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COMPATIBILITY OF TECHNOLOGIES WITH REGULATIONS
IN THE WASTE MANAGEMENT OF
H-3, I-129, C-14, AND Kr-85.
PART II. ANALYSIS

by

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NOMENCLATURE

AFR - Away from Reactor (spent fuel storage)

ALARA - As Low As Reasonably Achievable

ANL - Argonne National Laboratory

BACT - Best Available Control Technology

BWR - Boiling Water Reactor

CFR - Code of Federal Regulations

CHW - Cladding Hulls Waste

DCF - Dose Conversion Factor

DOE - Department of Energy

DOG - Dissolver Off-Gas

DOT - Department of Transportation

EPA - Environmental Protection Agency

FRP - Fuel Reprocessing Plant

GCD - Greater Confinement Disposal

HIP - Hot Isostatic Pressing

HLW - High-Level Waste

HEPA - High-Efficiency Particulate Air (filter)

IAEA - International Atomic Energy Agency

ICPP - Idaho Chemical Processing Plant

ICRP - International Commission on Radiological Protection

LLW - Low-Level Waste

LODC - London Ocean Dumping Convention

LWR - Light Water Reactor

MTHM - Metric Tons of Heavy Metal

NCRP - National Council on Radiation Protection

NEA - Nuclear Energy Agency

NOMENCLATURE (contd)

ar Regulatory Commission
ization for Economic Cooperation and Development
idge National Laboratory
ic Northwest Laboratories
urized Water Reactor
rch and Development
active Materials Packaging (data base)
ow-Land Burial
uranic (waste)

GLOSSARY

COLLECTION	The process of separating and pooling a waste radioactive isotope from the other components of the plant stream in a fuel reprocessing plant.
CANISTER	A vessel, usually cylindrical, that provides containment for radioactive waste. If additional attenuation of penetrating radiation is needed, the canister is surrounded and carried in an outer vessel called a shield or cask.
CONTAINMENT	The retention of radioactive material to prevent it from being released to the environment or to ensure that it is released at an acceptable rate only.
DISPERSION	The total effect of the processes (transport, diffusion, or mixing) which tend to distribute radioactive wastes into the environment.
DISPOSAL	Emplacement of radioactive wastes in some permanent, acceptable location, with no intention of retrieval.
EMPLACEMENT	Placing a radioactive waste in its final location.
ENCAPSULANT	An inert or nonradioactive phase which surrounds or incorporates a radioactive phase, giving it a monolithic form.
FINAL WASTE FORM	The form, with characteristic physical and chemical properties, into which the primary waste forms are converted and made acceptable for final emplacement or disposal.
GREATER CONFINEMENT DISPOSAL	Disposal of waste using natural and/or engineered barriers which provide a degree of isolation greater than that of shallow-land burial but less than that of a geologic repository.
G VALUE	A parameter expressing the extent of radiolysis of a material. The G value indicates the number of molecules of radiolytic product per 100 eV of energy absorbed by the material.
HIP	Hot isostatic pressing, a process for applying pressure uniformly on all surfaces of a sample at elevated temperature.
INTERIM STORAGE	Temporary storage of a radioactive waste awaiting further processing, transportation, or release after partial decay to the environment.
IMMOBILIZATION	Any process for converting radioactive wastes into a form that is less likely to become dispersed to the environment.

ISOLATION	Segregation of the radioactive wastes from the human environment and the restriction of their release into that environment in unacceptable quantities or concentrations.
LOW-LEVEL WASTE	Radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or by-product material as defined in section 11 e.(2) of the Atomic Energy Act (uranium or thorium tailings and waste).
MATRIX	An inert, nonradioactive material, synonymous with an encapsulant, used to incorporate a radioactive phase into a monolithic form, reducing its dispersibility.
MODEL	A representation of the chemical, physical, and biological behavior of radioactive waste in the natural environment.
NON-NEAR SURFACE DISPOSAL	Burial similar to SLB but at greater depths.
NORMAL FORM	A term, used in transportation regulations, applied to material in any form that does not meet the tests defined in transportation regulations to qualify as "special form" material. Materials that qualify as special form are capable of resisting dispersion if exposed to the environment.
OCEAN DISPOSAL	The process in which packages are allowed to settle onto the ocean floor. Drilling into the seabed is not involved, in contrast to concepts involving sub-seabed disposal.
ONSITE STORAGE	Storage of radioactive wastes on the site of the fuel reprocessing plant.
OVERPACK	An additional jacket for a radioactive waste package or packages, sometimes used only for transportation to protect waste packages from thermal or physical shocks.
PACKAGE	The sum of containers and contents.
PACKAGING	The containers only.
PLANT STREAM	A material, often a mixture of several components, on which the operations of the plant are performed.
PRIMARY CONTAINER	The container that is in contact with the radioactive waste.
PRIMARY WASTE FORM	The form in which radioactive wastes first appear after collection from a plant stream.
RADIOPHASE	The phase that contains the radioactive nuclides associated with a waste.

REPOSITORY	A facility that permits the isolation of radioactive wastes in a stable geologic formation at a depth sufficient to isolate them from biological species and phenomena.
SHALLOW-LAND BURIAL	Disposal of radioactive waste in the upper 50 ft of the earth's surface, with or without engineered barriers, with a covering of a few feet of soil.
SHIELD	A material used to attenuate penetrating radiation.
WASTE-FORM CONVERSION	The process of transforming primary waste forms to a form acceptable for final disposal.
WASTE MANAGEMENT	The entire set of operations applied to radioactive wastes, including collection, form conversion, immobilization, packaging, transportation, and disposal.

COMPATIBILITY OF TECHNOLOGIES FOR WASTE MANAGEMENT OF
H-3, I-129, C-14, AND Kr-85 WITH REGULATIONS
PART II. ANALYSIS

by

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ABSTRACT

Waste forms of H-3, I-129, C-14, and Kr-85 separated from fuel reprocessing streams and procedures for managing them were analyzed regarding compliance with regulations.

Transportation of these wastes in certain DOT-specification packagings would be permissible, but some of these packagings may not be acceptable in some disposal situations. Transportation of gaseous Kr-85 in a currently certified cylinder is possible, but a fuel reprocessor may wish to ship larger quantities per package. Disposal of tritium using a package designed by a DOE contractor and shallow land burial, in accord with the regulations of 10 CFR 61, seems practicable. Although 10 CFR 61 permits shallow land burial of I-129, the concentration limit requires distribution in a volume that may seem impractical to commercial fuel reprocessors. The concentration limit of 10 CFR 61 for shallow land burial of C-14 requires distribution in a lesser, although still large, volume. For both I-129 and C-14, management as high-level waste offers the advantage of smaller volumes. Similar advantages may be offered by greater confinement or non-near surface concepts for disposal. The concrete waste forms developed for these nuclides may not meet technical criteria being formulated for geologic disposal.

Although Kr-85 has had much attention in development of collection technology, this nuclide has not been provided for by disposal criteria. The lack of accommodation of Kr-85 at disposal facilities makes storage of the gaseous form at the fuel reprocessing plant, followed by dispersal after partial decay, seem attractive.

Ocean disposal of I-129 and C-14 by the rules of the International Atomic Energy Agency-London Ocean Dumping Convention offers advantages over shallow land burial: higher allowed concentrations, resulting in smaller volumes and fewer packages. These rules, however, thwart ocean disposal of Kr-85 since gaseous forms are banned, and for solid forms, concentration limits would require distribution of radioactivity in very large volumes.

Suggestions are made for technological development and for amendments to regulations to make disposal of these nuclides possible.

EXECUTIVE SUMMARY

The objective of this study was to determine if H-3, I-129, C-14, and Kr-85 wastes could be packaged, transported, and accepted for final emplacement in compliance with regulations. For each waste type, transportation and acceptance for shallow-land burial were analyzed first. The SLE* disposal mode was given priority because it is most capable of near-term application and perhaps lowest cost. For waste forms judged unacceptable for shallow-land burial, the possibility of disposal by other modes was examined. These other modes included interim storage on the site of an FRP, non-near surface and greater confinement disposal, deep geologic disposal, and ocean disposal. It was desired to identify at least one complete path for handling each waste form from collection to final disposal. For wastes where disposal could not be completed with present-day technology, the means to complete it, either by technological developments or amendments to regulations, are suggested.

In preparation for this analysis, reviews were made of (1) collection processes and their efficiencies, (2) preparation of final waste forms, (3) properties of the waste forms proposed for these nuclides, (4) regulations for release to the environment, (5) regulations on waste form properties and packaging for transportation and for final disposal, and (6) authorized packaging, including DOT-specification and NRC-certified designs.

The efficiencies of collection technologies promise to be more than adequate for the requirements of regulations concerning the release of H-3, C-14, and Kr-85. In contrast, the collection technology for I-129 may be hard pressed to provide compliance. This view follows from (1) a consideration of 40 CFR 190, which limits fuel cycle releases to about 0.5% of the amount in spent fuel, and (2) from uncertainties about the distribution of I-129 in a fuel reprocessing plant and the efficiency of collection processes in full-scale application.

Prescriptions for waste form properties in transportation regulations would not rule out transportation of any of the waste forms considered. The regulations determine what packaging is acceptable for materials having the properties expected for these wastes. Several packagings specified by the DOT could be used for transportation of the waste forms of H-3, I-129, and C-14. Department of Transportation (DOT) specification packagings were identified for transportation of Kr-85 in the form of zeolites and sputtered metal, but DOT-specification packagings would not be practicable for Kr-85 in the gaseous form. Of two containers certified by the NRC for transportation of Kr-85, only one may have adequate capacity to accommodate the quantities of Kr-85 resulting from fuel reprocessing. No packagings have a current NRC certificate applicable to the transportation of H-3, I-129, or C-14 resulting from full-scale fuel reprocessing.

*Acronyms are defined in the nomenclature list at beginning of report. A glossary for some of the terms used in this report is also included.

A complete disposal sequence includes collection, packaging, transportation, and emplacement at a final disposal site. It would, therefore, be convenient to package the waste form in a container that meets both transportation regulations and disposal regulations. Perhaps the cement waste forms would be judged capable, in licensing proceedings, of the 300-y stability required for either wastes or containers of classes B and C by 10 CFR 61. The unconsolidated lead zeolite waste form for I-129 does not seem to meet this requirement. The container must provide stability if the waste form cannot. Although the technical position papers of NRC broadly describe the characteristics of waste forms and containers, it is not entirely clear which of the currently available container designs, including DOT-specification containers, would meet the 300-y lifetime requirement; this is a question that should be more definitely resolved. One exception to this conclusion would be a waste form with sufficiently low radionuclide concentration to qualify under 10 CFR 61 as a class A waste for SLB. Dilution of fuel reprocessing wastes to make them fit into class A is not expected to be permitted. The intention of NRC seems to have been that 10 CFR 61 and the classification scheme, as well as the technical requirements of its subpart D, were developed for commercial radioactive wastes on the assumption that fuel-reprocessing wastes would not be a part of them. On the other hand, the intentions of NRC seem to be that use of the general performance objectives in subpart C could be applied to any waste-production scenario. Site inventory limits may be determined, according to 10 CFR 61, on a site-specific basis. For class A, the container need not have the 300-y lifetime required of classes B and C. However, the low concentration limit for class A would distribute the radionuclide in an impractically large volume of matrix material and number of containers. It is not evident that all DOT specification packagings would provide, for cases where the waste form cannot, the 300-y period of integrity required by 10 CFR 61 for SLB disposal of waste as classes B or C. Although some existing packaging designs can be identified for transportation in compliance with 49 CFR 171-178 for each of the waste forms considered, and although general guidelines for high integrity containers are given in the NRC technical position papers, which ones would also meet the 300-y lifetime requirement can probably only be determined by licensing proceedings. For other final disposal modes, specifications of acceptable packaging can be made only after criteria and regulations are put into final form by EPA and NRC.

Conclusions regarding disposal of these wastes at an SLB facility are as follows. Disposal by SLB of the waste forms for H-3, I-129, and C-14 could be done in compliance with 10 CFR 61, assuming waste forms or containers meet the 300-y lifetime requirement. The concentration limit required for I-129, however, is so low that the resulting volume of matrix material and number of drums may entail relatively high costs. For C-14, the volume of matrix material and number of containers that would result from meeting the concentration requirements of 10 CFR 61 might not be uneconomically large. Both I-129 and C-14 might be managed in more concentrated waste forms, and thus smaller volumes and numbers of containers, by HLW disposal. Choosing between SLB and HLW disposal for C-14, where high concentrations are allowed, would require an economic analysis.

A recent draft of 40 CFR 191 suggests HLW management of C-14 wastes and any other isotopes with a half-life >20 y (which would include I-129) in concentrations exceeding prescribed limits. But this rule would apply only

to spent fuel or to wastes from the first or subsequent extraction cycles of an FRP. In an FRP using the Purex flowsheet, significant amounts of C-14 and I-129 would not enter the extraction cycles; thus these rules seem to be paradoxical. Although the area of interest here is U.S. regulations, geologic emplacement of C-14 and I-129 in high concentrations seems to be promoted by the IAEA in its suggestion of deep-geologic disposal of all "intermediate-level, long-lived wastes." Furthermore, deep-geologic disposal is suggested for C-14 in a recent British review and for both I-129 and C-14 in a recent American review.

The I-129 waste forms developed in R&D efforts have volume concentrations of $15\text{--}26\text{ Ci/m}^3$, which are too high for SLB disposal by factors of $2\text{--}3 \times 10^2$. Management by HLW disposal of I-129 might be a more economic mode of disposal for I-129 than SLB, and the same conclusion may be true for C-14, but the latter is even less obvious. Concepts for the deep-geologic disposal of these two nuclides cannot be completely specified, however, because the waste forms that have been developed for them, ordinary concrete matrices, would probably not meet the fractional leach rate limit of $10^{-5}/\text{y}$ that would be required of materials emplaced in geologic storage if criteria and regulations now evolving are promulgated. The volumes of Kr-85 resulting from fuel reprocessing cannot be accommodated by SLB under 10 CFR 61. The question of disposal of this waste by SLB was postponed by NRC.

Possible advantages offered by intermediate-depth disposal methods (such as the non-near surface concept of NRC and the greater confinement disposal concept of DOE) were briefly examined for airborne waste nuclides. Perhaps the only advantage that GCD offers for disposal of H-3 is the possibility of a higher allowed inventory at a given disposal site. Greater confinement disposal should offer the advantage over SLB of disposal of I-129 and C-14 at higher concentrations and, therefore, lower volumes of matrix. Criteria for the disposal of Kr-85 by GCD have not been proposed. The initial suggestions for GCD criteria are now being revised on the basis of improved models of isotope behavior, and recommendations for concentration limits may be drastically changed. Also the concepts of non-near surface disposal expected from NRC are still being defined. Thus, the assessment of disposal of these wastes at intermediate depths was only briefly treated.

The possibility of ocean disposal was considered. Assessments were made using the rules developed by the IAEA in response to the London Ocean Dumping convention. The U.S. is committed to following these rules and also the policies set by the EPA. However, EPA has denied permits for ocean disposal, and a congressional moratorium has been placed on it. The concentration limits of the IAEA rules make disposal of the quantities of Kr-85 resulting from fuel reprocessing impracticable. For H-3, this mode of disposal probably offers no advantage over SLB and may be more costly. For C-14 and I-129, however, ocean disposal would offer an advantage over SLB in permitting much higher concentrations and, thus, lower volumes and numbers of containers. Advocates of ocean disposal have cited the advantage of dilution of radioactive nuclides by the stable nuclides of carbon and iodine that occur naturally in the ocean.

On-site storage at the FRP seems promising for Kr-85, since its disposal is not readily accommodated under the criteria of any of the common disposal modes. The prescription in 10 CFR 50 (Appendix F) is that radioactivity be removed from the FRP site by the time of decommissioning; this regulation can

probably not be satisfied by storing Kr-85 on-site in gaseous form and then releasing it after partial decay if the release limit is 50,000 Ci/GW_e-y and decommissioning occurs only 40 y after startup. On-site storage of H-3, as tritiated water to be released after partial decay, also appears attractive because of the short half-life of tritium and the relatively low cost of storage.

From this study, some needs for technological development are seen.

(1) The ability to collect these four nuclides would be more plausible if demonstrations of an integrated system for collection of all four nuclides were carried out. (2) Development of conceptual designs of full-scale collection systems may point out some of the difficulties that cannot be foreseen from small-scale demonstrations. (3) The leach rates of the solid waste forms containing Kr-85 should be characterized more precisely. (4) The ability to collect I-129 from plant streams with the efficiency needed to satisfy the very restrictive demands of 40 CFR 190 should be thoroughly demonstrated. Distribution of I-129 in the streams of an FRP should be clarified; perhaps this information cannot be obtained except on an operating FRP. (5) If I-129 and C-14 are to be disposed of by geologic emplacement, then waste forms other than the ordinary concrete matrices already developed may be needed at least to meet leach-rate requirements. (6) Verification that containers for disposal of wastes in SLB can meet the guidelines set forth in NRC technical position papers for high-integrity containers is needed. (7) If C-14 and I-129 were to be disposed of by HLW disposal, some work may be necessary to develop a suitable waste form and container; it is assumed that these nuclides would not be placed in the same type of package being developed for HLW, *i.e.*, vitrified mixed fission products in a steel canister. (8) Modeling studies are needed to predict dose rates from releases so that compliance with regulations expressed in terms other than the maximum individual dose (e.g., in 40 CFR 190) can be examined. (9) If C-14 and I-129 are managed by HLW disposal, experimental measurements of waste-form properties will be needed to obtain information that may be required by HLW acceptance criteria. (10) If I-129 is managed as HLW and waste forms of lower leachability than that of concreted barium iodate are required to meet the technical criteria of 10 CFR 60 and 40 CFR 191, then the waste form that involves AgI or I-sodalite in concrete should be considered. Since the iodine diffusivity at room temperature is two to three orders of magnitude lower in these waste forms than in concreted barium iodate, the leachability of the former waste form may be sufficiently low to satisfy the regulations for deep geologic disposal.

This study has also revealed some needs for amendments to regulations if separated airborne wastes are to be accommodated. (1) Although much development work has been devoted to collection of Kr-85 from fuel reprocessing, none of the criteria developed for acceptance of waste forms at disposal facilities seems to accommodate any of the Kr-85 waste forms. (2) Regulations that would govern storage of wastes on the FRP site, *e.g.*, Kr-85 and H-3, need to be formulated. (3) Criteria for GCD and non-near surface disposal are needed for all four nuclides. (4) Additional details on container qualifications for SLB, GCD, geologic disposal, and ocean disposal would be helpful. (5) Additional interpretation of the demands that ALARA and BACT will make on the management of these wastes is needed. (6) Clarification of future policies on ocean disposal is needed to determine if it is a

viable U.S. option for the disposal of I-129 and C-14. (7) If C-14 and I-129 are to be managed by HLW disposal, clarification of the acceptable waste-form properties is needed. (8) Parts 190 and 191 (Draft) to Title 40 of the Code of Federal Regulations seem inconsistent in their limits for control of the two long-lived airborne isotopes, C-14 and I-129. Part 190 treats I-129 more conservatively, setting a very restrictive release limit on I-129 but none on C-14. In contrast, Part 191, defining HLW by a source criterion (spent fuel or wastes of the first and subsequent cycles from fuel reprocessing) and a concentration criterion, treats C-14 more conservatively. It proposes to define wastes containing C-14 as HLW if the concentration exceeds the relatively small value of 8×10^{-6} Ci/g, but to define wastes containing isotopes with half-lives greater than 20 y (a category that includes I-129) as HLW only if the concentration exceeds a much greater value, 1×10^{-3} Ci/g.

I. INTRODUCTION

The perceived impact of H-3, I-129, C-14, and Kr-85 as wastes has caused sufficient concern to foster substantial efforts in the U.S., Britain, and the European community to develop technologies for their control. Several reviews of these waste management technologies, their operations, and their products have been published. The needs and technologies for management of these wastes were thoroughly reviewed and analyzed in two recent documents [BROWN-1983A, -1983B]. The subject of compliance of the technologies with regulations, however, has not been considered in detail heretofore. Such an analysis is the main objective of this report. A data base for this analysis has been assembled in another document (ANL-83-57:Part I), and the reader is referred to this data base regarding the sources of information used in this report. Furthermore, although substantial efforts have been invested in collection technologies, much less attention has been given to other elements of the complete waste management system needed to dispose of the wastes.

A. Objectives of this Analysis

The primary objective of this study, attained through an analysis of the capabilities of disposal technology and an analysis of the demands set by regulations, is to determine if disposal technology is capable of meeting the demands that can be inferred from the regulations. The task has required a brief review of (1) the characteristics of a selected set of disposal options imminently available and of other options still only remote concepts and (2) the regulations that may govern these options. Although future assessments may eventually conclude that some presently undefined disposal method is best, this work is aimed at the disposal of wastes by methods that could be available in the near future. One of the tasks is to decide which regulations apply to the management of these wastes. This question was resolved by considering the agencies responsible, the sections of the Code of Federal Regulations that would apply to management of these wastes, the topics included, and the specifications those regulations contain.

The topics covered in regulations considered in this work include release limits, dose limits, characteristics of packaging for transportation, and required disposal techniques. Recommendations are made on further needs in development of the technologies and amendments to regulations.

One of the primary questions of persons concerned with planning disposal of separated H-3, I-129, C-14, and Kr-85 is the best destination for final emplacement. Other established waste categories, e.g., LLW, spent fuel, HLW, and TRU, are recognized and are in the process of being assigned a final destination by DOE orders, EPA criteria, NRC regulations, and legislation. These four nuclides in the amounts, forms, and concentrations encountered in reprocessing do not obviously fit into any of the established categories. It has been assumed by some that, once these wastes are separated from fuel, they would no longer be associated with the type of materials that are to be managed by HLW disposal and, therefore, would be destined for the only other disposal mode currently practiced, i.e., SLB. The analysis reported here shows that for SLB some of these isotopes can be accommodated easily, some with difficulty, and one of them not at all. At any rate, one of the topics considered in this work, insofar as possible, is where packaged airborne nuclides would be accepted.

Discussion of the collection and disposal of the four nuclides should be taken neither as a recommendation that they be collected nor as a prediction that their collection will be required. This report is solely an analysis of compliance of waste management technologies with regulations.

B. Boundaries of this Analysis

At the onset, it is helpful to enumerate the topics that are not included, even though they may have been considered in other studies involving similar types of wastes. First, several other airborne nuclides are released to the atmosphere from fuel cycle facilities, but the impacts of these nuclides are considered to be sufficiently inconsequential, mainly because of short half-life, that collection and packaging are not required. Radon-222 is also an airborne nuclide that arises in some parts of the nuclear fuel cycle, but it has not been proposed to collect and package radon, which is therefore not considered here. Particulate solids, bearing a number of radioactive nuclides, including actinide elements and semi-volatiles (i.e., cesium and strontium), would also be collected from FRP gaseous streams. The management of particulate solids, however, is not considered here. These solids will be collected on High Efficiency Particulate (HEPA) filters whose development, efficiency, and deployment is a separate field. The HEPA filters, bearing the actinides and semi-volatile fission products from an FRP, will be classified as TRU wastes, for which DOE supports a separate program and for which a separate disposal destination will probably be assigned. In this work, the effects on compliance of contamination of any one of the separated nuclides with other fission products have not been addressed. The contamination of separated airborne wastes apparently has not been considered to be an important problem by waste-form developers. An accurate picture of the level of contamination of waste forms of these nuclides will not become available until significant amounts of data are obtained from full-scale fuel reprocessing in which the airborne wastes are collected. Furthermore, this study is not an exercise for selecting the best among several technological alternatives that have been suggested for each of the nuclides covered; also, economic evaluations of technologies are not made here.

In summary, for separated H-3, I-129, C-14, and Kr-85, this work presents the collection, the proposed waste form properties, and requirements of regulations for transportation and disposal and reports whether the technologies are capable of compliance with regulations. The work also describes how these wastes could be disposed of in the current climate of technological and regulatory development.

C. Generic Scheme for Disposal of the Airborne Wastes

Figure I-1 outlines the generic disposal process for the airborne wastes. The options for final disposal of packaged wastes were described in the preceding document (ANL-83-57:Part I). If spent fuel is not reprocessed, these four nuclides will accompany the fuel. In this case, the nuclides contained by the fuel would require no separate considerations of packaging, transportation, and emplacement. On the other hand, if spent fuel is ultimately reprocessed, these nuclides will become available almost in entirety at the FRP. All four nuclides appear in streams of gas and water vapor at the FRP. Even before their potential environmental impact had been fully assessed,

Fig. I-1. Generic Disposal Process for Airborne Wastes

much effort was devoted to development of technologies that would minimize, by collection and packaging schemes, their environmental dispersal. Much of the technology of disposal has been devoted to collection techniques.

The regulations and bases for the estimates presented in this analysis are summarized in the appendices. Regulations pertinent to airborne waste disposal are summarized in Appendix A; the properties of the nuclides, the fuel reprocessing plant parameters, and the quantities of nuclides are given in Appendix B. Categories of information for characterization of canisters (HLW) are given in Appendix C. Descriptions of currently feasible disposal systems for H-3, I-129, C-14, and Kr-85 are in Appendices D through G. The state of readiness of waste management operations is briefly summarized in Table I-1.

Table I-1. State of Readiness of Airborne Waste Disposal Operations to Meet Regulations

Disposal System	Transportation	SLB	NNS or GCD	Deep-Geologic Disposal	Ocean Disposal	Onsite Storage
H-3						
In Concrete	P	P	ND	N	N	-
Molecular Sieves in Concrete	P	P	ND	N	N	-
ZrT _x in Concrete	P	P	ND	N	N	-
Tritiated Water in Tank	-	-	-	-	-	O
Container Availability	P	P	N	N	N	-
I-129						
BaIO ₃ in Concrete	P	YNP	ND	N	P	-
AgI in Zeolite	P	P	ND	N	N	-
PbI in Zeolite	P	P	ND	N	N	-
Container Availability	P	P	ND	N	N	-
C-14						
CaCO ₃ in Concrete	P	P	ND	N	P	-
BaCO ₃ in Concrete	P	P	ND	N	N	-
Container Availability	P	P	ND	N	N	-
Kr-85						
Compressed Gas	YNP	N	N	N	N	O
Zeolite	P	N	N	N	YNP	-
Sputtered Metal	P	N	N	N	YNP	-
Container Availability	P	N	N	N	N	-

P = Can meet regulations and seems likely to be considered practicable by waste disposers.

YNP = Can meet regulations but seems unlikely to be considered practicable by waste disposers.

N = No, does not meet regulations.

ND = Not defined.

O = Option.

- = Not applicable.

