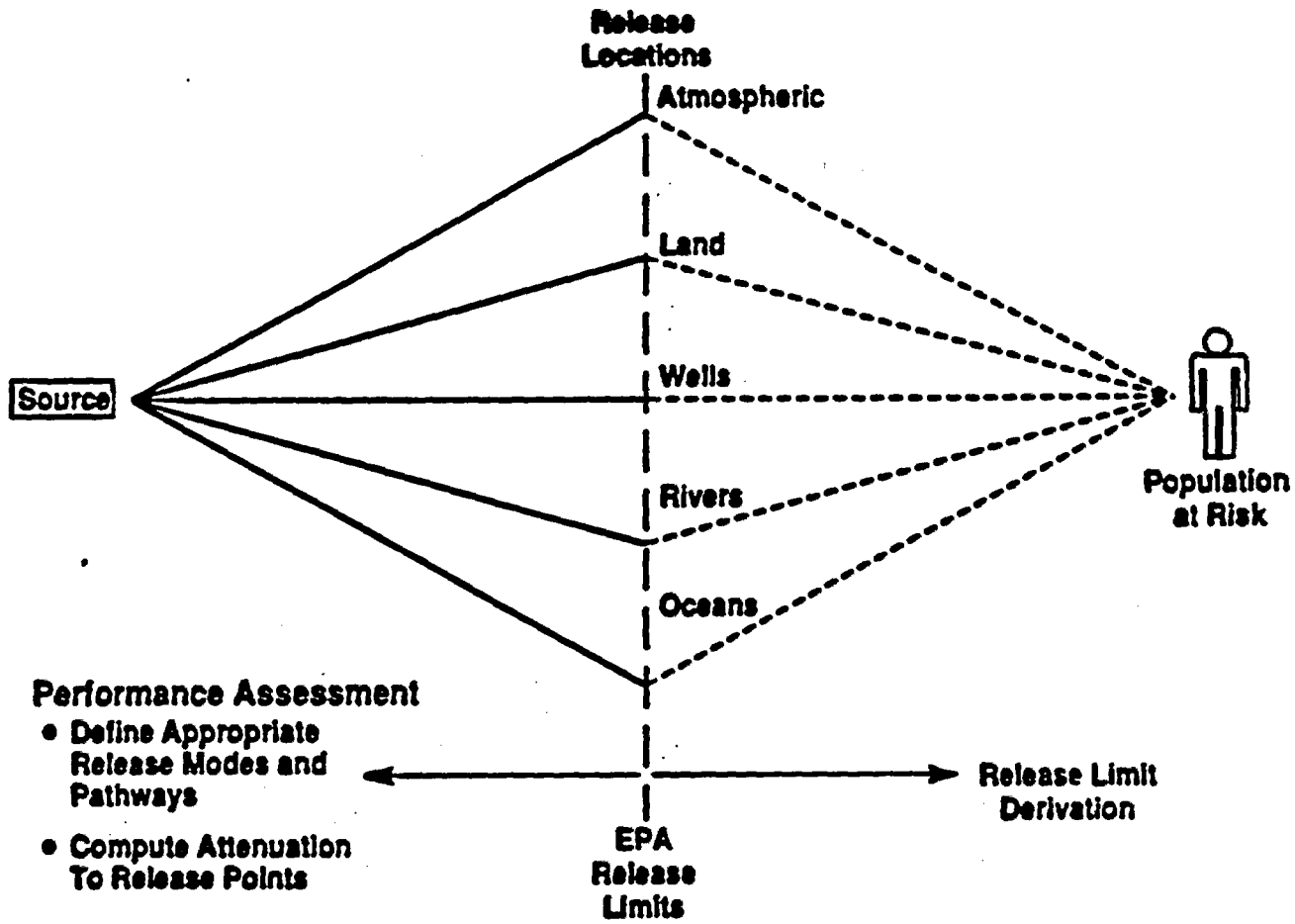

Figure 4. Multimode release limits in the risk assessment process.

Derivation and Implementation of Multimode Release Limits

The derivation of the present single generic table for release limits assumes that all the fresh water that is used comes from the world's rivers. Multimode release tables would separate fresh water into the actual sources, surface and groundwater. Surface water comes from lakes and rivers, but they can be combined into a river release mode to be consistent with EPA notation. The USGS publishes estimates of water sources and uses at 5-year intervals. Table 1 gives the 1985 percentages of water used for irrigation, livestock, and human drinking water that came from groundwater and surface water. Values are given for the Rio Grande Region (WIPP), the Great Basin (YMP), and United States. This table could be used to allocate water use to the well and river release modes. The values in Table 1 represent the percentages of each radionuclide that reach an aquifer by any means that would be available for well withdrawal or discharge to a river. It does not mean that all or any of these radionuclides will reach any points of release before they decay or during the 10,000 years of regulation. Some average value could be stated in the standards, or regulators or PA could select the percentages appropriate for each repository region.

Table 1. Fresh Water Sources in 1985 [Reference 28]

Region	Percentage	
	Groundwater	Surface Water
Rio Grande Region	28	72
Great Basin	19	81
United States	36	64

The EPA used world-average parameters to compute risk factors included in the present standards [14]. This approach is compatible with fundamental criteria for collective risk and can be used with multimode derivations. The pathways to humans include ingestion of drinking water, freshwater fish, food crops, milk, and beef; inhalation of resuspended material; and external exposure to ground contamination and air submersion. In order to be consistent with previous EPA usage, "river" includes all sources of fresh surface water.

Adjustments of Generic Release Limits

Generic or world average parameter values are used to compute multimode release limits, just as they were in the derivation of the present standards. Therefore they may not represent the actual radionuclide pathways or risk of specific repository locations. There are many site adjustment factors (SAFs) that could be applied to release limits for specific repositories to compensate for these generalizations. Alternatively, generic SAFs could be defined in the standards that would apply to all sites or the selection of site specific adjustment factors could be left to the implementing agency for each repository. Generic SAFs have the advantage of consistent use for all repositories, and an equitable selection of SAFs that increase and decrease the release limits would be predefined. The disadvantages of generic adjustment factors are they may overcorrect or undercorrect at any given site. The advantage of developing SAFs for each repository is local conditions such as repository location relative to rivers, oceans, agriculture, and populations at the time of assessment can be defined more precisely. The disadvantage is the potential for nonuniformity in the selection of SAFs and demands for an unreasonable number of SAFs.

Either option should produce more accurate predictions of actual risk than generic analyses with no site adjustments. The magnitude of the net adjustment would depend on site characteristics and may be insignificant for

some repositories. Generic SAFs for two of the most obvious cases are suggested for the river and well release modes in their respective sections. The alternative to SAFs for repositories that can not be adequately assessed with generic release limits is the use of collective dose limits, which do not require adjustments, but require additional site characterization and PA.

- River Release Mode

There are several generalizations and assumptions that are common to both the EPA derivation of river release limits [14] and the derivation of well release limits presented in this report. They are:

1. The ratio of local consumption of water and food to local water flow rates equals the ratio of world consumption to world water flow rates.

2. The ratio of local population at risk to local contaminated water flow rates equals the ratio of world population to world water flow rates.

All risk factor pathway equations for river and well release modes are of the form:

$$D'_i = \frac{Q'}{F} \cdot f(\text{individual use rates and conversion factors}) \qquad (3)$$

where

D_i = dose rate to individuals (rem/person-yr)

Q' = release rate of radionuclide to the environment (Ci/yr)

F = river or groundwater flow rate (km^3/yr).

Multiplying by population (P) and integrating over time produces the final form of the risk factor equations:

$$\frac{D_P}{Q} = \frac{P}{F} \cdot f(\text{individual use rates and conversion factors}) \qquad (4)$$

where

D_P = population dose (rem)

Q = total release of radionuclide to the environment (Ci).

Except for the fish consumption pathway, which applies to only the river mode, all biosphere pathway equations are the same for the river and well release modes. The only difference in the risk factor is caused by the radionuclide concentration in the water (Q'/F). With the assumptions used in Reference 14, the concentration is a linear function of total world volumetric flow rates. The total volumetric flow rates for both modes are computed by dividing the volumes of each part of the hydrosphere by their exchange activities. This information is available in a UNESCO report for all the major hydrosphere divisions [30] and is summarized in Table 2.

Table 2. World Hydrosphere Activities [Reference 30]

Part of Hydrosphere	Volume (km^3)	Exchange Activity (yrs)	Volumetric Flow (km^3/yr)
Rivers	1.2×10^3	.032	3.8×10^4
Lakes	2.3×10^5	10.	2.3×10^4
Active Groundwater	4.0×10^6	330.	1.2×10^4
Total Groundwater	6.0×10^7	5000.	1.2×10^4
World Oceans	1.4×10^9	3000.	4.6×10^5

Therefore, the risk factors for each biosphere pathway for the well mode will be the river mode values times the river flow divided by the ground water flow in Table 2. Except for fresh water fish consumption, which would be totally in the river mode, the risk factors for the river pathway would be weighted by the water source fractions in Table 1. The world-average radionuclide concentration in river waters was an independent variable in the risk factor equations for all pathways [14]. It was evaluated by dividing the reference release of each nuclide (1 curie) by the total volumetric flow rate of all rivers (3×10^4 km^3/yr). This flow rate is a good average of the lake and river divisions, which comprise surface water sources. Except for updating the release limits for the river mode with more recent data and removing ocean releases, which would be a separate release mode, this derivation is complete.

The derivation of the risk factors for the river release mode, using world-average parameters, assumes that the entire drainage system of all rivers is contaminated with the released radionuclides regardless of the repository location [12]. Figure 5 shows that, in reality, only the downstream section of the tributary that is fed by groundwater passing the

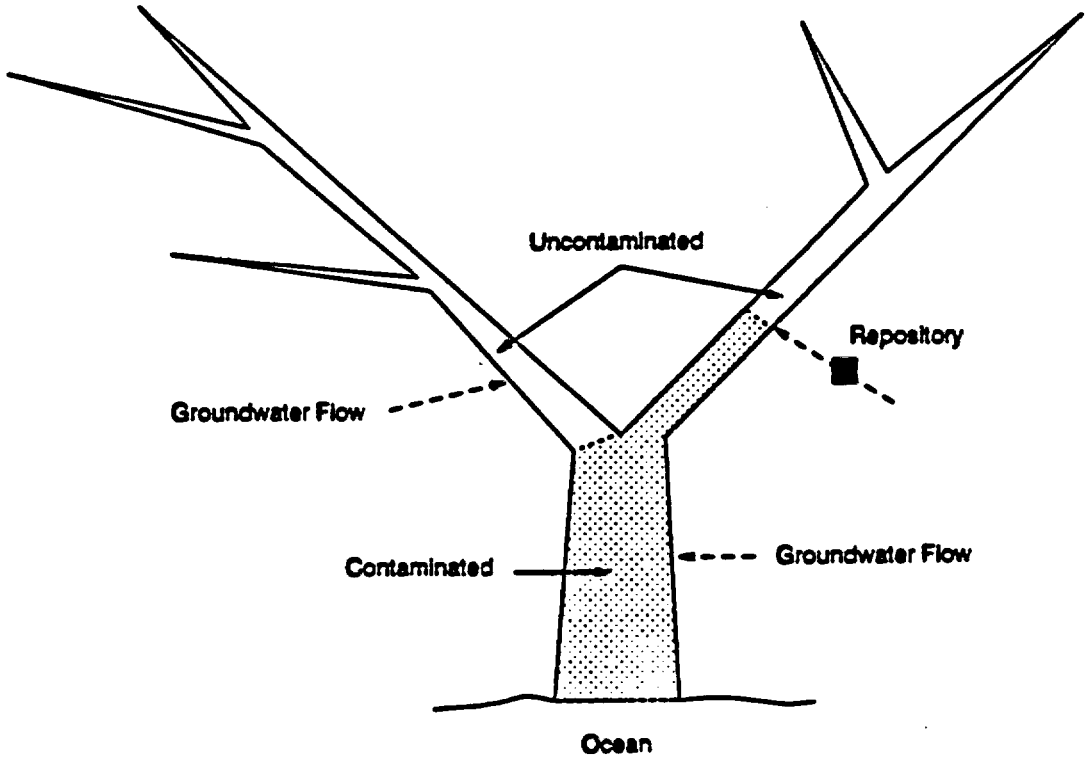

Figure 5. Generic river basin for the river release mode.

repository is contaminated. The ratio of the actual available contaminated water to the total available water in the drainage system can be approximated by dividing the sum of the products of contaminated tributary lengths and flow rates by equivalent sums of all tributaries:

$$SAF_R = \frac{\sum_{i=1}^{n} (L_{C(i)} * F_{C(i)})}{\sum_{i=1}^{n} (L_{C(i)} * F_{C(i)}) + \sum_{j=1}^{n} (L_{U(j)} * F_{U(j)})} \qquad (5)$$

SAF_R is the site adjustment factor used to correct the risk factors for the river release mode, L is the length of the river segments, and F is the volumetric flow rate of that segment. The subscripts C and U refer to contaminated and uncontaminated segments, respectively. If the correction is

applied directly to the release limits rather than to the risk factors, the reciprocal of the SAF_R is used. This definition of water availability would be compatible with the present derivation.

Attenuation factors (AFs) for nuclide transport in aquifers depend on flow rates, diffusion, dispersion, retardation, decay rates of the nuclides, the duration of regulation, and the performance of all preceding repository components [29]. Some examples of geologic risk attenuation between a repository and a river are given later in this section. The AFs for the river release mode would be an extension of the present assessments inside the controlled volume.

- Well Release Mode

Pathways for the well release mode would be the same as those for the river mode except for fish consumption. The radionuclide concentrations in groundwater used to compute risk factors for the well mode must be based on world averages, the same as the river mode, if the standards are to be consistent.

The present derivation of the river risk factors used a volumetric flow rate of 3×10^4 km^3/yr. The flow rates for groundwater are a factor of 2.5 lower, which means the radionuclide concentrations in groundwater are a factor of 2.5 higher. Because the risk factors in the EPA derivations [14] are linear functions of concentration, the risk factors for the two modes would scale with concentration. The ratio of release limits for the well release mode to those for the river mode would range from 0.400 for Zr-93 to 0.803 for Cs-137. This variation is caused by fish consumption in the river mode.

The development of the well release limits is parallel to that of the current river release limits. Both are based on world populations and flow rates. Neither depend on the actual size of the aquifer or river basin or the water velocities because of the linear hypotheses and the use of collective population doses in the criteria. In base case performance assessments, the rivers are assumed to be at their present location and the groundwater plume is computed based on present hydrology. Withdrawal wells can distort the contaminated plume by drawing uncontaminated waters into the plume as illustrated by Well 1 in Figure 6 or by enlarging the plume (Well 2). Over the 10,000 year regulation period, these effects should tend to cancel.

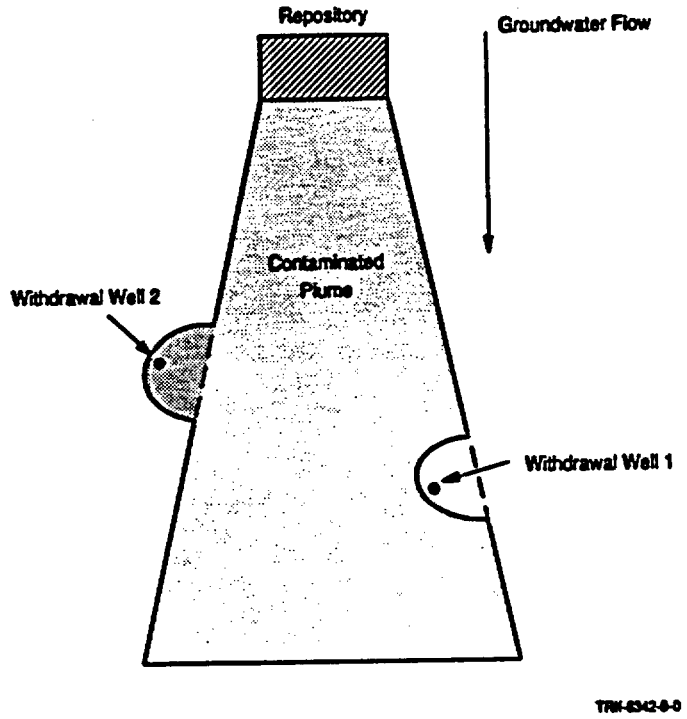

Figure 6. Deformation of a contaminated groundwater plume caused by withdrawal wells.

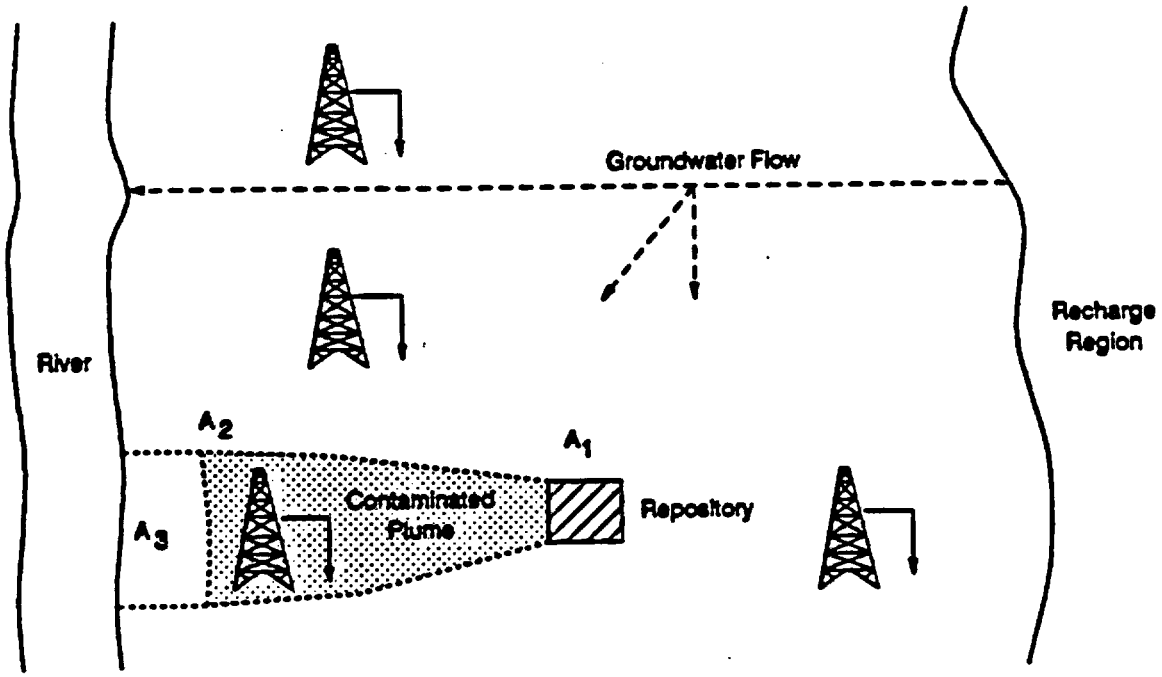

Figure 7. Generic groundwater diagram for the well release mode.

This derivation of the limits for the well release mode using world average parameters assumes all groundwater from the recharge area to the locations where it enters surface waters is contaminated. This is similar to the river derivation problem. Figure 7 shows that, in reality, wells up-gradient of the repository do not produce contaminated water. In addition, during the 10,000-year regulation period, the contaminated plume may not reach the discharge location, and some uncontaminated water also would be withdrawn down-gradient from the repository. The ratio of contaminated to total available water can be approximated by dating the water at the repository (A_1), at the point that the radionuclides are expected to reach in 10,000 years (A_2), and at the location where groundwater is discharged to a river (A_3), as shown in Figure 7. The site adjustment factor (SAF_W) can then be approximated by dividing the difference in the ages of the water at the farthest point of projected radionuclide migration in 10,000 years (A_2) and at the repository (A_1) by the age of the water at the point of discharge to the river (A_3):

$$SAF_W = \frac{A_2 - A_1}{A_3} \tag{6}$$

However, if the contaminated plume is projected to reach a river within 10,000 years, the SAF_W can be approximated by the following formula:

$$SAF_W = \frac{A_3 - A_1}{A_3} \tag{7}$$

The risk factors could be corrected by these ratios. If the correction is applied directly to the release limits rather than to the risk factors, the reciprocal of the SAF_W is used.

Computations of attenuation factors would be similar to those for the river release mode. Over a 10,000-year period, withdrawal wells could be located anywhere in the contaminated plume. Therefore, assuming uniform withdrawal in the plume for the entire time is reasonable. This is in contrast to the single fixed distance for the river release mode.

- Ocean Release Mode

Ocean risk factors in References 12 and 14 were compared with those computed with the MARINRAD [31] computer program and deep ocean and shelf models for the Subseabed Disposal Project [17,18]. The comparison showed

that the ocean risk factors used to derive the present release limits were up to a factor of 100 too low [11]. This difference was confirmed by a preliminary study of ocean risk factors that are defined in a letter from R.D. Klett (SNL) to D. Ensminger (TASC) concerning the "Ocean Model forRelease Limit Derivation," dated October 22, 1991. The preliminary study was conducted by TASC and explained in a letter from S. Oston (TASC) to R. Williams (EPRI) about "Ocean Pathway Modeling," dated December 10, 1991. A thorough study of the ocean mode should be conducted with a program such as MARINRAD and more detailed coupled shelf and deep ocean models.

No correction factors for repository location are required for the ocean mode. With the conservative assumptions of no risk attenuation in the rivers and the return of all irrigation water to the rivers, the same geologic AFs could be used for the river and ocean release modes for each repository.

- Land Release Mode

Changing the method of computing risk factors for the land mode is not necessary, but the risk factors could be updated using the latest data. No corrections for repository location and no computations of risk attenuation are required for the land release mode.