II. PLAN FOR ANALYSIS

The plan for this analysis consists of defining (1) the technologies proposed for disposal of H-3, I-129, C-14, and Kr-85 and (2) the important requirements in regulations that the technologies would have to meet; a comparison is then made between the expected performance of these technologies and the regulatory requirements. The technologies are to be evaluated against not only tangible technical criteria such as specifications on packaging and waste form, but also against less-tangible goals that underlie the technical criteria. Examples of such goals are limits on radiation dose to individuals or on the number of health effects from nuclear waste management sustained in a given time period. This type of analysis requires determining how the performance of technologies can be expressed to permit a comparison with limits and goals. From the comparison of expected technological performance with regulations, those technologies that seem to be capable of putting disposal of airborne waste into practice at this time are indicated. For cases where compliance seems impossible, the reasons are explained.

A. Characteristics of Waste Forms Derived from R&D Information

Much development of waste forms for the four nuclides was done before regulations that the waste forms must adhere to were written. For example, most of the work on waste form development has been done during the past ten years, whereas applicable regulations on SLB (i.e., 10 CFR 61) were only recently promulgated, and regulations on geologic disposal are still being developed. Further, intermediate-depth concepts such as greater confinement and non-near surface disposal are still in the early stages of definition and thus their criteria are only in an elementary stage. Lacking guidelines from regulations, developers endeavored to produce waste forms with properties generally considered to be desirable, i.e., limited dispersibility, low leachability, and low volatility. Thus, development has aimed at producing stable solid waste forms. For this reason, developers attempted to maximize the concentration of radiophase in a matrix for economy and yet remain within the limits required to maintain a stable solid.

Waste forms are characterized in some sections of this report according to the recommendations contained in the literature of R&D efforts. We have attempted to derive as many characteristics as possible from the results of development programs. Subsequently, these characteristics are compared to the requirements of transportation regulations and disposal criteria. Wherever the waste form characteristics based on R&D information do not meet transportation or emplacement criteria, the waste forms are re-characterized, e.g., in lower concentrations, to meet the regulations. Finally, characterizations of currently feasible disposal systems are made.

B. Identification of Applicable Regulations

The regulations prepared by EPA, NRC, DOT, and DOE likely to govern the management of H-3, I-129, C-14, and Kr-85 are listed in Table II-1. The jurisdiction of various regulations over the operations of the disposal process is indicated in Fig. II-1. Applicable elements of these regulations

are summarized in Appendix A. For some of these regulations, it is possible to judge whether airborne wastes management technologies could comply; for others, it is difficult or impossible to make such a judgment because of incomplete information.

Table II-1. Federal Rules That May Apply to Management of H-3, I-129, C-14, and Kr-85

Agency	Source	Title
NRC	10 CFR 20	Standards for Protection Against Radiation
NRC	10 CFR 50 (Appendix F)	Policy Relating to the Siting of Fuel Re-processing Plants and Related Waste Management Facilities
NRC	10 CFR 60	Disposal of High-Level Radioactive Wastes in Geologic Repositories
NRC	10 CFR 61	Licensing Requirements for Land Disposal of Radioactive Wastes
NRC	10 CFR 71	Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions
EPA	40 CFR 61	Clean Air Act, Section 112
EPA	40 CFR 141	Drinking Water Regulations
EPA	40 CFR 190	Environmental Radiation Protection Standards
EPA	40 CFR 191 (Draft)	Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level, and Transuranic Radioactive Wastes
DOT	49 CFR 171-178	Requirements for Transportation of Radioactive Materials
DOE	Order 5480, XI	Standards and Requirements for Radiation Protection
DOE	Order 5820	Radioactive Waste Management

Fig. II-1. Applicability of Regulations to Waste-Management Operations

C. Evaluation of Disposal Prospects from the Standpoint of Transportation Regulations

Regulations on transportation of radioactive materials are contained in 49 CFR, written by DOT, and in 10 CFR 71, written by NRC. Both 49 CFR and 10 CFR 71 apply to commercial wastes.

The need for assessment of management of these wastes against DOE regulations would be encountered only in processing of defense-production fuels. Although the HLW from past processing of defense fuels is in storage waiting to be disposed of, the H-3, C-14, and Kr-85* that these fuels contained have been dispersed. Therefore, the only impact of the DOE orders would be on disposal of the airborne nuclides from future processing of defense fuels. In the case of commercial fuels, however, irradiated fuels have not been processed since 1972 and are accumulating in storage basins with their airborne wastes encapsulated in fuel pins. Thus the transportation of significant quantities of airborne wastes is most likely to be associated with wastes from a commercial fuel reprocessing plant if these stored fuels are reprocessed.

For this analysis, applicable elements are selected from 49 CFR and characteristics of airborne wastes are examined in reference to them. The transportation regulations contain few direct requirements for the characteristics of waste forms. They are primarily a classification system specifying acceptable packaging for a given cargo [ANSI]. In general, the regulations for transportation of radioactive materials do not specifically prohibit any type of waste materials. Instead, the general scheme is to anticipate problems associated with some types of materials and to avoid problems by using packaging that meets specified performance standards. A waste material would be excluded from transportation only if it were impracticable to use the packaging that the regulations require. The analysis of transportation regulations presented in the preceding report (Part I, Section IV.E.1) is used in subsequent sections of this report in two ways: (1) waste form characteristics are reviewed against the regulations to see if there are any characteristics of the waste form that are not permitted by regulations and (2) characteristics of packagings acceptable for waste forms are described. One assessment made in this second case is whether the number of packages is impracticably high.

Since a general license is granted for shipment of licensed radioactive materials in the appropriate DOT-specification packaging, the use of such packaging would be the simplest course. Therefore, in selecting packaging, this possibility was examined first for each nuclide.

If DOT-specification packaging is not practicable for the amounts of nuclides considered here, three other options are open. The first option is packaging certified by NRC or DOE especially for one of the four nuclides of interest. A search for the existence of such packaging was made with

*Except for the relatively small quantities of Kr-85 that have been collected at the ICPP to supply the ORNL isotopes program.

the RAMPAC data base [EMERSON], maintained and searched by Sandia (ANL-83-57: Part I, Section E.1.d.). The second option is packaging certified for some other material or nuclide but adaptable to one of the airborne waste forms. This option would require amendment of the certificate since certificates are issued for a specific cargo. This option would also require a thorough study of the entire catalog of certified packages, a detailed examination of the list of packagings, and an engineering analysis of possible candidates. This is a strategy, however, that a consignor of wastes might use to avoid starting with a new design. The evaluation of packagings under this option was considered to be outside the scope of this study. If both options above are not viable, the design and certification of a new packaging would be required.

D. Evaluation of Disposal Prospects from the Standpoint of Final Emplacement

An appropriate destination for the airborne wastes is sought by comparing waste form properties with criteria for various modes of final emplacement. Since HLW has been defined by its origin (raffinate from solvent extraction of dissolved irradiated fuel) and TRU is defined by its actinide content, it has been assumed that airborne wastes must fall into the remaining category, LLW. Most LLW has been disposed of by SLB. The possibility of SLB for the airborne nuclides is considered first because it represents the most readily available and the most economic procedure. Therefore, the waste form characteristics are compared to the criteria outlined in 10 CFR 61 for SLB. If wastes are not clearly compatible with SLB criteria, then other possibilities are considered. At present, evaluating the possibility of using GCD or non-near surface disposal for the airborne waste is difficult because the development of criteria is still at an elementary stage and also may be site and waste specific. For those wastes for which conformance to the criteria on SLB is inconvenient, ocean disposal is also considered; however, the prospects of employing this mode of disposal is highly uncertain at this time. The rules devised by IAEA [IAEA-1980], following the instructions of the London Ocean Dumping Convention, can be used as a reference, but both the U.S. position and international regulations have not yet fully developed.

The IAEA tentative disposal scheme [IAEA-1982] suggests that low-level and intermediate-level wastes of long half-life be routed to geologic emplacement. On the chance that SLB may not be acceptable for I-129 and C-14 from fuel reprocessing, the characteristics of their waste forms are examined against proposed criteria for deep-geologic emplacement.

Criteria for final emplacement contain physical and chemical requirements on waste forms and also physical, mechanical, and design requirements on containers. An analysis of the significance of criteria for final emplacement of airborne wastes is initiated by selecting the elements of criteria against which airborne waste disposal systems must be checked.

The regulations for SLB (10 CFR 61) contain a classification system for waste forms. This system defines three categories: A, B, and C. Wastes that do not fall into these categories are generally unacceptable for SLB disposal. The bases for assignment of wastes to a category are the identity

of the radioactive nuclide and its concentration in the waste form. The category of a nuclide, in turn, determines the requirements on waste-form properties, packaging, and placement at the burial ground. For class A wastes, the requirements on waste form properties and packaging are least rigorous, and they must be segregated from other wastes in burial. For class B wastes, the requirements are more rigorous than for class A. For class C wastes, the requirements are the most rigorous, requiring (1) stability of the waste form or the container and (2) placement of the top of the container at least 5 m below the surface or use of engineered barriers that would prevent intrusion. For both class B and class C wastes, the waste forms or containers must maintain "gross physical properties and identity" for 300 y.

Regulations for LLW and HLW impose certain requirements on the waste form and packaging. The regulations in 10 CFR 61 stipulate that both the waste form and packaging be non-flammable and that no cardboard or fiberboard containers be used. In addition, these components are to maintain their gross physical properties and identity for over 300 y. In contrast, 10 CFR 60 approaches the requirements from a different standpoint. Containment is based on an engineered system of which the packaging requirements are retrievability for 50 y after emplacement, containment within the waste package for at least 1000 y following closure, and after 1000 y a release of $<1 \times 10^{-5}/y$ of the materials remaining.

E. Description of Currently Feasible Disposal Systems

In the discussions of the disposal of each nuclide (Sections III-VI), some descriptions are given for what is perceived at this time to be a currently feasible disposal system. These descriptions aim at characterizing the best compromises that can be made now between practical requirements and regulatory requirements, e.g., the maximum amount of a nuclide, consistent with regulations, that can be shipped in a currently licensed package and is acceptable for final emplacement. These descriptions are made after considering (1) waste characteristics with concentrations derived from R&D activities and (2) the characteristics of waste forms with concentrations altered to conform to regulations in both promulgated and draft status. In most cases, the characteristics of currently feasible waste forms differ considerably from those based on R&D information only. These descriptions represent a "best effort" on the basis of the current status of development of technologies and regulations.

The maturity of these descriptions should be judged from the perspective of the sequence of stages by which technological developments generally move from inception to application. The major stages are (1) R&D, (2) engineering-scale tests, (3) pilot-scale tests, (4) design and fabrication of full-scale equipment, and (5) routine operation of full-scale equipment. Although some of the parts of the technologies considered here have advanced as far as small-plant-scale testing, e.g., collection of Kr-85, others are still only at the R&D stage. Furthermore, some regulations are still in draft form. Even for regulations that have been promulgated, e.g., 10 CFR 61, their impact on practice and on the agreements to be drafted by regional

compacts,* eventually responsible for regulation of SLB, is not yet obvious; thus, all of these descriptions reflect the immaturity of the industry.

F. Comparison of the Performance of Disposal Systems with Limits

Limits are expressed by regulations in several ways, e.g., dose rates in mrem/y, release rates in Ci/GW_e-y, health effects or early mortalities, and concentration guides. The method of assessing compliance is made here in three different ways. (1) For limits expressed in terms of dose rate, a compliance analysis is made by estimating the dose rate from plant operations and comparing this estimate with the limit. The only operation for which sufficient information seems to be available to make an estimate is fuel reprocessing. (2) For regulations stated in terms of technical criteria, the assumption is made that, since the technical criteria are derived from the goals through modeling of the disposal process, meeting the technical criteria will result in meeting the underlying goals. (3) For regulations expressed as release limits, e.g., in Ci/GW_e-y, the expected efficiency of technology in restricting releases is compared to the efficiency required to meet the limits.

The efficiency required for isolating a given nuclide from the environment can be derived from the ratio of the maximum release allowed by regulations to the maximum release if the total amount of nuclide processed were dispersed to the environment. Thus,

$$F_{rel} = \frac{\text{maximum allowable release}}{\text{total amount of nuclide processed}} \quad (\text{II-1})$$

where F_{rel} is the fraction that can be released according to regulations. Then,

$$F_{isol} = (1 - F_{rel}) \quad (\text{II-2})$$

where F_{isol} is the fraction that by regulations must be isolated from the biosphere.

The total nuclide processed and the fraction collected can be used to derive the fraction that must be retained after packaging[†]:

$$\text{amount of nuclide collected} = A(F_{coll}) = \text{amount packaged} \quad (\text{II-3})$$

$$\text{amount retained after packaging} = (\text{amount packaged}) (F_{ret}) \quad (\text{II-4})$$

$$= A(F_{coll}) (F_{ret}) \quad (\text{II-5})$$

*A regional compact is an agreement on creation of a disposal facility and regulations for management of low-level waste within a region (one or more states) of the U.S.

[†]It is assumed that the amount packaged equals the amount collected.

where

A = total amount of nuclide processed

F_{coll} = fraction of total amount processed that is collected and packaged

F_{ret} = fraction of total amount collected and packaged that is retained over the total life of the package

The fraction of total amount processed that must be isolated from the biosphere, F_{isol} , can be derived from the permissible release of the nuclide per energy unit [40 CFR 190] and the amount of nuclide produced per energy unit (Eqs. II-1 and II-2). The rules of 40 CFR 190 specify maximum release limits; thus, the minimum amount that must be retained, $A(F_{isol})$, can be deduced as follows:

$$A(F_{coll})(F_{ret}) = A(F_{isol}) \quad (II-6)$$

$$F_{ret} = \frac{F_{isol}}{F_{coll}} \quad (II-7)$$

These equations are applied in assessing the compliance of the control technologies for I-129 (Section IV.H.1) and Kr-85 (Section VI.H.1) with release regulations.

The performance of technologies is examined against two types of regulations. One type is expressed as a release limit, e.g., the rules of 40 CFR 190 limiting Kr-85 and I-129 releases in units of Ci/GW_e-y. The other type is expressed as a dose limit. The task in this latter case is to compare the estimated dose rates from fuel cycle releases with the dose limits expressed in 40 CFR 190.

Judging whether the technology is capable of observing the dose limits involves three main steps: (1) identify the dose limits, (2) calculate a dose that would result from fuel cycle operations, and (3) compare the estimated dose against the limits.

The dose limits of 40 CFR 190 represent the maximum annual dose that any individual can receive from the fuel cycle. Several other methods of expressing the impact of radionuclide releases have been suggested, e.g., the estimation of health effects, risks, and hazard indexes; but EPA, the agency charged with the responsibility of setting criteria on environmental impact, has expressed these criteria in dose rates. Furthermore, dose rates have been calculated on a number of different bases (e.g., population dose in local, regional, or global areas) and over periods of 30 and 70 y. For comparison with regulations, however, an annual dose to the maximum exposed individual is the appropriate basis. Although the nuclides considered here will become distributed on a global scale and the effects over long time periods are of legitimate interest, regulations are not written in global terms.

Thus, the dose limit of interest in this assessment is an annual maximum individual dose. Conventionally, the maximum exposed individual is supposed to be a person residing at the boundary of the facility that is releasing the radionuclide to the environment. Incidentally, the customary boundary distance for modeling is 1 km from the release point. The comparison to be made here will thus be between the dose calculated for releases from a facility and the dose limits expressed in 40 CFR 190. The latter doses are 25 mrem/y to the whole body, 75 mrem/y to the thyroid, and 25 mrem/y to any other organ. Since the maximum individual dose is incurred at the boundary of a facility and the facility releasing significant amounts of H-3, I-129, C-14, and Kr-85 is the fuel reprocessing plant, the doses estimated in this work represent those to an individual residing at the boundary of a hypothetical fuel reprocessing plant.

The procedure chosen for calculating dose values here entails estimating the releases in curies for three levels of collection efficiency (0, 0.90, and 0.99). The release values in curies are then multiplied by the appropriate dose conversion factor (DCF) to obtain a result in terms of dose.

The total curies released (or source terms) can be obtained from

$$\begin{aligned} \text{Total Ci released} &= \text{Total Ci processed} (1 - F_{\text{coll}}) \\ &= A(1 - F_{\text{coll}}) \end{aligned} \quad (\text{II-8})$$

where F_{coll} is the fraction collected. Since source terms in curies are not compared in this work, the dose is calculated directly by

$$\text{maximum individual dose} = A(1 - F_{\text{coll}})(\text{DCF}) \quad (\text{II-9})$$

where DCF is in units of rem/(Ci emitted at the plant).

For this compliance analysis, doses are calculated by Eq. II-9 using not only the highest values claimed for collection efficiencies, but also lower ones.

The dose conversion factors, required by Eq. II-9, to convert a source term in Ci released at the plant to a dose to the maximum exposed individual in rem have been derived in studies by another organization [NEA]. They are based on assumed models, for a particular site, of the behavior of the individual nuclides, the pathways they would follow in the environment, and the biological-uptake mechanisms. Pathways and biological-uptake mechanisms are not discussed in the present work. Adoption of the dose conversion factors derived by NEA yields results that reflect the assumptions used in that modeling work.

The dose conversion factors are indicated in Table II-2. Even though site dependent, the NEA factors were selected for use here because, although several modeling studies of the release of airborne wastes at fuel reprocessing plants have been published [EPA, FINNEY, PALMS, VELURI, KNOX, TILL], the DCF's of NEA are the only ones found for directly estimating a value that can be compared with the limits expressed in 40 CFR 190. The NEA DCF's yield doses for a hypothetical, generic fuel reprocessing site. The DCF's listed in an EPA source [EPA] permit calculation of a dose in rem, given the

Table II-2. Dose Conversion Factors^a Used in Estimating Dose to the Maximum Exposed Individual from Release at a Hypothetical Fuel Reprocessing Plant

Isotope	Dose Equivalent Rate per Unit Emission Rate, ^b mrem/Ci
H-3	4.0×10^{-5}
I-129	42
C-14	3.8×10^{-3}
Kr-85	2.0×10^{-7}

^aFrom [NEA], tables 6', 7', 8', and 9'. Dose rates calculated from these factors will reflect the theoretical framework chosen by the NEA, which included use of tissue-weighting factors recommended by the International Council on Radiation Protection (ICRP) and quality factors of 1 for all nuclides.

^bFor H-3, C-14, and Kr-85, the dose calculated is to the whole body; for I-129, the dose calculated is to the thyroid.

concentration in Ci per volume of air. Therefore, calculation of the maximum individual dose at the boundary of a facility using the EPA data would require modeling to determine the curies per volume of air at the boundary resulting from a release at the plant.

Comparisons of calculated dose rates from a fuel reprocessing plant (substituting the parameters given in Table II-2 into Eq. II-9) with the regulatory limits are given in Sections III.F.2.a, IV.H.2.a, V.H.2.a, and VI.H.2 for H-3, I-129, C-14, and Kr-85 respectively. One of the difficulties in this comparison is that these regulations are for totals of all isotopes released at a facility, not just the isotopes of interest. Thus, some of the total 25 mrem/y whole body dose, for example, must be expected to be contributed by other isotopes. If the estimated dose from an airborne waste nuclide exceeds the total dose limit, a compliance problem is indicated. Since the DCF values are known [NEA] to have large uncertainties, the results of calculations using them are not presented here as decisive arguments for or against the need to collect any of the airborne isotopes at an FRP. If the calculated dose for a nuclide does not exceed the regulatory limit, it is not necessarily true that a compliance problem will not be encountered, depending on the dose attributed to releases of radionuclides other than the few here considered.

G. Method for Expressing Conclusions

The plan of this study is to determine whether technology for collection, packaging, transportation, and disposal of the airborne wastes is likely to be in accord with regulations that are being promulgated or in draft form at this time. In this analysis, the technological and regulatory factors involving each of the nuclides is examined against a similar set of questions:

- (1) Are the collection technologies capable of providing compliance with regulations that specify release limits?
- (2) Are there any problems in complying with transportation regulations?
- (3) Would it be practicable to use DOT specification packaging for transportation?
- (4) If DOT specification packaging is not practicable, is NRC-licensed packaging available? (DOT specification packaging is a generic packaging, not limited to a specific cargo. On the other hand, NRC-licensed packaging is licensed for a specific cargo. If NRC-licensed packaging, not specifically licensed for the airborne waste forms, were to be adapted, the license would have to be amended for the new cargo.)
- (5) Would the waste be acceptable for SLB according to 10 CFR 61? (This disposal mode represents the simplest option.)
- (6) If disposal by SLB is impractical, can the waste be disposed of by greater confinement or non-near surface* concepts?
- (7) If these disposal options seem unacceptable, what others should be considered?

*Non-near surface disposal is expected to be defined in a future NRC document, 10 CFR 61, Part B.

III. DISPOSAL OF H-3

Incentives for control of H-3 are tempered by its relatively short half-life and relatively high permissible concentration in releases specified by 10 CFR 20. Tritium (H-3) is not specifically addressed in the release regulations of 40 CFR 190. Extensive efforts have been made to develop processes for concentration of tritiated aqueous wastes to reduce the volumes in anticipation of the possibility that isolation from the biosphere might be required.

A. Origins of H-3 Waste Forms

The major operations in derivation of H-3 waste forms from spent fuel are shown in Fig. III-1.* A small fraction of the total H-3 is released into the shear off-gas, as THO, and even smaller fractions as TH, and combined with the off-gas stream from subsequent fuel-reprocessing operations. If voloxidation is used, the major fraction of the total H-3 would be collected on molecular sieves. If thermal outgassing is used, the major fraction of H-3 would be collected as ZrT_x . If neither voloxidation nor thermal outgassing is employed, the major fraction of the H-3 is collected as liquid tritiated water. Final waste forms for H-3 all consist of cement, involving either the incorporation of molecular sieve pellets or ZrT_x into cement or the formation of cement from tritiated water. Thus, all final waste forms for tritium considered here involve a cement matrix. As discussed in ANL-83-57:Part I (Section III.C.3.a), cement cannot reliably retain H-3; instead, retention must be insured by the containers holding it.

B. Characteristics of H-3 Waste Forms Based on R&D Information

The characteristics of H-3 waste forms based on R&D information are listed in Table III-1. These waste forms are prepared from tritiated water. The concentrations and volumes of tritiated water are based on concepts for management of aqueous streams in a fuel reprocessing plant, including recycle and the application of isotope separation [GRIMES]. The thermal outgassing example is based on the concept of [BRAY]. The information in this table is based on suggestions of waste-form developers only; no other characterizations for tritium wastes, e.g., in less-concentrated forms, are necessary to satisfy regulations for transport and SLB with regard to concentrations or quantities per package. All the waste forms listed in Table III-1 have the matrix of plaster and cement [described in ANL-83-57: Part I (Section III.C.3.a)] that was developed by Mound Laboratory. In this package, tritium is contained not by the plaster-cement but by the plastic and metal containers that hold the cement. The waste-form composition listed in Table III-1 is derived from the specifications for the Mound package, and the density, in turn, is derived from that composition. The specific activity of the waste form is estimated

*For simplicity, collection processes for H-3, I-129, C-14, and Kr-85 are shown separately in Figs. III-1, IV-1, V-1, and VI-1, respectively. In practice, these isotopes would appear together in some streams, and their collection would be carried out by some integrated off-gas system.

Fig. III-1. Operations in Disposal of H-3 Waste from Fuel Reprocessing. (Additional details appear in Appendix C.)

Table III-1. Properties of H-3 Waste Forms Derived from R&D Information

Composition Waste form	Tritiated Water in Plaster/Concrete ^a		Tritiated Water on Molecular Sieves in Plaster/Concrete ^a	
	Recycle Only ^b	Recycle Plus Isotope Separation on Recycle Side Stream ^c	From Voloxidation ^d	From Thermal Out-gas ^e
	plaster/concrete	plaster/concrete	water adsorbed on molecular sieves 3A plaster/concrete	water adsorbed on molecular sieves 3A plaster/concrete
H wt % of waste form	3.4	3.4	3.3	3.3
³ H atom fraction of total H	4.5×10^{-8}	2.0×10^{-6}	6.0×10^{-7}	1.8×10^{-5}
Phase of waste form	solid	solid	solid	solid
Density, g/cc	1.7	1.7	1.6	1.6
Particles, % of total waste form	0	0	0	0
Specific Activity ^f				
Ci/cc	1.2×10^{-4}	5.5×10^{-3}	8.2×10^{-4}	2.8×10^{-2}
Ci/g	7.1×10^{-5}	3.2×10^{-3}	5.0×10^{-4}	1.7×10^{-2}
Thermal loading ^f				
W/cc	4.1×10^{-9}	1.9×10^{-7}	2.8×10^{-8}	9.5×10^{-7}
W/g	2.4×10^{-9}	1.1×10^{-7}	1.7×10^{-8}	5.8×10^{-7}
Maximum absorbed dose ^f , rad/y	7.6×10^3	3.5×10^5	5.3×10^4	1.8×10^6
Radiolytic gas generation rate, cm ³ (STP)/g-y	4.8×10^{-5}	2.2×10^{-3}	3.3×10^{-4}	1.1×10^{-2}
Leak Rate, fraction/y				
by leaching ^g	$\sim 1 \times 10^{-7}$	$\sim 1 \times 10^{-7}$	1×10^{-7}	$\sim 1 \times 10^{-7}$
by volatilization	h	h	h	h

^a1:3 cement/plaster (volume) [MERSHAD].^bAssumes 85% of tritium in spent fuel is available for disposal in 3880 m³ water/y [GRIMES] at a 1500 MTHM/y plant.^cAssumes 85% of tritium in spent fuel is available for disposal in 86.4 m³ water/y [GRIMES] at a 1500 MTHM/y plant.^dAssumes 86% of tritium in spent fuel is available for disposal in 14.4 m³ water/y [GRIMES] at a 1500 MTHM/y plant.^eAssumes 99% of tritium in spent fuel is available for disposal in 0.558 m³ water/y [BRAY] at a 1500 MTHM/y plant.^fBased on an inventory of 1.1×10^6 Ci ³H in spent fuel containing 1500 MTHM.^gEstimate for package leak rate; tritiated water in plaster/concrete derived from [GIBBS].^hLeakage by volatilization mechanism only is not reported.

from the nuclide specific activity for the assumed mixture of hydrogen isotopes and the composition and density data. The thermal loading information is based on the same composition and density and the specific power for the mixture of hydrogen isotopes. The maximum absorbed dose was estimated from the definition of the rad (100 ergs absorbed in 1 g of waste) and the energy dissipation per gram of waste that follows from the composition and the specific power for the mixture of hydrogen isotopes. The radiolytic gas generation rate was estimated from the composition and the G values reported for a tritiated plaster-cement matrix [TADLOCK]. Leak rates were estimated from the leak rate determinations of Mound Laboratory [GIBBS]. These leak rates are characteristic not of the waste form material only but of the entire package.

C. Prospects for Management of H-3 Waste Forms from the Standpoint of Transportation Regulations

There is no indication that any of the H-3 waste forms considered here could not be transported to a site of final disposal. An overpack (e.g., the Super Tiger*) would probably provide the thermal insulation necessary to prevent overpressurization of containers carrying cement waste forms caused by release of water vapor from cement exposed to high temperatures in an accident involving fire.

1. Classification of H-3 for Transportation

The regulations of DOT, revised March 10, 1983, specify that up to 1000 Ci of H-3 in normal[†] form, adsorbed on solids, can be transported in Type A packages. Thus, quantities larger than 1000 Ci would be transported in Type B packages. It is assumed that all the waste forms of H-3 considered here would be equivalent to the "adsorbed on solids" category of the regulations. Table III-2 indicates the allowable quantities of H-3 per package for several types of authorized packaging.

2. Use of DOT-Specification Packaging for H-3

Table III-2 indicates that several DOT-specification packages could be used for transportation of the waste forms considered for H-3. Table III-3 characterizes H-3 wastes in the forms of tritiated water in cement, molecular sieves in cement, and ZrT in cement. An important indicator of the feasibility of a given packaging is the number of packages per year necessary to accommodate the output of a fuel reprocessing plant. Because the volumes for DOT-specification packagings are small, a large number of packages would be required to accommodate the H-3 produced, even with the use of isotope separation for volume reduction.

3. Use of Licensed Packaging for H-3

The RAMPAC data base [EMERSON] was searched by Sandia for packaging certified specifically for transportation of H-3. Several packagings certified for transportation are listed by RAMPAC. Having been designed for carrying gaseous samples, however, they would not be applicable to transporting the forms and quantities of H-3 considered here. The Mound package

*This overpack described in [JEFFERSON].

[†]Refer to definition in glossary.

Table III-2. Authorized Packaging Applicable to Transportation of H-3 Waste Forms^a

Authorized packaging ^{d,e} for shipments of radioactive materials in normal form inside U.S.	Allowed Quantities of H-3 per Package for Normal-Form ^b Materials Adsorbed on Solids, ^c Carried in Designated Packages	
	≤1000 Ci	>1000 Ci
DOT Spec. 7A Type A general packaging (including Spec. 6L, Spec. 6J) ^f	Y ^g	N ^h
Any foreign-made packaging that bears the marking "Type A" and complies with the regulations of the country of origin applicable to Type A packaging ^f	Y	N
Any Type B(U) or B(M) packaging that meets the applicable requirements in the regulations of NRC (10 CFR 71) ⁱ and that has been approved by that commission	Y	Y
DOT Spec. 6M metal packaging, only for solid or gaseous materials that do not undergo pressure-generating decomposition at temperatures up to 121°C and do not generate more than 10 W of decay heat	Y	Y
DOT Spec. 20WC wooden protective jacket, when used with a single, snug-fitting inner DOT Spec. 2R	Y	Y

^aDerived from DOT rules revised March 10, 1983 [FEDREG 10218].

^bNormal form defined in Glossary at front of report.

^cWaste forms for H-3 that are considered in this study are assumed to fit this category.

^dAll packaging employing Spec. 55 containers are omitted here. Since only existing Spec. 55 containers can be used (the construction of new ones is prohibited after Dec. 31, 1975), it is unlikely that they would be used in shipments of any airborne waste forms from fuel re-processing operations.

^ePackaging described in Part I, ANL-83-57.

^fType A packagings will be required to pass the more demanding tests outlined in 49 CFR 173.465-173.466 after July 1, 1985. Some packagings now acceptable may not pass these tests.

^gY = allowed.

^hN = not allowed.

ⁱStatus of 10 CFR 71 summarized in Appendix A.6.

Table III-3. Numbers of Containers and Activity Levels of Various Waste Forms for H-3 (1.08×10^6 Ci/y-1500 MTHM, 1.5-y cooled)

Item ^a	Process					
	Dissolution	Isotope Separation	Hulls	Voloxidation	Thermal Outgas	Thermal Outgas
Waste Form	THO + H ₂ O	THO + H ₂ O	Zr Hydride	THO + H ₂ O	Mol. Sieves/with	
Final Form	Cement	Cement	Cement	Cement	T ₂ O + H ₂ O	ZrT _{1.5}
Total Wt, kg	3.1×10^7	6.8×10^5	1.6×10^6	1.9×10^6	6.5×10^4	7.1
Total Vol, cc	1.5×10^{10}	3.3×10^8	1.0×10^9	1.0×10^9	3.9×10^7	4.8×10^3
No. Containers/y						
6L (2R)	1.6×10^6	3.6×10^4	1.1×10^5	1.1×10^5	4.2×10^3	
6M (2R)	2.7×10^6	5.9×10^4	1.8×10^5	1.8×10^5	7.0×10^3	4
6J (A)	1.2×10^5	2.8×10^3	6.0×10^5	7.3×10^3	1080	
Mound Pkg.	5.6×10^5	1.2×10^4	3.7×10^4	3.7×10^4	1.4×10^3	
Curies/Container			(max)			
6L (2R)	0.6	26	1.3	8.7	255	
6M (2R)	0.4	16	0.8	5.3	150	2.8×10^5
6J (A)	8.0	340	24	130	1.0×10^{3b}	
Mound Pkg.	1.7	79	3.9	26	7.6×10^2	
Watts/Container						
6L (2R)	2.0×10^{-5}	8.8×10^{-4}	4.4×10^{-5}	2.9×10^{-4}	8.6×10^{-3}	
6M (2R)	1.4×10^{-5}	5.4×10^{-4}	2.7×10^{-5}	1.8×10^{-4}	5.1×10^{-3}	10^c
6J (A)	2.7×10^{-4}	1.1×10^{-2}	8.1×10^{-4}	4.4×10^{-3}	3.3×10^{-2}	
Mound Pkg.	5.8×10^{-5}	2.7×10^{-3}	1.3×10^{-4}	8.8×10^{-4}	2.6×10^{-2}	
Waste Wt/Pkg, ^d kg						
6L (~136 kg max) (2R)	19	20	14	17.4	15	
6M (~273 kg max) (2R)	12	12	8.6	10.6	9.3	1.8
6J (~259 kg max) (A)	259	259	259	259	60	
Mound	55	57	43	51	46	

^a(A) denotes configuration "A" [EDLING]; 6L (2R) denotes DOT specification 6L packaging with 2R inner container. For the estimates given here, the volumes of the 2R containers were assumed to be typical of those that have been used--for the 6L package, 9.2×10^3 cm³, and for the 6M package, 5.6×10^3 cm³. The volume of the 6J package, which involves no 2R container, was assumed to be that of a 55-gal drum, 2×10^5 cm³.

^b6J container is restricted to radioactive quantities $\leq A_2$ for non-fissile materials, or 1000 Ci in the case of H-3.

^cLimit for Spec 6M pkg.

^dAllowable package weight for standard containers given in parentheses.

is designed for transportation and disposal of H-3. The performance of this package, especially with respect to leak rate, seemed to meet the requirements of the consignor and the potential consignee. However, it has apparently not been certified. Since the volume of H-3 carried by the Mound package is relatively small, the number of packages required if it were applied to carrying the amounts of H-3 produced in fuel reprocessing would be relatively large, as shown by Table III-3.

D. Compatibility of H-3 Waste Management Technologies with SLB

With regard to SLB of H-3, two important features of 10 CFR 61 are examined in the following sections: (1) criteria on waste form properties and (2) criteria on packaging. The criteria on waste form properties are not very specific, but are more detailed than criteria on packaging. The

difficult requirement of 10 CFR 61 is that, for class B and C wastes, the packaging or the waste form must be capable of maintaining "gross physical properties and identity" for 300 y.

1. Compatibility of H-3 Waste Form Properties with Criteria for SLB

The compatibility of H-3 waste forms with criteria for SLB is examined by listing (Table III-4) the expected points of compliance and noncompliance with the technical criteria of 10 CFR 61. This comparison is made with the reservation that these regulations were drafted with the assumption that reprocessing wastes would not appear in the near future. The entries in the table show that for the waste forms of H-3 there are no expected problems with the technical criteria of 10 CFR 61. If the H-3 waste is handled according to the requirements for Class B in 10 CFR 61, its permissible concentration is limited only by the specific activity of the nuclide.

2. Compatibility of H-3 Packaging with Criteria for SLB

Two types of possible packaging for transportation of H-3 wastes were identified in the preceding sections: (1) several DOT specification packagings and (2) the Mound packaging, which has not been certified but has been shown in tests to be capable of a very low leak rate of H-3. Perhaps the most important question with regard to both these types of packaging is, however, will they meet the criteria outlined in 10 CFR 61 for disposal at an SLB site even though they may be satisfactory for transportation? Since the outer container for both these packages is carbon steel, it seems unlikely that it would survive in contact with earth for 300 y. To provide the required stability, 55-gal drums could be overpacked by high-integrity containers designed to comply with the general descriptions in the NRC technical position papers on waste form and containers.

E. Description of Currently Feasible Disposal Systems for H-3

Table III-1 gives descriptions of H-3 waste forms that are in compliance with applicable regulations for transportation (49 CFR) and SLB (10 CFR 61) at their current stage of development. Waste packages for H-3 in compliance with the transportation regulations (49 CFR) are described in Table III-5. At the concentrations specified by Table III-1, the waste forms would fall into class B of 10 CFR 61 and, therefore, must either be shown capable of the stability requirement or be placed in a container capable of it. At concentrations $\leq 40 \text{ Ci/m}^3$, to qualify as a class A waste, the 300-y lifetime is not required, but the large volume of matrix necessary to accommodate this concentration limit would require an impracticably large number, $\geq 10^5$, of 55-gal drums.

F. Comparison of the Performance of H-3 Disposal Systems with Regulatory Limits

Information permitting the comparison of H-3 disposal technologies with limits expressed in regulations is available for some regulations but not for others. In the latter case, detailed modeling studies have to be performed to allow the comparisons. Tritium is specifically regulated by the

Table III-4. Compatibility of H-3 Waste Forms with Criteria for Shallow-Land Burial (10 CFR 61)

	Tritiated Water in Concrete	Molecular Sieves in Concrete	ZrT in Concrete
Sufficient absorbent to absorb twice the volume of any liquid?	$Y^{a,b}$	Y^b	Y^b
Liquid volume $\leq 1\%$ of total volume of waste when in a stable container or $\leq 0.5\%$ of total volume of waste for a stable waste form?	Y	Y	Y
Not capable of detonation, explosive decomposition, reaction at normal temperatures and pressures, or explosive reaction with water?	Y	Y	Y
Not pyrophoric?	Y	Y	Y
Are gases, if any, at pressure ≤ 1.5 atm at 20°C ?	Y	Y	Y
Are gases, if any, at total activity ≤ 100 Ci/package?	Y	Y	Y
No more than minimal amounts of biological, pathogenic, or infectious hazards?	Y	Y	Y
Structurally stable to effects of overburden weight, moisture, microbial activity, radiation effects and chemical change, either because of properties of waste form or container?	Y	Y	Y

^aY = yes, compliance expected. N = no, noncompliance expected.

^bAssumes the concrete has been prepared to eliminate any free-standing liquid.

Table III-5. Possible ^3H Waste Packages

	Source of Tritium					
	Dissolution Only	Dissolution and Isotope Separation	Hulls	Voloxidation	Thermal Out-gassing	
Capture Form	Water	Water	ZrTy	Molecular Sieves	Molecular Sieves	ZrTi _{1.5}
Waste Form	Cement	Cement	Cement	Cement	Cement	Cement
Curies/pkg	10	430	23	150	1.0×10^3 ^a	1.0×10^3 ^a
Curies/cc ^b	4.47×10^{-5}	1.39×10^{-3}	9.96×10^{-5}	6.5×10^{-4}	5.0×10^{-3}	5.0×10^{-3}
Number of Drums (max) per 1500 MTHM	9.0×10^4	2.1×10^3	7.1×10^3	6.4×10^3	9.1×10^2	9.1×10^2
No. of Mound Pkgs. per 1500 MTHM	5.6×10^5	1.2×10^4	3.7×10^4	3.7×10^4	1.4×10^3	--

^a 6L container is restricted to radioactive quantities $\leq A_2$ for non-fissile materials, or 1000 Ci in the case of H-3.

^b Based on volume of Spec. 64 (55-gal) drum, outside dimensions. Dimensions of inner container are assumed to be 51.3-cm dia and 75.9-cm length.

concentration guidelines of 10 CFR 20 and the Drinking Water limit of 40 CFR 141, but its environmental impact must also be judged against the general dose limits prescribed by 40 CFR 190. Published results of modeling studies permit an analysis of the compliance with the general maximum individual dose at the facility boundary.

1. Release Regulations and H-3 Control

There are no regulations specifically addressed to release of H-3 from the fuel cycle in Ci/GW_e-y, as in the cases of I-129 and Kr-85. Discharges of H-3 to the environment will have to be judged, therefore, against the general dose limits expressed in 40 CFR 190 and the limit of 20 nCi/L required by 40 CFR 141. Assessment of H-3 discharges in the light of the limits listed in 40 CFR 141 would involve an intricate model. This opinion is based on the expectation that discharges of tritium from a fuel reprocessing plant would involve dispersal of tritiated water as liquid and vapor through complex pathways to biological receptors.

2. Dose Regulations and H-3 Control

For this task, the comparison of performance of technology for H-3 control with dose regulations should be made at the fuel reprocessing plant and the disposal site.

a. General Dose Limits and H-3 Control

The question explored in this section is compliance of H-3 management with the dose limits of 40 CFR 190, i.e., 25 mrem/y to the whole body. This limit involves the dose incurred by an individual from all normal operations of the entire fuel cycle except for mining, transportation, and waste-management operations. Therefore, this dose can be used to judge the collection of the airborne wastes in fuel-reprocessing operations, but not the retention of the collected, packaged isotopes in subsequent waste management operations. Table III-6 presents the estimated doses, calculated from Eq. II-9 using the DCF for H-3 in Table II-2, from a 1500 MTHM/y plant. The data in Table III-6 show that, if collection technologies for H-3 perform at the highest efficiency they are thought to be capable of (0.99), or even at the lower efficiency (0.90), then the maximum exposed individual would incur only a fraction of the whole-body dose of 25 mrem/y permitted by 40 CFR 190. At the other extreme, that of collecting no tritium, the calculations indicate that the maximum exposed individual would incur the entire whole-body dose limit of 40 CFR 190. Since this result could be high by a factor of eight (judging from the estimated uncertainties of the DCF's [NEA]), it cannot be used as an indicator of the need to collect tritium from an FRP.

Table III-6. Estimated Dose to Maximum Exposed Individual from H-3

A, ^a Ci/Plant-y	DCF, ^b mrem/Ci	Maximum Individual Dose, ^c mrem/Plant-y		
		F _{coll} =0	F _{coll} =0.90	F _{coll} =0.99 ^d
9.1 x 10 ^{5e}	4.0 x 10 ⁻⁵	36	3.6	0.36

^aTotal amount of nuclide processed annually by 1500-MTHM/y fuel reprocessing plant.

^bDose conversion factor, see Table II-2.

^cCalculated from the expression $A(1 - F_{coll})(DCF)$.

^dCollection factors of 0.99 are assumed possible by the [NEA] evaluation. This high collection efficiency has not been demonstrated in practice.

^eMaximum quantity excluding hulls.

b. Dose Limits of Emplacement Criteria and H-3 Control

The limits on dose rates incurred from radioactive materials from a disposal facility underlying the development of 10 CFR 61 are the same as those expressed in 40 CFR 190, i.e., 25 mrem/y to the whole-body. These limits were used by NRC in modeling studies on movement of nuclides through the environment to arrive at the technical criteria, including concentration limits, expressed in 10 CFR 61 for SLB. It is assumed, therefore, that emplacement of H-3 in an SLB site in accord with the technical criteria of 10 CFR 61 (i.e., meeting the concentration limits, waste form properties, and packaging characteristics required) will ensure meeting the dose limits

on which 10 CFR 61 is founded. Caution is advised in accepting this conclusion in that the general attitude in development of 10 CFR 61 assumed non-existence of commercial reprocessing. As discussed above (Section III.D), it seems practicable to emplace H-3 wastes according to the rules of 10 CFR 61. However, SLB will be costly if the tritium-management scheme at the FRP results in a large volume of tritiated waste. Emplacement as a Class A (10 CFR 61) waste would be economically prohibitive because of the large volume of matrix and number of packages involved. Emplacement as a Class B (10 CFR 61) waste would require that either the waste form or the container "maintain gross physical properties and identity" for 300 y.

G. Conclusions on Disposal of H-3

Conclusions on disposal of H-3 are made with respect to three bases: compatibility with current regulations, needs for additional R&D on control technology, and needs for attention to tritium in any future development or amendment of regulations.

1. Compatibility of Systems for H-3 Disposal with Current Regulations

The physicochemical processes and the conceptual flowsheets and equipment for collection of tritium promise to be capable of preventing the release of tritium to satisfy the requirements of current regulations.

2. R&D Needs for H-3 Control Technology

The several concepts for tritium waste-management technology offer a wide range of collection efficiencies, but the operations become complex and costs promise to become high if the volume of waste is to be minimized. Furthermore, none of these technologies should be considered at the present time to be proven and available at full scale.

The use of some DOT-specification packaging is considered practicable for the transport of tritium wastes. The Mound package could, if it were licensed, also be used for transportation of tritium wastes from the fuel cycle. It seems quite likely that tritium wastes from fuel reprocessing could be disposed of by SLB in compliance with 10 CFR 61. In the future, 10 CFR 61 might be revised to accommodate the wastes of commercial fuel reprocessing, if it develops. It is not obvious, however, whether the packaging acceptable for transportation will also meet burial criteria on maintaining waste and site stability through container integrity if the waste form does not. For both DOT-specification packaging and the Mound package, the limited waste volumes per package result in distribution of the H-3 among a large number of packages. Perhaps within the guidelines given in the NRC technical position papers related to 10 CFR 61, packaging could be designed that would carry a larger quantity per package yet be of simpler construction than the Mound package. Such a package, however, must meet the weight limit for transportation and the requirements for burial at an SLB site.

Although the SLB disposal of H-3 has been characterized here, it may not be the most economic management mode. The short half-life and the large volumes of final waste form (except in the case of the product of

thermal out-gassing) are incentives to pursue the development of methods for on-site storage of H-3 wastes at the fuel reprocessing plant. After a period of 100-150 y, the hazards of this waste would have become small, but decommissioning of the FRP may be desired sooner.

Unrestricted disposal of radioactive materials may be done if the maximum release concentrations given in 10 CFR 20 (Appendix A-2, Table A-1) are not exceeded. This conclusion, however, is subject to application of the ALARA requirement. For disposal of liquids containing soluble radioactive species released to sewers, total yearly quantities are also given, e.g., 5 Ci/y for H-3 and 1 Ci/y for the other three airborne wastes. No such limits are given for releases to the air. After 40 y of FRP operation and a 100 y period of storage of H-3 on the FRP site, the residual activity of the H-3 from the first year of FRP operation would be $3.9 \times 10^{-2}\%$ of the initial activity, and the activity from the last year of FRP operation would be 0.366% of the initial activity.

3. Suggested Amendments or Additions to Regulations

Although there are no release values (Ci/GW_e-y) in regulations specifically addressed to tritium control, prevention of its uncontrolled release might be sought because of its contribution to the whole body dose limit of 40 CFR 190. Furthermore, even in the absence of specific rules, control might be imposed through future legal interpretations in licensing hearings of the ALARA requirement or of the BACT clause of the Clean Air Act.

The concentration limit for disposal of tritium as a class B waste in accord with 10 CFR 61 at an SLB facility is not restrictive; concentrations up to the specific activity of the nuclide are permitted. Thus, no other disposal mode now offers any advantage over SLB. If, however, at some future time, site inventory limits on individual nuclides are set for SLB, then some other disposal mode, e.g., intermediate-depth burial similar to GCD or non-near surface disposal, may appear more attractive since the site inventory limit for H-3 may be larger for these disposal modes than for an SLB site.

In view of these possible developments, interpretations of the significance of ALARA and BACT to management of the H-3 from fuel reprocessing are needed. More specific definitions of the allowed contributions of individual nuclides to the total 25 mrem/y whole body dose from the entire fuel cycle now specified by 40 CFR 190 would permit planning on what fraction of the H-3 needs to be collected from FRP streams. Definite conclusions on nuclide inventory limits for SLB sites (which may be site specific) and also firm criteria for GCD and non-near surface disposal would make it possible to decide on the relative advantages of disposal of H-3 by these two disposal modes. The requirement of 300-y stability of container or waste form regardless of half-life seems to unnecessarily penalize the disposer of pure tritium wastes. It is suggested that the rules might be amended to provide that, if the concentration of a nuclide decays to class A within the expected lifetime of the container, the 300-y container requirement would be waived. Alternatively, requirements for container lifetime could be specified as a function of half-life of the nuclides contained.

IV. DISPOSAL OF I-129

Iodine-129 has a long half-life, has a low concentration standard prescribed by 10 CFR 20, and tends to concentrate in the thyroid gland. These are the main factors steering the development of processes for the collection and isolation of I-129. An additional impetus for development of these processes is the release limit for I-129, prescribed by EPA criteria in 40 CFR 190, which requires a high efficiency of collection and retention.

A. Origins of I-129 Waste Forms

The major operations used to obtain I-129 waste forms are shown in Fig. IV-1. Caustic scrubbing has been described as a possible collection process; but because it is likely to be much less effective than the other processes, it is not considered here. A small fraction of the total amount of I-129 is released into the shear off-gas, perhaps as I_2 or CH_3I , which is combined with off-gas streams from subsequent fuel-reprocessing operations. If voloxidation or thermal outgassing is employed, a major fraction of the iodine might appear in the off-gas of these operations. If these operations are not used, most of the iodine present in spent fuel will proceed into the dissolver. The complex chemistry of iodine poses difficulties in its collection because iodine tends to distribute into several fuel-reprocessing streams. Without voloxidation or thermal outgassing, most of the iodine is expected to appear in the dissolver off-gas; most of the R&D work devoted to iodine collection has been aimed at this stream. An exception is the ICPP, where the dissolver off-gas is refluxed, with the result that a large fraction of the I-129 remains in the dissolver solution, eventually appearing as a dilute solution in the bottom of a subsequent evaporation process [MCMANUS]. The chemistry of the several proposed collection processes, involving gas-liquid and gas-solid reactions, is diverse and complex. Only three products are considered here: $Ba(IO_3)_2 \cdot H_2O$ in cement, silver zeolite, which may be incorporated into cement, and lead zeolite. The lead zeolite would be produced in a process that involves collection of I-129 on silver zeolite and subsequent reclamation of the silver zeolite by treatment with hydrogen to produce the hydrogen zeolite and release the iodine, which is then trapped on lead zeolite. The object of this process is to use the superior efficiency of the silver zeolite for collection and the relatively lower cost of the lead zeolite for disposal. Recently, however, it has been suggested [BURGER-1983B] that the ineffectiveness of lead zeolite in capturing the iodine released during the hydrogenation reaction, among other considerations, makes it seem unlikely that a lead zeolite would ever be chosen as a disposal form. Other waste forms believed to be less immediately practical than these are listed in ANL-83-57:Part I (Section III.D.3.e).

B. Characteristics of I-129 Waste Forms Derived from R&D Information

Table IV-1 lists characteristics of three I-129 waste forms derived from R&D information: (1) barium iodate in cement, (2) lead zeolite, and (3) silver zeolite in cement. The loadings of radioactive solids into cement are adapted from the R&D information described in ANL-83-57:Part I (Section III.D.3). Densities of cement waste forms are derived from the assumed composition of 9 vol % of solids added to cement and the literature values for densities of the barium iodate, zeolite, and cement. The specific activities of the waste forms are obtained from the nuclide specific activity

Fig. IV-1. Operations in Disposal of I-129 Wastes from Fuel Reprocessing. (Additional details of chemistry in Appendix E.)

Table IV-1. Properties of I-129 Waste Forms Derived from R&D Information

Properties	Barium Iodate in Cement	Lead Zeolite	Silver Zeolite in Cement
Composition			
Radiophase	Ba(IO ₃) ₂ ·H ₂ O	PbZI	AgZI
Encapsulant	portland cement	-	portland cement
I wt % of Waste Form	9.0	18	5.4
¹²⁹ I Atom Fraction of Total I	0.75	0.75	0.75
Phases	solid	solid	solid
Density, g/cc	2.2	1.0	2.1
Particles, % of Total	0	100	0
Particle Size	-	1/8 in. pellets	-
Specific Activity			
Ci/m ³	26	23	15
Ci/Mg	12	23	7.0
Estimated Gas Generation Rate			
cc(STP)/g-y	1.3 x 10 ⁻⁴	-	6.2 x 10 ⁻⁵
Maximum Dose Absorbed by Waste			
rad/y	1.7 x 10 ⁴	3.5 x 10 ⁴	1.0 x 10 ⁴
Thermal Load			
W/m ³	1.2 x 10 ⁻²	1.1 x 10 ⁻²	6.8 x 10 ⁻³
W/Mg	5.4 x 10 ⁻³	1.1 x 10 ⁻²	3.2 x 10 ⁻³
Compressive Strength, psi	6.4 x 10 ³	-	2.6 x 10 ⁻³
Leak Rates			
Leaching, fraction/y ^a	1x10 ⁻⁴ - 1x10 ⁻³	>>10 ⁻² ^b	1 x 10 ⁻⁴

^aBased on a monolithic cylinder with dimensions of a 55-gal drum.

^bEstimated from the solubility of silver iodide on zeolite, using the facts that the solubility of PbI₂ is 10⁵ times greater than that of AgI and that the surface area of the pellets is about 200 times greater than that of the monolith.

of the mixture of iodine isotopes (see ANL-83-57:Part I), the assumed composition, and the derived density. The calculated values of specific powers of the waste forms are based on the nuclide specific power of the mixture of iodine isotopes and the composition. The maximum radiation dose absorbed by the waste form is estimated from the definition of the rad (100 erg absorbed per gram of waste) and the estimated specific power of the waste. The radiolytic gas generation rate is estimated from G values given for the radiolysis of concrete [MOLECKE]* and the quantity of cement per weight of waste. Leach rates are estimated from reported experiments (ANL-83-57:Part I, Section III.D.3).

C. Prospects for Disposal of I-129 Waste Forms from the Standpoint of Transportation Regulations

None of the criteria in 49 CFR disqualifies I-129 wastes described in Table IV-1 from being transported in the appropriate packaging. The classification system of 49 CFR was used in this study to determine the general packaging types and also the DOT specification packaging permitted for materials with the characteristics of the I-129 waste forms. A search by Sandia National Laboratory of the RAMPAC data base showed that no packaging has been certified by NRC for transportation of I-129 specifically.