- Atmospheric Release Mode

The method of computing C-14 risk factors for release to the atmosphere is consistent with the other derivations in Reference 14. Updating the analysis with a later version of the global circulation model would only increase the release limit by a factor of 1.4. For completeness, a value for I-129 [32] could be added. One alternative would be to base the C-14 release limit on the Clean Air Act [33] and the corresponding regulations promulgated by the EPA [34]. However, 40 CFR 61, Subpart 1 states that it does not apply to facilities regulated by 40 CFR 191, Subpart B. It is also currently in litigation. No corrections for repository location and no additional computations of attenuation are required.

- Risk Factors

This section presents the derivation results in terms of risk factors, the premature fatal cancers induced over 10,000 years for each curie of the various radionuclides that may be released to the accessible environment. These risk factors could be used to develop the radionuclide release limits in Table 1 of 40 CFR Part 191. Risk factors in cancers per TBq are shown in Table 3, and risk factors in cancers per curie are shown in Table 4.

Table 3. Fatal Cancers per TBq Released to the Accessible Environment over 10,000 Years for Multiple Release Modes

Nuclide	Cancers per TBq				
	River[a]	Well[b]	Ocean[c]	Land[a]	Atmosphere
C-14	NA	NA	NA	NA	1.57E+00[a]
Ni-59	1.24E-03	3.03E-03		1.83E-05	NA
Sr-90	6.08E-01	1.51E+00	6.62E-04	1.02E-03	NA
Zr-93	4.08E-03	1.02E-02	1.06E-03	6.10E-04	NA
Tc-99	9.86E-03	2.41E-02	4.29E-05	1.53E-06	NA
Sn-126	2.84E+00	6.95E+00	2.89E+00	3.73E-02	NA
I-129	2.18E+00	5.43E+00	7.32E-03	1.07E-01	6.72E+00[d]
Cs-135	2.09E-01	4.69E-01	1.73E-03	1.55E-02	NA
Cs-137	2.89E-01	3.60E-01	1.33E-02	5.91E-04	NA
Sm-151	2.53E-04	6.14E-04		1.81E-06	NA
Pb-210	3.19E+00	7.03E+00		4.10E-03	NA
Ra-226	4.40E+00	1.05E+01		1.52E-01	NA
Ra-228	6.51E-01	1.52E+00		4.24E-04	NA
Ac-227	1.80E+00	4.34E+00	4.13E+00	3.35E-03	NA
Th-229	9.42E-01	2.30E+00	4.64E+00	5.13E-01	NA
Th-230	1.45E+01	3.60E+01		1.04E+01	NA
Th-232	9.18E+00	2.29E+01		1.02E+01	NA
Pa-231	4.00E+00	9.87E+00	1.60E+00	6.37E-01	NA
U-233	5.81E-01	1.44E+00	2.50E-02	2.03E-02	NA
U-234	5.29E-01	1.31E+00		1.77E-02	NA
U-235	5.86E-01	1.45E+00	2.26E-02	2.27E-02	NA
U-236	5.00E-01	1.24E+00		1.67E-02	NA
U-238	5.56E-01	1.38E+00		1.86E-02	NA
Np-237	2.15E+00	3.27E+00	3.89E-01	3.27E-03	NA
Pu-238	1.14E+00	2.82E+00		8.37E-03	NA
Pu-239	1.34E+00	3.32E+00	1.55E+00	1.68E-01	NA
Pu-240	1.31E+00	3.23E+00	1.55E+00	1.41E-01	NA
Pu-241	5.86E-02	1.45E-01	0.00E+00	6.75E-05	NA
Pu-242	1.29E+00	3.20E+00		1.71E-01	NA
Am-241	1.46E+00	3.28E+00	5.48E+00	2.84E-02	NA
Am-243	1.54E+00	3.49E+00	5.37E+00	6.62E-02	NA
Cm-245	2.73E+00	6.58E+00	8.07E+00	2.18E-01	NA
Cm-246	1.35E+00	3.25E+00		9.56E-02	NA

Sources:

[a] Reference 12

[b] This report

[c] Preliminary incomplete analysis by TASC using MARINRAD

[d] Reference 32 using 0.04 cancers per Sv

Table 4. Fatal Cancers per Curie Released to the Accessible Environment over 10,000 Years for Multiple Release Modes

Nuclide	Cancers per curie				
	River[a]	Well[b]	Ocean[c]	Land[a]	Atmosphere
C-14	NA	NA	NA	NA	5.83E-02[a]
Ni-59	4.61E-05	1.12E-04		6.79E-07	NA
Sr-90	2.25E-02	5.60E-02	2.45E-05	3.76E-05	NA
Zr-93	1.51E-04	3.77E-04	3.94E-05	2.26E-05	NA
Tc-99	3.65E-04	8.93E-04	1.59E-06	5.65E-08	NA
Sn-126	1.05E-01	2.57E-01	1.07E-01	1.38E-03	NA
I-129	8.07E-02	2.01E-01	2.71E-04	3.96E-03	2.49E-01[d]
Cs-135	7.73E-03	1.74E-02	6.39E-05	5.75E-04	NA
Cs-137	1.07E-02	1.33E-02	4.92E-04	2.19E-05	NA
Sm-151	9.38E-06	2.27E-05		6.71E-08	NA
Pb-210	1.18E-01	2.61E-01		1.52E-04	NA
Ra-226	1.63E-01	3.87E-01		5.62E-03	NA
Ra-228	2.41E-02	5.62E-02		1.57E-05	NA
Ac-227	6.67E-02	1.61E-01	1.53E-01	1.24E-04	NA
Th-229	3.49E-02	8.51E-02	1.72E-01	1.90E-02	NA
Th-230	5.38E-01	1.33E+00		3.86E-01	NA
Th-232	3.40E-01	8.47E-01		3.76E-01	NA
Pa-231	1.48E-01	3.66E-01	5.94E-02	2.36E-02	NA
U-233	2.15E-02	5.33E-02	9.25E-04	7.51E-04	NA
U-234	1.96E-02	4.86E-02		6.54E-04	NA
U-235	2.17E-02	5.38E-02	8.36E-04	8.42E-04	NA
U-236	1.85E-02	4.59E-02		6.18E-04	NA
U-238	2.06E-02	5.11E-02		6.90E-04	NA
Np-237	7.95E-02	1.21E-01	1.44E-02	1.21E-04	NA
Pu-238	4.23E-02	1.05E-01		3.10E-04	NA
Pu-239	4.97E-02	1.23E-01	5.73E-02	6.23E-03	NA
Pu-240	4.84E-02	1.20E-01	5.73E-02	5.22E-03	NA
Pu-241	2.17E-03	5.36E-03		2.50E-06	NA
Pu-242	4.79E-02	1.18E-01		6.34E-03	NA
Am-241	5.42E-02	1.22E-01	2.03E-01	1.05E-03	NA
Am-243	5.72E-02	1.29E-01	1.99E-01	2.45E-03	NA
Cm-245	1.10E-01	2.44E-01	2.99E-01	8.08E-03	NA
Cm-246	4.99E-02	1.20E-01		3.54E-03	NA

Sources:

[a] Reference 12

[b] This report

[c] Preliminary incomplete analysis by TASC using MARINRAD

[d] Reference 32 using 0.04 cancers per Sv

Table 5. Cumulative Release Limits for 10,000 Years (TBq per 100,000 MTHM) for Multiple Release Modes

Nuclide	Release Limit (TBq per 100,000 MTHM)				
	River[a]	Well[b]	Ocean[c]	Land[a]	Atmosphere
C-14	NA	NA	NA	NA	6E+02[a]
Ni-59	8E+05[a]	3E+05		5E+07	NA
Sr-90	2E+03	7E+02	2E+06	1E+06	NA
Zr-93	2E+05	1E+05	9E+05	2E+06	NA
Tc-99	1E+05	4E+04	2E+07	7E+08	NA
Sn-126	4E+02	1E+02	3E+02	3E+04	NA
I-129	5E+02	2E+02	1E+05	9E+03	1E+02[d]
Cs-135	5E+03	2E+03	6E+05	6E+04	NA
Cs-137	3E+03	3E+03	8E+04	2E+06	NA
Sm-151	4E+06	2E+06		6E+08	NA
Pb-210	3E+02	1E+02		2E+05	NA
Ra-226	2E+02	1E+02		7E+03	NA
Ra-228	2E+03	7E+02		2E+06	NA
Ac-227	6E+02	2E+02	2E+02	3E+05	NA
Th-229	1E+03	4E+02	2E+02	2E+03	NA
Th-230	7E+01	3E+01		1E+02	NA
Th-232	1E+02	4E+01		1E+02	NA
Pa-231	3E+02	1E+02	6E+02	2E+03	NA
U-233	2E+03	7E+02	4E+04	5E+04	NA
U-234	2E+03	8E+02		6E+04	NA
U-235	2E+03	7E+02	4E+04	4E+04	NA
U-236	2E+03	8E+02		6E+04	NA
U-238	2E+03	7E+02		5E+04	NA
Np-237	5E+02	3E+02	3E+03	3E+05	NA
Pu-238	9E+02	4E+02		1E+05	NA
Pu-239	7E+02	3E+02	6E+02	6E+03	NA
Pu-240	8E+02	3E+02	6E+02	7E+03	NA
Pu-241	2E+04	7E+03		1E+07	NA
Pu-242	8E+02	3E+02		6E+03	NA
Am-241	7E+02	3E+02	2E+02	4E+04	NA
Am-243	6E+02	3E+02	2E+02	2E+04	NA
Cm-245	4E+02	2E+02	1E+02	5E+03	NA
Cm-246	7E+02	3E+02		1E+04	NA

Sources:

[a] Reference 12

[b] This report

[c] Preliminary incomplete analysis by TASC using MARINRAD

[d] Reference 32 using 0.04 cancers per Sv

Table 6. Cumulative Release Limits for 10,000 Years (curies per 100,000 MTHM) for Multiple Release Modes

Nuclide	Release Limit (curies per 100,000 MTHM)				
	River[a]	Well[b]	Ocean[c]	Land[a]	Atmosphere
C-14	NA	NA	NA	NA	2E+04[a]
Ni-59	2E+07	9E+06		1E+09	NA
Sr-90	4E+04	2E+04	4E+07	3E+07	NA
Zr-93	7E+06	3E+06	3E+07	4E+07	NA
Tc-99	3E+06	1E+06	6E+08	2E+10	NA
Sn-126	1E+04	4E+03	9E+03	7E+05	NA
I-129	1E+04	5E+03	4E+06	3E+05	4E+03[d]
Cs-135	1E+05	6E+04	2E+07	2E+06	NA
Cs-137	9E+04	8E+04	2E+06	5E+07	NA
Sm-151	1E+08	4E+07		1E+10	NA
Pb-210	8E+03	4E+03		7E+06	NA
Ra-226	6E+03	3E+03		2E+05	NA
Ra-228	4E+04	2E+04		6E+07	NA
Ac-227	1E+04	6E+03	7E+03	8E+06	NA
Th-229	3E+04	1E+04	6E+03	5E+04	NA
Th-230	2E+03	8E+02		3E+03	NA
Th-232	3E+03	1E+03		3E+03	NA
Pa-231	7E+03	3E+03	2E+04	4E+04	NA
U-233	5E+04	2E+04	1E+06	1E+06	NA
U-234	5E+04	2E+04		2E+06	NA
U-235	5E+04	2E+04	1E+06	1E+06	NA
U-236	5E+04	2E+04		2E+06	NA
U-238	5E+04	2E+04		1E+06	NA
Np-237	1E+04	8E+03	7E+04	8E+06	NA
Pu-238	2E+04	1E+04		3E+06	NA
Pu-239	2E+04	8E+03	2E+04	2E+05	NA
Pu-240	2E+04	8E+03	2E+04	2E+05	NA
Pu-241	5E+05	2E+05		4E+08	NA
Pu-242	2E+04	8E+03		2E+05	NA
Am-241	2E+04	8E+03	5E+03	1E+06	NA
Am-243	2E+04	8E+03	5E+03	4E+05	NA
Cm-245	1E+04	4E+03	3E+03	1E+05	NA
Cm-246	2E+04	8E+03		3E+05	NA

Sources:

[a] Reference 12

[b] This report

[c] Preliminary incomplete analysis by TASC using MARINRAD

[d] Reference 32 using 0.04 cancers per Sv

- Development of Release Limits for 40 CFR Part 191

The analyses described in this chapter could be used to develop radionuclide release limits that correspond to the level of protection chosen for the containment requirements of 40 CFR Part 191 (Section 191.13). The 1985 BID [12] describes the procedure used to determine release limits from the risk factors. The maximum number of fatalities allowed by the fundamental criterion were divided by the fatal cancers per curie for each release mode and each radionuclide. The release limits in SI units are shown in Table 5, and the release limits in curies and associated units are shown in Table 6.

- Summed Normalized Releases

The four additional columns of release limits would be computed using the EPA method described in Reference 12. The maximum number of fatalities allowed by the fundamental criterion would be divided by the fatal cancers per curie for each release mode and each radionuclide. These computations would result in a maximum of five columns in a release limit table. The summed normalized release limit for each scenario or event would include the release fractions for each nuclide for each release mode.

$$Q_{A,a}/RL_{A,a} + Q_{A,b}/RL_{A,b} + \ldots + Q_{L,a}/RL_{L,a} + Q_{L,b}/RL_{L,b} + \ldots +$$

$$Q_{W,a}/RL_{W,a} + Q_{W,b}/RL_{W,b} + \ldots + Q_{R,a}/RL_{R,a} + Q_{R,b}/RL_{R,b} + \ldots +$$

$$Q_{O,a}/RL_{O,a} + Q_{O,b}/RL_{O,b} + \ldots + Q_{O,n}/RL_{O,n} < 1. \qquad (8)$$

Q is the computed 10^4 year release of a radionuclide for each release mode at the release location, and RL is the release limit for that nuclide and release mode. The subscripts A, L, W, R, and O refer to the atmospheric, land, well, river, and ocean release modes, respectively, and the subscripts a, b, . . ., n refer to the individual radionuclides listed in the tables. The effects of multimode release tables on the release CCDF would be to change the magnitude of the normalized release (R) for each scenario or event relative to the present single release method as illustrated in Figure 8. The probabilities of the individual scenarios or events that make up the CCDF would be unchanged.

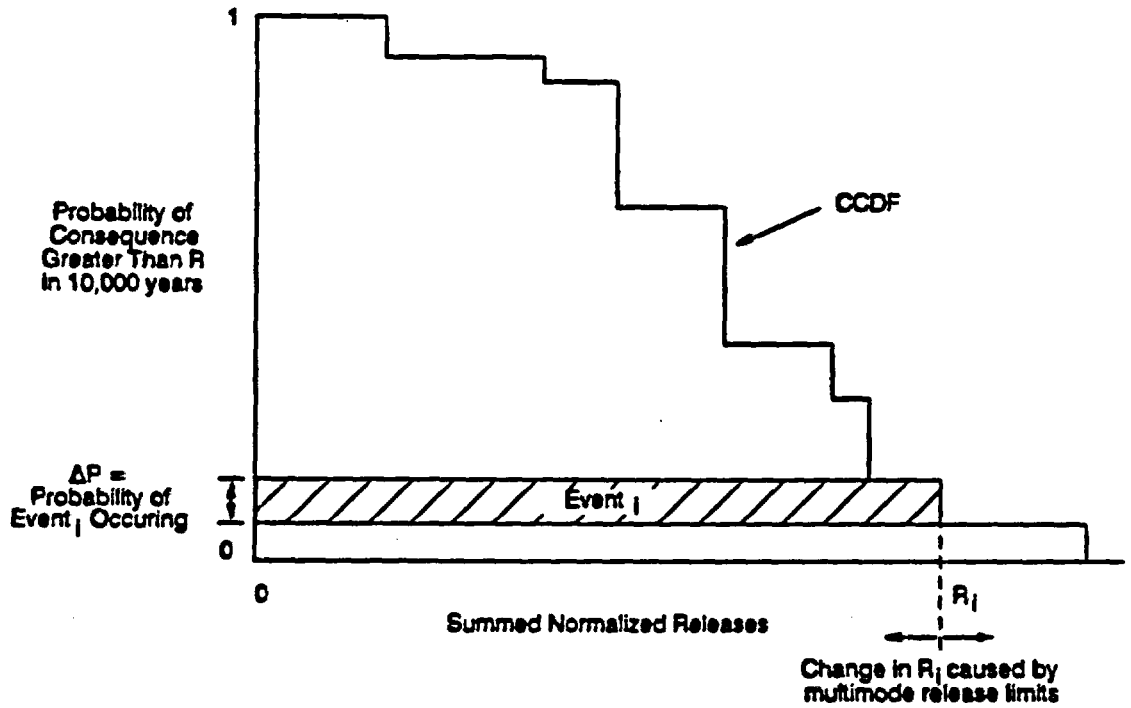

Figure 8. Effects of multimode release limits on the release CCDF.

- Geologic Risk Attenuation

A study of parametric geologic attenuation factors (AFs) was conducted to show the significance of the geologic component that is omitted in the present standards (controlled volume to location of release to the environment) and show the sensitivity of AFs to the input parameters. The input parameters were: groundwater velocity, retardation factor, dispersivity, distance from the repository in the direction of groundwater flow, duration of regulation, radionuclide half life, time of release from the repository, and rate of release. Figures 9 and 10 are examples of the results.

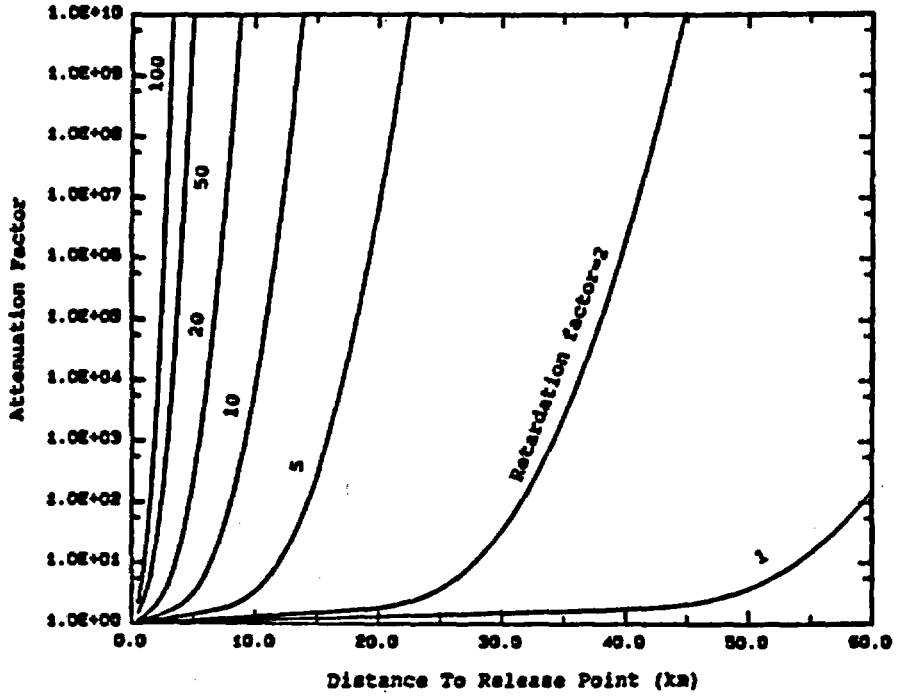

Figure 9. Attenuation factor sensitivity (Velocity = 5 m/yr, Half Life = 10,000 yrs, Dispersivity = 200 m, Instantaneous release at the time of repository closure).

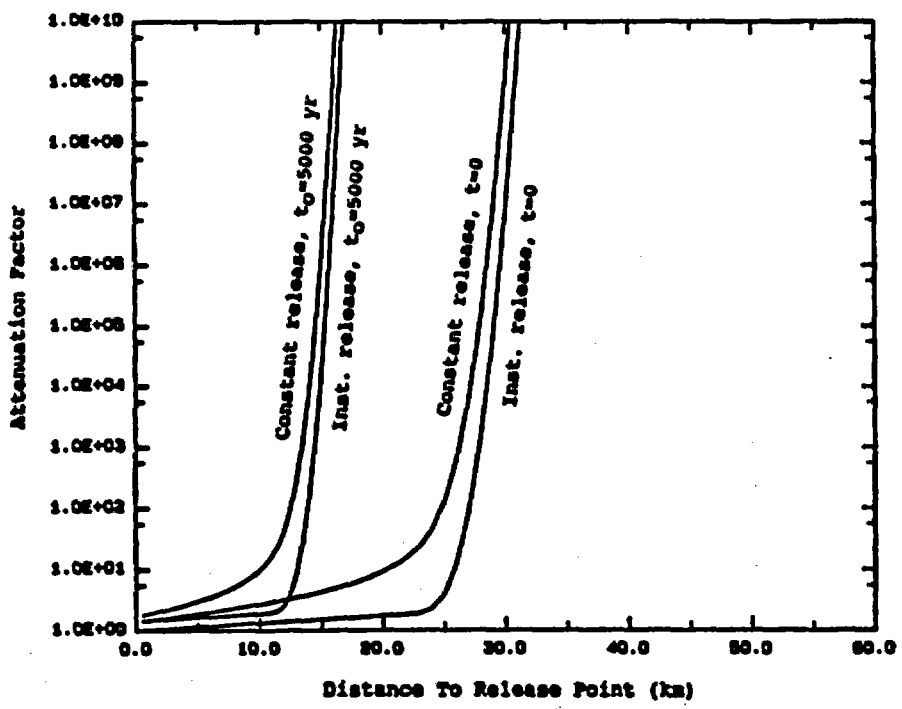

Figure 10. Attenuation factor sensitivity (Velocity = 5 m/yr, Half Life = 10,000 yrs, Dispersivity = 200 m, Retardation Factor = 2).

Figure 9 shows the retardation sensitivity when groundwater velocities are similar to those at the WIPP. Instantaneous release is assumed, which is unrealistically conservative even for human intrusion breaching an underlying brine pocket immediately after repository closure. The retardation factors for all the actinides in TRU waste are greater than 10 [35], so the longest travel distance for any nuclide would be about 12 km, indicating that the geologic component of the disposal system that is outside the controlled volume is too important to be omitted from performance assessments. Figure 10 shows the large effects of the time and rate of release from the repository on the distance traveled by a weakly retarded nuclide in 10,000 years. These important features of repository design cannot be evaluated when the geologic formation outside the controlled volume is not included in the assessments. The conclusions of this study are: all components of the disposal systems should be included in risk assessments unless it can be shown that their effects are negligible, and the attenuation factors are strongly dependent on too many variables to be included in the standards rather that being part of PA.

- Performance Assessments

Figure 4 illustrates the function of performance assessments using multimode release limits. Gaseous releases and some releases from human intrusion would be through the upper surface of the controlled volume as shown in Figure 3. For these pathways, the PA segment of the risk assessment would be unchanged, but the releases would be evaluated against atmospheric and land release limits instead of the present single generic release limits. For nuclide transport through an aquifer, the groundwater that is not withdrawn by wells would eventually reach rivers, lakes, and oceans. Computations of releases to wells, rivers, and oceans could require additional attenuation factor analyses [29] by PA, and some site characterization past the controlled volume might be required. Site characterization and analyses would only have to extend far enough to show compliance. The remainder of the disposal system could be considered an additional, but unquantified, margin of safety. If the standards do not specify average fractions of fresh water usage obtained from ground and surface water, regional values would be defined by PA. PA also would have to adjust the river and well release limits to account for the location of each repository relative to the recharge location and closest river or ocean.

References

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APPENDIX C

JUSTIFICATION FOR DEVELOPING A COLLECTIVE DOSE OPTION

This information is supplied as reference material only.

Excerpt from SAND92-0556 (To be published)

EXPANDED PROPOSED EXTENSIONS OF STANDARDS FOR HIGH-LEVEL AND TRANSURANIC RADIOACTIVE WASTE DISPOSAL

Robert D. Klett and Marilyn M. Gruebel

Task 4
COLLECTIVE DOSE OPTION

Justification for Developing a Collective Dose Option

The first part of this section discusses the uses of dose standards as an alternative to the present derived release limits. The second part describes the procedures for allowing the applicant or the regulating agency the option of using either dose or release limits, whichever is the most appropriate for the repository being regulated.

Dose Criteria

The alternative that has been most widely recommended, but would require the greatest change in regulation philosophy, would eliminate the generic derived release limits and replace them with dose limits. These limits can be placed on individual or population doses. An individual dose standard that limits peak rates to the maximally exposed group at any time has been recommended by the ICRP, NEA, and IAEA [1,2,3] and is being used with some modifications by other countries [4]. Population dose limits could be on peak rates, or total dose for the period of regulation. Since the fundamental criterion in 40 CFR 191 defines the maximum allowable cancer deaths per unit of waste during the time of regulation, collective dose per MTHM is the only type of dose limit that would be compatible.

The argument that risk assessments using dose standards require more predictive assumptions and computations, contain more uncertainties, and are less accurate than those using derived standards is invalid. If the use of approximations and predictions is valid for deriving release limits, they are also valid for dose analyses. The total risk assessment is the same in either case.

Dose standards are more versatile than derived standards because they apply to all repositories and all pathways. Though derived standards, such as generic release limits may poorly represent the actual attenuating process of some repositories, there is a reluctance to modify derived standards because of the complex and time consuming steps that are required. Another advantage of dose standards is they allow the risk assessment to be conducted sequentially from the waste source to humans. This is the only way that all attenuation functions of all components in the disposal system can be included in the assessment. The derivation of release limit standards does not start at the source and therefore cannot include all the retardation and temporal dispersion effects of all preceding components [5].

Dose standards regulate the entire disposal system whereas lower level standards regulate components or groups of components. The requirement to have efficient components is not sufficient; components have to work together to produce an effective disposal system. If dose standards are used, a reference person or a standard biosphere and a standard set of exposure pathways, similar to those used in release limit derivations, could be specified in the regulations. This would assure uniformity for all repositories, reduce the complexity of performance assessments, and maintain more control by the regulating agency. There are many advantages to using dose standards [1,6,7,8] and the change to a higher level standard would not require additional derivations by the EPA. However, it would require a change in regulatory philosophy, would increase the amount of site characterization, and additional PA analyses that would be required.

Release Limits with a Dose Standard Alternative

A recommendation was made during the first EPRI workshop on the technical basis for EPA HLW and TRU waste disposal criteria, September 24-26, 1991 in Arlington, Virginia, to allow the applicant or regulating agency the option of selecting either cumulative population dose standards or cumulative normalized release limits to satisfy the containment requirements of 40 CFR 191.12. The advantage of this option is the conservative but approximate generic release limits could be used for many repositories resulting in less site characterization and less complex performance assessments. For other repositories that are not adequately represented by generic release limits, the more precise dose standards could be used.

There are precedents for alternate methods of evaluation in EPA regulations. 40 CFR 191.17 permits the use of alternate provisions if the existing provisions of Subpart B appear inappropriate [9]. The use of alternative methods of regulation also appears in 40 CFR 264.94(b), which allows the use of alternative concentrations for chemicals [10] and 40 CFR 268.6, which allows the use of an alternative to the treatment of hazardous waste [11].

Three changes to 40 CFR 191 would be required to make this modification work effectively. First, it is essential to explain why the use of alternate criteria is acceptable. Second, a clear statement is needed that defines the optional dose standard method, guidance on when it should be used, and how it would be implemented. Third, the standard should include the fundamental safety criterion that is the basis for the dose limits, the maximum allowed 10,000 year collective dose, and a standard procedure and factors for computing the effective doses that are compatible with the rest of 40 CFR 191 and its supporting documentation. Future states consisting of some combination of a reference biosphere, reference demography, and reference human characteristics could also be defined. Suggestions for these changes are given in the remainder of this section.

Justification for alternative dose standards

The hierarchy of criteria levels is explained in Section 2.2 of this report. The top level fundamental criterion is the only level that explicitly defines the safety requirements of the repository. Some analyses are required to develop each criteria level below the fundamental requirement and each analysis adds uncertainty to the criteria. Derived standards are only used to facilitate regulation and therefore can always be replaced by more precise, higher level criteria without jeopardizing safety. Here, the more expedient, but ultra-conservative derived release limits can be replaced by the higher level, more exact dose limits. The Table 1 release limits were derived from dose limits and this derivation added considerable uncertainty because of the many predictive assumptions, generalizations, and simplifications. The derivation of the present generic single mode release limits contains many conservative assumptions and some important attenuating processes are omitted. The release limits are intended only to provide a simplified method of evaluation, and are not a true measure of risk. Complying with the release requirements is sufficient, but not necessary to prove compliance with the fundamental criterion. Dose limits provide a more accurate measure of actual risk but require more extensive site characterizations and performance assessments. An unsafe repository could not comply with either dose or release limits, so there is no advantage of using both standards.

A preliminary performance evaluation may be needed to select the most appropriate standard for a particular repository. Repository evaluations using release limits are less expensive and can be completed in less time because they require less site characterization and less complex PA. However, the conservative approximate release limits may not adequately represent the attenuating processes of some repositories and the more accurate dose standards may be required.

Description of the dose limit alternative

The information used to develop the dose limit is also used in the development of release limits. There are also many similarities in the implementation of dose and release limits.

The dose limits would be based on the fundamental criterion of 1,000 premature cancer deaths during the 10,000 year regulation period for the reference repositories (100,000 MTHM for HLW and a suggested 20 MCi for TRU waste). The premature cancer deaths in the fundamental criterion are converted to allowable effective doses using a conversion factor supplied by the ICRP [12] to produce the dose limits. The EPA could specify procedures for computing the effective doses for a repository by one of the methods suggested in the next section, or the procedure selection could be left to the implementing agency.

Consequences could be normalized for any event or scenario using dose limits, similar to the way they are normalized using release limits. The normalized dose consequence would be equal to the computed dose divided by the dose limit. Performance assessments with dose limit standards could produce the same type of normalized CCDF that is presently being produced using release limits. Therefore, the consequence CCDF based on dose and release limits could be regulated by the same containment requirements. The probabilities of events or scenarios in the CCDF would be the same with either limit. Only the values of individual normalized consequences (R for summed normalized release and D for normalized dose) would be different as illustrated in Figure 1. The CCDF could be constructed using all normalized releases, all normalized doses, or a combination of the two. The later option would be particularity advantageous for repositories that are expensive to characterize and analyses, and have only a few scenarios or events that cannot be represented properly by generic release limits.

The dose standard alternative could be used with either the present single generic release limits or the multimode release limits. The single generic release limits would be inappropriate for some repositories even if used with the dose standard alternative. It is also extremely conservative for most repositories, possibly making it necessary to use dose standards with added site characterization and analyses, when it would not be necessary with more appropriate release limits. The multimode release limit approach would produce more accurate predictions of risk for all repositories. Since the conservatism would be uniform for all repositories and the risk attenuation of all disposal system components could be included in the performance assessment, fewer repositories would have to use the more expensive and time consuming dose option.

Dose Criteria and Reference Future States

The consequences of radiation exposure that was used to develop the dose limits in Working Draft 4 of 40 CFR 191 [13] are the same as the latest ICRP recommendations [12]. The nominal probability coefficient for stochastic effects used to set the effective dose limits is 0.04 premature cancer deaths per Sv. When this coefficient is applied to the fundamental criterion of 1,000 premature deaths in 10,000 years for the reference HLW repository containing 100,000 MTHM, the effective dose limit is 25,000 person-sieverts per 100,000 MTHM (0.25 person-sieverts/MTHM). For the reference TRU repository containing 20 MCi that is defined in Chapter 7 of this report, the effective dose limit would be 25,000 person-sieverts per 20 MCi of radioactive waste(0.00125 person-sieverts/Ci).

The standards could provide two basic procedures to compute collective effective doses. The procedures for computing the effective dose in Appendix B of Working Draft 4 of 40 CFR 191 [13] are identical to those in Annex A of ICRP 60 [12]. The effective dose (E) is the sum of weighted absorbed

doses from all radiation types and energies, in all tissues and organs of the body. It is given by the expression:

$$E = \sum_R w_R \sum_T w_T \bullet D_{T,R}$$

where $D_{T,R}$ is the mean absorbed dose to organ T delivered by radiation R. The radiation is that incident on the body or emitted by a source within the body. Values for the radiation weighting factors (w_R) are given in Table 1 and values of the tissue weighting factors (w_T) are given in Table 2. This basic procedure is the most versatile, but it allows some variability in its use and would require detailed predictions of pathways and uptake of radionuclides.

The NEA used a modification of the ICRP procedures in the dose analyses for the Subseabed Disposal Program [14]. The average effective dose per unit intake of activity for the ingestion and inhalation pathways was computed for each radionuclide. Similar dose conversion factors were computed for external exposure. Most of the radioactive doses per unit intake for all the major radionuclides were taken from ICRP Publication 30 [15]. The exceptions are the doses per unit intake values for isotopes of plutonium and neptunium; these were calculated using values of the gut transfer fraction appropriate to the forms of these radionuclides found in environmental materials [16]. Tables 3 and 4 list the dose conversion factors for both systems of units. These tables simplify the dose calculations and assure uniform application.

An intermediate approach was taken by the U.S. Department of Energy. They have published dose conversion factors for internal and external exposure for each radionuclide and each exposed organ [17,18]. Reference 17 states that DOE/EH-0071 "is intended to be used as the primary reference by the U.S. Department of Energy (DOE) and its contractors for calculating radiation dose equivalents for members of the public, resulting from ingestion or inhalation of radioactive materials." It also states that "The use of these committed dose equivalent tables should ensure that doses to members of the public from internal exposures are calculated in a consistent manner at all DOE facilities." The series of ICRP publications starting with Publication 26 [19] provides the technical base used in calculating the dose equivalent factors listed in References 17 and 18.

When defining a reference biosphere, demography, or human characteristics, care must be taken not to obscure important site characteristics and to assure compatibility with the standards and their derivations. Except for world population, present processes and parameter values were assumed in the derivation and justification of the standards. Therefore, it

would be appropriate to assume that the reference future states are essentially as they are today. Changes in parameters could be assessed with sensitivity studies and stochastic analyses covering varying climatic, geologic, and hydraulic conditions. The present demographic pattern could be retained by multiplying local populations by the ratio of the 10^{10} world population used in the release limit derivation to the present world population. Human characteristics such as physiology, nutrition, technical and intellectual ability, medical resources, social structure, and values could be as they are now. Although the five basic release modes probably would still exist, they would not all apply uniformly to all repository sites. Geologic and hydraulic risk attenuations are site specific and it would not be appropriate to include them as part of the standards. The more that is included in definition of future states in the standards, the closer the dose limit alternative comes to the multimode release limit approach.

Performance Assessment

Dose based risk assessments, for repositories that do not have their attenuation processes adequately represented by either single mode or multimode release limits, could result in extensive site characterization and analyses. If release limits are inappropriate for only a few events that are responsible for the significant releases, it would be possible to analyze only these events using dose criteria. The predicted doses for each event could be normalized relative to the dose limits set by the EPA in the same manner as predicted releases. The dose fraction could then replace the summed release fraction for that event in the CCDF. The probability would remain the same so the only effect would be to change the consequence level for that event on the CCDF. Depending on the events, this could still be a large site characterization and analysis program, but it would be preferable to conducting dose based assessments for all events.

Summary and Conclusions

The selective substitution of dose limits for events or scenarios that cannot be represented accurately with generic derived release limits is a viable option. Substitution of higher level standards is always justified. This option could require additional site characterization and more analyses for PA. It would be possible to do dose analyses on only selected scenarios, and normalize them to EPA supplied dose limits. They would replace the corresponding normalized releases in the CCDF. All the information needed for dose limits is available so no development program is necessary. This option would require a thorough explanation and justification in the standards.

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Table 1. Radiation Weighting Factors, w_R[1]

Radiation Type and Energy Range[2]		w_R value
Photons, all energies		1
Electrons and muons, all energies		1
Neutrons, energy	<10 keV	5
	10 keV to 100 keV	10
	>100 keV to 2 MeV	20
	>2 MeV to 20 MeV	10
	>20 MeV	5
Protons, other than recoil protons, >2 MeV 5		5
Alpha particles, fission fragments, heavy nuclei		20

1 All values relate to the radiation incident on the body or, for internal sources, emitted from the source.
2 The choice of values for other radiation types and energies not in the table, see paragraph A14 in ICRP Publication 60 (Reference 6-2)

Table 2. Tissue Weighting Factors, w_T[1]

Organ or Tissue	w_T Value
Gonads	0.20
Red bone marrow	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Oesophagus	0.05
Thyroid	0.05
Skin	0.01
Bone surfaces	0.01
Remainder	0.05[2,3]

[1] The values have been developed from a reference population of equal numbers of both sexes and a wide range of ages. In the definition of effective dose, they apply to individuals and populations and to both sexes.

[2] For purposes of calculation, the remainder is comprised of the following additional tissues and organs: adrenals, brain, upper large intestine, small intestine, kidney, muscle, pancreas, spleen, thymus, and uterus. The list includes organs which are likely to be selectively irradiated. Some organs in the list are known to be susceptible to cancer induction. If other tissues and organs subsequently become identified as having a significant risk of induced cancer, they will be included either with a specific w_T or in this additional list constituting the remainder. The latter may also include other tissues or organs selectively irradiated.

[3] In those exceptional cases in which a single one of the remainder tissues or organs receives an equivalent dose in excess of the highest dose in any of the twelve organs for which a weighting factor is specified, a weighting factor of 0.025 should be applied to that tissue or organ and a weighting factor of 0.0225 to the average dose in the rest of the remainder as defined above.

Table 3. Dose Equivalent Factors for Humans (Curies and Related Units)

Nuclide	Ingestion (Rem/Ci)	Inhalation (Rem/Ci)	Immersion (Rem/Hr-Ci-M**3)	Exposure to Soil (REM/Hr-Ci-**3)
C-14	2.07E+03	2.07E+03	0.00E+00	0.00E+00
Ni-59	2.00E+02	1.33E+03	2.30E-03	0.00E+00
Sr-90	1.44E+05	1.26E+06	5.40E-04	0.00E+00
Zr-93	1.55E+03	3.18E+05	0.00E+00	0.00E+00
Tc-99	1.26E+03	7.40E+03	1.30E-04	0.00E+00
Sn-126	1.89E+04	7.40E+04	1.80E-02	9.00E+00
I-129	2.74E+05	1.74E+05	1.70E-02	4.50E-01
Cs-135	7.03E+03	4.44E+04	6.60E-05	0.00E+00
Cs-137	5.18E+04	3.22E+04	1.00E+00	4.20E+00
Sm-151	3.37E+02	2.81E+04	2.60E-04	4.80E-02
Pb-210	5.18E+06	1.30E+07	3.00E-03	1.30E-02
Ra-226	1.15E+06	7.77E+06	1.80E+00	6.40E+00
Ra-228	1.22E+06	4.44E+06	6.75E+00	2.60E+01
Ac-227	1.41E+07	6.66E+09	1.69E+00	8.21E+00
Th-229	3.70E+06	2.11E+07	5.80E-01	2.20E+00
Th-230	5.55E+05	3.18E+08	1.80E+00	6.50E+00
Th-232	2.74E+06	1.63E+09	4.00E+00	1.56E+01
Pa-231	1.07E+07	1.26E+09	5.00E-01	2.20E+00
U-233	2.66E+05	1.33E+08	5.90E-01	2.30E+00
U-234	2.63E+05	1.33E+08	1.18E-03	7.32E-03
U-235	2.52E+05	1.22E+08	2.96E-01	1.31E+00
U-236	2.48E+05	1.26E+08	2.97E-06	2.06E-04
U-238	2.33E	1.18E+08	7.30E-02	3.52E-01
Np-237	4.07E+06	4.81E+08	3.60E-01	1.40E+00
Pu-238	1.85E+06	4.44E+08	1.50E-04	1.30E-03
Pu-239	2.22E+06	5.18E+08	1.20E-04	7.90E-04
Pu-240	2.22E+06	5.18E+08	1.40E-04	1.30E-03
Pu-241	4.44E+04	1.04E+07	6.10E-05	4.60E-03
Pu-242	2.04E+06	4.81E+08	1.10E-04	1.10E-03
Am-241	2.22E+06	5.18E+08	3.90E-02	1.80E-01
Am-243	2.18E+06	5.18E+08	3.10E-01	1.30E+00
Cm-245	6.66E+04	1.74E+07	3.40E-04	5.50E-03
Cm-246	1.11E+06	2.74E+08	2.60E-04	2.90E-03

Table 4. Dose Equivalent Factors for Humans (TBq and Related Units)

Nuclide	Ingestion (Sv/TBq)	Inhalation (Sv/TBq)	Immersion (Sv/Hr-TBq-M**3)	Exposure to Soil (Sv/Hr-TBq-M**3)
C-14		5.60E+02	0.00E+00	0.00E+00
Ni-59	5.40E+01	3.60E+02	6.21E-04	0.00E+00
Sr-90	3.90E+04	3.40E+05	1.46E-04	0.00E+00
Zr-93	4.20E+02	8.60E+04	0.00E+00	0.00E+00
Tc-99	3.40E+02	2.00E+03	3.51E-05	0.00E+00
Sn-126	5.10E+03	2.00E+04	4.86E-03	2.43E+00
I-129	7.40E+04	4.70E+04	4.59E-03	1.22E-01
Cs-135	1.90E+03	1.20E+04	1.78E-05	0.00E+00
Cs-137	1.40E+04	8.70E+03	2.70E-01	1.13E+00
Sm-151	9.10E+01	7.60E+03	7.02E-05	1.30E-02
Pb-210	1.40E+06	3.50E+06	8.10E-04	3.51E-03
Ra-226	3.10E+05	2.10E+06	4.86E-01	1.73E+00
Ra-228	3.30E+05	1.20E+06	1.82E+00	7.02E+00
Ac-227	3.80E+06	1.80E+09	4.56E-01	2.22E+00
Th-229	1.00E+06	5.70E+06	1.57E-01	5.94E-01
Th-230	1.50E+05	8.60E+07	4.86E-01	1.76E+00
Th-232	7.40E+05	4.40E+08	1.08+00	4.21E+00
Pa-231	2.90E+06	3.40E+08	1.35E-01	5.94E-01
U-233	7.20E+04	3.60E+07	1.59E-01	6.21E-01
U-234	7.10E+04	3.60E+07	3.19E-04	1.98E-03
U-235	6.80E+04	3.30E+07	7.99E-02	3.54E-01
U-236	6.70E+04	3.40E+07	8.02E-07	5.56E-05
U-238	6.30E+04	3.20E+07	1.99E-02	9.50E-02
Np-237	1.10E+06	1.30E+08	9.72E-02	3.78E-01
Pu-238	5.00E+05	1.20E+08	4.05E-05	3.51E-04
Pu-239	6.00E+05	1.40E+08	3.24E-05	2.13E-04
Pu-240	6.00E+05	1.40E+08	3.78E-05	3.51E-04
Pu-241	1.20E+04	2.80E+06	1.65E-05	1.24E-03
Pu-242	5.50E+05	1.30E+08	2.97E-05	2.97E-04
Am-241	6.00E+05	1.40E+08	1.05E-02	4.86E-02
Am-243	5.90E+05	1.40E+08	8.37E-02	3.51E-01
Cm-242	1.80E+04	4.70E+06	9.18E-05	1.49E-03
Cm-244	3.00E+05	7.40E+07	7.02E-05	7.83E-04

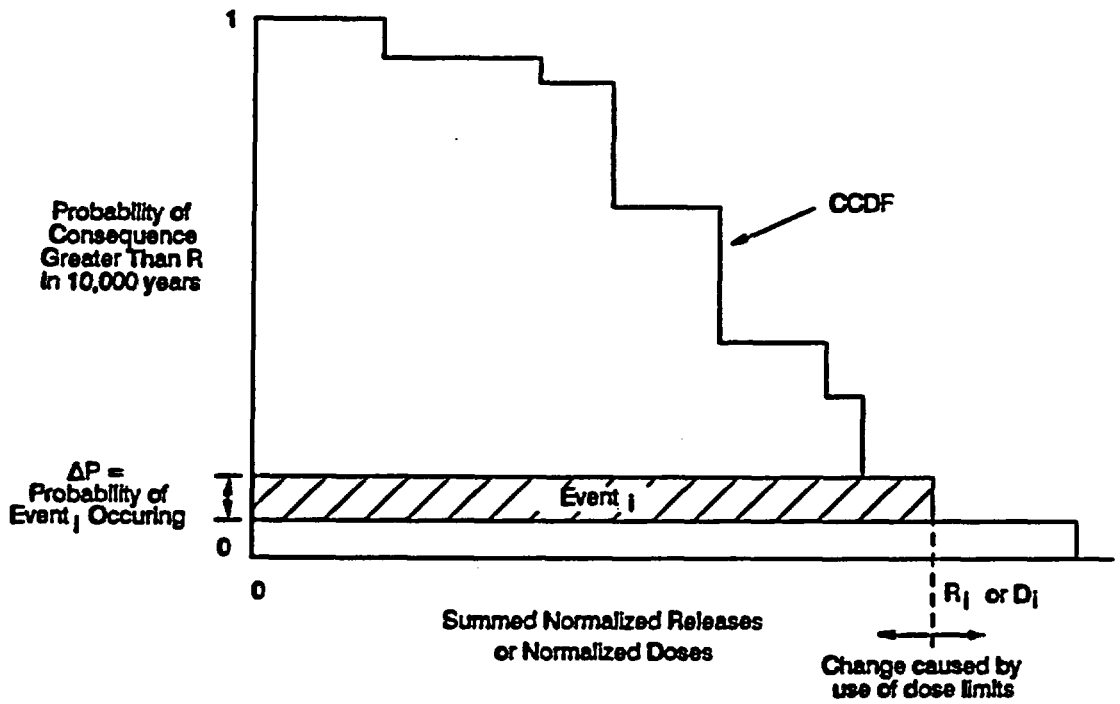

Figure 1 - CCDF made up of normalized doses or normalized releases.