1. Classification of I-129 for Transportation

The recent revisions[†] to the DOT regulations (49 CFR) specify that up to 2 Ci of I-129 in normal form can be transported in Type A packaging. Thus, I-129 in quantities larger than 2 Ci, in normal form, must be carried in Type B packaging.

2. Use of DOT-Specification Packaging for I-129

Table IV-2 lists several DOT-specification packages authorized for transportation of radioactive materials represented by the I-129 waste forms considered here. Table IV-3 characterizes packages of I-129 wastes utilizing some of these containers. The number of packages per year necessary to accommodate the I-129 throughput of a fuel reprocessing plant is one of the most important indicators of feasibility. The limited volumes of DOT-specification packaging would result in the need for a large number of packages to accommodate the I-129 output of fuel reprocessing. The number of packages of I-129 seems practicable if the concentrations of Table IV-1 are used. These concentrations are too high, however, for SLB, by the rules of 10 CFR 61. This problem is considered further below.

*From the range of G-values for total gas generation rate estimated by Molecke from the gamma radiolysis studies of [BIBLER-1977, -1978].

[†]March 10, 1983 [FEDREG 10218].

Table IV-2. Authorized Packaging Applicable to
Transportation of I-129 Waste Forms^a

Authorized packaging ^{c,d} for shipments of radioactive materials in normal form inside U.S.	Maximum Quantities of I-129 per Package for Normal- Form Materials ^b Carried in Designated Package	
	≤2 Ci	>2 Ci
DOT Spec. 7A Type A general packaging (including Spec. 6L, Spec. 6J) ^e	Y ^f	NG
Any foreign-made packaging that bears the marking "Type A" and complies with the regulations of the country of origin applicable to Type A packaging ^e	Y	N
Any Type B(U) or B(M) packaging that meets the applicable requirements in the regulations of NRC (10 CFR 71) ^h and that has been approved by that commission	Y	Y
DOT Spec. 6M metal packaging only for solid or gaseous materials that do not undergo pressure-generating decompo- sition at temperatures up to 121°C and do not generate more than 10 W of decay heat	Y	Y
DOT Spec. 20WC wooden protective jacket, when used with a single, snug-fitting inner DOT Spec. 2R	Y	Y

^aDerived from DOT Rules revised March 10, 1983 [FEDREG 10218].

^bNormal form defined in Glossary.

^cAll packaging employing Spec. 55 containers are omitted here. Since only existing Spec. 55 containers can be used and the construction of new ones is prohibited after Dec. 31, 1975, it is unlikely that such containers would be used in shipments of any airborne waste forms from fuel reprocessing operations.

^dPackaging described in ANL-83-57:Part I (Section IV.E.1.c).

^eType A packagings will be required to pass the more demanding tests defined in 49 CFR 173.465-173.466 after July 1, 1985.

^fY = allowed.

NG = not allowed.

^hStatus of 10 CFR 71 summarized in Appendix A.6.

Table IV-3. Numbers of Containers and Activity Levels of Various Waste Forms for Iodine-129 Adapted from R&D Information (47 Ci/y-1500 MTHM, 1.5-y cool)

Item ^a	Barium Iodate in Cement	Silver Zeolite in Cement	Lead Zeolite Pellets
Form	compound	zeolite	zeolite
Final Form	in cement	in cement	pellets
Total Weight, kg/y	3.9×10^3	6.5×10^3	2.0×10^3
Total Volume, cc/y	1.8×10^6	3.1×10^6	2.0×10^6
No. of containers/y			
6L (2R)	190	340	220
6M (2R)	320	560	360
6J (A)	24	25	24
Curies/container			
6L (2R)	0.24	0.14	0.22
6M (2R)	0.15	0.084	0.13
6J (A)	2 ^b	1.9	2 ^b
Watts/container			
6L (2R)	1.1×10^{-4}	6.5×10^{-5}	1.0×10^{-4}
6M (2R)	6.9×10^{-5}	3.9×10^{-5}	6.0×10^{-5}
6J (A)	9.2×10^{-4}	8.8×10^{-4}	9.2×10^{-4}
Waste Form Weight/pkg, ^c kg			
6L (~136 kg max) (2R)	20	19	9.1
6M (~273 kg max) (2R)	12	11	5.5
6J (~259 kg max) (A)	160	259	83

^a(A) denotes configuration "A" [EDLING]; 6L (2R) denotes DOT specification 6L packaging with 2R inner container. For the estimates given here, the volumes of the 2R containers were assumed to be typical of those that have been used--for the 6L package, $9.2 \times 10^3 \text{ cm}^3$, and for the 6M package, $5.6 \times 10^3 \text{ cm}^3$. The volume of the 6J package, which involves no 2R container, was assumed to be that of a 55-gal drum, $2 \times 10^5 \text{ cm}^3$.

^b6J container is restricted to radioactive quantities $\leq A_2$ for non-fissile materials, or 2 Ci in the case of I-129.

^cAllowable package weight for standard containers given in parentheses.

D. Compatibility of I-129 Control Technologies with SLB

The two main features of 10 CFR 61--criteria on waste form properties and criteria on packaging acceptable for final SLB emplacement--are considered in the following sections. The criteria on waste form properties are not very specific, and criteria on packaging are even less so. An outstanding criterion on packaging in 10 CFR 61 is that, for Class B and Class C wastes, either the waste forms or containers must be capable of maintaining their "gross physical properties and identity" for 300 y.

1. Compatibility of I-129 Waste Form Properties with Criteria for SLB

The compatibility of I-129 waste forms with 10 CFR 61 criteria for SLB is indicated in Table IV-4. An obvious difficulty for compliance of I-129 waste forms with SLB criteria stems from the very low concentration allowed by 10 CFR 61. The waste forms characterized in Table IV-1 have concentrations that follow suggestions of waste form developers, but exceed the limit of 0.08 Ci I-129/m³ allowed for SLB by 10 CFR 61. Although there are no physical or chemical reasons that waste forms with concentrations meeting this limit could not be prepared, such a limit results in the distribution of I-129 in a large amount of matrix material, which would require a large number of packages and a large burial space (discussed later). Characteristics of I-129 waste forms based on the concentration allowed by 10 CFR 61 are summarized in Table IV-5.

2. Compatibility of I-129 Packaging with Criteria for SLB

Several packages capable of bearing I-129 wastes were identified in Table IV-2. Wastes with I-129 concentrations $>0.008 \text{ Ci/m}^3$ * fall into Class C and are therefore subject to the requirement of 300-y stability for waste form or containers by 10 CFR 61. Although the packages described in Table IV-2 could carry I-129 wastes in compliance with the transport regulations of 49 CFR, the concentrations of I-129 involved require that the waste forms or containers be stable for 300 y. Although the cement waste forms will probably meet this stability requirement, the lead zeolite pellets cannot and therefore must rely on a container to do it. DOT specification packages may have to be overpacked in high integrity containers to meet the 300-y requirement.

E. Compatibility of I-129 Control Technologies with Geologic Emplacement

The only disposal regulations specific for I-129 appear in 10 CFR 61, relating to management of LLW. Although 40 CFR 191 Draft would define as HLW any waste containing a nuclide with $t_{1/2} > 20 \text{ y}$ in concentrations exceeding a given limit, these rules apply only to spent fuel or to wastes derived from

*10 CFR 61 specifies that concentrations in excess of 0.1 times the value in its Table 1 (0.08 for I-129) fall in Class C. This class is subject to the 300-y stability requirement.

Table IV-4. Compatibility of I-129 Waste Forms with Criteria for Shallow-Land Burial (10 CFR 61)

	Ba(IO ₃)•H ₂ O in Cement	Lead Zeolite	Silver Zeolite in Cement
Sufficient absorbent to absorb twice the volume of any liquid?	ya	ya	ya
Liquid volume $\leq 1\%$ of total volume of waste when in a stable container or $< 0.5\%$ of total volume of waste for a stable waste form?	Y	Y	Y
Not capable of detonation, explosive decomposition, reaction at normal temperatures and pressures, or explosive reaction with water?	Y	Y	Y
Not pyrophoric?	Y	Y	Y
Gases restricted to pressure < 1.5 atm at 20°C ? Gases restricted at total activity ≤ 100 Ci/package?	Y	Y	Y
No more than minimal amounts of biological, pathogenic, or infectious hazards?	Y	Y	Y
Structurally stable to effects of overburden weight, moisture, microbial activity, radiation effects, and chemical change, either because of properties of waste form or container?	Y	N ^b	Y
Concentration ≤ 0.08 Ci/m ³ ? ^c	Y	Y	Y

^aAssumes cement prepared to exclude free-standing liquid. Y = Yes.

^bN = No.

^cThis concentration could be prepared as readily as the higher concentrations described in Table IV-1, without detrimental effects on structural properties of the waste form. Meeting this concentration limit, however, would result in distributing the amounts of I-129 to be disposed of throughout a large volume of matrix and, consequently, a large amount of burial space and a large number of packages.

Table IV-5. Properties of I-129 Waste Forms with Concentrations Limited to Comply with 10 CFR 61

Properties	Barium Iodate in Cement	Lead Zeolite	Silver Zeolite in Cement
Composition			
Radiophase	Ba(IO ₃) ₂ ·H ₂ O	PbZI	AgZI
Encapsulant	portland cement	-	portland cement
I wt % of Waste Form	3.0 x 10 ⁻²	6.1 x 10 ⁻²	3.0 x 10 ⁻²
Radiophase wt % of Waste Form	6.0 x 10 ⁻²	100	1.9 x 10 ⁻¹
¹²⁹ I Atom Fraction of Total I	0.75	0.75	0.75
Phases	solid	solid	solid
Density, g/cc	2.0	0.99	2.0
Particles, % of Total	0	100	0
Particle Size	-	1/8 in. pellets	-
Specific Activity			
Ci/m ³	0.080	0.080	0.080
Ci/Mg	0.040	0.081	0.040
Estimated Gas Generation Rate			
cc(STP)/g-y	5.3 x 10 ⁻⁷	-	5.3 x 10 ⁻⁷
Maximum Dose Absorbed by Waste			
rad/y	5.7 x 10 ¹	1.2 x 10 ²	5.7 x 10 ¹
Thermal Load			
W/m ³	3.7 x 10 ⁻⁵	3.7 x 10 ⁻⁵	3.7 x 10 ⁻⁵
W/Mg	1.8 x 10 ⁻⁵	3.7 x 10 ⁻⁵	1.8 x 10 ⁻⁵
Compressive Strength, psi	1.1 x 10 ⁴	-	1.1 x 10 ⁴
Leak Rate			
Leaching, fraction/y ^a	1x10 ⁻⁴ - 2x10 ⁻³ ^b	>>10 ⁻² ^b	1 x 10 ⁻⁴

^aBased on a monolithic cylinder with dimensions of a 55-gal drum.

^b[BURGER-1983A].

^cBasis for leach rate estimate given in footnote b of Table IV-1.

the first or subsequent extraction cycles. In an FRP operated in the usual manner, however, the I-129 is expected to be released into gaseous streams from which it would be separated and collected in operations preceding the extraction cycles. Therefore, the I-129 collected at a typical FRP would not meet any of these qualifications, being neither spent fuel nor a waste from extraction cycles, and would not be qualified as HLW by 40 CFR 191 Draft. Present U.S. regulations seem to imply that separated I-129 would be managed as LLW. In order for I-129 to be managed as LLW, its primary waste forms, isolated from FRP streams, would have to be diluted with sufficient volumes of matrix (e.g., the cement matrices considered in this report). If the increase in waste volume resulting from this dilution is found to be unacceptable, then management of I-129, in greater concentrations and lesser volumes, by HLW systems might be chosen. A recent American review [CROFF-1983] suggested geologic disposal for I-129. Because of this possibility, the geologic disposal of I-129 wastes is examined here by comparing the properties of I-129 waste forms against criteria on waste form properties and packaging for geologic disposal.

Although U.S. policies and regulations are the focus of this analysis, it is relevant to take note of the disposal methods being recommended for I-129 by other groups. In a recent expression of tentative criteria, the IAEA generally recommended that low- and intermediate-level wastes containing nuclides with long half-lives be disposed of by geologic emplacement [IAEA-1982]. Also, in the recent study by the NEA [NEA] of the management of the nuclides considered here, the disposal options for I-129 were limited to emplacement in the seabed or in a geologic formation.

1. Compatibility of I-129 Waste Form Properties with Criteria for Geologic Emplacement

If I-129 is to be emplaced in a geologic site, its waste forms and containers must be evaluated with respect to the policies, criteria, and regulations that are being developed for HLW.

Although many suggestions have been made on the characteristics that may have to be specified to describe HLW for acceptance at a disposal site (ANL-83-57:Part I, Section IV.C.2), the specifications are still in a state of evolution. The NRC has the ultimate responsibility of determining the criteria for acceptance of waste for geologic emplacement; however, DOE has recently drafted a set of criteria to be used in the interim to guide its efforts in development of waste form, packaging, and repository for geologic emplacement of HLW [ONWI-OOC]. Because criteria on geologic emplacement are not in final form, an assessment of the capability of I-129 waste forms to comply with geologic emplacement criteria must be tentative. Furthermore, the draft criteria seem focused on vitrified, mixed fission products, and do not cover such possibilities as separated I-129. A list of waste form criteria given by NRC [FEDREG 58196] (Table IV-6) and also items from a draft of the criteria prescribed by DOE [ONWI-OOC] (Table IV-7) are used below to illustrate the capability of the I-129 waste forms to meet possible requirements for geologic emplacement. These tables list the waste-form properties that may have to be described in applications for waste disposal.

Table IV-6. Assessment of I-129 Waste Forms against Tentative NRC Criteria for Geologic Emplacement

	Barium Iodate in Cement	Lead Zeolite	Silver Zeolite in Cement
≤1500 W/container (1700 W/m ³)?	Y ^a	Y	Y
Retain performance characteristics up to 500°C?	P	P	PN
Fission product concentration <10 ⁵ Ci/L of waste form?	Y	Y	Y
Release rates after container failure <0.01%/y?	PN	PN	P
Withstand loading of 18 mPa in triaxial compression at emplacement temperature?	P	N	P
Does not contain material that would increase mobility of the radionuclides?	Y	Y	Y
Waste form and container withstand drop of 2 times container height without breaching?	P	P	P
In drop of 2 times container height, are fractions of waste form converted to particles ≤200 micron and ≤10 micron limited?	P	P	P
Canister has sufficient strength to resist breaching in 800°C fire, 15 min?	P	P	P
Release of volatile materials in fire of 800°C, 15 min, is limited?	Y	Y	Y
Waste form not readily soluble in water?	Y	Y	Y
Waste form is a solid that retains its physical shape under repository conditions?	Y	Y	Y
Free liquids released during operational period are limited?	Y	Y	Y
Weight ≤5000 lb per package:	Y	Y	Y
Not combustible in fire 800°C, 15 min?	Y	Y	Y
Any chemicals added to retard fire do not corrode container?	Y	Y	Y
Class A and B explosives (defined by 49 CFR 173) present in subhazardous concentrations?	Y	Y	Y
Class A and B poisons (defined by 49 CFR 173) contained in waste form are present in subhazardous concentrations?	Y	Y	Y
No Class A and B poisons that are not incorporated into waste form?	Y	Y	Y
Package will remain subcritical?	Y	Y	Y

^aY = yes, compliance expected; N = no, noncompliance expected.

P = possible.

PN = probably not, needs testing.

**Table IV-7. Assessment of I-129 Waste Form Properties
by Information Requirements Anticipated
by DOE for Geologic Emplacement of HLW**

	Barium Iodate in Cement	Lead Zeolite	Silver Zeolite in Cement
Isotopic Content, Ci/cc	$2.6 \times 10^{-5}^a$	$2.3 \times 10^{-5}^a$	$1.5 \times 10^{-5}^a$
Chemical Composition and Form	BaIO ₃ ·H ₂ O, 9 vol % in cement	Iodine and iodides fixed on lead zeolite	Iodine and iodides fixed on silver zeolite, incorp. in cement, 9 vol %
Characterization of Phases Containing Radionuclides	Inorganic solid	Inorganic solid	Inorganic solid
Amount of Following:			
Poisons A and B ^b	none	none	none
Pyrophoric materials	none	none	none
Explosives	none	none	none
Flammable inorganic solids	none	none	none
Fire retardant materials	none	none	none
Free liquids	none	none	none
Physical Properties:			
Dimensions, ^c cm	~60 OD, ~90 ht.	~60 OD, ~90 ht.	~60 OD, ~90 ht.
Geometry	cylinder	cylinder	cylinder
Weight, kg	<259	<259	<259
Density, g/cc	2.2 ^d	1.0 ^d	2.1 ^d
Structural support of fissile material	NA ^d	NA	NA
Particle size distribution	monolith	pellets	monolith
Particulate amounts before and after drop test	NT ^e	NT	NT
Impact resistance	NT	NT	NT
Thermal Decomposition Rate	Complete at 660°C	NT	NT
Source Term for Radionuclide Release ^f	1×10^{-4} – 1×10^{-3}	$>> 10^{-2}$	1×10^{-4}
Volatility	none	10^{-6} – 10^{-8} atm pressure @ 20°C	NT
Flammability	none	none	none
Flash Point	none	none	none
Behavior of Fire Retardant Materials	none	none	none
Thermal Properties			
Thermal conductivity	NT	NT	NT
Specific heat	NT	NT	NT
Coefficient of thermal expansion	NT	NT	NT
Transition temperature	NT	NT	NT
Mechanical Properties			
Tensile strength	NT	NT	NT
Compressive strength, psi	$\sim 6.4 \times 10^3$	NA	$\sim 2.6 \times 10^3$
Shear strength	NT	NT	NT
Young's Modulus	NT	NT	NT
Shear modulus	NT	NT	NT
Bulk modulus	NT	NT	NT
Elongation	NT	NT	NT

^aBased on concentrations suggested by RSD information (Table IV-1).

^bPoisons A and B defined by the DOT regulations in 49 CFR 173.

^cAssumes 55-gal drum outer container.

^dNA = not applicable.

^eNT = Needs testing.

^fNeeds further definition; the table entries are leach rate as fraction/y.

The temperature limit of 500°C (Table IV-6) reflects the concern over self-heating of the HLW. Iodine wastes would have much lower heat generation rates than vitrified mixed fission products and, if placed in a geologic site, would probably be isolated from other wastes. Concern about heating from adjacent wastes and self-heating suggests that the temperature of the waste be limited such that steam is not formed from the non-chemically bound water of concrete waste forms and that pressure is not sufficient to cause a breach of the container. Although the entries in Table IV-6 indicate compatibility with a number of the suggested criteria, the only conclusion that seems to follow from this comparison is that, if data to fill all these categories are required, then much testing of waste forms must be carried out before their compliance with these criteria can be determined.

The list of properties in Table IV-7 illustrates how much information may ultimately have to be supplied if one is seeking acceptance for geologic emplacement of I-129. The requirements being developed for geologic emplacement of HLW, although not in final form, promise to be more detailed and specific than those that are available in 10 CFR 61 for SLB.

2. Compatibility of I-129 Packaging with Criteria for Geologic Emplacement

Assessment of the requirements for packaging to contain I-129 waste forms, if they are managed by geologic disposal, must be made against the tentative requirements proposed in concepts, drafts, and interim guidelines. The NRC and EPA have the responsibility for expression of general criteria that must be ultimately observed. These criteria are not prescriptive--details of what is acceptable will be determined in licensing proceedings.

A packaging for I-129 has not been characterized, except in the present work involving DOT specification packaging. These DOT packagings have all involved 55-gal drums. Some characterizations of operation of a geologic repository have assumed that TRU waste would be placed in 55-gal drums, whereas HLW would be placed in canisters of about 2-ft dia and 10-ft length. Iodine-129 would probably be emplaced in 55-gal drums rather than the canisters being designed for vitrified mixed fission products.

Two of the properties listed in Table IV-6 [ONWI-OOC] depend jointly on the waste form and the container: (1) withstanding a drop of two times the container height without breaching and (2) resisting a fire of 800°C for 15 min without breaching. Since there is also a requirement that containers be standardized [ONWI-OOC], a standard container will have to be defined when decisions are made on whether I-129 will go to a repository. It has been suggested [ONWI-OOC] that the list of properties given in Appendix C be supplied for any canister proposed for disposal of HLW. In the present state of uncertainty, it is not possible to judge packaging of any I-129 that would be sent to a repository. The properties given in Appendix C are presented, therefore, only as a possible list that may have to be supplied on the packaging of I-129 if it is sent to geologic emplacement.

F. Compatibility of I-129 Control Technologies with Other
Final Emplacement Modes

Other possible modes for final emplacement of I-129 are examined here because, as indicated in Sections IV. D and E, the desirability of either shallow-land burial or geologic emplacement of separated I-129 wastes that would emerge from fuel reprocessing cannot be definitely judged at this time. Other possibilities--non-near surface, greater confinement, and ocean disposal--are examined below, although they too are associated with uncertainties concerning regulations and criteria.

1. Prospects for Non-Near Surface Disposal of I-129

Criteria for non-near surface disposal (an intermediate-depth concept of NRC for commercial wastes) are not available. In their absence, reference to criteria that have been suggested for a similar disposal method, greater confinement disposal (an intermediate-depth concept of DOE for DOE wastes), seems to be the only basis for any discussion. Greater confinement disposal is still in its infancy, not only in development of concepts of site design, procedures, and technology, but also in development of disposal criteria. Some criteria have been proposed [CARD-1981, HUNTER] as discussed in ANL-83-57:Part I (Section IV.B). These criteria were based on modeling studies of nuclide pathways to the biosphere carried out by Ford, Bacon, & Davis Utah, Inc. The models are being revised and significant changes are expected [CARD-1983]. It is anticipated that intermediate-depth methods will offer the opportunity to dispose of I-129 in concentrations greater than those expressed by the limits in 10 CFR 61. From initial suggestions for GCD criteria (ANL-83-57:Part I, Section IV.B.2) it is inferred that an I-129 concentration four times higher than the SLB limit in an arid site, but seven times lower than the SLB limit in a humid site, might be permitted.

A compliance analysis requires criteria or regulations against which the performance of disposal technology can be judged. In the case of GCD, although primitive criteria have been proposed, they are being revised and are now in a state of such uncertainty that a compliance analysis based on them would not be meaningful.

Thus, it is expected that intermediate-depth disposal of I-129 may offer the advantage over SLB of higher permitted concentrations; however, this expectation cannot be supported by documentation at present. Although what has been considered the reference work on criteria for GCD was discussed in some detail in Part I (Section IV.B), that work is in the stage of being redone and the revision is not yet available. Since it would be misleading to base an evaluation on the earlier work, no detailed assessment of the possible disposal of I-129 can be given at this time. Perhaps the only conclusion possible is that, if SLB will permit disposal of I-129, then it seems reasonable to expect that an intermediate-depth concept, being regarded as a more secure disposal mode, would permit this disposal in higher concentrations, providing the additional convenience and economy of lower volumes.

2. Prospects for Ocean Disposal of I-129

As explained in ANL-83-57:Part I (Section IV.D), the IAEA's response to the LODC [IAEA-1980], along with the policies of the EPA, provide the current direction on ocean disposal for U.S. waste consignors. The IAEA's response [IAEA-1980], therefore, is used here to judge the compliance of ocean disposal of I-129 with regulations. Only low-level waste is acceptable for ocean disposal according to the LODC. Furthermore, the IAEA, in response to the LODC, has defined HLW and, thereby, the materials unacceptable for ocean dumping. For beta-gamma emitters, with half-lives of >0.5 y, the limit is 10^{-4} Ci/g of waste package. All the I-129 waste forms characterized in Table IV-1 with I-129 concentrations ranging from 0.7×10^{-5} to 2.3×10^{-5} Ci/g fall within the LODC (IAEA) criteria for concentration. An assessment of the properties of I-129 waste packages against the general criteria listed by the IAEA is outlined in Table IV-8. Except for the need to qualify or design a container that would retain its contents during descent to the seabed, Table IV-8 gives positive indications that I-129 wastes could comply with most of the general criteria of the LODC (IAEA).

Table IV-8. Assessment of I-129 Waste Forms Based on R&D Recommendations with the General Criteria on Ocean Disposal Developed by IAEA (LODC)^a

	Barium Iodate on Cement	Lead Iodide on Zeolite	Silver Iodide on Zeolite in Cement
I-129 Concentration $<10^{-4}$ Ci/g?	Y ^b	Y	Y
Is waste form either solid or absorbed liquid?	Y	Y	Y
Specific gravity ≥ 1.2 ?	Y	Y	Y
Container capable of retaining contents during descent to depths of 4000 m or more?	P	P	P
Package does not contain buoyant material that, if returned to surface, would constitute a radioactive hazard with or interfere with legitimate uses of the sea?	Y	Y	Y
Radiophase is relatively insoluble in a relatively insoluble matrix?	Y	N	Y

^aFormulated by the IAEA in response to the London Ocean Dumping Convention [IAEA-1980].

^bY = Yes, compliance expected.

P = Probable.

N = No.

The ocean disposal of I-129 has an advantage over SLB because it is much less restrictive on concentrations of I-129. For comparison, the IAEA (LODC) allows 10^{-4} Ci/g for ocean disposal, whereas the I-129 waste forms based on concentrations that would satisfy 10 CFR 61 (Table IV-5) are restricted to 4×10^{-8} to 8×10^{-8} Ci/g. Although the I-129 waste forms seem capable of compliance with most of the criteria of the IAEA (LODC), the policy of the EPA will ultimately decide whether ocean disposal of I-129 by U.S. consignors will be permitted. Recent EPA policy asserts that "permits for ocean disposal will be issued only under the most pressing of circumstances and ... the applicant would be required to make a most compelling demonstration of need before an application would be considered complete" [NORMAN]. In a recent development on ocean disposal of LLW, Congress has passed a two-year moratorium on ocean dumping for the two years 1983-84 [RN]. After that time, a resolution passed by both the houses of Congress will be necessary to obtain a permit. Near-term plans for management of I-129 wastes should not rely on this mode of disposal.

G. Description of Currently Feasible Disposal Systems for I-129

Descriptions for I-129 disposal systems acceptable for transportation and burial are indicated in Table IV-9. This table also points out the need for additional information.

Table IV-9. Directory of Descriptions of Waste Form and Packaging for Disposal of I-129

Criteria	Waste Form Specification	Packaging Specification
Transportation	a	Tables IV-10, IV-11
SLB	Table IV-5	b
Geologic Disposal	Not Defined	Not Defined
GCD or Non-Near Surface	Not Defined	Not Defined
Ocean Dumping	Table IV-1	Not Defined ^c

^aTransportation regulations specify packaging, not waste form characteristics.