APPENDIX D

JUSTIFICATION FOR DEVELOPING ALTERNATIVE TRU CRITERIA

This information is supplied as reference material only.

Excerpts from SAND92-0556 (To be published)
EXPANDED PROPOSED EXTENSIONS OF STANDARDS FOR HIGH-LEVEL AND TRANSURANIC RADIOACTIVE WASTE DISPOSAL
Robert D. Klett and Marilyn M. Gruebel

Task 5-6
CRITERIA FOR TRANSURANIC WASTE DISPOSAL

Justification for Developing Alternate TRU Criteria

Since 40 CFR 191 was not developed as an integrated system, it contains many provisions and constraints that prevent effective modifications and additions such as criteria for TRU waste disposal. Although far from ideal, the following are the only feasible options that have been suggested for regulating TRU waste disposal. The options are: two types of fundamental criterion developed specifically for TRU waste [1,2], and a family of procedures that use a reference TRU waste unit with commercial HLW criteria [1,2,3]. This chapter compares these options. Background information pertinent to both options is covered in the following sections on the functions of fundamental criteria, and a description of the HLW fundamental criterion and release limits presently in 40 CFR 191. The section on criteria specifically for TRU waste suggests a methodology for developing or adapting fundamental and derived criteria that are consistent with all other aspects of the standards. The section on reference TRU waste units covers all the parameter variations that have been suggested for this option. The technical bases of each approach are reviewed, implementation is discussed, and their relative attributes and deficiencies are evaluated.

TRU repositories will contain some radioactive wastes that are not officially classified as transuranic waste. Actinides and daughters of short lived transuranics can be a significant part of the risk potential. Since all radionuclides constitute a potential risk, the standards for TRU repositories should be based on and apply to all radionuclides in the inventories.

1 Fundamental Criteria

Fundamental criteria (Level 1) are the only standards that explicitly define the radiological safety requirements of the repositories. Level 1 criteria control risks to the populace, have a significant effect on the cost of repositories, and are the basis for other levels of radiological criteria. To have any radiological risk significance, all other levels of criteria must be traceable to an appropriate fundamental criterion [1]. The position of the NEA is, "The general risk limit should be considered as the lower boundary of a region of unacceptable risks rather than as the upper demarcation of a region of unchallenged acceptability. Therefore, the level at which these objectives are set should be based as far as possible on a scientific assessment of risk in relation to well established radiation protection standards. Where exposures could arise from various sources,

there will be a need to take this into account by an apportionment of the general limit" [4].

Appropriate fundamental criteria are needed for all repositories and for each waste category. These criteria should be based on established principles and set at the lower boundary of unacceptable risk. The regulatory philosophy for any fundamental criteria that is added to 40 CFR 191 should be consistent with that of the HLW fundamental criterion and it should be compatible with existing release limits.

1.1 HLW Fundamental and Derived Criteria in 40 CFR 191

The present fundamental criterion for HLW and SF allows no more than 1,000 premature cancer deaths over the first 10,000 years from disposal of the wastes from 100,000 metric tons of reactor fuel (average of 10^{-6} HE/MTHM-yr). This is a risk/benefit criterion that allows the risk from waste disposal to be proportional to the amount of power generated. Power is equated to the amount of fuel used to generate the power (MTHM) for convenience of analyses. It is also based on collective world population risk over the 10,000-year period of regulation with no constraints on population risk rates. Derived standards for HLW must follow this format. The HLW release limits were derived by computing risk factors (fatal cancers per curie released) for each radionuclide for several release modes [5]. The fundamental criterion was divided by each of these risk factors to produce a table containing release limits for each radionuclide [6], which is compatible with the risk/benefit, collective population risk fundamental criterion.

The allowable risk level for HLW disposal was based on predicted capabilities of the reference HLW repository in several geologic media instead of the lower bound of unacceptable risk. This accounts for the high level of stringency compared with standards for other carcinogens. The 100,000 MTHM size of the reference repository was selected because it was the estimated cumulative inventory by the year 2000 [7]. Reference 3 states this is the quantity of existing US HLW waste plus the future wastes from all currently operating US reactors. There has never been a clear and consistent statement of the basis or rationale for the HLW fundamental criterion, nor has it been shown that it assures an acceptable level of risk to the populace. TRU waste was not considered in the development of this fundamental standard and therefore it does not apply to TRU waste disposal.

1.2 Fundamental and Derived Standards for TRU Waste Disposal

The present version of 40 CFR 191 contains no fundamental criteria for TRU waste disposal, and no safety requirements have been established that apply to TRU waste. This may be the only major waste disposal process without a fundamental safety requirement. Military TRU waste is not associated with commercial reactor fuel, does not have a unit comparable to a MTHM of fuel, and does not have a comparable risk/benefit relationship. These TRU waste characteristics preclude direct application of the HLW

fundamental criterion, but a fundamental criterion can be developed specifically for TRU waste disposal.

One approach would be to develop a fundamental criterion for TRU waste based on acceptable risk to the populace and the expected quantity of TRU waste. This is the general approach recommended by NEA and ICRP [4,8]. Assuming collective population risks will continue to be used as the basis for the fundamental criteria and derived release limits will be used to show compliance in 40 CFR 191, neither the recommended ICRP standards nor the EPA standards for chemical carcinogens could be used for TRU waste. The ICRP fundamental standards are based on a peak individual risk rate, which is not compatible with collective risks or release limits. The standards for chemical carcinogens are based on individual risks as a function of the number of people at risk. This method is also incompatible. Since there is no quantifiable benefit associated with military TRU waste, the EPA would have to develop a new absolute collective risk limit. This TRU fundamental criterion would be completely independent of the HLW fundamental criterion and based solely on expected quantities of TRU waste and acceptable levels of risk. One difficulty with an absolute TRU criterion is the uncertainty in predicting the total quantity of TRU waste that will be generated, which is needed to allocate a risk for each repository. New release limits would also have to be developed based on the absolute TRU fundamental criterion. Developing the new absolute collective risk limit, agreeing on the total future TRU inventory, and developing new release limits could be a very time-consuming process. There also would some inconsistencies in regulatory philosophy between the risk/benefit HLW criteria and an absolute TRU criterion and these differences would have to be justified. It is probably not practical to develop this form of fundamental criteria for TRU waste disposal at this time.

Another approach would be to develop a TRU fundamental criterion that is related to the allowable risk for HLW repositories. There is a straightforward and simple method of developing a TRU fundamental criterion using rationale and analyses that are parallel to that used by the EPA to develop the HLW standards [5,6]. No new release limits would be needed and it would be compatible with the HLW criteria and all other requirements in 40 CFR 191. Although this TRU fundamental would not be a true risk/benefit criterion, the allowable risk would be scaled relative to repository size, making the TRU waste allowable risk units comparable to those of HLW. This would allow either single mode or multimode HLW release limits to be used for TRU waste with no changes or additions.

One justification used by the EPA for the level of the HLW fundamental risk criterion was it assures adequate protection for the EPA's reference HLW repository (10^5 MTHM). The reference repository was used in derivations [6] and in comparison studies of waste disposal systems and undisturbed ore bodies [3,9]. It should then be acceptable to establish a reference TRU repository and equate the risk to that of the reference HLW repository. With this risk level as the basis, the only task remaining would be to define the size of the reference TRU repository.

The EPA based the reference HLW repository size on the expected inventory in the year 2000 including all existing HLW and projected waste from then operating reactors. A consistent TRU reference repository size could be defined using the same guidelines. The Integrated Data Base for 1991 [10] lists the total known remote-handled (RH) and contact-handled (CH) TRU waste in the year 2000 as 9.8 MCi. This value is 14.3 MCi in 2013, which is the last year listed. Following the rationale used to select the HLW reference repository size, a conservative size for the TRU reference repository would be 20 MCi including RH and CH waste. Given the conservatism built into the HLW criteria; this size would give the TRU reference repository a very conservative allowable risk. The allowable risk for either the HLW or TRU reference repositories would then be 1,000 premature cancer deaths over the first 10,000 years for an average of 1 premature death every 10 years. The allowable risk for smaller TRU repositories, such as the WIPP, would be scaled down proportional to their size relative to the reference repository.

The size of the proposed TRU reference repository is based on current inventory predictions. If larger quantities of TRU waste are generated because of changes in waste management strategy such as decommissioning an decontamination of DOE facilities, the size of the TRU reference repository could be increased. However, there is no parallel provision in the HLW criterion that would increase the HLW reference repository size if new reactors are built or new sources of HLW arise.

If this approach is adopted, no new release limits would have to be derived. The risk factors used to derive the release limits were computed for individual radionuclides and apply to any inventory or waste category. Presently the fundamental HLW criterion and dose limits in Working Draft 4 of 40 CFR 191 [11] are based on 100,000 MTHM, but the release limits are based on 1,000 MTHM. The standards would be more consistent and less scaling would be required if the 100,000 MTHM for HLW (20 MCi for TRU waste) base is used throughout the standards. Scaling the release limits to different size TRU repositories could be the same as the method defined in the present version of 40 CFR 191. Release limits for both HLW and TRU repositories would be the values in a 100,000 MTHM (20 MCi) based release limit table, multiplied by the ratio of repository size to the reference repository size. For example, for a TRU repository with an inventory of 5 MCi and a TRU reference repository of 20 MCi, the release limits applicable to the repository would be 5/20, or 0.25 the values in the release limit table. The purpose of this scaling is to prevent compliance by using the strategy of making repositories small instead of well designed.

The characteristics of this approach to TRU waste disposal regulations are:

1. It is based on repository safety and applies equally to all release modes, all repositories, all inventories, and at all times.

2. It uses the same format and regulatory philosophy as the HLW standards so additional justification is not needed.
3. It is completely compatible with other aspects of the standards.
4. No new derivations are required.
5. There is no need for a quasi-equivalent TRU waste unit.
6. It is as conservative and defensible as the HLW standard.
7. Repository risks can be computed because the release limits are traceable to a fundamental criterion.

The parallelism of TRU and HLW criteria with this approach is shown in Table 1.

Table 1 - Features of HLW and TRU criteria when parallel development of the fundamental criterion is used

Feature	Waste Type	
	HLW/SF	TRU Waste
Maximum deaths from the reference repository in 10,000 years	1000	1000
Basis for reference repository size	Cumulative inventory by year 2000 [7] Waste from currently operating reactors [3] 100,000 MTHM	Cumulative inventory by year 2013 from existing facilities rounded up to 20 MCi
Fundamental Criterion	Deaths per 10,000 years / Reference repository size	Deaths per 10,000 years / Reference repository size
Release limit values	40 CFR 191, Table 1	40 CFR 191, Table 1
Scaling factor for release limits	Actual repository size / Reference repository size	Actual repository size / Reference repository size

This is definitely not an endorsement of the present HLW fundamental criterion. If the HLW criterion is improved, a combined HLW/TRU criterion should be considered, or the same new procedure should be used to develop both the HLW and TRU fundamental criteria.

2 Reference TRU Waste Units

The present version of 40 CFR 191 contains no fundamental criterion or safety requirements that apply to TRU waste. Instead, TRU waste repositories are evaluated using HLW/SF criteria and a TRU waste unit (1 MCi) that is "equivalent" to 1000 MTHM of commercial HLW.

There is a whole family of quasi-equivalent TRU waste units that have been suggested for use with the regulations developed for HLW disposal. These include the one that is presently in 40 CFR 191. Combinations of four or five groups of parameters could be used to compute the quasi-equivalent TRU waste unit, with varying degrees of realism. These parameter groups are:

1. Reference inventory - High-level waste, spent fuel, or some average.
2. Included nuclides - All radionuclides in the inventories or only actinides with half-lives greater than 20 years.
3. Time - Initial, averaged over the time of regulation, or end of the regulatory period.
4. Metric - Risk potential, untreated dilution index, or activity.

If the risk potential metric is selected, the release mode parameters - rivers, oceans, withdrawal wells, land, and atmosphere - also would be included. There is also variability in the reference HLW/SF inventories and the TRU inventories that are equated. Sample analyses of 12 of the possible 126 combinations have shown that these reference TRU units could vary by at least a factor of 525 depending on the parameters selected. This large spread makes the selection of parameters difficult to justify. None of these parameter combinations produce a true equivalent unit and subjective judgment must be used in the selection of the best combination. Any of the reference TRU waste units can be equivalent to only one HLW or SF inventory, at a single time, for one repository, and for only one parameter that is not proportional to actual risk. A clear statement of the basis for equating waste units, including substantiated reasons for the selection, would be required.

The parameters selected from each parameter group affect the value of the reference waste unit. In the following discussion of two of the 126 combinations, the effects of each parameter will be discussed separately but the effects of all parameters must all be viewed together to see the net result.

One of the above combinations was used to compute the reference TRU unit presently in the standards. Initial activities of the actinides with half-lives greater than 20 years in a reference TRU waste inventory were equated to those in a reference 1,000 MTHM HLW inventory. This quasi-equivalent TRU waste unit was computed to be 3 MCi, which was rounded to 1 MCi in the standards.

Appendix A of 40 CFR 191 limits the summed normalized release fractions of both transuranics <u>and</u> fission products. Risk and performance assessments [12,13] have shown that releases and doses from undisturbed HLW repositories during the first 10,000 years would be completely dominated by the more mobile fission products (Tc-99, I-129, SE-79, C-14, etc.), with almost no contribution from transuranics. Since the radionuclides in the summed normalized release from HLW repositories are almost entirely fission products and those from TRU repositories are almost entirely transuranics, this aspect of the present regulations

requires a higher degree of control and higher retention fraction of transuranics in a HLW repository than in a TRU repository.

Another inconsistency arises from the selection of initial activities as the parameter that was equated for the two types of waste. Most of the releases and risks come near the end of the regulation period, not the beginning. Using the initial value of any parameter for equating risk potentials would not typify actual releases to the environment. Also the risk from each radionuclide depends on the dose equivalent weighting factor, pathways to humans, and risk attenuation of the entire disposal system [14], not just the activity of the nuclide.

Reference 1 suggests a combination of parameters that partially rectify some problems with the present reference TRU waste unit. An approximation to the risk potential was chosen as the metric for equating the entire inventory of a reference TRU inventory to an average HLW/spent fuel reference inventory. The time varying risk potential was approximated by multiplying the activity of each radionuclide by the risk factors (cancers per curie) for surface water release given in Table 7.8-1 of Reference 6. This accounts for nuclide transport pathways from a generic river to humans and resulting biological effects, but does not include the risk attenuation between the repository and the river, which is assumed to be the release location. This risk potential is both unsuitable and incomplete for computing an equivalent waste unit. Actual release modes of planned repositories include atmospheric, land, and wells - not surface water [15]. It is incomplete because risk attenuation between the repositories and release locations is not included and is different for each repository, each release scenario, and each radionuclide, so actual risk potentials would not be comparable to the risk potential used in this analysis, and risk potential ratios between repositories would be far from uniform. The inappropriateness could be eliminated by using the five risk potentials from the multimode release limit derivation. However, this would result in five different equivalent TRU waste units, one that is appropriate for each release mode.

These approximate risk potentials for both the HLW and TRU inventories were then integrated over the time of regulation. Actual risks could occur any time, but a time averaged value is a better representation than the initial value. The resulting average risk potentials were equated to define a reference TRU waste unit. This quasi-equivalent TRU waste unit was computed to be 8.1 MCi. The same analysis was later repeated by other investigators using different pairs of HLW and TRU inventories [16,17], which made their results slightly different as expected.

Although this time averaged parameter combination appears to be more logical and appropriate than the method used to derive the present TRU standards, it does not produce a TRU waste unit with the same risk as the HLW unit. The ability to equate risks could be worse than the present TRU "equivalent" waste unit for some repositories and scenarios. The problem is not in the combination of parameters selected or the method of analysis. The entire concept of trying to equate risks by matching repository component parameters, using specific inventories, at specific times is

unsound. There are no generic equivalencies between any waste categories that apply uniformly to all repository designs and locations, to all inventories, and at all times. The variability and inconsistency of this approach can be illustrated with the following parametric example. All the cases used risk potential as the metric. Only two release modes (river and land) were analyzed. The inventory pairs were spent fuel and reprocessed HLW, used with a single TRU inventory. All radionuclides in the inventories were included in the analyses. The TRU and HLW risk potentials were equated at times 0 and 10,000 years and averages over the 10,000-year duration. The results are shown in Table 2. The variability is demonstrated by the factor of 485 separating the highest and lowest "equivalent" TRU waste units. The variability with time of evaluation can be a factor of 261, with waste form a factor of 43, and with release mode a factor of 17. There are also interactions between parameters. Time factors vary from 1.96 to 261 depending on the waste form and release mode selected. Similarly, waste form factors vary from 1.05 to 43 and release mode factors vary from 1.07 to 17.4.

Table 2 - Examples of TRU Waste Units (MCi) that are "equivalent" to 1,000 MTHM HLW based on risk potential metrics.

Reference HLW Waste	Release Mode	Initial Time	10,000 Yr Average	10,000 Yrs
Spent Fuel	River	63.	12.	6.0
Spent Fuel	Land	11.	6.9	5.6
Reprocessed HLW	River	60.	4.0	0.23
Reprocessed HLW	Land	5.3	0.23	0.13

All quasi-equivalent TRU waste units would make the acceptable risk proportional to the amount of waste placed in a repository, similar to the HLW criteria. However, none of these reference TRU waste units equate repository risks and there would be no rationale for using the HLW/SF criterion. There would still be no fundamental safety criterion for TRU waste.

3 Evaluation of the Options

The three basic options for regulation of TRU waste disposal are so different that there can only be a limited characteristic-by-characteristic comparison. These are compared in Table 3. The remaining characteristics are summarized separately.

Table 3 - Characteristics of TRU criteria options

Characteristic	Option: Absolute TRU Fundamental Criterion	Option: Fundamental TRU Criterion Parallel to HLW Criterion	Option: Reference TRU Waste Unit
Based on acceptable risk	Yes	Yes	No
Applies uniformly to all repositories and scenarios	Yes	Yes	No
Defensible	Yes	Yes	No
Scales to repository size	Yes	Yes	Yes
New derivations required	Major	None	None or minor
Follows 40 CFR 191 format	No	Yes	Yes
Uses HLW release limits	No	Yes	Yes

TRU fundamental criteria could be related to the allowable risk from a HLW repository or could be completely independent of the HLW fundamental criterion. Either method of developing separate fundamental standards for TRU waste would base them on repository safety and acceptable risk. Both methods would apply to all release modes and all repositories, would scale with repository size, and would apply any time during the regulatory period. This allows easy computations of repository risk from release analyses. The method that equates the allowable risk from the reference TRU repository to the risk from the reference HLW repository would require no new derivations of risk criteria or release limits and is more compatible with HLW criteria and other requirements in the standards. It would be as conservative and defensible as the HLW standards. The method that develops an independent absolute risk limit for TRU waste disposal would require the derivation of a new fundamental criterion and different derived release limits. It would be at least as defensible as the HLW standards.

A TRU repository risk limit is not used by the family of 126 reference TRU waste units. Instead, several combinations of parameters are used to equate MCi units of TRU waste to MTHM units of HLW. HLW criteria are then used to evaluate TRU repositories. All the quasi-equivalent TRU waste units scale with repository size and are compatible with HLW criteria and other requirements in the standards. However, collective risk is

not evaluated or equated to HLW risk. Other parameters that do not scale linearly to risk are equated at a single time during the regulation period or the average is equated over a specified time interval. The reference units can differ by factors of 525 depending on the combination of parameters selected. It would be difficult to justify the selection of any parameter combination, and to rationalize the use of any reference waste unit instead of fundamental safety criteria. Protection provided by reference TRU waste units is far from uniform. It is different for each repository, scenario, pathway, release mode, and inventory. The present standards give no rationale for using this method of regulating TRU waste disposal or for equating the initial activity of only some radionuclides. If this method of regulation is retained, a detailed explanation of how it assures repository safety is needed. Development time would be insignificant even if a different combination of parameters is selected.

In addition to the technical arguments concerning uniformity, appropriateness, and defensibility, it is also useful to put the risks allowed by each regulatory option in perspective. References 18 and 19 reviewed and analyzed U.S. regulations governing exposure to environmental carcinogens, which were promulgated by several regulating agencies. Both found a high degree of consistency in the agencies' implicit definition of de minimis levels of lifetime individual risk as a function of the population size at risk. Using the results of their studies, the total de minimis risks from a carcinogen over a 10,000 year period were computed for three population sizes significant to the WIPP. If the carcinogen placed the entire U.S. population at risk, the de minimis number of premature cancer deaths would be 26,000 for 10,000 years. If only the population of New Mexico would be at risk, the de minimis level would be 2,570 premature deaths. If only the residents of Eddy county, where the WIPP is located and where any human intrusion and well water withdrawal would take place, would be at risk, the de minimis level would be 514 premature deaths.

The projected total inventory for the Waste Isolation Pilot Plant (WIPP) used in the 1991 WIPP performance assessment, including all radionuclides of both CH and RH waste, was 11.1 MCi [20]. More recent inventories given in the draft report "The Radionuclide Inventory for the Waste Isolation Pilot Plant," DOE/WIPP 91-058 by H. M. Batchelder define the total WIPP inventory to be 7.7 MCi. It is unlikely that any future TRU repository would be more than twice the size of the WIPP because of geologic limitations. Table 4 lists the allowable premature cancer deaths for HLW repositories and for TRU repositories using the most recent inventories, with requirements based on a fundamental criterion and on reference waste units. It compares the WIPP requirements to the de minimis risk level if all the risk would be confined to Eddy county. The 8.1 MCi equivalent TRU waste unit would allow the WIPP to have only one percent of the risk allowed for the YMP, and the risk level would be a factor of 50 lower than de minimis. The TRU fundamental criterion also would be conservative, but the level of protection would be more realistic.

Table 4 - Relative stringency of several radioactive waste disposal criteria alternatives.

Source of risk	Standard	Reference for standard	Allowable premature deaths per 10,000 yrs	Ratio of allowable risk to Eddy Co. de minimis
HLW reference repository (100,000 MTHM)	HLW fundamental criterion	3	1000	NA
YMP (70,000 MTHM)	HLW fundamental criterion	3	700	NA
TRU reference repository (20 MCi)	TRU fundamental criterion	This report	1000	NA
WIPP (7.7 MCi)	TRU fundamental criterion	This report	385	0.75
WIPP (7.7 MCi)	1 MCi equivalent waste unit	3	77	0.15
WIPP (7.7 MCi)	8.1 MCi equivalent waste unit	1	9.5	0.02

Table 4 shows the allowable risks that are computed using methods prescribed in the standards. Actual risks would be orders of magnitude lower because of the present conservatism in the release limits used for human intrusion, the absence of aquifer risk attenuation for the well, river, and ocean release modes, and the conservatism in the stepped containment requirements. Also, actual releases from repositories would be far below the limits for most scenarios.

In establishing the reference TRU repository, the size could be increased to 60 MCi based on projected inventories of DOE facilities that might be decommissioned. This would be equivalent to 8 WIPP repositories. The factor of three increase in the base for the TRU fundamental criterion would decrease the allowable risk for any given TRU repository by a factor of three. The HLW criteria does not have this flexibility to account for changes in expected inventories, so the option to change to a 60 MCi TRU reference repository could create an inconsistency in the standards.

Figure 1 shows another way to put the alternate TRU criteria in perspective. The bar graph shows the amount of TRU waste that would be required to produce the same risk as the 100,000 MTHM

reference HLW repository with each of the proposed TRU criteria. These values are compared to the amount of TRU waste that is predicted for the year 2000. This is the same year that was used to define the inventory for the HLW reference repository. All proposed TRU criteria are higher (more conservative) than the year 2000 inventory. However, the 1 MCi equivalent waste unit is factor of 10.2 higher than the year 2000 TRU inventory and the 8.1 MCi equivalent waste unit is a factor of 82.7 higher making these criteria inconsistent with projected TRU inventories and risk levels set for HLW. Also shown in Figure 1 is the inventory for eight repositories the size of WIPP, which is one of the higher estimates of TRU waste from decommissioning and decontamination of DOE facilities. Both values for equivalent waste unit criteria are even greater than this inventory, while the 60 MCi based fundamental TRU criterion has the same value.

The above discussions and the Table 3 summary show that all the reference TRU waste units are nonuniform, inappropriate, and indefensible, and Table 4 and Figure 1 show that their risk limits are unrealistic. The fundamental TRU standard is more defensible, is consistent with the development of the HLW fundamental criterion, and the levels of protection are more realistic. However, as stated earlier, it is not possible to develop a TRU standard that is completely defensible because of the unintegrated and unique structure of 40 CFR 191.

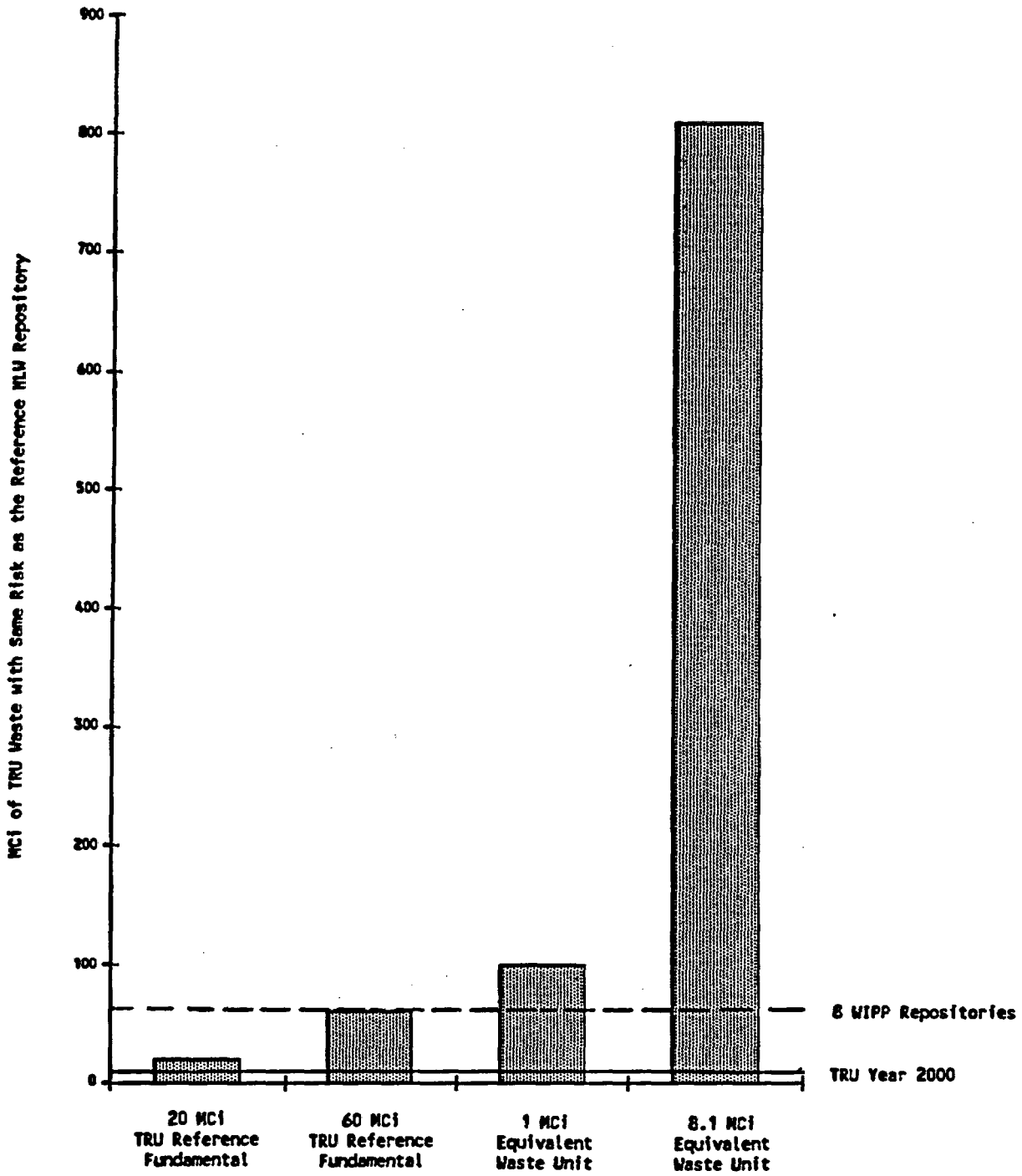

Figure 1 - Comparison of TRU criteria with the reference HLW risk and two quantities of TRU waste.

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APPENDIX E

REGULATORY OVERVIEW AND RECOMMENDATIONS ON A REPOSITORY'S RELEASE OF CARBON-14

This information is supplied as reference material only.

REGULATORY OVERVIEW AND RECOMMENDATIONS ON A REPOSITORY'S RELEASE OF CARBON-14

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SUMMARY

The release of gaseous carbon-14 (C-14) dioxide from a potential Yucca Mountain repository to the accessible environment, with the current design of waste packages, could exceed the release limits set by the U.S. Environmental Protection Agency (EPA) and U.S. Nuclear Regulatory Commission (NRC). The amount released depends on the sources of C-14, mechanisms to free C-14 from the sources, and transport mechanisms to the accessible environment, each of which is in turn affected by many parameters in the natural geologic environment. This paper examines the current information on the amount, the sources, and the transport of carbon-14. From this information, the paper assembles a coherent conceptual model for C-14 release and transport. It is shown that the uncertainties in our knowledge and data are so large that we must conclude there is a significantly high probability of exceeding both the NRC and EPA release limits, and consequently violating both NRC and EPA regulations. The uncertainties are in both the source term (engineered) and transport (natural), of which the former may be more dominant. The source term, however, is also so strongly influenced by the natural system, primarily the hydrology of the site, that even after site characterization the residual uncertainties may still be unacceptably high. This may force the U.S. Department of Energy (DOE) to look for an expensive solution to the source term (costing billions of dollars and years of delay).

Analyses done by the DOE contractors and others have been reviewed, including the regulatory implications of the preliminary results. It has already been demonstrated that the additional expenditures that would be required to contain C-14 would not measurably benefit the public health and safety. Several regulatory alternatives have been discussed. The gaseous release of radionuclides could be regulated by the Clean Air Act (CAA) requirements, either through EPA's National Emission Standards for Hazardous Air Pollutants (NESHAP) or by a rulemaking in consultation with the NRC. It is recommended that the currently existing NESHAP Subpart I be used, which exempts the facilities regulated by 40 CFR Part 191.

In terms of the gaseous emission standard, there are several options available whose pros and cons are discussed in detail. Among them, the following option seems to be most reasonable in terms of providing a technical basis for the numerical criteria and regulatory consistency with the CAA requirements.

> "The gaseous release of radionuclides shall not exceed the amounts that would cause any member of the public to receive an effective dose equivalent of 5 mrem/yr, except that any combined releases that would cause an effective dose equivalent of 0.1 mrem/yr or less need not be regulated."

Although the implementation was considered in recommending the alternatives, other political considerations may have to be factored into the final formulation of the emission standard applicable to the gaseous releases. There is no one solution that will solve all the problems and satisfy all the parties involved. In addition, the problem is a global one and may require a global solution.

I. INTRODUCTION

The release of carbon-14 (C-14) from potential high-level nuclear waste repositories in the U.S. is regulated by the NRC's 10 CFR Part 60. This regulation implements environmental standards specified in the EPA's 40 CFR Part 191. When these regulations were promulgated, major candidate sites for repositories were in saturated zones in different geologic formations. Although an unsaturated zone in tuff was also considered before 40 CFR Part 191 was finalized in 1985, no specific consideration for the release of gaseous radionuclides was made. The only gaseous radionuclide that could be released in any significant amount from a potential repository in the unsaturated zone at Yucca Mountain, Nevada, is C-14 in the form of carbon dioxide (Ref. 1).

Recent performance assessment studies conducted by the DOE (Ref. 2) and the Electric Power Research Institute (EPRI) (Ref. 3) show that Yucca Mountain's compliance or non-compliance with the regulations is largely dominated by the uncertainties associated with the release of C-14. Among the radionuclides regulated by the EPA and NRC, C-14 is the <u>only</u> radionuclide that is a part of our essential environment, is in our daily diet, is present everywhere on earth and in the atmosphere (even in the human body), is abundant in nature (global inventory of 230 million curies: 7.5 million curies in land biosphere and humus, and 3.8 million curies in the atmosphere) (Ref. 4, 5), and gives a very small exposure to any individual from a very large inventory. The expected release rate from a potential repository at Yucca Mountain (less than a few curies per year) is so small that it would hardly affect the radiation dose that any individual on Earth would receive naturally during his or her lifetime. Yet this release could violate the EPA and NRC regulations unless very costly design alternatives are adopted or a significant amount of additional site characterization work is done with great cost and significant project delays. A more robust design of the waste package will undoubtedly enhance the confidence that the regulations are met for other, more soluble radionuclides. However, the requirements on C-14 are more severe than on other radionuclides, as evidenced in the DOE's Performance Assessment Calculation Exercise (PACE). The inappropriateness of regulating such a low release as that expected from a geologic repository has been expressed by many scientists (Ref. 6, 7).

This paper reviews what DOE Yucca Mountain Project (YMP) researchers know about C-14; i.e., measurements made and analyses performed to date by YMP scientists and others. It also discusses regulatory aspects of C-14 releases through both liquid and gaseous pathways, lays out possible alternative regulatory standards for C-14, and recommends a technical position on C-14 for the DOE to consider. Attempts were made to use references extensively in order to avoid unnecessary duplication of information readily available in the literature.

II. REVIEW OF ANALYSES

Since the current regulations governing the geologic repositories are expressed in terms of cumulative release, individual doses and release rate, the main questions to be addressed are how much C-14 has been emplaced (inventory), how much and how fast it can be freed from the various confinements (source term), how fast it can travel toward the accessible

environment (transport), and what it will do to the individual in the population (radiological exposure). These questions are examined individually with our current knowledge and understanding, based on actual measurements or analyses with ranges of assumptions where these are available or on pure speculation where they are not. An effort was made to identify the sources of information so one can trace the original source of information and make a reasonable guess on the associated uncertainties.

A. Inventory

Carbon-14 is produced as an activation product during reactor operation by neutron reactions with nitrogen-14 (N-14) impurities in the fuel, cladding, hardware and coolant, and with oxygen-17 (O-17) in the oxide fuel and coolant. Production of C-14 by ternary fission can be safely ignored (Ref. 8). The amount produced is directly proportional to the neutron flux and the duration of irradiation time provided the latter is much shorter than one-tenth of the half-life of activated product, which is the case for C-14. In other words, the amount of C-14 in the spent fuel depends on the amount of power generated from the fuel. For this reason, most literature values of C-14 production in the reactor are expressed in terms of curies per gigawatt-year of electricity produced. Since not all fuel elements are exposed to the same level of neutron flux and nitrogen impurity content varies, the amount of C-14 in each fuel element can vary substantially. Calculations based on average burnup and expected level of nitrogen impurities and O-17, therefore, can provide as reasonable an estimate of the total C-14 inventory in the spent fuel as those based on the few available laboratory measurements of samples.

The most comprehensive calculations for U.S. fuel were done by Davis at Oak Ridge National Laboratory (ORNL) (Ref. 9), and have subsequently been updated by others (Ref. 10, 11). The values in the studies have been used as a base in the Yucca Mountain Site Characterization Plan (SCP) and other regulatory analyses (Ref. 6), shown in Table 1.

TABLE 1

Estimated C-14 Content of Spent LWR Fuel (Ci/MTHM)

	Burnup (MWd/MTHM)	UO_2	Zircaloy	Fuel Assembly Hardware	Total
BWR	27,500	0.54	0.76	0.23	1.5[illegible]
PWR	33,000	0.60	0.35	0.60	1.55

The estimated C-14 content in the UO_2 fuel matrix agrees with actual measurements made by the Materials Characterization Center at Pacific Northwest Laboratories. Van Konynenburg documented available measured data on C-14 content in the spent fuel (4 Pressurized Water Reactor and 1 Boiling

Water Reactor fuel assemblies) (Ref. 12). Using the actual measured concentrations of C-14 and more recent data on nitrogen impurities, he revised the estimate of C-14 content in spent Light Water Reactor fuel (Ref. 7) as shown in Table 2.

TABLE 2

Revised Estimate of C-14 Content in Spent LWR Fuel (Ci/MTHM)

	Burnup (MWd/MTHM)	UO_2	Zircaloy	Fuel Assembly Hardware	Total
BWR	27,500	0.54	0.38	0.10	1.02
PWR	33,000	0.60	0.18	0.22	1.00

He then adjusted these numbers for higher average burnups of 29,500 and 37,500 MWd/MTHM for BWR and PWR, respectively, and a total inventory of 70,000 metric tons of initial uranium equivalent, which consisted of 22,500 MTHM of BWR, and 40,500 MTHM of PWR fuel elements, and 7,000 MTHM equivalent of defense waste, to get an average of 1.12 Ci/MTHM in the spent fuel and a repository total of 71,000 curies of C-14.

A more global review of C-14 production from nuclear industries, including seven different types of power reactors and fuel reprocessing, was done by Bush et al. for the Commission of the European Communities (Ref. 13). Their numbers were also based on actual measurements and calculations, including those from the U.S. Since the purpose of their review was to address the total C-14 production from the nuclear industry that will eventually have to be managed, they also included estimates of C-14 in the reactor hardware, which will become low or intermediate level wastes after decommissioning. Table 3 summarizes the values for BWR and PWR. Since the C-14 production is expressed as Ci/GWe-yr in the report, the numbers have been converted to Ci/MTHM using nominal values of 40.2 and 33.5 MTHM/GWe-yr for BWRs and PWRs, respectively.

TABLE 3

Total Production of C-14 from nuclear power generation (Ci/MTHM)

	UO_2	Zircaloy and Fuel Hardware	Reactor Off-Gas	Reactor Hardware	Total
BWR	0.5	0.5	0.25	1.11	2.36
PWR	0.6	0.6	0.15	0.75	2.10

The C-14 inventory in the uranium fuel matrix, cladding, and hardware compares well with those given by Van Konynenburg (Ref. 7). The latter are used as a reference inventory for the following burnup adjustment.

The Table 1 release limits for containment requirements in 40 CFR Part 191 apply to the wastes containing 1,000 MTHM exposed to a burnup between 25,000 and 40,000 MWd/MTHM (Ref. 14). If the burnup is higher, a credit is given. In other words, more release per MTHM is allowed for fuels with higher burnup (where more energy is produced) only if the burnup is higher than 40,000 MWd/MTHM; likewise, a penalty (less release per MTHM) is imposed on those with a burnup less than 25,000 MWd/MTHM. The table does not make any distinction between the BWR and the PWR, and the burnup credit is calculated in reference to a standard burnup of 30,000 MWd/MTHM. Any fuel with a burnup higher than the nominal values of 27,500 (BWR) and 33,000 (PWR) MWd/MTHM but below 40,000 MWd/MTHM will have a higher C-14 inventory than those in Table 2, but will not be allowed with a commensurate increase in the release limit. This would penalize fuels with a higher burnup than the nominal one in terms of allowable fractional release of C-14 if we used the inventory of C-14 in fuels with a nominal burnup as a reference. It is true that any fuel with a burnup below the nominal values but higher than 25,000 MWd/MTHM will benefit in terms of allowable fractional release of C-14 inventory. However, the general trend is toward higher burnups for both the BWRs and PWRs. In addition, the actual measurements for the PWR fuels with high burnups show a substantially higher C-14 content than those in Table 2 (Ref. 7). For those fuels, even after the burnup credit the use of the values in Table 2 will not be conservative. For the purpose of regulatory compliance analysis in this review, the values in Table 2 have been adjusted upward toward higher burnups as shown in Table 4.

TABLE 4

Adjusted C-14 Content in Spent Fuel (Ci/MTHM)

	Burnup (MWd/MTHM)	UO_2	Zircaloy	Fuel Assembly Hardware	Total
BWR	35,000	0.69	0.48	0.13	1.30
PWR	40,000	0.73	0.22	0.26	1.21
Weighted Average		0.72	0.31	0.21	1.24

The 70,000 MTHM to be emplaced in the first repository will consist of 22,500 MTHM of BWR and 40,500 MTHM of PWR spent fuel, and 7,000 MTHM equivalent of high-level defense waste. The average C-14 content for both the BWRs and PWRs is shown in Table 4. The high-level defense waste is the liquid waste generated in fuel reprocessing that has subsequently been solidified in a glass matrix. Because of an almost complete removal of C-14 during the fuel reprocessing and the subsequent vitrification process, these contain hardly

any C-14. The total repository C-14 content will then be 78,000 curies, almost entirely from spent fuel.