^bAt the concentrations specified by Table IV-5, the I-129 waste forms would fall into class C (10 CFR 61), requiring that the waste form or the container be capable of a 300-y lifetime. Guidelines for acceptable waste form and container properties are given in the technical position papers of NRC. At concentrations ≤ 0.008 Ci/m³, the 300-y lifetime is not required, but the volume of matrix needed to produce this low concentration would result in about 3×10^5 55-gal drums per year to dispose of the I-129 from 1500 MTHM fuel.

^cNot defined except for the general requirement that the container must maintain its integrity during descent to ocean depth of at least 4000 m.

Three different waste forms have been proposed for I-129. Two of these utilize zeolites for capture. Two of the final waste forms incorporate cement into the system for immobilization and the third consists of pellets. Optimization of these waste forms with respect to disposal limits results in the packages described in Table IV-10, which fit into a DOT-specification packaging 6L drum (55 gal).

Table IV-10. Specifications of ^{129}I Waste Packages to Meet 10 CFR 61

	$\text{Ba}(\text{IO}_3)_2$	AgZI	PbZI
Capture Form	-	Zeolites	Zeolites
Final Form	Concrete	Concrete	Pellets
Curies/pkg	0.018	0.018	0.018
Size of 2R, cm			
dia	28.4	10.1	8.7
length	12.4	14.9	12.9
Curies/cc ^a	7.9×10^{-8}	7.9×10^{-8}	7.9×10^{-8}
Number of 6L Drums, per 1500 MTHM	2612	2612	2612

^aWithin Spec. 6L Drums.

The waste packages have been maximized in content to meet the 0.08 Ci/m^3 disposal limit. This limit places the packages well within presently allowable transportation quantities. However, one cannot load a drum to the allowable transportation limit and still comply with the disposal limit for SLB.

The I-129 waste packages in Table IV-10 meet the Type A quantity ($<2 \text{ Ci/pkg}$) of DOT and the SLB Type C disposal level ($<8 \times 10^{-8} \text{ Ci/cc}$) of NRC, and require 2600 drums for disposal of the I-129 from processing 1500 MTHM. This is a large number of drums, but comparable volumes of LLW are shipped annually from a single 1000-MW_e reactor [PHILLIPS]. Although the driving force behind the design of these packages is the LLW criteria, disposal at an HLW site would permit a concentration per package up to a DOT Type B quantity. Thus, if the waste forms alone were to limit the curies per 6J container to the values given in Table IV-11, a much smaller number of shipping containers would result. The actual number of drums would probably be greater than that shown in Table IV-11--perhaps up to twice the amount depending on whether a 2R container is required and what its size is.

Table IV-11. Specification of Alternative I-129 Waste Packages

	Ba(IO ₃) ₂ in Cement	Ag ₂ I in Cement	Pb ₂ I
Transportation Type	B	A, B	B
Maximum Curies/pkg	2	1.9	2
Number of 6J Drums, per 1500 MTHM	24	25	24

H. Comparison of the Performance of I-129 Disposal Systems with Regulatory Limits

The effectiveness of I-129 disposal technologies should be compared against the limits expressed in a number of ways in several regulations and criteria, but all the applicable regulations cannot be tested in this way because only some of the modeling studies and dose conversion factors are available. Modeling studies are outside the scope of the study reported here. Information in the literature does, however, allow estimates of dose so that an analysis of the compliance with the general maximum individual dose at the boundary of an FRP can be made. With respect to release criteria, I-129 is specifically cited in 40 CFR 190 and in the concentration guidelines of 10 CFR 20.

1. Release Regulations and I-129 Control

The very small release limit for I-129 specified by 40 CFR 190 (Table IV-12) makes extreme demands on technologies of collection and retention for this nuclide. Furthermore, the assessment of whether the technologies will comply with this limit is affected by three uncertainties.

First, the uncertainty in the nuclide content of spent fuel puts some uncertainty in the estimate of the fraction that must be isolated.

Second, the uncertainty in the distribution of iodine in the plant causes an uncertainty in how many separate iodine-collection technologies must be used in a fuel-reprocessing plant. The distribution of iodine in the plant will depend on how the dissolver is operated [MCKAY]. If the dissolver is operated with reflux,* a major fraction of the iodine is retained in the dissolver, appearing eventually as a very dilute solution in a voluminous evaporator condensate [MCMANUS]. Alternatively, if the dissolver is thoroughly sparged with air, the fraction of I-129 routed to the dissolver off-gas is expected to be >99% [MCKAY]. For the reference fuel reprocessing plant characterized in the environmental impact statement on management of commercially generated radioactive waste [DOE/EIS-0046F], the fraction of

*Not the usual case, but as it is practiced in the ICPP [MCMANUS].

Table IV-12. Fraction of I-129 That Must be Isolated from the Environment as Indicated by 40 CFR 190

Release Limit 40 CFR 190, Ci/GWe-y	Quantity Processed in Spent Fuel, Ci/GWe-y ^a	Fraction ^b that Must be Isolated from Environment (F _{isol})
5 x 10 ⁻³	1.1	0.996

^aBased on expected nuclide inventories in fuel from a 2/1 ratio of PWR/BWR power plants [DOE/NE-0017] with a 1.5-y cooling period.

^bCalculated from the expression of Eq. (II-2), $1 - \left[\frac{\text{release limit}}{\text{quantity processed}} \right]$.

I-129 appearing in the DOG is tabulated with one significant figure as 1. Development work on iodine-collection technology in the U.S., aimed at removal of iodine from gas streams onto solid trapping media, is based on the assumption that a dissolver would be operated in this second mode. Even a small uncertainty in the distribution of iodine, however, is significant in view of the small release permitted by regulations.

Third, the uncertainty of the efficiency of the technology for collecting I-129 from gas streams is especially significant in accurately predicting whether disposal technology will conform to regulations; the efficiency that follows from the release limit of 40 CFR 190 must be somewhere between 99 and 100%. The NEA assessment of airborne waste control technologies concluded that a 99% efficiency for collection of I-129 is possible [NEA]. An assessment of the environmental impact of I-129 released by a fuel reprocessing plant [PALMS] assumed that removal efficiencies for silver zeolite systems would be 99.5-99.9%. Although the iodine collection methods most recently developed have received considerable attention in laboratory testing, their efficiency has not been demonstrated in routine, plant-scale operation.

Another type of release regulation would apply to I-129 if it were managed by HLW disposal. The draft regulations of NRC, proposed for 10 CFR 60, prescribe (without reference to individual isotopes) that after the end of container lifetime releases from the waste form shall be restricted to 10⁻⁵/y. Furthermore, the draft regulations of EPA on HLW management (40 CFR 191) propose that any waste containing nuclides not specifically identified in the tables of Part 191, but with $t_{1/2} > 20$ y should have a rate of release restricted to (1) 500 Ci/1000 MTHM in 10⁴ y for releases defined as reasonably foreseeable* and (2) 5000 Ci/1000 MTHM in 10⁴ y for releases defined as very unlikely.† Since 47 Ci of I-129 is associated with 1500 MTHM

*Definition of reasonably foreseeable release: chance of occurring in 10⁴ y > 1/10².

†Definition of very unlikely release: 1/10² > chance of occurring in 10⁴ y > 1/10⁴.

(Table B-3), these limits would allow an average release of $1.6 \times 10^{-3}/y$ for reasonably foreseeable releases of I-129 and $1.6 \times 10^{-2}/y$ for very unlikely releases of I-129. Analysis of the possibilities of compliance of I-129 control with the release regulations proposed by 40 CFR 191 is complicated by the need to estimate the probability of occurrence and, presumably, to anticipate the type of occurrence, e.g., contact by underground water. A modeling study to make such an analysis is beyond the bounds of the present work. The estimated leach rate of I-129 waste forms described here (Table IV-1) may not provide the leach resistance required by 10 CFR 60,* but seem to be within the limit of 40 CFR 191 Draft. If this shortcoming is borne out and I-129 is routed to geologic emplacement, then waste forms other than ordinary cement would be needed to satisfy the requirement of 10 CFR 60 Draft. Perhaps FUETAP cements with incorporated iodates might be used for additional resistance to leaching.

2. Dose Regulations and I-129 Control

The comparison of the performance of technology for I-129 management with dose regulations should be made for the fuel reprocessing plant and the site of final emplacement. Doses from I-129 incurred at these two facilities are discussed below.

a. General Dose Limits and I-129 Control

The question explored in this section is compliance of I-129 management with the thyroid dose limit to an individual from the fuel cycle operations given in 40 CFR 190, i.e., 75 mrem/y to the critical organ. The dose rates of a fuel reprocessing plant were calculated from Eq. II-9 using the DCF for I-129 listed in Table II-2; the values given in Table IV-13 can be easily compared with the thyroid dose limit specified by 40 CFR 190, 75 mrem/y. The derived values in Table IV-13 show that, if the collection technologies perform at the highest efficiency of which they are claimed to be capable, 99% [NEA], then the releases of I-129 will account for almost one-third of the dose of 75 mrem/y that is permitted to the maximum exposed individual by 40 CFR 190. At 90% collection, the I-129 release would deliver eight times the thyroid dose limit of 40 CFR 190. For complete release, the estimated dose rate is about 25 times the thyroid dose limit. The uncertainties of the DCF values, however, preclude using this result as an argument for iodine collection. Similar calculations show that 96% of the I-129 must be collected to correspond to the maximum individual dose limit of 75 mrem/y. The thyroid dose limit seems to be less demanding of collection technology than is the release limit of 5×10^{-3} Ci/GW_e-y, which requires >99% collection as discussed in Section IV.H.1. This observation is consistent, however, with the general philosophy of EPA, stated in 40 CFR 190, in which dose-rate limitations are aimed primarily at short-lived isotopes, and release-rate limitations are aimed primarily at long-lived isotopes. The EPA philosophy is that release rates of long-lived isotopes should be kept relatively low in fairness to future generations who will experience the effects of the radiation from these isotopes but not the benefits of the energy of which they are the by-products.

*Table IV-1 shows that the estimated leach rate exceeds the $10^{-5}/y$ limit specified by [10 CFR 60].

Table IV-13. Estimated Dose to Maximum Exposed Individual from I-129 at a 1500-MTHM/y Fuel Reprocessing Plant

A, ^a Ci/Plant-y	DCF, ^b mrem/Ci	Maximum Individual Dose, ^c mrem/Plant-y		
		F _{coll} =0	F _{coll} =0.90	F _{coll} =0.99 ^d
47	42	2000	200	20

^aTotal amount of nuclide processed annually by 1500-MTHM/y fuel reprocessing plant.

^bSee Table II-2.

^cCalculated from the expression $A(1 - F_{coll})(DCF)$. This dose is incurred by the thyroid in the [NEA] model.

^dCollection factors of 0.99 are assumed possible by the NEA evaluation [NEA]. This level of efficiency has not been demonstrated.

b. Dose Limits of Emplacement Criteria and Waste Management of I-129

The regulations in 10 CFR 61 are based on the same numerical goals as the regulations in 40 CFR 190, i.e., limiting individual exposure to 25 mrem/y to the whole-body, 75 mrem/y to the critical organ, and 25 mrem/y to any other organ (10 CFR 61 applies to the disposal site, whereas 40 CFR 190 applies to the fuel cycle except for mining, transportation, and waste management). These goals were used by NRC in modeling studies on movement of nuclides through the environment, from which NRC derived the concentration limits and technical criteria expressed in 10 CFR 61 for SLB [EIS]. It is, therefore, assumed here that emplacement of I-129 in an SLB facility in accord with the technical criteria (waste form properties and packaging characteristics) of 10 CFR 61 will achieve the goals on which 10 CFR 61 is based, with the reservation that 10 CFR 61 was not intended to accommodate fuel reprocessing. Thus, if I-129 is emplaced according to the technical criteria of 10 CFR 61 and if the models are realistic, then the goals will be met. As discussed above (Section IV.D), emplacement of I-129 wastes in SLB may not be chosen because of the large volume and number of packages of I-129 wastes that compliance with 10 CFR 61 would require.

The dose-avoidance goals underlying the development, by the EPA, of regulations for geologic emplacement of nuclear waste are expressed in Draft 40 CFR 191, i.e., 10^3 health effects in 10^4 years. It is the responsibility of NRC to develop technical criteria and regulations that will prescribe the practical conditions that would result in meeting these EPA limits. Whether fuel reprocessors will choose to manage I-29 by HLW disposal will depend strongly on their economic goals. The criteria are not in final form, but the absence of restriction on concentration limits and, therefore, the possibility of great reduction in waste volume and number of drums in which the I-129 would be distributed might make management of I-129 as HLW seem attractive compared to SLB as LLW. Some general container requirements, if I-129 were managed by geologic emplacement, can be inferred from the discussion in the last paragraph of Section II.D, e.g., capability of retrieval for 50 y and maintenance of integrity for 1000 y.

I. Conclusions on Disposal of I-129

Conclusions on disposal of I-129 are made in the following sections with respect to compatibility with current regulations, needs for additional R&D on control technology that are apparent from this analysis, and needs for attention to the management of I-129 in any future activities on development or amendment of regulations.

1. Compatibility of Conceptual Systems for I-129 Control with Current Regulations

Collection processes for I-129 are judged by published evaluations [NEA] to be capable of 99% collection or better. Collection efficiencies of 99.5-99.9% have been assumed for silver zeolites [PALMS], and collection efficiencies of 99% have been concluded from a technical assessment of iodine collection capabilities [NEA]. The claims for these high collection efficiencies are not based on plant-scale experience. Whether I-129 collection efficiency on a plant scale can consistently remain at >99% is crucial to determining whether the technology for managing I-129 can comply with the release rate limit of 5×10^{-3} Ci I-129/GW_e-y prescribed by 40 CFR 190. The transportation of I-129 collected in fuel reprocessing operations, in compliance with current transportation regulations, appears practicable. Furthermore, DOT-specification packaging could be used for transportation of this waste. It is questionable, however, whether DOT-specification packaging, although suitable for transportation, will meet criteria for final emplacement. If not, the I-129 waste can probably be overpacked in a container that meets disposal criteria. There are no NRC-licensed containers that have been designed and certified for transportation of I-129 specifically.

Disposal of I-129 wastes at SLB facilities could be done, but the concentration limit of 10 CFR 61 is so low that the waste would have to be distributed among a large volume of matrix and number of containers. Thus, the concentration limits on I-129 imposed by 10 CFR 61 could lead reprocessors to choose management of I-129 by HLW disposal. For HLW, no restrictive concentrations are involved, and the I-129 could be placed in a much smaller number of containers than LLW; the number of 55-gal drums, for example, would be reduced by a factor of 10^2 . However, requirements for release rates from the waste packages and container integrity are economic factors not fully evaluated for deep geologic disposal.

2. R&D Needs for I-129

The >99% control necessary to meet the release limits (5×10^{-3} Ci/GW_e-y) of 40 CFR 190 should be demonstrated on a plant scale. Although opinions indicate that iodine-collection concepts are adequate to meet this level of collection, these concepts have not been shown in practice. Increased knowledge on the distribution of I-129 in plant streams is necessary before it can be concluded that a sufficiently high fraction of total I-129 to satisfy 40 CFR 190 can be collected. In fact, additional information on the distribution of I-129 may show that, if 99% is to be collected, attention must be given to other plant streams besides those that have occupied research efforts, i.e., the DOG stream and, in the case where the DOG is refluxed, the bottom of the liquid waste concentrator. The very restrictive

limit on release of I-129 requires attention to those streams that carry only a very small fraction of the total. Perhaps a complete understanding of the distribution of I-129 in plant streams will only come from experience in full-scale fuel reprocessing. Predicting the distribution in plant streams is more difficult for I-129 than for the other isotopes considered here because of the tendency of iodine to assume several oxidation states and chemical forms.

The planning for scale-up of the collection processes thus far investigated in only laboratory-scale studies would be helped by the conceptual design of equipment to carry out iodine collection on a plant scale. Difficulties of applying the laboratory-scale concepts would become apparent in such design studies, providing information on which collection process and, consequently, which iodine waste forms should be selected for application. If geologic emplacement of I-129 is considered, then waste forms with lower leach rates than those of barium iodate might be required. Since the iodine diffusivity in cemented iodine sodalite is two to three orders of magnitude lower than that in barium iodate [BARNES], its leachability may be more likely to satisfy the regulations for deep geologic disposal. Since the leach rate of AgI in cement is about equal to that of iodine sodalite [SCHEELE], it may also be an acceptable waste form if geologic disposal of I-129 is considered. Regardless of which waste forms are used, needed property values [ONWI-OOC] will have to be determined by experiment, e.g., thermal expansion coefficient, uniaxial compressive strength, bulk moduli.

Since an economic criterion may be the most important factor in future decisions on the final destination of I-129 wastes, a study resulting in estimates of relative costs for disposal of I-129 by SLB, intermediate-depth methods, and deep geologic disposal would provide some predictions of which management scheme fuel reprocessors are likely to prefer.

3. Suggested Amendments or Additions to Regulations

Since tentative suggestions of the IAEA indicate geologic disposal for long-lived, low- or intermediate-level wastes [IAEA-1982], and also since the low concentration of I-129 for SLB as required by 10 CFR 61 may result in this disposal mode being burdensome, geologic emplacement of I-129 probably should not be ruled out. Since geologic emplacement of I-129 separated from spent fuel is a possibility, it should receive some individual attention in regulations for this disposal mode. Whether I-129 must meet the same requirements on waste form and packaging as other items destined for repository emplacement (such as spent fuel, vitrified mixed fission products, and TRU wastes) will depend on the hazards of the I-129 waste forms relative to these other waste forms, their release rates and their behavior in environmental pathways. This question requires clarification in rule making. Furthermore, it has been suggested [ALTOMARE, MCKAY, PROUT, SMITH, BURGER-1983B] that, based on chemistry, concentrations, isotopic ratios, and expected migration pathways, ocean disposal may be the best choice for final emplacement of I-129 wastes. Two factors are common in these recommendations. First, geologic processes over time periods comparable to the half-life of I-129, almost 16 million years, are predicted to transport most of the I-129 placed in geologic formations to the oceans. The environmental impact incurred in this transport could be avoided by initially [SIMON] placing the I-129 in

the ocean. Second, the isotope dilution afforded by the naturally occurring iodine in the ocean would diminish exposure to I-129 placed there. Specific attention should, therefore, be given to this isotope in any future developments of regulations on ocean disposal of radioactive wastes.

V. DISPOSAL OF C-14

Because C-14 has a relatively long half-life, decay during storage plays no part in schemes for its disposal. This nuclide is not addressed in the EPA regulations on exposure from the nuclear fuel cycle that specify a release limit in Ci/GW_e-y. Estimates show (Section V.H.2.a) that if the quantities that would become available in fuel reprocessing were released, the maximum exposed individual would incur only a minor fraction of the dose limit prescribed by 40 CFR 190. The nuclide has been addressed specifically in the NRC regulations for SLB (10 CFR 61) and also in the criteria proposed by EPA for geologic emplacement (40 CFR 191), but the latter is addressed to selected waste origins. Collection processes and the preparation of waste forms have been developed on the laboratory-scale only.

A. Origin of C-14 Waste Forms

The sequence of major operations in derivation of C-14 waste forms from spent fuel is shown in Fig. V-1. For voloxidation or thermal out-gassing, predictions on the fraction of total C-14 that would appear in the off-gas from those operations is not available. The conditions of both these operations probably favor the formation of ¹⁴CO₂, the same species expected if C-14 appears in dissolver off-gas. The several collection processes for C-14, some gas-liquid and some gas-solid reactions, have all involved removal of CO₂ from gas streams in which the main constituent is air. Form-conversion processes involve the incorporation of either CaCO₃ or BaCO₃ into cement. Thus, the only final waste form considered here for C-14 is cement.

B. Characteristics of C-14 Waste Forms Derived from R&D Information

Table V-1 summarizes characteristics of C-14 waste forms derived from R&D information, regardless of regulations. Two types of waste forms are listed: barium carbonate in cement and calcium carbonate in cement. These waste forms were probably developed with the assumptions that they would be treated as LLW and be emplaced in an SLB facility. Disposal regulations appeared, however, after the R&D work that developed these waste forms.

The compositions given in Table V-1, *i.e.*, loading of carbonates in cement matrix, are based on information from R&D carried out at PNL and described in ANL-83-57:Part I (Section III.E.3.a). The carbonates are assumed to make up about 9 vol % of the total waste form. This volume and the literature values for densities of the components are used to determine densities of the waste forms. The nuclide specific activity of C-14 (Table B-1) adjusted for atmospheric carbon isotopes and the composition and density data were used to estimate the specific activities of the waste forms. The C-14 nuclide thermal power (W/Ci, Table B-1) adjusted for atmospheric carbon isotopes and the composition were used to estimate the thermal powers of the waste forms. The maximum dose absorbed by this waste was estimated from the definition of the rad (100 erg absorbed per gram of waste) and the thermal power estimated for the waste forms. The rate of gas generation by radiolysis was obtained from the G values estimated for β radiation by Bibler [BIBLER-1977, -1978] from observed alpha and gamma radiolysis of cement and the cement fraction of the waste. Leach rates were estimated from experiments of PNL [SCHEELE].

Fig. V-1. Operations in Waste Disposal of C-14 Wastes from Fuel Reprocessing. (Additional details of chemistry are in Appendix F.)

Table V-1. Properties of C-14 Waste Forms, Derived from R&D Information

	CaCO ₃ in Cement	BaCO ₃ in Cement
Composition		
Encapsulant	cement	cement
Radiophase	solid CaCO ₃	solid BaCO ₃
C wt % of waste form ^a	1.4	1.1
¹⁴ C atom fraction of total C	7.6×10^{-4}	7.6×10^{-4}
Phase of waste form	solid	solid
Density, g/cc	2.1	2.2
Particles, % of total waste form	0	0
Specific activity		
Ci/cc	1.2×10^{-4}	9.7×10^{-5}
Ci/g	5.6×10^{-5}	4.4×10^{-5}
Thermal loading		
W/cc	3.5×10^{-8}	2.8×10^{-8}
W/g	1.6×10^{-8}	1.3×10^{-8}
Maximum absorbed dose, rad/y	5.0×10^4	4.1×10^4
Radiolytic gas generation rate, cc/g-y	4.1×10^{-4}	3.1×10^{-4}
Compressive strength, psi	7×10^3	7×10^3
Leach rate, fraction/y ^b	$4.5-10 \times 10^{-6c}$	$1-2 \times 10^{-5c}$

^aBased on an air sparge of 100 ft³/min through the dissolver.

^bBased on a monolithic cylinder with dimensions of a 55-gal drum. The loading of carbonate for these leach tests was about twice that used in this table.

^c[BURGER-1983A].

C. Prospects for Disposal of C-14 Waste Forms from the Standpoint of Transportation Regulations

Reviews of the transportation regulations and the properties of C-14 waste forms indicate that, with the use of the appropriate packaging, C-14 wastes could be transported in compliance with the regulations. This conclusion does not provide assurance, however, that the packaging acceptable to regulation and currently available is economically satisfactory. The desirability of using currently available packagings can be judged from the characteristics, given below, of conceptual packages involving cemented carbonates.

1. Classification of C-14 for Transportation

The revisions of March 10, 1983 [FEDREG 10218] to the regulations of the DOT (49 CFR) specify that up to 60 Ci of C-14 in normal form can be transported in Type A packaging. Type B packaging must be used for transportation of quantities larger than 60 Ci.

2. Use of DOT-Specification Packaging for C-14

Several DOT-specification packagings, listed in Table V-2, could carry, in compliance with transport regulations, the C-14 waste forms described in Table V-1. Characteristics of such packages, involving DOT-specification packaging and waste forms with the C-14 concentrations taken from R&D information, are summarized in Table V-3. As for the other airborne nuclides, the number of packages needed to accommodate the volume of C-14 wastes produced by a full-scale fuel reprocessing plant is used as an index of practicability. The restricted volumes of the specification 2R inner container in the 6L(2R) and 6M(2R) packagings result in numerous packages, but the 6J container in configuration A [EDLING] would provide transport of the C-14 wastes with a reasonable number of packages. Although this latter package offers an acceptable way to transport the C-14 wastes, the question of its acceptability for final emplacement would be answered definitely in licensing proceedings.

3. Use of Certified Packaging for C-14

The RAMPAC data base was searched by Sandia National Laboratories for descriptions of Type B packagings specifically certified for transportation of C-14. This search showed that the data base contained no reference to any packaging certified specifically for C-14.

D. Compatibility of C-14 Control Technologies with SLB

Two significant elements of 10 CFR 61 are used here to evaluate the compatibility of C-14 waste management technology with SLB: the criteria on waste form properties and prescriptions for acceptable packaging. Both these elements are treated briefly in 10 CFR 61. In the following sections, the probability of complying with waste-form criteria listed in 10 CFR 61 are qualitatively examined item-by-item, and then the more quantitative characteristics of packages designed mainly to conform to the concentration limits for C-14 of 10 CFR 61 are summarized.

1. Compatibility of C-14 Waste Forms with Criteria for SLB

The compatibility of C-14 waste forms with criteria from 10 CFR 61 for SLB is examined in Table V-4. The only obvious difficulty in compliance with SLB criteria seems to arise from the low concentration of C-14 required by 10 CFR 61. The concentrations of C-14 in the wastes, prepared according to waste-form developer's concepts, i.e., 97-120 Ci/m³ (Table V-1), are greater than the limit for SLB, 8 Ci/m³, according to 10 CFR 61. There are no physical or chemical reasons why waste forms of C-14 could not be prepared to comply with the required concentration limit.

Table V-2. Authorized Packaging Applicable to Transportation of C-14 Waste Forms^a

Authorized packaging ^{b,c} for shipments of radioactive materials in normal form inside U.S.	Allowed Quantities for C-14 per Package for Normal-Form ^d Materials Carried in Designated Packages	
	<60 Ci	>60 Ci
DOT Spec. 7A Type A general packaging (including Spec. 6L, Spec. 6J) ^e	Y ^f	N ^g
Any foreign-made packaging that bears the marking "Type A" and complies with the regulations of the country of origin applicable to Type A packages ^e	Y	N
Any Type B(U) or B(M) packaging that meets the applicable requirements in the regulations of the NRC (10 CFR 71) and that has been approved by that commission	Y	Y
DOT Spec. 6M metal packaging, only for solid or gaseous materials that do not undergo pressure-generating decompo- sition at temperatures up to 121°C and do not generate more than 10 W of decay heat	Y	Y
DOT Spec. 20WC wooden protective jacket, when used with a single, snug-fitting inner DOT Spec. 2R	Y	Y

^aDerived from DOT rules revised March 10, 1983 [FEDREG 10218].