At present, it is not clear how the burnup credit is going to be applied to the defense waste. If the burnup credit is given on each radionuclide, the defense waste may not be allowed to release any C-14, since all C-14 in the fuel has already been released to the atmosphere during the processing, or at best it could be treated as a waste with the lowest burnup (i.e., 5,000 MWd/MTHM) allowed in 40 CFR Part 191 and get one-sixth (5,000/30,000) of release credit. In other words, the Table 1 limit for the 7,000 MTHM equivalent defense waste will be either zero or 117 (700/6) curies. Note 4 in Appendix A (Table for Subpart B) of 40 CFR Part 191, however, strongly indicates that no credit may be taken for C-14 for the defense waste, since the release during reprocessing of the fuel already exceeds the release limit of the spent fuel had it not been reprocessed. The release limit for the nominal spent fuel (25,000 to 40,000 MWd/MTHM burnup) for 63,000 MTHM is 6,300 curies. The total release limit for the entire repository would then be 6,300 curies, which represents approximately eight percent of the total inventory.

B. Source Term

C-14 in the spent fuel is distributed in the UO_2 matrix, Zircaloy cladding, and other fuel hardware. A small but significant amount has also been found on or near the surface of the cladding (Ref. 15, 16). Compared to the uncertainty in the inventory of C-14 discussed in the previous section, there is a tremendous uncertainty about the amount of C-14 that will become mobile and be released out of the waste package and Engineered Barrier System (EBS); i.e., the source term for transport to the accessible environment. In fact, this uncertainty may become the main source of difficulty in determining the compliance or non-compliance of the repository system with the regulations. The source term depends on many factors, including the container failure rate, fuel cladding failure rate, fuel oxidation rate, and fuel dissolution rate, all of which in turn depend on conditions in the repository environment such as temperature, amount of water, and water chemistry. Detailed discussion of these subjects is beyond the scope of this paper; only a brief analysis of relevant studies on C-14 is provided below.

1. Waste Container Failure

The container failure rate, as well as the cumulative container failures in 10,000 years, must be known to assess compliance with both the NRC and EPA regulations. At present, our knowledge of both is preliminary. The container material has not yet been selected and the design of waste packages for the spent fuel and defense waste is only at the conceptual stage. The problem, however, is more fundamental than that. There is no established method of predicting, with any certainty, the performance of any man-made material tens of thousands of years into the future. Efforts are being made to develop methods to project the life of containers that far into the future.

It has been shown that, for the release of radionuclides by the aqueous pathway, extending container life beyond 300 years and up to 1,000 years does not improve the total system performance (Ref. 17). 10 CFR Part 60

requires only 300 to 1,000 years of substantially complete containment. The SCP reference strategy for meeting the NRC regulation for the gradual release of radionuclides after the containment period does not rely on the integrity of the containers. Therefore, unless the containers are designed for a longer lifetime to contain gaseous nuclides, the probability of failure of a large fraction of the current reference design containers in 10,000 years is assumed to be high if water comes in contact with the containers, primarily because of a large uncertainties in our knowledge.

The container failure rate depends greatly on the environmental conditions. It is believed that the current candidate repository horizon has remained unsaturated for more than one million years (Ref. 18). Even during the hot period immediately after waste emplacement, when there could be much refluxing of moisture around the waste packages, the DOE's near field performance assessment show that the rock around the waste packages would not become saturated. In addition, there is no known mechanism by which the water in the pores can cross the air gap between the containers and the host rock other than through diffusion across contact areas that might develop or by fracture flows. Depending on the climate, the containers may or may not fail completely during the next 10,000 years. Uncertainties in predicting the climate and repository environment may be so great that the DOE must assume that all containers will fail in 10,000 years. Even with an expensive, more durable container, it would be difficult to guarantee its integrity with any "reasonable assurance."

2. Release from the Waste Container

When a container fails, the spent UO_2 fuel is normally still protected by the Zircaloy fuel cladding, but C-14 on the surface of Zircaloy cladding is not protected and can be released in the form of carbon dioxide. This C-14 is termed the "rapid release fraction of C-14" in the SCP. One measurement of C-14 released from the cladding surface by this mechanism was obtained from an intact PWR spent fuel assembly with 204 rods in it (Ref. 15). The fuel assembly was stored in a test canister filled with air and radiated about 10^{+4} Rad/hr. The canister was heated to 275°C and slowly cooled. A gas sample taken at 118°C during the heating period indicated very little release of C-14. A second gas sample was taken 38 days later at 275°C and contained 1.5 mCi of C-14. It was not reported how long the fuel had been at 275°C before the sample was taken. A third gas sample taken a month later at 270°C indicated an additional release of 0.3 mCi of C-14. It also indicated that one fuel rod out of 204 had breached, as evidenced by the presence of the fission product gas Kr-85. It is, however, believed that the additional C-14 also originated from the external surface of the fuel assembly, based upon later analyses of fuel rod fill gas from other assemblies (Ref. 12). The total release of 1.8 mCi is 0.26 percent of the estimated total inventory of 690 mCi in the sample. Since the estimated total inventory was based on high values of nitrogen content in the fuel and Zircaloy, the actual fractional release may have been somewhat higher than 0.26 percent. Samples taken four months later contained little C-14.

Additional laboratory tests were conducted to determine the magnitude of the rapid release fraction of C-14 and its distribution in the Zircaloy. The results showed that the concentration of C-14 in the 10-micron thick oxide layer is up to five times higher than that in the bulk cladding

(Ref. 19). Release tests were performed with a small piece of Zircaloy sample heated in both air and argon atmospheres at different temperatures. The results indicated that most of the C-14 was released in the form of carbon dioxide from the oxide layer. A release as high as about three percent of total inventory in eight hours was observed at 350°C in air. After eight hours at 350°C, the release appeared to be relatively complete. Considering the variations in the C-14 inventory among different fuel assemblies, H. Shaw of Lawrence Livermore National Laboratories (LLNL) believes that as much as five percent of the inventory could be rapidly oxidized and released (Ref. 20).

It was also observed that a much smaller, but still significant, amount of C-14 was released in an argon environment. It was speculated that some C-14 might have been present in an oxidized form or could be oxidized even in the absence of air before the container was breached (Ref. 21). The implication of this speculation is significant. Since the rate of oxidation of C-14 strongly depends on temperature, the size of the fast release fraction of C-14 could decrease significantly as the waste package cools. However, if the C-14 was oxidized before the container breached, then the amount of rapid release would not depend much on when the breach occurred. This speculation still must be confirmed. The argon gas used in the experiment contained approximately 10 to 50 ppm (vol) of oxygen, an amount far in excess of what would be required to oxidize all the carbon in the sample used (Ref. 21). The presence of other, preferred oxygen-getters such as zirconium may not have completely blocked the oxidation of C-14. Further tests with ultra-pure argon gas were planned but not carried out due to a reduction in funding. [Note: R. Van Konynenburg, LLNL, informed me that a more recent German experiment conducted in an ultra-pure argon environment indicated that an external supply of oxygen would be needed to oxidize the C-14.]

In a different experiment in a saline environment at 200°C, German researchers found that about 50 percent and 95 percent, respectively, of the C-14 inventory in cladding samples of PWR and BWR fuel could be released by corrosion (Ref. 22). This suggests that in addition to the rapid release fraction of C-14 from the oxide layer of Zircaloy cladding, C-14 can also be released as carbon dioxide after the cladding corrodes. The corrosion rate of Zircaloy cladding under conditions at Yucca Mountain is not known. An initial evaluation of samples from two-, six- and twelve-month electrochemical corrosion experiments indicated no Zircaloy-4 corrosion at a detection sensitivity of 1 to 2 microns of corrosion per year (Ref. 23). Further study also indicates that for the storage conditions investigated, the outer zirconium oxide layer is in a state of compression, thus making it unlikely that stress corrosion cracking of the exterior surface will occur (Ref. 24). However, the uncertainty in the long-term corrosion rate of cladding remains. It is assumed, therefore, that once the container is breached, the cladding will also likely breach within a 10,000-year time frame. For this reason, the SCP states that credit will be taken for the cladding as a barrier only if analyses could support it. Even if the cladding does not breach, corrosion processes could release some C-14. In the absence of any data on the corrosion rate of the cladding, Park and Pflum speculated that the combined release in 10,000 years from the rapid release fraction and cladding corrosion could reach ten percent of the total inventory (Ref. 6).

A different type of analysis measured gaseous C-14 through penetrations that corroded through the canisters (Ref. 25, 26, 27). The flow of gases in and out of the container through the penetrations was modeled as a function of time after emplacement, size of the penetration, time of the breach, and internal packaging pressure. The results show that small penetrations will limit the rate of escape of gas from the container. These analyses are useful in analyzing the release of C-14 during the substantially complete containment period. However, they all show that 10,000 years is enough time for the oxygen to diffuse into the container and oxidize C-14 in the Zircaloy oxide layer, and for the C-14 dioxide to escape from the container. In addition, the uncertainties involved in these analyses are too great for the results to be directly useful. It is not presently possible to predict how many and what size penetrations would be created by metal corrosion, and when they would occur. Studies in this area are still very preliminary and the uncertainties involved in the predictions, even if they were possible, would be very large. For the C-14 analysis, therefore, we assume C-14 can move freely through the penetrations once the container is breached.

3. Release from the Fuel Matrix

After the container and cladding are breached, the UO_2 fuel matrix will be altered and dissolved when contacted by the water. Data on the long-term matrix alteration rate are not available. Although a value of 5.3×10^{-6}/yr was used in the Fiscal Year 1990 PACE exercise (Ref. 28), the uncertainty is very large. A value as high as 10^{-3}/yr was used in source term calculations for the tuff repository (Ref. 29). Any contact with water would be limited by the small amount of water flux at the repository horizon, even if a pluvial climate developed in the future, and it is highly likely that the site will remain unsaturated for the next 10,000 years. The earlier study at 25°C indicated a saturated dissolution rate of less than 10^{-5}/yr (Ref. 30). More recent studies indicated, however, that the rate could be two orders of magnitude higher at higher temperatures (Ref. 31). At the flux assumed in the SCP (20 liter/yr/waste package), the entire spent fuel inventory could be dissolved in 10,000 years if the container and cladding breached. This, of course, is a very unlikely scenario, especially in view of the fact that the SCP assumed a flux rate 80 times higher than the 0.5 mm/yr considered a reasonable and conservative upper bound for a Yucca Mountain repository (Ref. 32). It should be noted that the nominal flux used in the FY 90 PACE exercise is 0.01 mm/yr. Nonetheless, in the presence of high water flux, a substantial portion of spent fuel and hence C-14 could be dissolved and transported in water. Due to an extremely low diffusion coefficient in unsaturated rubble around the waste package (Ref. 33) and low flux, the liquid would travel very slowly and would be exposed to gas flow moving upward. The heat from the emplaced wastes in an unsaturated site could induce a large-scale air and gas convective movement (Ref. 34).

The C-14 in the water will reach thermodynamic equilibrium between gaseous CO_2 and aqueous HCO_3^-. Once the C-14 transfers to the gaseous phase it will go through the same process as the gaseous C-14 released from the Zircaloy surface. It should be noted that the conditions above and below the repository level are almost identical in terms of the CO_2 environment, so the CO_2 will partition between the liquid and gas regardless of the origin. The C-14 in the gaseous phase will move upward much faster than the liquid will travel downward. The net result is that most of the C-14 in the water, after

some time delays due to retardation, could end up being released in gaseous form to the environment. Therefore, the net source term for the gaseous transport of C-14 would be the sum of the rapid release fraction from the Zircaloy surface and a significant fraction of the C-14 dissolved in water. While the former is a one-time release from the breached container, the latter is a continuous and cumulative release from all breached containers as long as the fuel continues to dissolve. The cumulative release of C-14 from spent fuel dissolution could provide a much larger source term than the rapid release fraction, depending on the amount dissolved and the degree of thermodynamic equilibrium (partitioning between the gas and liquid).

Carbon-14 may exist in various chemical forms in spent fuel and hardware. Release of C-14 from reactor off-gas was observed to be in the form of carbon dioxide, carbon monoxide, and hydrocarbons including methane (Ref. 12). The C-14 in the Zircaloy surface is oxidized first, before it is released. The actual release measured from the test fuel assembly was in the carbon dioxide form with no measurable amount of carbon monoxide, except for one sample that contained an insignificant amount (Ref. 15). During the dissolution of chopped spent LWR fuel rods with air sparging at ambient temperature (in nuclear fuel reprocessing plants), almost all C-14 is released into the dissolver off-gas in the form of carbon dioxide. Therefore, it appears that the gaseous release of C-14 from the tuff repository would most likely be in a carbon dioxide form.

C. Transport of C-14

The transport of gaseous C-14 from the repository to the environment would be controlled primarily by the flow of gas through fractures and rock pores. The gas interacts with the water trapped in rock pores or on the fracture surface. C-14 in the gas will exchange with the C-12 in the pore gas, which is in equilibrium with the bicarbonate ions in the pore water, which in turn may be in equilibrium with calcite in the rock. The net result is an effective retardation of C-14 movement through the rock. The degree of retardation depends on the degree of deviation from a thermodynamic equilibrium between the gas and liquid in the pores.

1. Gas Flow Through the Mountain

Gas moves through the deep unsaturated zone at appreciable velocities (Ref. 18). This is a convective movement caused by the density difference in gases with depth due to the geothermal temperature gradient, as well as by diurnal and seasonal changes in barometric pressure (Ref. 35, 36). Substantial air flow has been observed in several wells drilled in the vicinity of Yucca Mountain and a section of open hole above the water table. In one well, the observed flow rates are so great they can only be explained as fracture flow phenomena (Ref. 37). Nearly 40 percent of the actual flow from one observation well is generated by wind effects. The flow log also indicates that the midpoint for flow entering the well is at a depth of 20 meters (Ref. 37). Although the observed gas flow velocity -- ranging from negative to +7 m/s at the top of the well -- has been modeled, gas flow throughout the mountain is not known well, especially at the repository depth.

If high-level waste is placed in an unsaturated repository, the heat generated by the waste will provide a driving force that moves large volumes of gas. Tsang and Preuss estimated the velocity of heat-driven gas flow from a hypothetical repository and natural geothermal temperature gradient (Ref. 38). Their results show that gas phase convection could take place with appreciable velocity, of the order of 22 m/yr. This average velocity has been used by others to calculate the rate of C-14 transport through Yucca Mountain (to be discussed later). More detailed simulation of gas flow velocities as a function of depth shows a range from 4.5 to 1174 m/yr at 100 years after waste emplacement, with the highest velocity at the repository level. Other studies of the potential repository at Yucca Mountain indicate that the temperature disturbance resulting from emplacing the waste will be significant even after 10,000 years (Ref. 39, 40). In a recent study, Tsang simulated the temperature and gas velocity field up to 10,000 years after waste emplacement using the layered stratigraphy at Yucca Mountain and the reference heat load of 57 kW/acre at the time of emplacement (Ref. 41). The results still show a wide range of velocities through the different strata, from a fraction of a meter per year in Paintbrush tuff (gas flow only through matrix pores with porosity of 0.4 was assumed) to over 200 m/yr at the repository level. The average velocity near the top of Tiva Canyon still approaches 40 m/yr at 100 years, 20 m/yr at 1,000 years, but then decreases to a few m/yr at 5,000 years. Due to a buoyancy effect, the locus of the fastest velocity moves toward the top of the Topopah Spring tuff.

Water vapor movement produced by the heat pipe near the waste package could affect the migration of gaseous radionuclides. Zhou et al., however, show that for the equivalent waste sphere the heat pipe exists from eight days to 40 years after emplacement (Ref. 42). In addition, they also conclude that the heat pipe extends from the waste surface to about three meters from the center of the equivalent waste sphere. For a large-scale gas movement for 10,000 years, therefore, we can safely ignore the heat pipe effect.

2. Retardation

The movement of gaseous C-14 can be retarded by complex chemical interactions with the pore water and the solid rock. Ross describes a general chemical model for C-14 retardation at Yucca Mountain and estimates the bounds of the retardation factor to be 2 to 2,000 (Ref. 43). In a more recent study, he calculated the retardation factors for three different stratigraphic layers as a function of temperature, obtaining a range of 30 to 70 with an approximate median at 50 (Ref. 44). Ross used the PHREEQE computer code to obtain the equilibrium distribution coefficient. Others used data from the literature, expressed as a function of pH and temperature, to account for the retardation of gaseous C-14 movement in their transport equations (Ref. 45, 46). Although they did not calculate retardation factors explicitly, their numbers are of the same order of magnitude but higher than those calculated by Ross. While Knapp used the equilibrium distribution coefficients at pH 8 as a function of temperature, Light et al. used a fixed value at pH 7 and 50°C to get an equilibrium distribution coefficient of 3 (Ref. 47).

Many implicit assumptions have been made in calculating the retardation factor, of which the most important is that of a thermodynamic equilibrium between the gas and liquid. On the time scale involved in the repository C-14 travel, Ross justifies the validity of such an assumption. Yang

analyzed pore fluid and pore gas to determine the extent of the water-rock interaction and gas travel time at Yucca Mountain using an isotope ratio of carbon, oxygen, and tritium (Ref. 48). His preliminary finding suggests a lack of thermodynamic equilibrium between the gas and liquid but the results are not conclusive. Although the gas and pore liquid were obtained generally from the same geologic strata, the actual gas sample was collected from the UZ-1 hole while the pore water was extracted from UZ-4 cores. The data still strongly suggest the possibility of a very low retardation, especially if the liquid is confined to small pores (high suction pressure) and the gas flows through the path of least resistance (fractures and large pores) with minimal contact with pore liquid.

All this suggests that the degree of retardation may also strongly depend on the degree of saturation. With a pluvial scenario, more liquid flux to the repository (still unsaturated) may accelerate the corrosion of containers, thus increasing the source term for C-14. A possibly lower temperature resulting from more cooling may also reduce the rate of oxidation, but in the long run the total release may not be affected much. On the other hand, a higher saturation may enhance liquid-gas contact, hence increasing the retardation. No quantitative data are available on the relative contribution of these two counteracting effects from increased flux. However, it can be seen that the source term and transport strongly depend on the expected hydrology.

3. Far Field Transport

A nominal travel time of gaseous C-14 from the repository to the accessible environment can be obtained from the gas flow velocity through the mountain and the retardation coefficient of C-14. As mentioned earlier, the unretarded gas travel time through the mountain is relatively short -- from tens to hundreds of years -- which means the retarded travel time could be from less than 1,000 years to over 10,000 years. Since the half-life of C-14 is 5,730 years, the effect of retardation can become significant with a long travel time. Although this view of gas travel time is very simplistic, it clearly indicates that the travel time is neither very short nor very long and more accurate estimates are needed.

Ross first modeled the C-14 transport at Yucca Mountain (Ref. 43). His preliminary calculations based on the governing equation and order of magnitude estimates indicated that a substantial portion of C-14 could reach the surface in less than 10,000 years. Knapp solved an analytic equation for gas phase transport of a C-14 kinematic wave, incorporating advection, isotope exchange between CO_2 in a flowing gas phase and HCO_3^- in a stastic aqueous phase, and radioactive decay (Ref. 45). His calculations indicate that the C-14 wave takes about 5,900 years to reach the surface. This implies that about half of the C-14 released from the repository during the first 4,000 years will reach the surface during the regulatory time frame of 10,000 years. His calculation is based on an estimated gas Darcy velocity of 1 m/yr and no diffusion, with dispersion and temporal and spatial variations in rock and fluid properties taken into consideration.

Lerman also estimated the travel time of gas through an unsaturated rock zone based on the expanding gas volume and the density gradient caused by the heat generated in the repository and diffusional flux (Ref. 49). He estimated an

average gas velocity of 2 m/yr, using a gas permeability three orders of magnitude lower than the values reported by Montazer et al. (Ref. 32). Although his analysis made the point that some gaseous radionuclides might reach the surface in a relatively short time, his model grossly lacked the complexities needed; e.g., no geochemical retardation was modeled.

Light et al. also solved the governing equation using an equivalent porous-medium approach and calculated the gas concentration at the ground surface as a function of time and gas flow velocity in the mountain (Ref. 46). They used the Darcy velocity of gas calculated by Tsang and Preuss (Ref. 38) as a reference, and calculated the gas travel time for 0.1, 1.0, and 10 times the reference Darcy velocity. A fixed equilibrium distribution coefficient of 3 at pH 7 and 50°C was used to calculate the retardation. The results show C-14 travel times to the surface to be in hundreds to thousands of years for the assumed parameter values.

The most rigorous and comprehensive modeling was done by Ross et al. (Ref. 44). A two-dimensional, steady state numerical model of rock-gas flow driven by temperature and humidity differences, called TGIF (Thermal Gradient Induced Flow), was developed to determine flow paths by particle tracking and to calculate C-14 travel time. The model takes into consideration the different geologic strata with different permeabilities, tilting of the bed, Yucca Mountain topography, and geochemical equilibrium between the gas and liquid. The model treats the fractured tuff as a homogeneous medium. C-14 travel times were calculated for three different repository temperatures two levels of permeability contrast between the Paintbrush nonwelded tuff and the Tiva Canyon and Topopah Spring welded units at four east-west cross sections. Fixed repository temperatures were used instead of the actual time dependent heat generation rate of the waste. The temperature profiles generated using a waste heat load of 57 kW/acre by Tsang indicate the repository temperature could be higher than the values used by Ross, especially during earlier times, even up to several thousand years (Ref. 41). The C-14 travel times calculated were shown in histograms. As expected, the unretarded travel times range from tens to hundreds of years, and the retarded travel times are generally in thousands of years. His calculations also show that at lower temperatures and higher permeability contrasts, many or most of the retarded travel times exceed the C-14 half-life of 5,730 years and the regulatory time frame of 10,000 years. On the other hand, with a low permeability contrast and a repository temperature of 330°K, almost all C-14 escapes to the atmosphere in less than 2,000 years.

Overall, these calculations show that the expected C-14 travel time is generally several thousands of years or less, including retardation. These calculations assume the maximum retardation possible using thermodynamic equilibrium, but do not take into account the effects from wind and barometric pumping. Analysts used a retardation factor of about 50, which is a very high retardation for gas movement. In many other geologic media, the retardation results from physical or chemical sorption of C-14 on the media itself. There is little information on the sorption of C-14 on various kinds of rocks. There are some indications, however, of the magnitude of retardation that sorption provides. Bush et al. used a value of 8 for retardation in a clay medium, which is highly sorptive (Ref. 13). The high retardation at Yucca Mountain is due to the geochemical interaction of C-14 dioxide with HCO_3^- in the pore water, which is in equilibrium with an

abundant amount of calcite in the rock. Other geologic media may not have as high a retardation factor as Yucca Mountain; therefore, it appears that the relatively short C-14 travel times may not be unique to Yucca Mountain, but may apply to most generic unsaturated sites in the U.S. Ross also states that the general conditions used in his simulation would apply to most other unsaturated sites (Ref. 44).

D. Health Effects of C-14

Carbon is one of the most abundant elements on earth and in the biosphere. It constitutes over 22 percent of the human body by weight (Ref. 8) and is abundant in our daily diet. Natural carbon contains about 1.4×10^{-12} g C-14/g C. A reference human being weighing 70 kg contains 0.1 microcurie of C-14, from which he receives 1.3 mrem/yr of radiation exposure (Ref. 8, 50). The global inventory is estimated to be 230 million curies, which are distributed as follows: 90 percent in deep ocean more than 100 m from the surface; 8 percent in surface waters, sediment and biosphere; and two percent in the atmosphere (Ref. 51). In addition to the large inventory of C-14 already existing in the natural system from cosmic ray production, additional C-14 is continuously produced in the atmosphere by the interaction of cosmic ray neutrons with nitrogen. The amount in the atmosphere is estimated to be 3.8 million curies, and the annual natural production of 28,000 curies in the atmosphere (Ref. 52) balances the loss by radioactive decay.

C-14 released from a repository in gaseous form would enter the atmosphere and mix completely in about four years to become part of the global inventory. C-14 in the human body also comes to an equilibrium with the atmospheric C-14 after a lag time of 1.4 years (Ref. 53). Once it is released, C-14 becomes a part of the global inventory and any increase in concentration in the atmosphere could affect the entire world population, provided the assumption of a linear no-threshold relationship between the health effect and radiation exposure holds. It should be noted that this assumption is not well established at a low level of radiation.

The C-14 in the atmosphere exchanges with carbon in the ocean surface waters, which in turn exchanges with carbon in other reservoirs such as deep ocean, land biosphere, and humus; most of the radioactive decay occurs in the ocean, where it stays longest during the global circulation cycle. As a result, the effective half-life of C-14 in the biosphere is much shorter than its natural half-life of 5,730 years.

The potential health effects of C-14 from both the natural and man-made sources have been studied extensively (Ref. 50). Infinite time (effectively about eight half-lives or 46,000 years) population dose commitment of C-14 has been calculated by many studies (Ref. 13, 50). The numbers range from 370 to 620 man-rem/Ci (divide the number by 100 to get person-Sievert/Ci) based on a projected steady world population of 10 to 12 billion. In a more recent study, McCartney et al. reported a value of 460 man-rem/Ci for the 100,000-year dose commitment based on a steady world population of 10 billion (Ref. 54). The biological effect per unit population dose also varies depending on the pathway model and other assumptions used. Reported values range from 100 to 200 cancers for $1 \times 10^{+6}$ man-rem (Ref. 8, 50). The EPA used a value of 146 cancers per $1 \times 10^{+6}$ man-rem exposure in their analysis, although they also indicated the value probably was lower by a factor of 1.5

based on newer data (Ref. 55). [Note: They are using 400 cancers per $1x10^{+6}$ man-rem now, according to Mr. Galpin at the 11/91 EPRI Workshop]. We use a number of 200 cancers per $1x10^{+6}$ man-rem, which is consistent with the value recommended by the ICRP. The number of genetic effects from C-14 exposure is estimated to be about one-tenth to two-third of the total cancers (Ref. 50, 55). Using these numbers, the limit of 6,300 Curies to comply with the EPA regulation from the 70,000 MTHM repository equates to a total cancer death of 580 over 10,000 years. It is to be compared with 370,000 cancers from natural C-14 and 37 million from total natural background radiation over the same time period.

The health effect of a release of C-14 from a potential repository at Yucca Mountain has been calculated by Daer under two different scenarios (Ref. 56). Under the first bounding scenario, a release of 1,000 curies in one year from the ground surface was assumed. It was also assumed that the entire projected surface area of the repository was covered by an invisible confinement 2 meters high, and the C-14 inside stayed within this volume fc_ the entire year. People lived inside the confinement eating contaminated food grown inside and drinking contaminated water. Under this ultra-conservative, almost implausible scenario, the maximum exposure was calculated to be about 2 mrem/yr. Obviously C-14 would not be trapped locally, the annual release would be almost three orders of magnitude lower, and there would not be much vegetation near the Yucca Mountain area. Ingestion dose dominated over submersion and inhalation doses, as expected.

The second analysis was only for internal and external doses from air containing C-14, and was based on a uniform release of 1,000 curies from the ground surface of the repository in one year and currently prevailing climate conditions, such as wind velocity, direction, dispersion of the plume, etc., at Yucca Mountain. Under this still conservative scenario, the exposure to the maximally exposed individual was calculated to be 0.05 mrem/yr. Under the allowable release limit of an average of 0.63 Ci/yr (6,300 Ci per 10,000 years), the corresponding exposure would be $3x10^{-5}$ mrem/yr. The second analysis did not include the dose from ingestion. In areas with much vegetation, the ingestion dose from the food chain dominates over the dose from inhalation and immersion by about two orders of magnitude. At Yucca Mountain, however, the ingestion dose is expected to be only one order of magnitude larger than the inhalation dose, primarily due to the low potential for vegetation (Ref. 57). If we include the dose from ingestion in the second scenario, the total dose from C-14 from the potential Yucca Mountain repository would be $3x10^{-4}$ mrem/yr, which is about one one-millionth of what an average individual receives from natural background and one ten-thousandth of what an individual receives from natural C-14 from the atmosphere.

In a more recent study, done as a part of the FY 91 PACE by Pacific Northwest Laboratory and Sandia National Laboratories, the potential dose from the repository C-14 was calculated (Ref. 57). The ground surface source term of C-14 for the dose calculation was estimated probabilistically for different container failure times, two different gas flow modes; i.e., matrix and fracture flow, an average wind speed of 3.3 m/sec with no vertical or horizontal dispersion, and different matrix gas permeabilities. The overall scenario, including the source term from the EBS, was very conservative. Under this scenario, the calculated dose to a hypothetical, maximally exposed individual living on the surface of Yucca Mountain ranged from $2.3x10^{-17}$ to

1.2×10^{-1} mrem/yr. No attempt was made to calculate an average or median value in this preliminary study. The numerical values calculated deterministically by Gary Daer fall within the range of this study.

For the purpose of the regulatory analysis in the next section, we will use 3×10^{-4} mrem/yr as the basis.

E. Uncertainties

Among the factors influencing the release of C-14 to the accessible environment, the inventory estimates have the least uncertainty, though they are still significant. Considering the accuracy of the ORIGEN code used in the calculation of isotope generation in the reactors, the amount of nitrogen impurities in the fuel, cladding, hardware, variability among fuels, the fact that two-thirds of the spent fuel to be emplaced in the repository doesn't even exist today, and the trend toward ever higher fuel burnups, the uncertainties in the inventory are probably at least -50 to +100 percent.

The largest uncertainty, however, is in the source term, which in part stems from the uncertainty in the post-closure near field environment. Container failure rate is largely unknown and uncertainties will remain even after the material and design are fixed. If the near field environment remains unsaturated and relatively dry, the container failure rate would be very small and a large fraction of the waste containers will survive for 10,000 years. If the climate changes to a pluvial condition, fracture flows dominate at the repository level, and a large amount of water comes into contact with the waste containers, then, conservatively, with the current design of the waste package it should be assumed that most of the containers would fail during the first 10,000 years. The uncertainties in the container failure rates would be at least one order of magnitude and could be higher, depending on the degree of site characterization and material testing. The uncertainties in container failure rate could be reduced by employing more robust, long-life waste package design, but presently there is no regulatory need for a long-life (10,000 years or longer) waste package to meet the EPA performance requirements other than that for C-14. Compliance with the NRC's subsystem performance requirements on waste packages and EBS may necessitate a long-life waste package because of the need to contain gaseous radionuclides and several other readily soluble radionuclides. Among these, the requirement for C-14 would still be the most imposing.

Data on the C-14 release from the surface of fuel assemblies; i.e., the rapid release fraction, are extremely limited, so more experimental measurements are needed. The value assumed for the rapid release fraction in the SCP; i.e., one percent, appears low in view of more recent laboratory experimental results. Two to ten percent may be a reasonable range, although there is a possibility that it may even exceed ten percent. Again, it should be mentioned that these figures are based on a limited number of observations and are speculative at best.

Release of C-14 from the fuel matrix would be strongly influenced by the alteration rate of the fuel. Current assessment indicates a possible range of at least two to three orders of magnitude. There is an additional uncertainty in the fraction of C-14 released in liquid form initially that might eventually be released to the accessible environment in a gaseous form.

Depending on the scenarios for the near and far field environment, the fraction could vary from almost 0 to 100 percent.

Most calculations on the C-14 travel time at Yucca Mountain indicate that it may be relatively short compared to the 10,000-year regulatory timeframe and the half-life of C-14. The natural barriers at Yucca Mountain may not be able to delay the movement of C-14 past the regulatory time limit or until it decays by a significant amount, even with the retardation due to geochemical interactions. It would be reasonable to assume that all C-14 released from the waste packages within the 10,000-year timeframe would reach the accessible environment quickly, without much radioactive decay. Aside from the uncertainties in the retardation factor, from one to an average of 50, the travel time is strongly influenced by rock permeabilities that vary in different strata. C-14 could reach the surface in a few years to tens of thousands of years, although a few thousand years seems the most likely.

The long-time population dose commitment of C-14 is generally well established. Models for the global carbon circulation cycle have long been in existence, from the simple three-reservoir models of earlier days to recent, more sophisticated multi-reservoir models. Most of the models currently in use are variations of the six-reservoir model by Bacastow and Keeling (Ref. 58). Results from different models generally agree well because the deep ocean acts as the primary reservoir, holding more than 90 percent of global C-14 and dominating the circulation cycle.

The overall combined uncertainties are so large, including those for the disturbed scenarios, that from almost 0 up to 50 percent of the total inventory in the repository (up to 40,000 curies) could be released in the gaseous form over the 10,000-year period. Of course, this is a very high estimate, and most likely the probability distribution of release would be highly skewed toward lower values. The big question is what would be the probability of the release exceeding eight percent of the total inventory. Due to the uncertainties discussed above, it would be reasonable to assume a ten percent probability that the gaseous release would exceed eight percent of the C-14 inventory.

F. Need for Additional Analyses

The results of most analyses are uncertain because of lack of data, especially long-term data that may or may not be fully obtainable. Some uncertainties could be reduced by site characterization data and laboratory and field experimental measurements, but there will always be residual uncertainties from both the known and unknown unknowns. Since the transport of C-14 is relatively fast, what is needed most is more data on the source term, not only for Yucca Mountain but for other unsaturated sites as well. Analyses that could reduce the uncertainty band in the source term should be emphasized.

It might be worthwhile to solicit expert opinions in each of the categories discussed above to narrow the range of uncertainties, then to run a simple model to obtain a probability distribution of C-14 gaseous release by employing time-distributed container failure, range of retardation and travel time, etc. The results, however, would still be speculative at best since we are limited more by the lack of real data than by reliable means of analysis.

Additional data needs have already been discussed in the Yucca Mountain SCP and briefly in the sections above, and will not be repeated here.

The analyses in the previous sections deal only with undisturbed performance of the geologic repository. Based on our preliminary knowledge of volcanism scenarios, it was assumed in this study that any gaseous release of radionuclides under disturbed conditions of the repository would be insignificant. This, however, should be investigated further.

III. REGULATORY IMPLICATIONS AND POSSIBLE ALTERNATIVES

A. Regulatory Implications

The NRC's subsystem performance requirements in 10 CFR Part 60 require that the containment of radionuclides in the waste packages be substantially complete for 300 to 1,000 years, and that after containment the annual release rate of any radionuclide from the EBS not exceed one part in one hundred thousand of the inventory of that nuclide at 1,000 years after emplacement with an exclusion limit for radionuclides with an extremely small release potential. The regulatory term "substantially complete containment" has not yet been defined quantitatively. The NRC made it clear in its Site Characterization Analyses that the term should be interpreted to mean that the release during the containment period be much less than that allowed during the post-containment period (Ref. 59). Design goals were established in the SCP with a goal of achieving a C-14 release rate of less than 10^{-6}/yr of the 1,000 year inventory, which would correspond to 7.8×10^{-2} Ci/yr. Even if we assume the rapid release fraction to be two percent of the inventory in the container, failure of two or fewer containers per year would exceed the SCP goals and the 10 CFR Part 60 requirements even if we ignore the C-14 released through the aqueous phase. If we take a more conservative number of ten percent for the rapid release fraction, then it takes only a fraction of one waste container to violate the requirement in a given year. The 10 CFR Part 60 requirements could also be violated if 2 to 20 waste containers breach in a given year. If we include the cumulative release from all failed containers that will cross the EBS boundary in either a gaseous or liquid form, the number of containers that can breach annually would be even less. This level of containment may be possible if an expensive waste package design with multiple barriers is employed. Nevertheless, it would be almost impossible to guarantee such a low level of failure on an annual basis as the NRC regulations require.

The EPA regulation, 40 CFR Part 191, does not specify any requirement on the performance of subsystems. It is an overall environmental standard, and as such it only limits cumulative release to the accessible environment. The limit for C-14 is 100 curies per 1,000 MTHM over 10,000 years with better than 90 percent probability that the level would not be exceeded, provided no other radionuclides are released at the same time. If other radionuclides are released concurrently, the release limit must be prorated (i.e., reduced) by a formula given by the EPA. The release limits were conceived to limit the number of fatal cancers to 1,000 over 10,000 years from a repository containing 100,000 MTHM. As shown in the previous section, the final number used for C-14 is equivalent to 570 fatal cancers over 10,000 years from a repository containing 70,000 MTHM, of which 63,000 MTHM are spent fuel. The level of risk; i.e., 1,000 cancers over 10,000 years, was considered easily

achievable at the time based on performance assessment of generic sites, and was also considered to be comparable to the risk from the unmined uranium ore (Ref. 55, 60).

The EPA limits total release of C-14 to 6,300 curies in 10,000 years and the NRC limits the release to about 0.63 Ci/yr. If only eight percent of the C-14 inventory at emplacement escapes to the accessible environment, we could violate the EPA and NRC regulations. The current lack of data and high uncertainties also reduce the confidence that we can meet the regulations.

It has also been shown in the preliminary performance assessment of the Yucca Mountain repository that the complementary cumulative distribution function (CCDF) of the release is largely dominated by the release of C-14 (Ref. 2, 3). Although the results show that the CCDF curve is still within the bound of the EPA limit it is very close to violating it, even without taking into account all the uncertainties discussed in the previous section.

A few alternative waste package strategies have been proposed in the SCP that could be very expensive and still might not be able to provide reasonable assurance that the release would be within the EPA and NRC limits. Some of the proposed technologies have not yet been fully developed or demonstrated. They are discussed below in conjunction with regulatory alternatives.

B. Discussion of Regulatory Alternatives

The EPA conclusion that its release limits were easily achievable was based on assessments of several hypothetical repositories (Ref. 61). Unsaturated repositories and gaseous radionuclides were not considered in determining whether the release limits could be met. The hypothetical repositories were also simpler than the real sites the DOE has studied, making the validity of the EPA's conclusions questionable.

An apparent basis for the EPA limits is hidden in their comparison of repository risks to the risk from unmined uranium ore:

> "Accordingly, the Agency has promulgated environmental standards that would restrict projected releases from high-level waste disposal system -- for 10,000 years after disposal -- to levels that should keep the risks to future generations less than the risks they would have been exposed to from the unmined ore if these wastes had not been created." (Ref. 55)

The level of risk from unmined uranium ore was calculated for a few real and one hypothetical uranium mine (Ref. 62). Using a hypothetical uranium mine as a basis is unreasonable in view of the fact that most of the uranium mines from which the first 70,000 MTHM fuel would be produced could be identified (both domestic and foreign), and the risks from unmined uranium ore body could also be obtained from environmental documents. The probability limits EPA assigned to the release; i.e., 0.1 and 0.001, also have no basis, since the probability of releasing the calculated amount from a real mine is almost 1.0, because those assessments are based on actual measurements. These facts have been pointed out in testimony to the Advisory Committee on Nuclear Waste

(ACNW) by U. Park (Ref. 63). A subsequent ACNW evaluation confirmed that the EPA release limit was at least one order of magnitude more stringent than the limit that would produce the same risk from real unmined uranium ores in terms of release probability, and three orders of magnitude in terms of the associated health effects due to radionuclide releases (Ref. 64). In explaining why the EPA did not choose higher (less protective) release limits, they state:

> "... The differences in costs for different levels (of protection) are much smaller than the overall uncertainties in waste management costs. For example, consider the increased costs of complying with the release limits we have proposed, rather than release limits 10 times less stringent. The potential increase ranges from zero to 50 million (1981) dollars per year.... As discussed above, setting the release limits at the level we chose -- as opposed to a level 10 times less or 10 times more stringent -- appears to cause only very minor effects on the costs of high-level waste disposal. This is why we did not choose higher (less protective) release limits." (Ref. 65)

The EPA was mistaken. Costs are very sensitive to the level of protection, especially when the requirements push the design of waste packages to the limits of practical engineering and science. If costs were properly considered, the release limit could be justified at 10 times higher than what was finally set by the EPA and the public health and safety would still be fully protected.

Given this general background on the EPA regulation, the following approaches to develop an alternative standard for allowable release of C-14 would seem to merit consideration:

- o Keep the current regulation and
  - use longer-life containers
  - release the C-14 before emplacement
  - use fuel reprocessing
- o Relax the current release limit for all radionuclides by a factor of ten.
- o Give special consideration to C-14 because of its unique nature and because it produces an individual dose that falls well below regulatory concern (dose truncation).
- o The same as above, except base the truncation on the affected population (geographic truncation).
- o Change the basis of the standard from population dose to individual dose.
- o Regulate repository gases under the National Emission Standards for Hazardous Air Pollutants Act (NESHAP) (40 CFR Part 61).

- o State that the release limits in Table 1 of Subpart B, Appendix A do not apply to gaseous release of radionuclides and hold the regulation of gaseous releases in reserve.

These options are discussed individually below in terms of their advantages and disadvantages from a scientific point of view and, to the extent possible, from their political implications.

1. Keep the Current Regulation

The current regulation was promulgated based on three basic premises: (1) it is easy to meet the limits; (2) the risk is comparable to the risk from unmined uranium ore, which is acceptable to the public; and (3) more stringent regulation does not incur any significant additional cost. However, what may have been a reasonable assessment based on the state of knowledge 10 to 15 years ago is no longer valid. The regulation is outdated and should be changed.

There appears to be a high probability that it will not be possible to satisfy the EPA and NRC regulations because of overwhelming uncertainties in the source term. The preliminary performance assessment clearly showed that the main reason for potential violation of the regulations is the gaseous release of C-14. This has been foreseen by YMP scientists for a long time, and the DOE has proposed several alternative approaches in the SCP in case the reference waste package cannot meet the requirements due to uncertainties in the site conditions (Ref. 18). The alternatives were presented primarily to address the NRC's 10 CFR 60.113 requirements. They include the use of alternative container design and materials, use of 10 CFR 60.113 (b) (variation in containment period and post-containment release rate), release of C-14 from the surface of fuel assemblies prior to emplacement, taking more credit for cladding if this could be supported by more testing, and inclusion of part of the host rock in the EBS. Among these, only two could address the C-14 problem for both the EPA and NRC requirements: a long-life waste package using alternative material, and the pretreatment of fuel assemblies to release the rapid release fraction of C-14. These are discussed in more detail below.

a. Use of long-life waste packages

The current reference design for the waste packages is a thin-walled, single wall metallic container that capitalizes on the unsaturated nature of the site. In the absence of any significant water movement at the repository level, this design would be adequate to protect the public health and safety. Under any scenario that would allow the breach of waste containers in any significant quantity during 10,000 years, the reference design and the current candidate materials may not be adequate or may be adequate but cannot be so proven. Since the rapid release fraction of C-14 is on the outside surface of the fuel cladding, the waste container wall must be gas tight for 10,000 years. Most metals have only a short performance history and are susceptible to various failure mechanisms. Ceramics such as alumina were considered in combination with metal inner or outer layers. The additional cost over that of the reference design is estimated to be in the billions of dollars for 35,000 waste packages. In addition,

the technology exists only in small-scale applications. Significant technology development would be needed involving an additional cost and schedule delay. The DOE is studying other long-life waste package designs that rely on multiple barriers to increase reliability, but no firm designs have evolved yet. The high cost of developing and fabricating long-life waste packages compared to the negligible gain in public health and safety has already been pointed out (Ref. 6,7).

b. Pretreatment of the fuel assemblies

The existing regulations do not regulate the release of C-14 from nuclear power reactors and other nuclear fuel cycle facilities. The operating PWRs and BWRs release gaseous C-14 at the rate shown in Table 3. Each reactor releases 5 to 10 curies every year. If the C-14 on the surface layer of Zircaloy cladding is released prior to emplacement, it would not violate any regulation. The rate of release from this operation would be much higher than the release of C-14 from the repository, since at least two percent of the total inventory (1,500 curies) would be released in less than 50 years. Assuming that the linear dose-response model is valid, the resulting health effects would be much higher than the effects produced by the expected release from the repository, although both would still be very low.