^bAll packaging employing Spec. 55 containers are omitted here. Since only existing Spec. 55 containers can be used, and the construction of new ones is prohibited after Dec. 31, 1975, it is unlikely that such containers would be used in shipments of any airborne waste forms from fuel reprocessing operations.

^cPackaging described in ANL-83-57:Part I (Section IV.E.1.c).

^dNormal form defined in Glossary.

^eType A packagings will be required to pass the more demanding tests defined in 49 CFR 173.465-173.466 after July 1, 1985; some packagings now acceptable may not pass these tests.

^fY = Allowed.

^gN = Not allowed.

Waste forms designed to meet the concentration limit prescribed by 10 CFR 61 for C-14, i.e., 8 Ci/m³, are described in Table V-5. The dilution of wastes by a factor of >10 to fit a disposal method, however, may not be acceptable to licensing bodies.

Table V-3. Number of Containers and Activity Levels of Waste Forms for C-14 (860 Ci/y-1500 MTHM, 1.5-y cool)

Item ^a	CaCO ₃	BaCO ₃
Form	compound	compound
Final Form	in cement	in cement
Total Weight, kg	1.5×10^4	1.9×10^4
Total Volume, cc	7.4×10^6	8.8×10^6
No. of containers/y		
6L (2R)	810	960
6M (2R)	1300	1600
6J (A)	58	73
Curies/container		
6L (2R)	1.06	0.8
6M (2R)	0.64	0.54
6J (A)	15	12
Watts/container		
6L (2R)	3.1×10^{-4}	2.3×10^{-4}
6M (2R)	1.9×10^{-4}	1.6×10^{-4}
6J (A)	4.4×10^{-3}	3.5×10^{-3}
Waste Form Weight/pkg, ^b kg		
6L (~136 kg max) (2R)	18	20
6M (~273 kg max) (2R)	11	12
6J (~259 kg max) (A)	259	259

^a(A) denotes configuration "A" [EDLING]; 6L (2R) denotes DOT specification 6L packaging with 2R inner container. For the estimates given here, the volumes of the 2R containers were assumed to be typical of those that have been used--for the 6L package, $9.2 \times 10^3 \text{ cm}^3$, and for the 6M package, $5.6 \times 10^3 \text{ cm}^3$. The volume of the 6J package, which involves no 2R container, was assumed to be that of a 55-gal drum, $2 \times 10^5 \text{ cm}^3$.

^bAllowable package weight for standard containers given in parentheses.

2. Compatibility of C-14 Packaging with Criteria for SLB

Table V-3 described packages involving three DOT-specification packaging. The concentrations of C-14 could be reduced from those listed in Table V-3 to the limit necessary for SLB, i.e., 8 Ci/m^3 (10 CFR 61). As a result, the number of packages would be increased by more than an order of magnitude over that listed in Table V-3, but this number may still be practicable. The rules of 10 CFR 61 also prescribe that containers or waste forms with concentrations in excess of 0.8 Ci C-14/m^3 * must be capable of

*10 CFR 61 specifies that concentrations in excess of 0.1 times the value in its Table 1 (8 for C-14) fall in class C. This class is subject to the 300-y stability requirement for waste form or container.

Table V-4. Compatibility of C-14 Waste Forms with Criteria for Shallow-Land Burial (10 CFR 61)

	CaCO ₃ in Concrete	BaCO ₃ in Concrete
Sufficient absorbent to absorb twice the volume of any liquid?	Y ^{a,b}	Y ^b
Liquid volume $\leq 1\%$ of total volume of waste when in a stable container or $\leq 0.5\%$ of total waste for a stable waste form?	Y	Y
Not capable of detonation, explosive decomposition, reaction at normal temperatures and pressures, or explosive reaction with water?	Y	Y
Not pyrophoric?	Y	Y
Any gases at pressure ≤ 1.5 atm at 20°C?	Y	Y
Any gases at total activity ≤ 100 Ci/package?	Y	Y
No more than minimal amounts of biological, pathogenic, or infectious hazards?	Y	Y
Structurally stable to effects of overburden weight, moisture, microbial activity, radiation effects, and chemical change, either because of properties of waste form or container?	Y	Y
Concentration ≤ 8 Ci/m ³ ?	c	c

^aY = Yes, compliance expected. N = No, noncompliance expected.

^bAssumes the concrete was prepared to exclude free-standing liquid.

^cThe concentrations of C-14 in the waste forms based on R&D information are greater than this limit (Table V-1). Waste forms could be tailored to comply (see Section V.D.2).

a 300-y lifetime for SLB. The technical position papers of NRC, presenting guidelines for waste form and container properties, specify that the waste form must be capable of the ANS standard [ANS 16.1] on leachability of solidified LLW. It seems likely that the cement waste form can meet this test. If not, high-integrity containers, defined also by guidelines in the technical position papers, might be used to overpack the DOT-specification containers. It is reasonable to expect that, if designed for a 300-y life and the transportation criteria, the resulting package would be accepted for both transportation and shallow land burial. In this discussion, it is assumed that the preparation of C-14 waste forms with concentrations of < 0.8 Ci/m³, which qualifies as class A waste (300-y lifetime not required), would be undesirable, since this practice would spread the C-14 among a large volume of waste and a large number of packages.

Table V-5. Properties of C-14 Waste Forms with Radioactive Concentrations Limited to Comply with 10 CFR 61

	CaCO ₃ in Cement	BaCO ₃ in Cement
Composition		
Encapsulant	cement	cement
Radiophase	solid CaCO ₃	solid BaCO ₃
C wt % of waste form	1.1×10^{-1}	1.0×10^{-1}
¹⁴ C atom fraction of total C	7.6×10^{-4}	7.6×10^{-4}
Phase of waste form	solid	solid
Density, g/cc	2	2
Particles, % of total waste form	0	0
Specific activity		
Ci/cc	8.0×10^{-6}	8.0×10^{-6}
Ci/g	4.0×10^{-6}	4.0×10^{-6}
Thermal loading		
W/cc	2.3×10^{-9}	2.3×10^{-9}
W/g	1.2×10^{-9}	1.2×10^{-9}
Maximum absorbed dose, rad/y	3.7×10^3 ^a	3.7×10^3 ^a
Radiolytic gas generation rate, cc/g-y	3.5×10^{-5}	3.5×10^{-5}
Compressive strength, psi	1.1×10^4	1.1×10^4
Leach rate, fraction/y ^b	$4.5-1 \times 10^{-6}$ ^c	$1-2 \times 10^{-5}$ ^c

^aThe NRC technical position paper suggests a limit of 10^8 rads in 300 y.

^bBased on a monolithic cylinder with dimensions of a 55-gal drum.

^c[BURGER-1983A].

E. Compatibility of C-14 Control Technologies with Geologic Disposal

The rules of 10 CFR 61 seem to be the sole location of regulations on disposal of separated C-14. Although C-14 is cited in 40 CFR 191 Draft, defining concentrations above which certain nuclides should be disposed of as HLW, this definition is to be applied only to spent fuel or to wastes from the first or subsequent extraction cycles. In the operations of a typical FRP, however, the C-14 in spent fuel is expected to be converted to CO₂ that is incorporated into gaseous streams, from which it would be separated and collected before the extraction operations are carried out. Thus, the C-14 collected in typical operations would not be spent fuel, nor would it be a waste from extraction cycles. Further, regardless of concentration,

it could not be defined as HLW under the proposed rules of 40 CFR 191 Draft. From these considerations, it is inferred that present U.S. regulations intend that separated C-14 be managed by LLW disposal systems. To qualify as LLW, the primary waste forms of C-14 would require some dilution, e.g., by the cement matrices considered here. If either economics or policies on waste-form preparation determine that the increased volume attending this dilution is unacceptable, HLW disposal might be chosen for C-14 wastes in the form of materials of higher concentration and lower volume. Consequently, disposal of C-14 wastes is evaluated here by examining the properties of C-14 waste forms in the light of criteria for deep geologic disposal, relevant to waste form properties and packaging.

A recent American review [CROFF-1983] suggested geologic disposal for C-14. Although this analysis is concerned with the regulations that U.S. waste disposers would be committed to, consideration of the disposal methods being considered for C-14 in foreign studies is worthwhile. The tentative general recommendation by the IAEA [IAEA-1982] of geologic disposal for all low- and intermediate-level wastes of long half-life is applicable to C-14. Furthermore, the NEA study of management of these wastes [NEA] limits the methods of disposal of C-14 to "sub-surface shafts," seabed emplacement, and geologic emplacement. Also, a recent British report on the management of C-14 wastes [BUSH] considers a number of disposal options, including discharge to environment, ocean-bed emplacement, and geologic emplacement. Although this report [BUSH] suggests that geologic disposal of C-14 may have some advantages, it does not present a recommendation.

1. Compatibility of C-14 Waste Form Properties with Criteria for Geologic Emplacement

The possibility of management of C-14 wastes by HLW disposal at concentrations greater than 8×10^{-6} Ci/g apparently has not been incorporated into the planning of DOE or NRC. Therefore, there is little information available that can be used to judge the acceptability of geologic emplacement of this nuclide. Although separate criteria are being developed for geologic emplacement of spent fuel and vitrified mixed fission products, regulatory agencies may decide that these criteria should not apply to geologic emplacement of separated C-14 wastes. At this time, a compliance analysis of the possible geologic emplacement of C-14 can only be done by comparison against general criteria that have been drafted for geologic disposal [ONWI-OOC, 40 CFR 191 Draft]. The criteria expressed in [ONWI-OOC] were assembled by DOE contractors to guide DOE development activities until regulations become available from NRC and EPA. An analysis of the prospects of C-14 waste forms satisfying these DOE guidelines is summarized in Tables V-6 and V-7. The tables illustrate some of the information that must be supplied if geologic emplacement of C-14 in the form of carbonates incorporated into cement is done and if the suggested EPA criteria for repository regulation are adopted. The data requirements for geologic disposal will be much more detailed and demanding than for SLB if the information suggested by the items in Tables V-6 and V-7 is demanded.

Table V-6. Assessment of C-14 Waste Forms against Tentative Criteria for Geologic Emplacement

Tentative Criteria	Judgment of Capability of Complying with Criterion	
	Calcium Carbonate in Cement	Barium Carbonate in Cement
≤1500 W/container (1700 W/m ³)?	Y	Y
Retain performance characteristics up to 500°C?	P	P
Fission product concentration ≤10 ⁵ Ci/L of waste form?	Y	Y
Release rates after container failure ≤0.01%/y?	N	N
Withstand loading of 18 mPa in triaxial compression at emplacement temperature?	P	P
Does not contain material that would increase mobility of the radionuclides?	Y	Y
Waste form and container withstand drop of two times container height without breaching?	P	P
In drop of two times container height, are fractions of waste form converted to particles ≤200 micron and ≤10 micron limited?	P	P
Canister has sufficient strength to resist breaching in 800°C fire, 15 min?	P	P
Release of volatile materials in fire of 800°C, 15 min, is limited?	Y	Y
Waste form not readily soluble in water?	Y	Y
Waste form is a solid that retains its physical shape under repository conditions?	Y	Y
Free liquids released during operational period are limited?	Y	Y
Weight ≤5000 lb per package?	Y	Y
Not combustible in fire 800°C, 15 min?	Y	Y
Any chemicals added to retard fire do not corrode container?	Y	Y
Class A and B explosives (defined by 49 CFR 173) present in subhazardous concentrations?	Y	Y
Class A and B poisons (defined by 49 CFR 173) contained in waste form are present in sub- hazardous concentrations?	Y	Y
No Class A and B poisons that are not incorporated into waste form?	Y	Y
Package will remain subcritical?	Y	Y
Y = yes, compliance expected.		
N = no, noncompliance expected.		
P = probably, needs testing.		

Table V-7. Assessment of C-14 Waste Forms Properties by Information Categories That May be Demanded for Geologic Emplacement as HLW

	Calcium Carbonate in Cement	Barium Carbonate in Cement
Isotopic Content, Ci/cc	$1.2 \times 10^{-4}^a$	$9.7 \times 10^{-5}^a$
Chemical Composition and Form	CaCO ₃ , 9 vol % in cement	BaCO ₃ , 9 vol % in cement
Characterization of Phases Containing Radionuclides	Inorganic solid	Inorganic solid
Amount of Following:		
Poisons A and B (49 CFR 173)	none	none
Pyrophoric materials	none	none
Explosives	none	none
Flammable inorganic solids	none	none
Fire retardant materials	none	none
Free liquids	none	none
Physical Properties:		
Dimensions, cm	~60 OD, ~90 ht ^b	~60 OD, ~90 ht ^b
Geometry	cylinder	cylinder
Weight, kg	<259	<259
Density, g/cc	2.1 ^a	2.2 ^a
Structural support of fissile material	NA ^c	NA
Particle size distribution	monolith	monolith
Particulate amounts before and after drop test	NT ^a	NT
Impact resistance	NT	NT
Thermal Decomposition Rate at T	stable	stable to 1000°C
Source Term for Radionuclide Release, fraction/y	~10 ⁻⁴	~10 ⁻³
Volatility	calcite, not in cement, begins to lose CO ₂ at ~500°C	BaCO ₃ , not in cement begins to lose CO ₂ at ~1000°C
Flammability	none	none
Flash Point	none	none
Behavior of Fire Retardant Materials	none	none
Thermal Properties:		
Thermal conductivity	NT	NT
Specific heat	NT	NT
Coefficient of thermal expansion	NT	NT
Transition temperature	NT	NT
Mechanical Properties:		
Tensile strength	NT	NT
Compressive strength, psi	7 x 10 ³ , NT	7 x 10 ³ , NT
Shear strength	NT	NT
Young's modulus	NT	NT
Shear modulus	NT	NT
Bulk modulus	NT	NT
Elongation	NT	NT

^aBased on concentrations suggested by R&D information (Table V-1).

^bAssumes 55-gal drum outer container.

^cNA = not applicable. NT = Needs testing.

2. Compatibility of C-14 Packaging with Criteria for Geologic Emplacement

Although several packages for C-14 have been identified (Table V-2) as acceptable to transportation regulations, current information is insufficient to decide whether these packages would also be acceptable for geologic emplacement, since repository criteria are still in development. Hence, it is difficult to judge the acceptability for geologic emplacement of a given container listed in Table V-2. With the criteria and regulations drafted by EPA and NRC only in initial stages of development, DOE contractors have suggested a list of data (see Appendix C) to guide the design of containers in the interim.* There has been no consideration, however, of a container specifically designed for C-14 waste forms. In other concepts of repository management, only two container types were suggested: (1) canisters for vitrified mixed fission products and (2) 55-gal drums for TRU. If C-14 were treated as HLW, it would probably be contained in 55-gal drums rather than in the canisters being designed for vitrified mixed fission products. Two criteria in Table V-6 refer to container performance, i.e., the drop test and the fire test. Another significant criterion by which a C-14 container would be judged is the lifetime. A criterion on lifetime, debated at some length, is that the container must retain its contents for 1000 y. The usual carbon-steel (55-gal) drum is unlikely to meet this criterion, and it is unlikely that any of the packagings listed in Table V-2 would retain their integrity for even 300 y, unless the container material was selected for survival in repository conditions. The criterion that containers be standardized should narrow the choice of packagings. It seems likely that the same type of container could serve for both I-129 and C-14 waste forms. At the repository, contact handling† should be possible for both C-14 and I-129 waste containers, unless the contamination by other fission products is high enough that it raises the gamma emission rate to unacceptable levels. The possible data requirements on containers of waste to be accepted for geologic emplacement, listed in Appendix C, are no more than guidelines for the information that may be demanded to describe containers for geologic emplacement.

F. Compatibility of C-14 Control Technologies with Other Disposal Modes

Two other modes for final emplacement of C-14 are examined here; the best destination for C-14 wastes is not entirely clear. Greater confinement disposal, as a representative of an intermediate-depth method, and ocean disposal are considered, though criteria for these disposal modes are still unsettled.

1. Prospects for Intermediate-Depth Disposal of C-14

Initial concepts of GCD were reviewed in ANL-83-57:Part I (Section IV.B), because of the potential of an intermediate-depth system for disposal of some of the airborne wastes. A considerable effort is being

* See Part I, Tables IV-4, IV-5, and IV-6 for concepts involving HLW canisters.

† Manual handling, not requiring remote control equipment or special shielding. Surface dose is ≤ 200 mrem/h.

supported by DOE [LARGE] to develop GCD. An intermediate-depth disposal mode called non-near surface disposal is also in the process of being defined by NRC. The GCD concept is still being defined with respect to site characteristics, technology, operating procedures, and criteria for acceptance of wastes. Hence, a compliance analysis is difficult because technical criteria against which the characteristics of waste forms and management technology can be judged are not available.

A concentration limit for C-14 was cited in initial suggestions by a DOE contractor [CARD-1981] on criteria for greater confinement disposal. Reinforcing the expectation that this disposal concept would permit concentrations greater than those permitted by SLB, these initial criteria permit a C-14 concentration of five times the SLB limit in an arid facility and three times the SLB limit in a humid facility.

These suggested concentration limits for nuclides to be emplaced in GCD facilities were based on modeling studies of nuclide pathways to the biosphere. Revision of the model, now in progress, is expected to result in changes in these concentration limits [CARD-1983]. Thus, although greater confinement disposal may offer greater economy and safety for disposal of C-14 wastes than SLB, uncertainties in the feasibility of the concept and in the rules governing it prevent a detailed analysis from being undertaken at this time.

2. Prospects for Ocean Disposal of C-14

Since the U.S. position on ocean disposal of LLW follows the principles of the LODC and the policies of the EPA, this analysis of possible compliance of ocean disposal of C-14 is based on the criteria defined by the IAEA in response to the LODC [IAEA-1980]. Only LLW can be considered for ocean disposal now, and the IAEA has defined concentration limits of LLW for this mode of disposal. By these rules, C-14 could be emplaced on the seabed in concentrations up to 10^{-4} Ci/g of waste. This limit would permit ocean disposal of waste forms described by R&D data (Table V-1), with concentrations of about 4 to 6×10^{-5} Ci C-14/g. Thus, for waste forms with the density of cement, C-14 could be disposed of by ocean disposal in a concentration about 10 times greater than that acceptable for SLB. Estimates of how much this advantage is offset by the higher costs of ocean disposal, compared to SLB, are outside the bounds of this study but should be evaluated.

Judgments of the ability of C-14 waste packages to meet the criteria listed by the IAEA are summarized in Table V-8. Of the items examined in Table V-8, only the need to identify or design a container that would hold its cargo during descent to at least 4000 m would prevent C-14 waste disposal technology from meeting the criteria of the IAEA (LODC).

The moratorium of two years on ocean disposal, recently passed by Congress [RN], and the requirement that a permit after the moratorium expires would need a joint resolution by both Houses of Congress, present additional deterrents to the ocean disposal of C-14. Because the opinion among U.S. consignors of LLW is that ocean disposal costs more than SLB, and because the attitude of the EPA is that permits for ocean disposal would be granted for

Table V-8. Assessment of C-14 Waste Forms Based on R&D Recommendations against the General Criteria for Ocean Disposal Developed by IAEA (LODC)^a

	Calcium Carbonate in Cement	Barium Carbonate in Cement
C-14 concentration $<10^{-4}$ Ci/g?	Y ^b	Y
Is waste form either solid or absorbed liquid?	Y	Y
Specific gravity >1.2 ?	Y	Y
Container capable of retaining contents during descent to depths of 4000 m or more?	c	c
Package does not contain buoyant material that, if returned to surface, would constitute a radioactive hazard or interfere with legitimate uses of the sea?	Y	Y
Radiophase is relatively insoluble or in a relatively insoluble matrix?	Y	Y

^aFormulated by the IAEA in response to the London Ocean Dumping Convention [IAEA-1980].

^bY = compliance expected.

N = noncompliance expected.

^cIt is assumed that compliance could be achieved through appropriate design.

very special needs only, the chances that this disposal mode would be practiced are slim. Because of these impediments, ocean disposal of C-14, although some advantages are promised and technical barriers to its implementation are outmoded, is unlikely to be practiced in the foreseeable future.

G. Summary of Currently Feasible Disposal Systems for C-14

Table V-9 is a directory of descriptions of conditions that could be used to comply with the criteria for the transportation and disposal of C-14. Although qualitative measures are readily expressed, quantitative descriptions of C-14 waste form and packaging have a very limited basis in experience and must, therefore, be accepted with the reservation that they are liable to change. These descriptions are not presented as the only means of C-14 disposal, nor are they meant to imply that technological and regulatory developments are complete. In fact, one of the functions of Table V-9 is to point out the need for additional data.

Table V-9. Directory of Descriptions on Feasible Waste Form and Packaging for Disposal of C-14

Criteria	Waste Form Specification	Packaging Specification
Transportation	a	Table V-10
SLB	Table V-5	b
Geologic Disposal	Table V-1	Not Defined
GCD or Non-Near Surface	Not Defined	Not Defined
Ocean Dumping	Table V-1	Not Defined ^c

^aTransportation regulations contain specifications for packaging, but waste form specifications are made only in the sense of defining categories to which packaging types are assigned.

^bAt the concentrations specified by Table V-5, the C-14 waste forms would fall into class C (10 CFR 61), requiring that the waste form or container be capable of a 300-y lifetime. Guidelines for acceptable waste-form and container properties are given in the technical position papers of NRC. At concentrations $\leq 0.8 \text{ Ci/m}^3$, the 300-y lifetime is not required, but the volume of matrix needed to produce this low concentration would involve more than 5×10^3 (55-gal) drums per year to dispose of the C-14 from 1500 MTHM.

^cNot defined except for the general requirement that the container must maintain its integrity during descent to ocean depth of at least 4000 m.

The two waste forms proposed for C-14, CaCO_3 and BaCO_3 , are each to be immobilized in a cement matrix placed in a container for final disposal. Adaptation of these waste forms to disposal limits results in the packages described in Table V-10, involving a 6L packaging (55-gal drum).

The designs in Table V-10 meet the Type A quantity of 49 CFR ($<60 \text{ Ci/pkg}$) for transportation and the Type C of 10 CFR 61 ($<8 \times 10^{-6} \text{ Ci/cc}$) for SLB, resulting in ~478 drums for each waste form. This number seems reasonable, although economic analyses are required to fully justify this conclusion.

H. Comparison of the Performance of C-14 Disposal Systems with Regulatory Limits

Regulations expressing or implying goals in the form of dose limits, release limits, health effect limits, or concentration guidelines for C-14 control have been searched. Regulations written in terms of dose rates require modeling studies, and some of these related to C-14 have been done. Limits for C-14 are explicit only in the concentration guides of 10 CFR 20 and in the standards for HLW disposal recently drafted by EPA [40 CFR 191 Draft]. Comparison of goals or limits with expected technology performance are made here by comparing estimated doses from a fuel reprocessing plant employing three levels of collection efficiencies with the general dose

Table V-10. Possible C-14 Waste Packages

Waste Form	CaCO ₃	BaCO ₃
Final Form	cement	cement
Curies/pkg	1.8	1.8
Size of 2R, cm		
dia	25.1	23.5
length	37.2	34.7
Curies/cc ^a	7.9×10^{-6}	7.9×10^{-6}
Number of Drums per 1500 MTHM	478	478

^aCalculated for DOT-Specification 6L drum.

limits of 40 CFR 190. Because information is limited on the behavior of C-14 in the environmental pathways of a fuel reprocessing plant, the only parameter that can be estimated at present is the whole-body dose to the maximum exposed individual.

1. Release Regulations and C-14 Control

There is no specific regulation in 40 CFR 190 on the release of C-14 from the fuel cycle similar to the regulations for I-123 and Kr-85. If C-14 is managed by HLW disposal, however, the release regulations specified by 40 CFR 191 and 10 CFR 60 are assumed to apply. The regulations of NRC, 10 CFR 60, include a release-rate regulation that after the end of container lifetime, at least 1000 y, the releases from the waste form shall be restricted to $10^{-5}/y$.^{*} Furthermore, [40 CFR 191 Draft] specifically suggests a release rate limit for C-14 if it is managed by HLW disposal. These release limits are (1) <200 Ci C-14/1000 MTHM in 10^4 y for releases defined as reasonably foreseeable and (2) <2000 Ci C-14/1000 MTHM in 10^4 y for releases defined as very unlikely.[†] On the basis that 860 Ci of C-14 is associated with 1500 MTHM (Table B-3), one can calculate that these limits allow an average release of $\sim 3.5 \times 10^{-5}/y$ for reasonably foreseeable releases and $\sim 3.5 \times 10^{-4}/y$ for very unlikely releases. The estimated leach rates of the cement waste forms described in Table V-1 are in

^{*}"Performance of the engineered system (waste package and underground facility) following permanent closure is specified to require containment of the wastes within the waste package for at least 1000 y following closure..." Waste package means airtight, watertight, sealed container which includes the waste form and any ancillary enclosure including shielding, discrete backfill, and overpacks. Release from the engineered system after the first 1000 y into the geologic setting must be $<1/100,000$ [10 CFR 60].

[†]See Section IV.H.1 for definitions of "reasonably foreseeable" and "very unlikely" releases.

the same range as the leach-rate limits required by 10 CFR 60 and 40 CFR 191 Draft. If the ordinary concrete waste forms cannot meet these criteria, perhaps FUETAP concrete would. The performance of FUETAP concrete with incorporated carbonates would require evaluation.

2. Dose Regulations and C-14 Control

Dose limits in 40 CFR 190 represent the maximum dose that an individual should receive from the entire fuel cycle, except for mining, transportation, and waste management. That this maximum dose would be incurred by an individual residing at the boundary of one of the fuel cycle facilities is a scenario commonly used. Therefore, compliance with 40 CFR 190 should be analyzed with respect to the fuel reprocessing plant. The modeling work carried out by an expert group representing the NEA on the behavior of airborne wastes released at a hypothetical fuel reprocessing plant [NEA] provides the basis for calculating a dose that can be compared with the regulatory limit.

Comparing an estimated dose with regulations to analyze the compliance with limits on doses incurred from final emplacement facilities is not so easily carried out. In this case, the information needed for readily estimating dose is not available. Instead, a judgment is made on which set of technical criteria a disposer of C-14 wastes is likely to find convenient and, therefore, which disposal method associated with that set of technical criteria is likely to be selected.

a. General Dose Limits and C-14 Control

Compliance of C-14 management with the appropriate dose limit of 40 CFR 190, i.e., 25 mrem/y to the whole body, is examined here. This limit can be interpreted as the dose to the maximum exposed individual at a single fuel cycle facility, as mentioned in the previous section. This analysis, therefore, compares the dose calculated for C-14 discharged from a fuel reprocessing plant at assumed levels of collection efficiency with the dose limit of 25 mrem/y. The dose estimates are obtained by the aid of a published dose conversion factor [NEA] and represent the dose to an individual residing at the boundary of a hypothetical, generic fuel reprocessing plant that is releasing C-14 in amounts that depend on the efficiency of the collection technology.

Three dose estimates are derived using Eq. II-9 with the three levels of efficiency, the 99% level representing the highest efficiency claimed by developers of C-14 collection processes and also by the technical assessment of an independent group of experts [NEA]. Two other estimates are also made, based on a 90% assumed collection efficiency and also on no collection. The dose estimates reflect the particular model of environmental behavior of C-14 at a hypothetical fuel reprocessing plant used by the NEA group [NEA].

The three calculated dose rates of C-14 from a fuel reprocessing plant, ranging from 0.033 to 3.3 mrem/y, are to be compared with 25 mrem/y, the whole body dose limit of 40 CFR 190. Although the estimated dose from C-14, even in the case of complete release, is much smaller than

the dose-rate limitation, this observation should not be used to make any conclusion about whether collection should be carried out or will be required. This caution is based on two considerations: (1) the dose-rate estimate has a large uncertainty; and (2) any limit set on C-14 by the EPA, although one does not presently exist, may be relatively restrictive (see the general philosophy on limiting the release of long-lived isotopes discussed in Section IV.H.2.a). Other bases that might be used for justifying C-14 control technology are the general demands of ALARA and the BACT policy of the Clean Air Act.

The results in Table V-11 suggest that, if the collection technologies for C-14 perform at the highest assumed efficiency, 99%, then the releases of C-14 will account for only a small fraction of the dose limit of 40 CFR 190. Even if only 90% of the C-14 is collected, the dose would still be a small fraction of the limit set by 40 CFR 190. For that matter, the estimated dose for complete release of C-14 is slightly more than 10% of the limit expressed in 40 CFR 190. It is concluded that no compliance problems in reference to dose regulations for C-14 can be foreseen by these results.

Table V-11. Estimated Dose to Maximum Exposed Individual from C-14

A, ^a Ci/Plant-y	DCF, ^b mrem/Ci	Maximum Individual Dose, ^c mrem/Plant-y		
		F _{coll} =0	F _{coll} =0.90	F _{coll} =0.99 ^d
860	3.8 x 10 ⁻³	3.3	0.33	0.033

^aTotal amount of nuclide processed annually by 1500-MTHM/y fuel reprocessing plant.

^bUpper bound of the dose conversion factor (see Table II-2).

^cCalculated from Eq. II-2.

^dCollection factors of 0.99 are assumed possible by the technical assessment of an NEA group. This level of efficiency has not been proved at plant-scale.

b. Dose Limits of Emplacement Criteria and Waste Management of C-14

Since information needed for a simple estimate of dose rate from C-14 released from sites of final emplacement is not available, compliance is analyzed indirectly. The basic dose limit of 10 CFR 61 is the same as that of 40 CFR 190: 25 mrem/y to the whole body. This limit and the results of modeling studies on movement of C-14 through the environment were factored, by NRC, into development of the concentration limits and other technical criteria of 10 CFR 61 for SLB. Therefore, if C-14 wastes are emplaced in an SLB site in accord with the requirements on waste form properties, packaging, and emplacement location of 10 CFR 61, then performance following emplacement should satisfy the goals on which 10 CFR 61 is founded. The question of whether the technology complies with regulations is transformed

into the question of whether a waste disposer chooses to be obligated to the set of technical criteria attached to a disposal mode, in this case, SLB. Waste disposers may choose SLB for C-14 as a practical possibility; if the technical criteria listed in 10 CFR 61 are sufficient, the dose limits underlying this regulation should be met.

The goals chosen by EPA as a basis of regulations for geologic emplacement, summarized in 40 CFR 191 Draft, are expressed not in units of dose but health effects. It is assumed that management of C-14 wastes by HLW systems in accord with the technical criteria of 40 CFR 191 will ensure meeting the goal of incurring less than 10^3 health effects in 10^4 y. Then, in analogy to the consideration of SLB above, the question to be examined is to which set of technical criteria (and thus which disposal mode) will the waste disposer choose to be obligated. The answer to this question is subject to change with the development of the criteria, now in draft form. If the information on waste forms suggested by Table V-7 is required, much information on the waste forms considered here remains to be determined before the answer can be given. On the other hand, these waste forms seem to be just on the borderline of meeting the leach-rate criteria. Thus, if C-14 must be emplaced in a repository, some other waste form may have to be developed before it becomes obvious whether geologic emplacement has any advantage over SLB for C-14 disposal.

I. Conclusions on Disposal of C-14

Conclusions on disposal of C-14 are grouped into three categories: compatibility with current regulations and criteria, needs for additional R&D on technology, and needs for regulatory amendment.

1. Compatibility of Conceptual Systems for C-14 Disposal with Current Regulations

The conceptual collection processes for C-14 are considered to be capable of satisfying current requirements for limiting the environmental release of C-14. The requirements, however, are not stringent; the estimated maximum individual dose resulting from complete release of C-14 is only a minor fraction of the whole body dose limit given by 40 CFR 190.

The transportation of C-14 wastes is not expected to present compliance problems. Although there are no containers designed and licensed especially for the transportation of C-14 wastes, transportation of C-14 wastes collected from fuel reprocessing in DOT-specification packaging seems practicable.

2. R&D Needs for C-14 Disposal

The conceptual technologies for C-14 control were judged, by the NEA group of experts, to be capable of collecting 99% of the C-14 from FRP gas streams. Although the collection technology has had much attention in experimental work during the past decade, except for collection of CO_2 by fluorocarbon absorption [LITTLE], it has not been pursued beyond laboratory-scale testing. If C-14 wastes are to be placed in a repository as HLW and if the data requirements for HLW forms suggested by Table V-7 are eventually

adopted, then the table indicates the chemical and physical properties of the waste forms that will require measurement. If C-14 is to be managed as HLW, the waste forms considered here may not be acceptable because their leach rates are in the same range as the limits; then other waste forms and the processes by which they would be prepared must be developed.

3. Suggested Amendments or Additions to Regulations

Regarding disposal of C-14 wastes by SLB, the concentration limit of 10 CFR 61 would require distributing the C-14 among a large number of containers. If economics or policies indicate that SLB involves an impracticably large number of drums, then the disposal of C-14 in a geologic repository might be considered. With geologic disposal of C-14, involving no limiting concentration, waste could be disposed of in a much-reduced number of containers. The number of drums of C-14 for SLB disposal is not so great as to obviously preclude this disposal mode. Whether SLB would be chosen for C-14 by waste disposers may be decided by economic analysis. Since this is a borderline case, two possibilities for accommodating C-14 by LLW disposal could be explored in regulatory developments: (1) raising the concentration limit for C-14 in SLB and (2) providing a concentration limit for C-14 in GCD or non-near surface disposal that would make this mode more attractive than HLW management.

VI. DISPOSAL OF Kr-85

Although control of the Kr-85 from fuel reprocessing will be necessary for compliance with the release limit of 40 CFR 190, criteria for final emplacement of nuclear wastes do not give specific guidance on the disposal of this nuclide. Nevertheless, the development of collection processes for Kr-85 has advanced to the pilot scale. Satisfaction of not only final emplacement criteria but also transportation regulations will be difficult due to the large amounts of Kr-85 liberated in fuel reprocessing and the incomplete development of the packaging technology.

A. Origins of Kr-85 Waste Forms

The sequence of major operations in the derivation of Kr-85 waste forms from gaseous fuel reprocessing streams is shown in Fig. VI-1. A large fraction of the Kr-85 in spent fuel may be released into the shear off-gas, which will be combined with the off-gas from subsequent operations. Any Kr-85 remaining in sheared fuel will be released during dissolution to the dissolver off-gas stream. The main process in Kr-85 collection involves separation of gaseous krypton from air. The subsequent separation of krypton from xenon is an additional option. The operations for collection of Kr-85 considered here involve (1) absorption in a liquid fluorocarbon, (2) organic condensation and distillation, and (3) adsorption on solids. The form-conversion processes involve incorporation into microcavities of zeolite pellets or into deposits of sputtered metal. No conversion process, except compression, is involved if the gaseous form is chosen for waste management. Thus, the forms considered for Kr-85 are gas, gas in metal, and gas in inorganic solid.

B. Characteristics of Kr-85 Waste Forms Derived from R&D Information

Table VI-1 summarizes characteristics of Kr-85 waste forms. These data were obtained either directly from published R&D information or by estimates from R&D results. The data were assembled on the basis of the physics and chemistry of the waste forms, and do not reflect the requirements of regulations. The three waste forms listed--compressed gas, sputtered metal, and zeolite--were probably developed without plans for any particular final emplacement mode, but mainly with the goal of immobilizing and retaining krypton. The characteristics of the compressed gas form are those for 90% krypton. The composition of Kr-85 collected from a fuel reprocessing plant will vary with the collection process used. For the present study, however, it is considered sufficient to deal with a single composition chosen as 90% krypton and argon and xenon plus trace elements. The composition of the sputtered metal waste form is based on studies at PNL [TINGEY-1982A, -1982B], and the composition of the zeolite waste form is based on studies at Exxon Nuclear Idaho [CHRISTENSEN]. The design and size of sputtering chambers were taken from the conceptual study by the Ralph M. Parsons Co. [PARSONS].

C. Prospects for Disposal of Kr-85 Waste Forms from the Standpoint of Transportation Regulations

A review of the transportation regulations and the characteristics of Kr-85 waste forms suggests that compliance with transport regulations may be more difficult for Kr-85 than for any of the other airborne wastes.

Fig. VI-1. Operations in Disposal of Kr-85 Wastes from Fuel Reprocessing. (Additional details of chemistry appear in Appendix G.)

Table VI-1. Properties of Kr-85 Waste Forms Derived from R&D Information

	Compressed Gas	Sputtered Metal	Zeolite Monolith
Composition			
Radiophase	gas	gas	gas
Encapsulant	none	Ni-La (9:1)	zeolite 5A ^a
Kr wt % of waste form	90 ^b	13 ^c	16 ^d
Kr-85 atom fraction of total Kr	6.0x10 ⁻²	6.0x10 ⁻²	6.0x10 ⁻²
Phase of waste form	gas	solid	solid
Density, g/cc	3.7x10 ⁻³ (STP) 1.2x10 ⁻¹ (25°C, 500 psia) 4.6x10 ⁻¹ (25°C, 2000 psia)	8.1 ^e	1.4 ^{e, f}
Particles, % of total	0	0	0
Specific Activity			
Ci/cc	8.1x10 ⁻² (STP) 2.5 (25°C, 500 psia) 1.0x10 ¹ (25°C, 2000 psia)	25	5.3
Ci/g of waste form	2.2x10 ⁻¹	3.1	3.8
Thermal loading			
W/cc	1.2x10 ⁻⁴ 3.8x10 ⁻³ (25°C, 500 psia) 1.5x10 ⁻² (25°C, 2000 psia)	3.8x10 ⁻²	8.1x10 ⁻³
W/g of waste form	3.3x10 ⁻²	4.7x10 ⁻³	5.8x10 ⁻³
Maximum absorbed dose, rad/y	1.0x10 ¹¹	1.5x10 ¹¹	1.8x10 ¹⁰
Leak Rate			
Leaching	-	No major release on exposure to water ^g	None observed in storage under water ^g
Volatility	-	0.5% in 10 y @ 350°C.	0.03% in 10 y @ 350°C.

^aZeolite is an alkali-metal aluminosilicate. The designation 5A indicates that the microcavities that trap krypton have a diameter of 0.5 mm.

^bBalance argon and xenon plus trace elements.

^cBased on 34 cm³ (STP) Kr/g sputtered metal waste form [TINGEY-1982A].

^dBased on 50 cm³ (STP) Kr/g zeolite [CHRISTENSEN].

^eWeight per volume of waste form, not volume of package.

^fZeolite compacted by HIP processing.

^gCarefully regulated observations on leach rate have not been carried out.

For shipment of Kr-85 in gaseous form, regulations would be readily met if the gas were at 1 atm pressure. However, the comparatively large amounts produced in fuel reprocessing would occupy inconveniently large volumes ($\sim 10^5$ L per 1500 MTHM processed) at 1 atm pressure. As discussed below, shipment at higher pressures would require more sophisticated technology to comply with transport regulations.

The sputtered-metal waste forms would release the Kr-85 in an accident involving fire. Therefore, transportation of this waste form requires a package design to meet the fire test (800°C, 30 min) of 49 CFR since the waste form will not. The data of [PENZHORN] suggest that the zeolite waste form, which does not release the bulk of its Kr-85 loading below temperatures of $\sim 900^\circ\text{C}$, may be able to meet the fire test of 49 CFR.

Although the transportation of Kr-85 is discussed here, it must be noted that the criteria of none of the final emplacement modes will accommodate the disposal of Kr-85. Hence, no destination for Kr-85 is offered by disposal regulations at this time.

1. Classification of Kr-85 for Transportation

According to the regulations of the DOT, recently revised [FEDREG 10218], Type A packaging can carry a maximum of 5 Ci of Kr-85 in compressed form. The regulations further stipulate that Type A packaging can carry a maximum of 1000 Ci of Kr-85 in uncompressed form. Quantities larger than these must be transported in Type B packaging. Table VI-2 summarizes the DOT regulations on which packagings are allowed for given quantities of Kr-85 in compressed and uncompressed forms. The DOT regulations for uncompressed Kr-85 are assumed to apply to the zeolite and sputtered-metal waste forms.

2. Use of DOT-Specification Packaging for Kr-85

In Table VI-3, characterizations of Kr-85 in DOT-specification packagings are presented (1) to illustrate the range of properties expected from these packagings and (2) to permit evaluation of the consequences of their use. For zeolites, because 6J and 6L containers would be limited to 10^3 Ci ^{85}Kr , the 6M container would seem the most practical. For sputtered metal, the restricted volume of the inner 2R container in the 6L and 6M packagings is too small to receive the sputtering chambers proposed and the curie limit on the 6J container requires a large number of packages. The design and licensing of a special packaging to carry the sputtered metal waste form seemed to be needed. Although the DOT-specification packagings could be used for carrying the solid waste forms, sputtered metal, and zeolites, they are unlikely choices for carrying Kr-85 in gas form. Consequently, Table VI-3 also depicts the packaging of the gas form in cylinders suitable for containing gas at 500 and 2000 psig.

3. Use of Certified Packaging for Kr-85

The RAMPAC data base, searched by Sandia, lists two certified packagings for transportation of Kr-85, both designed to carry gaseous krypton. Not surprisingly, no certified packagings are on record for the solid waste forms of Kr-85 since they are recent developments. For disposal of the relatively large amounts of Kr-85 from fuel reprocessing, one packaging is definitely too small; the other, although not designed for full-scale application, might be utilized if it were necessary.

Table VI-2. Authorized Packaging Applicable to Transportation of Kr-85 Waste Forms^a

Authorized packaging ^{c,d} for shipments of radioactive materials in normal form inside U.S.	Quantities of Kr-85 per Package for Normal-Form ^b Materials Carried in Designated Package			
	Compressed		Uncompressed	
	≤ 5 Ci	> 5 Ci	≤ 1000 Ci	> 1000 Ci
DOT Spec. 7A, Type A general packaging (including Spec. 6L, Spec. 6J) ^e	Y ^f	N ^g	Y	N
Any foreign-made packaging that bears the marking "Type A" and complies with the regulations of the country of origin applicable to Type A packaging ^e	Y	N	Y	N
Any Type B(U) or B(M) packaging that meets the applicable requirements in the regulations of the NRC (10 CFR 61) and that has been approved by that commission	Y	Y	Y	Y
DOT Spec. 6M metal packaging, only for solid or gaseous materials that do not undergo pressure-generating decomposition at temperatures up to 121°C and do not generate more than 10 W of decay heat	Y	Y	Y	Y
DOT Spec. 20WC wooden protective jacket, when used with a single, snug-fitting inner DOT Spec. 2R	Y	Y	Y	Y

^a Derived from DOT Rules revised March 10, 1983 [FEDREG 10218].

^b Normal form defined in Glossary.

^c All packaging employing Spec. 55 containers are omitted here. Since only existing Spec. 55 containers can be used and the construction of new ones is prohibited after Dec. 31, 1975, it is unlikely that such containers would be used in shipments of any airborne waste forms from fuel reprocessing operations.

^d Packaging described in ANL-83-57:Part I.

^e Type A packagings will be required to pass the more demanding tests defined in 49 CFR 173.465-173.466 after July 1, 1985. Some packagings now acceptable may not pass these tests.

^f Y = Allowed.

^g N = Not allowed.

Table VI-3. Characteristics of Packages of ^{85}Kr Wastes
(1.25×10^7 Ci/y-1500 MTHM, 1.5-y cool)

Item ^a	Compressed Gas	Sputtered Metal	Zeolites
Gas Volume, L (STP)	1.5×10^5		
Total Waste Weight, kg	-	4.1×10^3	3.3×10^3
Number of Cyl.			
500 psig	100	-	-
2000 psig	25	-	-
Curies (W)/Cyl.			
500 psig	1.2×10^5 (1.8×10^2)	-	-
2000 psig	4.6×10^5 (7.0×10^2)	-	-
No. of Containers/y			
6L (2R)	-	-	1.2×10^4
6M (2R)	-	-	1.9×10^3
6J (A)	-	1.2×10^4	1.2×10^4
Curies (W)/Container			
6L (2R)	-	-	1.0×10^{3b} 1.5
6M (2R)	-	-	6.6×10^3 (10) ^c
6J (A)	-	1.0×10^{3b} 1.5	1.0×10^{3b} 1.5
Waste Weight/Pkg, ^d kg			
6L (~136 kg max) (2R)	-	-	0.27
6M (~273 kg max) (2R)	-	-	1.7
6J (~259 kg max) (A)	-	0.34	0.27

^a(A) denotes configuration "A" [EDLING]; 6L (2R) denotes DOT specification 6L packaging with 2R inner container. For the estimates given here, the volumes of the 2R containers were assumed to be typical of those that have been used--for the 6L package, 9.2×10^3 cm³, and for the 6M package, 5.6×10^3 cm³. The volume of the 6J package, which involves no 2R container, was assumed to be that of a 55-gal drum, 2×10^5 cm³.

^b6L and 6J packagings limited to quantities $\leq A_2$ for non-fissile materials, or 1×10^3 Ci in the case of Kr-85.

^c6M packaging limited to ≤ 10 W per package.

^dAllowable package weight for standard containers given in parentheses.

One package listed by the RAMPAC data base, certificate No. DOT SP 5958, is certified for a relatively small quantity of Kr-85 in gaseous form. This packaging adapts a standard 220 ft³ oxygen cylinder within a cylindrical lead-shielded cask. An outer container has a double steel wall, with phenolic foam and oak in the annular space between walls. The limits of 1 Ci and 1 W for this package make it inapplicable to managing the 1.25×10^7 Ci produced annually by a 1500 MTHM/y fuel reprocessing plant.

The other packaging for Kr-85 listed by the RAMPAC data base, certificate No. USA/5958/B, is certified for carrying 74,000 Ci of Kr-85 in gaseous form. This packaging consists of a DOT-specification 3A gas cylinder carried in a lead-filled shield within phenolic foam insulation and a spacing structure of tubular steel. The limit of 74,000 Ci per cylinder permits disposal of the Kr-85 throughput of a 1500 MTHM/y fuel reprocessing plant in 170 packages. Although this packaging was not designed to be used in full-scale fuel reprocessing, the management of 170 packages per year seems within reason.

D. Compatibility of Kr-85 Control Technologies with SLB

The compatibility of Kr-85 waste forms with criteria for SLB is not examined here against the technical criteria of 10 CFR 61, as was done for the other three airborne nuclides, because the discussion of comments preceding 10 CFR 61 states that "the requirements of Part 61 do not contemplate the disposal of millions of Curies of Kr-85 as suggested by the DOE." This action was taken by NRC in the belief that "there is ample time," presumably before fuel reprocessing is instituted, "to assess the still-emerging technology for krypton fixation and establish suitable disposal requirements through future technical guidance or rulemaking action." The relatively low limits permitted by 10 CFR 61 for emplacement of any gaseous wastes in SLB sites (1.5 atm at 20°C and 100 Ci/pkg) would not efficiently accommodate the large quantities from fuel reprocessing. The discussion preceding 10 CFR 61 recognizes the solid waste forms developed for Kr-85 storage, but refrains from any specific regulations on them.

E. Compatibility of Kr-85 Control Technologies with Geologic Emplacement

Early concepts of repository management, in advance of the development of regulations, have pictured transport of krypton containers to a repository, where they are buried [CRANDALL]. Recent developments in criteria, however, do not suggest that Kr-85 would be managed as HLW. The criteria for management of HLW recently presented by EPA [40 CFR 191 Draft] are based on the philosophy that nuclides with half-lives <20 y would not be managed by HLW disposal.* This position would exclude Kr-85, with its half-life of 10.7 y, from management as HLW. Furthermore, no other criteria or regulations being developed at this time for geologic emplacement refer specifically to disposal of Kr-85. Disposal of Kr-85 in a geologic repository has no regulatory basis now, and there is no indication that any is expected for the future. In short, no basis exists for a compliance analysis of the possibility of geologic emplacement of Kr-85 at this time.

*Deep Geologic Disposal of Kr-85 is not precluded, however, by 10 CFR 60; moreover, it contains the statement "...the Commission does not know what other radioactive wastes, if any, will be emplaced."

F. Compatibility of Kr-85 Control Technologies with Other Modes of Emplacement

The preceding sections have shown that accommodation of Kr-85 from fuel reprocessing by SLB is impossible by present regulations. Further, based on criteria for geologic emplacement, receipt of Kr-85 wastes at a repository is not expected. Since these two main modes of final emplacement of radioactive wastes do not accommodate Kr-85, consideration of other disposal modes is appropriate. Accordingly, the prospects of disposal by the three other modes of the five possibilities in Fig. I-1 are briefly reviewed below. Even if one of these eventually proves to be the best disposal mode for Kr-85, the conditions and procedures for the disposal operation and acceptance for emplacement in the disposal facility are indefinite at this time.

1. Prospects for Intermediate-Depth Disposal of Kr-85

The initial proposals for criteria for greater confinement disposal [CARD-1981] recommended acceptable concentrations for "mobile" isotopes, including H-3, C-14, and I-129. Neither these proposed criteria nor other recent discussions of greater confinement disposal [LARGE], however, have referred to any possibility of accepting Kr-85 wastes. There is presently no basis for assessing disposal of Kr-85 against even proposed criteria for non-near surface or greater confinement disposal.

2. Prospects for Ocean Disposal of Kr-85

The uncertainties in making an assessment of compliance with regulations on ocean disposal for any radionuclide have been noted in previous sections. The only regulatory basis available now for ocean disposal is the set of rules devised by the IAEA in response to the instructions of the LODC [IAEA-1980] and supported by the U.S. Thus, ocean disposal of Kr-85 is assessed by comparing the properties of the Kr-85 waste forms described in Table VI-1 against significant items of the IAEA criteria for ocean disposal. This comparison is presented in Table VI-4.

At least two conclusions are evident from Table VI-4, both disfavoring the ocean disposal of Kr-85 from fuel reprocessing. First, the IAEA criteria require that waste forms must be either solids or liquids absorbed in solids. This requirement excludes ocean disposal of the gaseous waste form, even though feasibility studies [VAN DALEN] have recommended it. The sputtered-metal and the zeolite forms, however, are assumed here to qualify as solids. Second, although the sputtered-metal and zeolite forms may meet the requirement that the waste form be solid, their Kr-85 concentrations, in forms prepared in R&D efforts (Table VI-1), are factors of more than 10^4 higher than the limit for β - γ emitters with $t_{1/2} > 0.5$ y set by the IAEA. The preparation of sputtered-metal or zeolite waste forms containing Kr-85 concentrations that are a factor of 10^4 lower than those shown in Table VI-1 would require the Kr-85 released in fuel reprocessing be confined in impracticably large volumes of solid matrix.

It appears, therefore, that ocean disposal of the Kr-85 from fuel reprocessing in compliance with the regulations currently effective for nations of the OECD is impracticable.

Table VI-4. Assessment of Kr-85 Waste Forms Based on R&D Recommendations with the General Criteria on Ocean Disposal Developed by IAEA (LODC)^a

Criteria	Waste Form		
	Compressed Gas	Sputtered Metal	Zeolite
Kr-85 Concentration $<10^{-4}$ Ci/g?	N ^b	N	N
Is waste form either solid or absorbed liquid?	N	Y	Y
Specific gravity >1.2 ?	Y ^c	Y	Y
Container capable of retaining contents during descent to depths of 4000 m or more?	d	d	d
Package does not contain buoyant material that, if returned to surface, would constitute a radioactive hazard or interfere with legitimate uses of the sea?	Y ^e	Y	Y
Radiophase is relatively insoluble or in a relatively insoluble matrix?	N	Y	Y

^aFormulated by the IAEA in response to the London Ocean Dumping Convention [IAEA-1982].

^bY = yes, compliance expected, N = noncompliance expected.

^cSpecific gravity of total package.

^dIt is assumed that design of an appropriate container is possible.

^eAssumes that Kr-85 released at the seabed would dissolve and return to the surface only by convection, as a dilute solution.

3. Prospects for On-Site or Interim Storage of Kr-85

The preceding sections have shown that transportation of Kr-85, in the amounts expected from full-scale fuel reprocessing, presents some difficulties. It was concluded that a final emplacement mode for Kr-85 is not offered by disposal criteria for either SLB or geologic emplacement. Furthermore, disposal of Kr-85 has not been addressed in planning for non-near surface or greater confinement disposal, and disposal of Kr-85 in compliance with current regulations for ocean disposal was considered to be impracticable. Thus, the mode of disposal of Kr-85, including final emplacement, is not obvious. One option is to avoid transportation and any requirement not associated with the fuel reprocessing plant site by providing for storage

on the site. This arrangement seems attractive for a short-lived nuclide ($t_{1/2} = 10.7$ y for Kr-85) since a significant fraction of it would decay during the lifetime of the plant. Furthermore, it has been suggested [DOE/ET-0028] that onsite storage be practiced as an interim measure; the waste would be stored until decay of a sufficient fraction of it would permit release of the remainder to the environment, or until decommissioning, employing sufficient dilution by the atmosphere to observe the limits of 10 CFR 20 for atmospheric concentrations. Onsite storage of Kr-85 as a compressed gas would fit the concept of interim storage.