To release C-14, the fuel must be heated to about 275°C for an undetermined length of time. Under laboratory conditions with a purge gas flow, the release was almost complete after 8 hours. However, the only actual test done with an intact fuel assembly indicates up to two months might be needed. The cost of performing this operation, even if it was technically feasible, would be extremely high. The annual spent fuel receiving rate is twice as high as the rate at a full-scale, 50 GWe/yr fuel reprocessing plant, and the fuel would then have to be stored for up to two months at 275°C. The fuel assemblies would have to be cooled before transport to the repository. The cost of such a facility, operated remotely, would be prohibitively high when the off-gas treatment and other handling facilities are included.

In addition, the effect of heating the fuel in a dry condition is not known. One out of the 204 fuel rods failed during the test. Other technical problems include finding a method of heating the fuel assemblies uniformly without overheating to prevent cladding failure, and the treatment of radioactive off-gases Kr-85 and I-129 from breached fuel rods. Both Kr-85 and I-129 are regulated under current regulations. The C-14 gas from heating would have to be vented to the atmosphere, since it would be diluted so much with air it could not be recovered economically.

It should be noted that releasing the C-14 at a higher rate just to circumvent the repository regulations may not be acceptable to the public regardless of the low health affects.

c. Fuel reprocessing

Fuel reprocessing is not a real solution to the C-14 problem, since the decision to reprocess will involve many considerations and C-14

may not be an important one. Although the release of C-14 from a fuel reprocessing plant (FRP) is not regulated at present, primarily because there is no FRP in the U.S. except for the defense facilities, the off-gas stream is concentrated enough to warrant its collection from a cost/benefit point of view (Ref. 66).

Technologies are available to collect the C-14 (diluted with C-12 to increase the efficiency the fixation process). The problem is what to do with the waste containing C-14. Most fixation processes capture the C-14 in a carbonate matrix. The release rate of C-14 from such waste forms packaged in a less stringent container buried in a shallow or deep geologic disposal may be significantly higher than the release rate from a repository. [See Radiation Physics & Chemistry, Vol. 37, No. 2, pp. 363-365, (1991) on radiolytic decomposition of $Ca^{14}CO_3$.]

2. Relax the Stringency by a Factor of 10

The stringency of the current regulation does not have its basis on a firm need to protect the public health and safety. The ACNW showed that the EPA used a factor of 10 conservatism in the probability and three orders of magnitude in the associated health effects (Ref. 64). In 1984 the EPA's Science Advisory Board (SAB) recommended that EPA relax the risk objective for all nuclides by an order of magnitude (Ref. 67).

There is plenty of justification to relax the regulation by a factor of 10 based on a realistic estimate of risks from unmined uranium ores, difficulty for any generic site to meet the current regulation under real repository conditions (all unsaturated sites may be penalized), and the high cost of meeting the regulation with little benefit to the public health and safety. On the other hand, it might be perceived by the public that the public health and safety would be compromised, if the regulation were relaxed.

3. Dose Truncation

It has already been shown that the expected radiation exposure from C-14 by the repository release is very small, even to the maximally exposed individual; i.e., on the order of $3x10^{-4}$ mrem/yr. Although the no-threshold linear dose assumption is well accepted by the scientific community, its applicability to low levels of radiation dose has been questioned continuously. The current acceptance of the no-threshold assumption at low doses is not because of demonstrated validity but because it is believed that it will not make much difference, since most sources of such low doses are not regulated. Most other EPA regulations allow a lifetime risk factor of 10^{-4} to 10^{-6}, and the EPA's NESHAP allows an exposure of 10 mrem/yr, which corresponds to an individual risk of $3.3x10^{-4}$. The NCRP also recommends the exclusion of any exposure of 1 mrem/yr ($3.3x10^{-5}$ individual risk) or less from the assessments (Ref. 68).

The $3x10^{-4}$ mrem/yr radiation exposure from the repository would be 3 to 4 orders of magnitude lower than the level for below regulatory concern (1 mrem/yr). This level of exposure is equivalent to an additional exposure to cosmic rays caused by reduced shielding when one wears a pair of shoes with heels of an inch higher than normal. Evidence does not suggest a higher rate of cancers at higher altitudes, even at several thousand feet higher than sea

level. Women are not reluctant to wear high heels because of higher exposures to radiation, nor are people reluctant to live in the "Mile High City" of Denver, Colorado. The public should readily accept this level of imaginary risk.

Some people may be concerned over the possibility that most of the radionuclides in Table 1 may be excluded under this rule since, depending on the scenario, the expected exposure of the public to many radionuclides may be very small. It should be noted, however, that the low exposure from gaseous C-14 is generic; i.e., it is almost independent of scenarios. The low exposure is the result of the abundant presence of non-radioactive carbon everywhere on earth, especially in the biosphere. The number of potential health effects from the release of one curie of C-14 used in developing the EPA regulation is based on applying the inherently low dose to over 1.4 trillion people over their lifetime (70 years). No other radionuclide was applied to such a large critical population base, so a stronger case can be made for dose truncation for C-14 than for other radionuclides.

4. Geographic Truncation

Carbon-14 in the global inventory affects the total world population, which is the basis of the EPA regulation. The EPA model is valid when the release is large, such as that expected from a commercial nuclear fuel reprocessing plant; i.e., 860 Ci/yr from a 50 GWe/yr plant, if no treatment is done, as it is not required under the current regulation. When the release level is low, it would be within the natural level of variation among different regions. (The C-14 concentrations in the Pacific and the Atlantic oceans are different, and the difference is used to measure the communication between them under the North Pole.) At that low level, the potential effect would be localized. Eventually, the C-14 would become a part of the global inventory, but its residence time in the ocean is so long that its global impact on other regions of the world would for all practical purposes be nil. The health effect should therefore be calculated based on regional population, such as that of the U.S. or North America.

This logic is not meant to ignore the health impact outside the region. Rather, it is based on the premise that at an extremely low level of release, at a "noise" level, the actual impact would be limited to the regional population. It should not be confused with dose truncation, since the population dose, no matter how small, would still be calculated based on the regional population. This would have the same effect as relaxing the release limit for C-14 (but not for other radionuclides), by an order of magnitude.

5. Change to an Individual Dose Basis

This was strongly advocated by the Waste Isolation Systems Panel (WISP) of the National Research Council (Ref. 69). The current EPA philosophy is based on protecting both the population and the individual, not one or the other. Although most European countries have adopted individual dose as the basis for regulation, it was done for reasons more applicable to them, such as a high population density in the region, which makes for less difference between population and individual protection.

If a standard based on individual dose is adopted, gaseous C-14 will no longer be of concern. If, however, C-14 is released in liquid form through fracture flows, such a standard would penalize sites with no means of diluting the radionuclides, as was shown in the WISP report (Ref. 69).

It should also be noted that the geologic repository system relies on favorable geologic conditions, which may include a lack or slow movement of the media that would carry the radionuclides and a significant retardation of movement of radionuclides by sorption and precipitation. Since for most sites (except probably those in the salt media) there are groundwater flows that could carry the radionuclides, the retardation by sorption would play an important role in limiting the release. The sorption process, however, concentrates the radionuclides in the media by a similar process to that used in chromatographic separation and concentration. The irony is that the better the site is, the longer it delays the release, but the more it concentrates the radionuclides and the higher the dose to the maximally exposed individual becomes when the concentrated peak finally reaches the accessible environment unless the retardation is so large that the radionuclides decay by a significant amount. For most sites the peak dose would appear after the 10,000-year regulatory time frame, and for some sites the peak dose may not appear for over 100,000 years. Concerns about the delayed appearance of the peak dose have been expressed (Ref. 69). Since the only alternatives to a high peak dose, aside from a perfect site with no carrier media, are no retardation (earlier release) and dilution (more population exposure), the truncation of the regulatory timeframe before the appearance of a delayed peak dose would be a justifiable and better alternative.

Because this is an alternative with far more impact on all other radionuclides than on C-14, its consideration is outside the scope of this paper.

6. Apply Clean Air Act

Neither the EPA's 40 CFR Part 191 nor the NRC's 10 CFR Part 60 were intended to regulate radioactive gases released from the repository after closure. When the initial analysis was done for the EPA standards, gaseous releases were not considered credible by the NRC nor the DOE (Ref. 70). It now appears that only the Clean Air Act (CAA) provides a general framework for the regulation of gaseous release of radionuclides from the repository after closure. In 1979, the EPA listed radionuclides as hazardous air pollutants under Section 112 of the CAA (Ref. 71). As a result, the EPA was required by Section 112(b)(1)(B) of the CAA to establish the National Emission Standards for Hazardous Air Pollutants (NESHAP). Following their earlier attempts not to regulate NRC-licensed facilities (including the high level radioactive waste facilities), the EPA in 1991 published Subpart I of the NESHAP for radionuclide emissions from facilities licensed by the NRC, but exempted facilities regulated under 40 CFR Part 191, which include the high level radioactive waste repository (Ref. 72). The EPA estimated the individual risk from the HLW disposal facilities to be very small, 7×10^{-8}, much less than the 1×10^{-4} benchmark, and determined no NESHAP was needed (Ref. 72). In this determination, however, the EPA did not consider the gaseous release after permanent closure of the repository (Ref. 73). In essence, the NESHAP never addressed the gaseous release of radionuclides from the repository

after closure. Regulatory implications of this omission of post-closure gaseous release of radionuclides is discussed below in conjunction with the 1990 amendments to the CAA. It should be noted that the CAA has not exempted the gaseous releases from the HLW repository from the CAA requirements. It provided the EPA two options: (1) promulgate emission standards (NESHAP) for the HLW repository, or (2) exempt it from the NESHAP by rulemaking after consultation with the NRC, provided the program established by the NRC provides ample margin of safety. Since the CAA does not delegate the regulation of gaseous release of radionuclides to 40 CFR Part 191, any regulation of gaseous release from the repository added to 40 CFR Part 191 would have to be made consistent with the CAA. This is in keeping with the court ruling that remanded the 40 CFR Part 191, Subpart B because of the inconsistency of the groundwater protection requirement with the Safe Drinking Water Act.

Section 112(d)(9) of the CAA, addressing the emission standards for NESHAP, states:

> "No standard for radionuclide emissions from any category or subcategory of facilities licensed by the Nuclear Regulatory Commission (or an Agreement State) is required to be promulgated under this section if the Administrator determines, by rule, and after consultation with the NRC, that the regulatory program established by the NRC pursuant to the Atomic Energy Act for such category or subcategory provides an ample margin of safety to protect the public health." (Ref. 74)

Since the EPA (Administrator) has not determined by rule that the regulatory program established by the NRC provides an ample margin of safety to protect the public health, and since the NRC regulation 10 CFR Part 60 did not consider gaseous release of radionuclides in the analysis during promulgation, the CAA still requires the gaseous release to be regulated under the NESHAP until the Administrator makes the determination mentioned above in regard to the regulatory program established by the NRC. In fact, Section 112(f)(2)(B) further states:

> "Nothing in subparagraph (A) or in any other provision in this section shall be construed as affecting, or applying to the Administrator's interpretation of this section, as in effect before the date of enactment of the Clean Air Act Amendments of 1990 and set forth in the Federal Register of September 14, 1989 (54 Federal Register 38044)."

The (EPA) Administrator's interpretation of the gaseous release of radionuclides has been reflected in 40 CFR Part 61 (NESHAP), including the background analyses and records of promulgation. Within this regulatory framework, the EPA has a few options to regulate gaseous release of radionuclides under the CAA.

a. Repromulgate the NESHAP to include the HLW repository. Since the current NESHAP, Subpart I, exempted the HLW repository with no consideration of gaseous release of radionuclides after closure of

the repository, it did not fully implement the mandate of the CAA. Under this choice, the EPA would promulgate an emission standard in the NESHAP, Subpart I, that would apply to the repository after closure and the standard would be consistent with the standards in other subparts of the NESHAP.

b. Regulate repository gases under the current NESHAP. However, since the current NESHAP, Subpart I, exempts the facilities regulated by 40 CFR Part 191, and delegates the responsibility to 40 CFR Part 191, the EPA would have to add a new performance standard to 40 CFR Part 191 that would apply to gaseous release of radionuclides. This new performance standard for gaseous nuclides could be any of the alternatives already discussed or the standard in (6)a. above.

c. Consult with the NRC and amend the NRC regulation 10 CFR Part 60 to include performance standards for gaseous release of radionuclides for the post-closure period. Then no NESHAP would be required. The NRC could also consider the alternatives already discussed.

Under the first option, the standard would be consistent with those in other subparts of NESHAP. In establishing the policy for setting NESHAP, the EPA determined that emissions resulting in a lifetime maximum individual risk (MIR) no greater than approximately $1x10^{-4}$ are presumptively acceptable (Ref. 72). The subparts of NESHAP involving radionuclide emissions are all based on an MIR equal to or greater than $1x10^{-4}$. Subparts B, H, and I limit the emissions to a level that would cause 10 mrem/yr effective dose equivalent (ede) exposure, which is equivalent to an MIR of $3.3x10^{-4}$; Subpart K limits the release of Po-210 from elemental phosphorus plants to 2 Ci/yr, which is also equivalent to an MIR of $3.3x10^{-4}$; and the Subparts Q, R, T, and W limit the release of Rn-222 to 20 pCi/m^2-sec, which is equivalent to an MIR of $1x10^{-3}$ (Ref. 72). Therefore, a consistent standard for gaseous release of radionuclides from the repository could be set in the NESHAP at 10 mrem/yr (MIR=$3.3x10^{-4}$) or 3 mrem/yr (MIR=$1x10^{-4}$). It should be noted that 3 mrem/yr is based on the EPA's own dose conversion factors (Ref. 72). If we use the dose conversion factor of 200 cancers for $1x10^{+6}$ man-rem, discussed in section II-D, then the 10 mrem/yr exposure would correspond to an MIR of $1.4x10^{-4}$ and a MIR of $1x10^{-4}$ would represent about 7 mrem/yr. The discrepancy between the two numbers representing different dose conversion factors, can be resolved by averaging the two numbers -- namely use 5 mrem/yr for a MIR of $1x10^{-4}$.

No additional explanation is necessary for the second and third options, except to say that the same degree of individual protection would be incorporated in 40 CFR Part 191 under the second option.

If the EPA does not defer to NRC regulations and exempt the HLW repository from the NESHAP regulation per Section 112(d)(9), the EPA may be subject to Section 112(f) requirements. Although there is no advantage to any party involved, it would be detrimental for the DOE to proceed with no clear regulatory criteria for gaseous releases. If the EPA decides to use Section 112(f), it may be forced to comply with the Section 112(f) by default if they do not take any of the actions discussed above; i.e., the three options. It is interesting to note that Section 112(f) indirectly provides a minimum MIR cutoff level at $1x10^{-6}$ for lifetime, above which the EPA is mandated to

promulgate standards if the pollutants are classified as known, probable, or possible carcinogens. This risk level corresponds to an annual exposure of 0.03 mrem, using the EPA's own dose conversion factors. If we use the dose conversion factor discussed in Section II.D, the same risk level would correspond to an annual exposure of 0.07 mrem, or approximately 0.1 mrem, since these are not exact numbers.

The $1x10^{-6}$ risk cutoff is consistent with other regulatory precedents. Analyses of regulatory decisions based on risk showed that every chemical that presents an individual risk of $4x10^{-3}$ was regulated (Ref. 75). Except for one case, no action was taken to reduce the risk below $1x10^{-6}$. Similar cutoffs for lifetime risk for individuals, typically $1x10^{-6}$ for large populations like that of the U.S. and $1.5x10^{-3}$ for smaller populations, were noted by others (Ref. 76). [Note: The information in this paragraph was provided by Robert Wilems, RAE.]

7. Hold the Regulation of Gaseous Release of Radionuclides in Reserve

As discussed in the previous section, the EPA will have to comply with the requirements in the CAA either through the NESHAP or by exempting the HLW repository from the NESHAP process by complying with the requirements in Section 112(d)(9). In either case, the EPA has the option of not making any decisions or taking any actions immediately. This would temporarily relieve the EPA from the gaseous C-14 problem without affecting the court-mandated repromulgation of 40 CFR Part 191, Subpart B. This alternative could also be treated as a fourth option under the CAA, which was discussed in the previous section. It has been separated because it does not provide any solution, but avoids the problem by deferring any action on it.

This alternative, however, should be considered as a last resort. It is clear that the implementation of the current (court-vacated) regulation to gaseous radionuclides is impractical, although not impossible, as was discussed earlier. To have the EPA state that the current Table 1, Subpart B does not apply to gaseous radionuclides and that regulations governing their release will be held in reserve would provide the EPA grounds for future actions. While not providing the DOE any advantage over the current regulation, and the uncertainties about future regulation would be so great that the DOE would be forced to assume the worst case scenario, resulting in unnecessary expenditures and schedule delays.

IV. DISCUSSION AND RECOMMENDATIONS

A. Regulation of Gaseous Release of Radionuclides

The regulation of gaseous release of radionuclides certainly falls under the CAA, and it leaves the EPA with only two choices: Alternatives 6 and 7 in the previous section. Alternatives 1 through 5 are possible options only through Alternatives 6a through 6c.

Among these possible alternatives, the most logical choice would be 6c, which has its basis in the 1990 amendments to the CAA. It would provide the EPA and NRC the highest flexibility, although it does not provide them any technical basis to develop quantitative criteria unless they borrow the same basis used in Alternatives 6a and 6b. Both 6a and 6b employ the NESHAP as a

vehicle to regulate the gaseous release of radionuclides, the difference being that 6b takes the EPA out of using NESHAP through the existing interpretation of NESHAP, Subpart I, which is allowed in the CAA. In terms of quantitative criteria, both 6a and 6b would have to rely on the same type of risk assessment used in the NESHAP as discussed in 6a. Alternative 6b would have Alternatives 1 through 5 available to the EPA. For this reason, it is strongly recommended Alternative 6b be adopted.

Under 6b, the EPA has six options altogether, namely Alternatives 1 through 5 and adoption of the same numerical values used in 6a, since both 6a and 6b employ the NESHAP process. Adoption of the same risk criteria as NESHAP (Alternative 6a) through the 6b process would be my first recommendation, followed by the Alternatives 4, 3, 2, and 1a, in that order.

The preferred option can be stated as follows:

> Per the 1990 Amendments to the CAA, the EPA determines to uphold the current NESHAP (40 CFR Part 61, Subpart I) and regulate the gaseous release of radionuclides by adding a new standard to 40 CFR Part 191, which would apply to the gaseous releases only. The new standard shall be consistent with the requirements in the CAA and the risk assessment methodology used in other subparts of the NESHAP; i.e., the release of gaseous radionuclides shall not exceed those amounts that would cause any member of the public to receive an effective dose equivalent to 5 mrem/yr, except that any combined release that would cause no greater than 0.1 mrem/yr effective dose equivalent need not be regulated. In addition, since the CAA/NESHAP already insures public health with an ample margin of safety, the release of gaseous radionuclides need not be included in the probabilistic calculation of releases required in 40 CFR 191.13.

B. Exempt C-14 Release from Regulation

As mentioned earlier, C-14 has unique characteristics. As long as there are sources of neutrons in the presence of nitrogen, the production of C-14, whether in a reactor or in the atmosphere, will continue. Once it is produced it can only decay away, but never disappears. Therefore, the best management of C-14 from a public health point of view would be the one that would minimize the exposure of the public (decay in isolation) and slow the release to reduce the individual dose to a noise level, at which there is no evidence of discernible health effect. The geologic repository provides such a solution.

As the use of nuclear energy increases, the generation of C-14 will also increase, even with the efforts to minimize the C-14 production per unit energy produced. In addition, there are other technical reasons why the production of C-14 per unit energy produced may even increase substantially in order to gain other benefits (Ref. 13). In one estimate, the annual C-14 release to the atmosphere from envisaged global nuclear power production could even approach the same level as the natural production of C-14 in the atmosphere (28,000 Ci/yr), twice as much accumulating in solid wastes. At present, the release of C-14 from nuclear power plants and fuel reprocessing

plants is not regulated anywhere in the world. Even if some control measure is adopted to capture it in solid waste forms, the resulting waste forms do not provide the same degree of isolation as the spent fuel emplaced in the geologic repository. It should be noted that the release would be significant in terms of curie amount but, not in terms of health effect.

Restricting a repository's release of C-14 to less than 1 Ci/yr, which is less than the annual release from a single operating reactor, is almost meaningless compared to the global release of C-14 into the atmosphere. This is a global problem, if it is a problem, and requires a global solution. Spending billions of dollars to keep the repository release below 1 curie per year while others are pouring thousands of curies into the atmosphere simply does not make any sense. It would be prudent for the EPA to exempt the gaseous release of C-14 from 40 CFR Part 191.

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If there are errors or omissions, they are all mine.

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