Two concepts of onsite storage of Kr-85 have been suggested. The first concept involves above-ground engineered storage of compressed gas in cylinders within a concrete structure [WARNECKE, DOE/ET-0028, PROUT]. This concept has been criticized as being too costly to be practicable [BROWN-1980]. It does, however, offer the possibilities of the control that would be needed for release after several years of storage as described above. The second concept involves stacking cylindrical waste containers into dry wells or caissons vertically oriented underground [KLETT]. The study of Klett [KLETT] shows this concept to be technically feasible and less costly than above-ground, engineered storage.

The criteria or regulations that are likely to apply to onsite storage are not readily identified; some might develop in response to a proposal to practice the method. In the absence of any regulations specifically directed at onsite storage, the sections of the Code of Federal Regulations applying to operation of a fuel reprocessing plant, 10 CFR 50, contain the only applicable guidelines presently available. These guidelines are found in Appendix F of 10 CFR 50, "Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities." The main instructions for waste management in these regulations apply to high-level waste. One brief guideline, however, applies to other types of waste: "A design objective for fuel reprocessing plants shall be to facilitate decontamination and removal of all significant radioactive wastes at the time the facility is permanently decommissioned." If this policy is applied to the onsite storage of Kr-85, it follows that the Kr-85 would have to be either released to the environment or shipped to some final emplacement facility by the time of decommissioning. The EPA regulation [40 CFR 190] tolerating the release of only 50,000 Ci Kr-85/GW_e-y is assumed to apply to release after interim storage of gaseous Kr-85 at a fuel reprocessing plant that is decommissioned 40 y after it begins to operate. Simple calculations indicate that disposal of the Kr-85 from a 1500-MTHM fuel reprocessing plant probably cannot be completed in accord with this 50,000 Ci Kr-85/GW_e-y limit. A conceptual schedule for Kr-85 disposal by this method from a 2000 MTHM/y fuel reprocessing plant was suggested [DOE/ET-0028] and proposed that the disposal be completed by 80 y after startup. Appendix F of 10 CFR 50 also promises that further criteria for the extent of decontamination will be developed at a later date.

The concern for corrosion of the primary container of Kr-85 by the rubidium produced in its radioactive decay was briefly reviewed in ANL-83-57: Part I (Section III.A.3.h). This problem is not important for relatively short time periods after packaging and, therefore, is probably not of concern for transportation planning. It is a concern, however, that requires choice of suitable materials for containment of Kr-85 either in final emplacement or in interim onsite storage.

G. Summary of Currently Feasible Disposal Systems for Kr-85

In view of the lack of provision for Kr-85 in criteria for final emplacement of radioactive wastes (see Table VI-5), it must be concluded that disposal of the large amounts of Kr-85 from fuel reprocessing cannot be carried out at the present time. Krypton-85 has been shipped via gas cylinders, modified 3A bottles, at pressures of no more than 2 atm. Descriptions of the possible waste forms are summarized in Table VI-6 with the allowable levels based only on transportation criteria alone.

Based on this information, onsite storage using any of the methods seems feasible, with proper selection of rubidium-corrosion resistant materials. Gas cylinders at 500 psi would require ~100 cylinders, and sputtered metal contained in a 6J packaging would require 1.2×10^4 drums. Zeolites would require a minimum of 1.9×10^3 6M packages depending on requirements for a DOT-specification 2R container and its size (see Appendix G).

Table VI-5. Directory of Descriptions on Waste Form and Packaging for Disposal of Kr-85

Criteria	Waste Form Specification	Packaging Specification
Transportation	a	Table VI-6
SLB	Not Defined	Not Defined
Geologic Disposal	Not Defined	Not Defined
GCD or Non-Near Surface	Not Defined	Not Defined
Ocean Dumping	Not Feasible ^b	Not Defined

^aTransportation regulations specify packaging, not waste form characteristics.

^bThe criteria on ocean dumping [IAEA-1980] do not permit gaseous waste forms and would require concentrations four orders of magnitude lower than those proposed for solid waste forms (Table VI-1). Therefore, ocean dumping of the amounts of Kr-85 resulting from fuel reprocessing seems ruled out.

H. Comparison of the Performance of Kr-85 Disposal Systems with Regulatory Limits

From a consideration of Section II.F, it is apparent that goals against which the performance of Kr-85 disposal technologies should be compared are contained in 40 CFR 190. This section of the code prescribes the limit on release of Kr-85 from the nuclear fuel cycle. It also prescribes the whole-body dose limit to which estimated doses can be compared.

Table VI-6. Proposed Kr-85 Waste Packages

	Gas	Gas in Sputtered Metal	Gas in Zeolites
Container	Gas cylinder (500 psi)	Pipe & caps (25 x 30 cm ^a)	Spec. 2R (22.7 x 33.5 cm ^a)
Drums	No	55-gal (Spec. 6J)	55-gal (Spec. 6M)
Curies/Pkg	1.2×10^5	1.0×10^3	6.6×10^3
Number of cyl- inders or drums to meet DOT rules	100	1.2×10^4	1.9×10^3

^aDiameter by length.

^bCompared to transport regs. 49 CFR 173, Type B quantity.

1. Release Regulations and Kr-85 Control

According to Table VI-7, the fraction of Kr-85 that must be isolated to meet the release limit of 40 CFR 190 is about 87%. The simple chemistry of Kr-85 causes little uncertainty about its distribution in FRP streams, a factor that contributes to collecting a high percentage of it. Furthermore, the technology of collection of Kr-85 is more advanced than for any of the other airborne wastes. Although not proved on a plant scale, a collection efficiency of 99% is assumed by technical assessments [NEA]. This efficiency is much better than the 87% that would be required for adherence to 40 CFR 190 by collection processes applied in the processing of 1.5-y cooled fuel.

Table VI-7. Fraction of Kr-85 That Must Be Isolated from the Environment for Compliance with the Release Limit of 40 CFR 190

Release Limit 40 CFR 190, Ci/CW _e -y	Quantity Processed in Spent Fuel, ^a Ci/CW _e -y	Fraction ^b that Must be Isolated from Environment (F _{isol})
5×10^4	3.3×10^5	0.87

^aBased on expected nuclide inventories in fuel from a 2/1 ratio of PWR/BWR power plants [DOE/NE-3017], with a 1.5-y cooling period.

^bCalculated from the expression of Eq. (II-2), $1 - \left[\frac{\text{release limit}}{\text{quantity processed}} \right]$.
Being dependent on the quantity processed, the fraction quoted in this table applies only to the conditions of footnote a.

2. Dose Regulations and Kr-85 Control

For the other airborne nuclides, comparison of performance of the technology with dose regulations is considered at two facilities, the fuel reprocessing plant and the site of final emplacement. In the case of Kr-85, a quantitative comparison can be made at the fuel reprocessing plant but not the site for final emplacement.

In analogy to the treatment of the other airborne nuclides, a published value for the dose conversion factor is used to calculate doses (by Eq. II-9) from Kr-85 releases for three collection efficiencies. The result, given in Table VI-8, is three estimates of the maximum individual whole body dose incurred from three Kr-85 quantities released from a hypothetical fuel reprocessing plant. The 99% efficiency is the highest that Kr-85 collection technology is assumed to be capable of by evaluators of those processes [NEA]. Two other estimates, based on lower collection efficiencies, are also made. The resulting dose estimates reflect the assumptions of the NEA modeling study on environmental behavior of Kr-85 at a hypothetical fuel reprocessing plant. The three estimates of expected dose rates of Kr-85 from a fuel reprocessing plant range from 0.025 to 2.5 mrem/y, much lower than the 25 mrem/y whole body dose of 40 CFR 190.

Table VI-8. Estimated Dose to Maximum Exposed Individual from Kr-85

A, ^a Ci/Plant-y	DCF, ^b mrem/Ci	Maximum Individual Dose, ^c mrem/Plant-y		
		F _{coll} =0	F _{coll} =0.90	F _{coll} =0.99 ^d
1.25 x 10 ⁷	2.0 x 10 ⁻⁷	2.5	0.25	0.025

^aTotal amount of nuclide processed annually by 1500-MTHM/y fuel reprocessing plant.

^bDose conversion factor (see Table II-2).

^cCalculated from the expression $A(1 - F_{coll})(DCF)$, where F_{coll} is the fraction of the nuclide in spent fuel matrix that is collected and packaged for disposal.

^dCollection factors of 0.99 are assumed possible by the NEA evaluation [NEA]. They have not been proved in plant-scale operation.

The results presented in Tables VI-7 and VI-8 suggest two conclusions: Kr-85 collection technology must have an efficiency of at least 87% to satisfy the 50,000 Ci/GW_e-y prescription of 40 CFR 190, but its use to satisfy the 25 mrem/y whole body dose limit of 40 CFR 190 is not indicated by the result derived here. The calculation is not offered as an argument for or against collection of Kr-85 because of the admitted uncertainty [NEA] in the DCF's. The above conclusion that the release-rate limit is more restrictive than the dose-rate limit is not surprising, in view of the discussion in Section IV.H.2.a of the EPA philosophy of setting more conservative limits for release rates than for dose rates.

I. Conclusions on Disposal of Kr-85

The results of our attempts to assess the compliance with regulations of the technologies that would manage Kr-85 wastes from fuel reprocessing are grouped, in the following sections, under three categories: compatibility with current regulations, needs for additional R&D, and suggestions for regulatory amendment. Krypton-85, having been generally ignored in the development of criteria for final emplacement, presents more difficulties than any of the other airborne wastes in the definition of a basis for reaching conclusions on the compatibility of a proposed disposal system with regulations. Development of technology for collection of Kr-85 has been given considerable attention, though planning and criteria writing for final emplacement has not; there are some definite needs for additional development work if the technology is to be brought to the stage where it could be considered ready and available.

1. Conclusions on Compatibility of Conceptual Systems for Kr-85 Disposal with Current Regulations

The efficiencies of the processes proposed for collection of Kr-85 promise to be more than adequate, but have not been proved on plant scale for the control required by current regulations. The maximum individual dose resulting from uncontrolled release of all the Kr-85 in spent fuel would be only a small fraction of the dose limit in 40 CFR 190, but the curie release limit requires retention of about 87% of the total.

Transportation of the large amounts of Kr-85 produced in fuel reprocessing would present some difficulties in complying with regulations. The use of DOT-specification packaging for shipment of the quantities of Kr-85 produced in fuel reprocessing would not be practicable for the gas form, but it does seem practicable for the solid forms. One of the shielded packagings authorized for shipment of Kr-85 by NRC certification, although not designed for full-scale use, might be utilized for disposal of the large quantities from full-scale processing operations, but may not be acceptable for final disposal.

Disposal of any of the Kr-85 from fuel reprocessing by SLB would not be possible at this time, regardless of the waste form chosen. The discussion of comments preceding 10 CFR 61 explains that the disposal of the quantities of Kr-85 formed in fuel reprocessing has not been addressed in the development of these regulations. Only minimal amounts could be disposed of in SLB in accord with 10 CFR 61. Some future provision for disposal of Kr-85 by this means, however, is not discounted, as the preamble to 10 CFR 61 suggests that there is ample time (before fuel reprocessing gets under way) to assess the proposed waste forms.

The perceived difficulties in meeting the transportation regulations of Kr-85 in the form of compressed gases, zeolite, or sputtered metal suggest that storage on the site of the fuel reprocessing plant is the most attractive concept for current planning. The heat associated with decay of Kr-85 is expected to require special provisions, to be derived in design work, for cooling in any storage situation. Without such provisions, the temperature from decay heat could keep the temperatures of the solids at levels sufficiently high to

cause a significant rate of decomposition, releasing gas into the container and resulting in a reversion of the waste form to simply a compressed gas. Placing Kr-85 waste forms in an annular shape has been suggested [KLETT] as one means of reducing the temperature without external cooling. In the case of the sputtered metal, the waste form is normally deposited in the shape of an annulus, but structuring the zeolite in the shape of an annulus requires a double-walled container. Onsite storage of cylinders in engineered concrete bunkers above ground and emplacement as a stack of cylinders underground in wells have been suggested [KLETT].

The disposal of Kr-85 in a geologic repository seems unlikely at this time; one reason is that management by HLW disposal is not being advocated for nuclides with half-lives of less than 20 y.

2. R&D Needs for Kr-85 Disposal

The collection technologies have been tested on increasingly larger scales and on simulated plant streams, but have not been proved on a plant scale. Demonstrations of collection on larger scales and on more realistically simulated plant streams will be needed. The compatibility of the product of the collection process with subsequent operations for preparation of the final waste form should be demonstrated. A detailed estimate of composition of the product of fluorocarbon absorption that includes all impurities does not seem to be available.

If the solid waste forms of Kr-85 are eventually recognized as candidates for disposal, then additional data characterizing the waste forms will be required. The leach rates of the solid waste forms of Kr-85 have been observed in only very brief tests. Although much attention has been given to volatilization of Kr-85 from sputtered metal and zeolite waste forms, the behavior in contact with water is not well characterized. The requirements will be defined by the technical criteria in the regulations applying to the mode of disposal selected.

Krypton-85 is the only one of the airborne wastes that will require shielded packaging, even when uncontaminated by other fission products. The need for shielding is not an unusual problem, but it does introduce the need for design work. The question of whether a shielding should be buried with the waste or removed and re-used will also enter design considerations.

3. Suggested Amendments or Additions to Regulations

The discussion of comments preceding 10 CFR 61 only suggests that the possibility of SLB for the Kr-85 from fuel reprocessing would be considered in future rule-making activities. A re-assessment of part 61 to include rules for SLB of Kr-85 is recommended. The need to decide on an appropriate limit for Kr-85 wastes requires attention also by rule-making bodies involved in formulation of acceptance criteria for greater confinement and non-near surface disposal. Changes in regulations that might make ocean disposal of Kr-85 feasible would require increases in allowed concentrations by 10^4 for the sputtered metal and zeolite waste forms at the highest Kr-85 concentrations they are capable of. Rule-making bodies on ocean disposal should consider revising concentration limits for separated nuclides, such as Kr-85, based on the individual elements, in place of the present system of

applying blanket limits to all β - γ emitters of $t_{1/2} > 0.5$ y. Until it can be shown, however, that Kr-85 waste forms can meet the requirements of transportation regulations (especially that the waste forms and packagings can meet the fire test), it would be difficult to transport Kr-85 wastes to any type of disposal site. Although regulations may not be necessary, in view of the small number of FRP's that may be built, guidelines that would apply to storage of Kr-85 at the site of the fuel reprocessing plant should be formulated if planning of the onsite storage option is to proceed any further.

APPENDIX A.

APPLICABLE REGULATIONS

Regulations prepared by NRC (10 CFR), EPA (40 CFR), DOE (Orders), and DOT (49 CFR) that may apply to disposal of airborne wastes were discussed briefly in the data base document (ANL-83-57:Part I). An overview of regulations that may apply and a summary of their areas of governance are presented here.

1. 10 CFR 20, Standards for Protection Against Radiation

The rules of 10 CFR 20 establish standards for protection of individuals against radiation in all activities licensed by the NRC. Therefore, they apply to all commercial nuclear fuel cycle activities involving the nuclides of interest here. Two goals underlie these rules. First, the dose to individuals in restricted areas (within boundaries of a facility) is limited to (a) 1-1/4 rem/calendar quarter to whole body, head and trunk, active blood-forming organs, lens of eye, and gonads, excepting that the whole body dose may go as high as 3 rem/quarter, depending on the age of the individual; (b) 18-3/4 rem/calendar quarter to hands and forearms; and (c) 7-1/2 rem/calendar quarter to skin of whole body. Limits for concentrations of radionuclides in air and water above natural background in restricted areas are given in Appendix B, Table I of 10 CFR 20; the values for pertinent isotopes are summarized in Table A-1 of this report. Second, the dose to individuals in unrestricted areas (beyond boundaries of a facility) is limited to 500 mrem/y. The limits for concentrations in air and water above natural background in unrestricted areas are given in Appendix B, Table II of 10 CFR 20; these values are also summarized in Table A-1 of this report.

Table A-1. Maximum Permissible Concentrations for Selected Airborne Nuclides from 10 CFR 20

	Restricted ^a		Unrestricted ^b	
	Area Limit, $\mu\text{Ci/mL}$		Area Limit, $\mu\text{Ci/mL}$	
	Air	Water	Air	Water
H-3				
Soluble	5×10^{-6}	1×10^{-1}	2×10^{-7}	3×10^{-3}
Insoluble	5×10^{-6}	1×10^{-1}	2×10^{-7}	3×10^{-3}
Submersion	2×10^{-3}		4×10^{-5}	
I-129				
Soluble	2×10^{-9}	1×10^{-5}	2×10^{-11}	6×10^{-8}
Insoluble	7×10^{-8}	6×10^{-3}	2×10^{-9}	2×10^{-4}
C-14				
Soluble	4×10^{-6}	2×10^{-2}	1×10^{-7}	8×10^{-4}
Submersion (CO_2)	5×10^{-5}		1×10^{-6}	
Kr-85				
Submersion	1×10^{-5}		3×10^{-7}	

^aWithin boundary of facility.

^bBeyond boundary of facility.

In addition to adhering to these specific requirements, licensees are instructed to make every reasonable effort to limit radiation exposures and releases in unrestricted areas to levels that are consistent with the ALARA philosophy. Except for the operation of LWR's as described in 10 CFR 50, Appendix I, quantitative interpretations of the ALARA philosophy have not been developed.

2. 10 CFR 50, Appendix F, Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities

This section prescribes policy relating to the siting of fuel reprocessing plants and related waste management facilities. The only part of this section that may be applicable to the present analysis is the paragraph calling for decontamination and removal of all significant radioactive wastes at the time the facility is decommissioned. This ruling would seem to apply to any plans for onsite storage of the two airborne nuclides that have relatively short half-lives, *i.e.*, H-3 and Kr-85. Adherence to this rule may require that, if these nuclides are stored on the site of the fuel reprocessing plant, they would have to be removed (*e.g.*, by release into the atmosphere or transportation) by the time the plant is decommissioned.

3. 10 CFR 50, Appendix I, Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Power Reactor Effluents

This section of the Code of Federal Regulations is listed here only because it has been considered a bellwether for the interpretation of ALARA. Nevertheless, as specifically stated in this section of the rules, these guides are appropriate only for light-water-cooled nuclear power reactors and not for other types of nuclear facilities. Therefore, they are not applicable to operations at the facility that is significant for the management of the H-3, I-129, C-14, and Kr-85 from the uranium fuel cycle, *i.e.*, the reprocessing plant, and are not considered in this analysis.

4. 10 CFR 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories; Technical Criteria

For this section of the Code of Federal Regulations [10 CFR 60], the NRC has proposed rules for disposal of HLW in saturated geologic formations. The most important parts of these proposed regulations to this analysis indicate that, if I-129 and C-14 are placed in geologic storage, (1) the package must contain the waste for 1000 y and (2) the annual release rate must be a maximum of 10^{-5} of the amount remaining in the repository.

These rules are expected to be consistent with the EPA criteria in 40 CFR 191, which is now in proposal form only. In 10 CFR 60 general criteria are presented to guide the development of waste form and container. These general criteria are as follows: the waste form should be solid, in a sealed container, consolidated and noncombustible; the waste package should contain no materials that are explosive, pyrophoric, or chemically reactive, and no free liquids; the package should maintain containment during transportation, emplacement, and retrieval; the package must be uniquely marked to provide identification to the end of a chosen retrievable period.

5. 10 CFR 61, Licensing Requirements for Land Disposal of Radioactive Waste

The elements of 10 CFR 61 most important to this task are the tables that determine, for individual nuclides, permissible concentrations and outline some packaging and handling considerations. The requirement that some classes (B,C) of waste must have either waste forms or containers capable of maintaining "gross physical properties and identity" for 300 y has a serious effect on planning.* Another pronouncement of importance is that any consideration by NRC of SLB applied to the Kr-85 resulting from fuel reprocessing has been postponed.

6. 10 CFR 71, Packaging of Radioactive Material for Transport and Transportation of Radioactive Materials Under Certain Conditions

In these rules, the NRC establishes requirements for transportation of materials licensed by the commission. These rules set forth prescriptions for packaging, shipping, standards, procedures, and methods of obtaining approvals. These rules have been under revision in recent years, and proposed changes were published in the Federal Register in 1979 [FEDREG 48234]. Since these changes were proposed, however, it was decided to rewrite the regulations to be in accord with those of the IAEA, just as the DOT rules have been rewritten to agree with those of the IAEA. At the time of this writing, the final revision of 10 CFR 71 has not been issued. This lack does not affect the present task seriously, however, because the final revision of the DOT rules (49 CFR 171-178), which are expected to be fully consistent with those that will eventually appear in 10 CFR 71, has been available for use since March 10, 1983.

7. 40 CFR 61, National Emission Standards for Hazardous Air Pollutants

This section of the Code of Federal Regulations is expected to eventually express rules adopted from the Clean Air Act. Adoption of some sections of this act is presently being discussed by Congress. Perhaps the most significant effect of the Clean Air Act on the interests of the present analysis is the prescription that "best available control technology" (BACT) must be employed. This is a qualitative criterion that, in analogy to the ALARA criterion of NRC, could lead to requiring the collection of some nuclides (such as H-3 and C-14) for which no other, quantitative regulation is in effect.

8. 40 CFR 141, National Interim Primary Drinking Water Regulations

In these rules, the EPA has prescribed a limit of 2×10^{-8} Ci/L for H-3 in drinking water. The EPA estimates that this concentration would result in a dose of 4 mrem/y to an individual. This ruling focuses on the end rather than the source of the nuclide pathway to the biosphere. Predicting the demands that this rule would make on the management of H-3 at the FRP would require accounting for the complexities of the travel of H-3 from the FRP to the drinking water; this pathway will be unique for each plant and its environment.

*The design and development of high-integrity containers proposed for meeting the required 300-y lifetime have been pursued by at least two corporations, the Hittman corporation [WILLIAMS] and the NUS Process Services Corp. [NWN].

9. 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operations

These standards govern planned emissions from the facilities of the commercial uranium fuel cycle except for mining, transportation, and waste management. Two rules expressed in these standards are especially important to the present analysis: (1) annual individual dose commitments from fuel cycle operations are to be limited to 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to other organs; and (2) limits on releases of radio-nuclides from fuel irradiated after 1982 include 5×10^4 Ci Kr-85/GW_e-y and 5×10^{-3} Ci I-129/GW_e-y.

10. 40 CFR 191, Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes

These standards are presently being circulated in draft form [FEDREG 58196] by the EPA for comment. The underlying goal is to limit the health effects from these disposal activities to 10^3 premature deaths in 10^4 y. To implement this goal, a table of release limits for a few individual nuclides and all others with half-life >20 y, in Ci/ 10^3 MTHM, for a period of 10^4 y are given. These limits are expected, on the basis of modeling studies, to restrict the health effects to the goal stated. Another item of importance in these proposed standards is a new definition of HLW. In 40 CFR 191 Draft concentrations for certain nuclides above which they should be disposed of as HLW are defined. A limit for C-14 is defined here. Also, a general limit is defined for all other nuclides not specifically mentioned that have a $t_{1/2} > 20$ y, which is presumably applicable to I-129. These limits defining HLW, however, are to be applied only to spent fuel or to wastes from the first or subsequent extraction cycles. In the operations of a typical FRP, both C-14 and I-129 contained in spent fuel are expected to be released from spent fuel into gaseous streams, from which they would be separated and collected before the extraction operations are carried out. Thus, the C-14 and the I-129 collected in operations of a typical FRP would not be spent fuel, nor would they be wastes from extraction cycles. Therefore, regardless of concentration, they could not be defined as HLW by the proposed rules of 40 CFR 191 Draft. Also, neither Kr-85 nor H-3 qualifies as HLW by these standards.

11. 49 CFR 171-178, Requirements for Transportation of Radioactive Materials

These sections of the Code of Federal Regulations have been recently revised to be consistent with the IAEA rules on transportation of radioactive materials [FEDREG 10218]. These regulations deal with package-handling procedures, nuclide-classification schemes that assign characteristic limits on total quantity of nuclide per package, specifications for authorized packaging for each nuclide according to two characteristic assigned values (A_1 and A_2), specifications on packaging permitted depending on the dispersibility of the radioactive material, descriptions of DOT-specification packaging, performance requirements for other types of packaging, and a few criteria on properties of the radioactive material.

12. DOE Order 5480 Chapter XI, Requirements for Radiation Protection

The purpose of this chapter is to establish standards for protection from radiation for operations of DOE and its contractors, based on recommendations of the EPA and NCRP. The chapter contains guidelines on several considerations, including dosimetry in accident situations, quality factors in determining exposure, permitted exposure in rescue and recovery activities, guidance on maintaining exposures to ALARA levels, and the conduct of internal audits of radiation control programs. Two topics are of interest to the present analysis. The first topic is radiation protection standards for controlled (occupational) areas. These standards consist of (a) maximum doses to parts of the body for individuals and (b) maximum permissible concentrations in air and water. These concentrations are listed in tables that are equivalent to those presented in 10 CFR 20. The second topic is radiation protection standards for uncontrolled (public) areas. These standards consist of (a) maximum doses to body parts, including 500 mrem/y for whole body, gonads, and bone marrow and 1500 mrem/y for other organs, and (b) maximum permissible concentrations in air and water. These concentrations are also equivalent to the corresponding limits presented in 10 CFR 20.

13. DOE Order 5820, Radioactive Waste Management

This order is currently still in draft form. Its purpose is to establish guidelines for handling, storage, and disposal of radwaste for DOE operations and programs. The underlying goals include 25 mrem/y to the whole body, 75 mrem/y to the thyroid, and 25 mrem/y to any other organ, in agreement with 40 CFR 190. They also require observance of ALARA and the exposure limits expressed in DOE Order 5480, Chapter XI (see item 12 above). Order 5820 outlines DOE policies on the management of HLW, TRU, LLW, and materials contaminated with naturally occurring nuclides. In its present form, this order offers no guidance on the management of airborne waste forms. It contains neither any specific reference to the nuclides of interest nor any means of deciding which categories they would fit into. The order requires field organizations to develop site-specific acceptance criteria for wastes in order to achieve the goals stated above. These criteria must be developed before judgments can be made on how they would affect the management of airborne wastes.

APPENDIX B.

BASIC DATA

Three sets of data provide the bases for quantitative estimates in this analysis. Although these data are given in Part I of this study, they are summarized below for convenience in referring to the numbers on which the estimates in Part II are based.

1. Properties of the Nuclides

Values for the properties of the four nuclides are summarized in Table B-1.

Table B-1. Relevant Properties of the Airborne Waste Nuclides

	^3H	^{14}C	^{85}Kr	^{129}I
Decay half-life, y	12.35 ± 0.01	5730 ± 40	10.72 ± 0.01	$1.57 \pm 0.04 \times 10^7$
Decay mode	β^-	β^-	β^-, γ	β^-, γ
Stable decay product	^3He	^{14}N	^{85}Rb	^{129}Xe
Specific activity, Ci/g nuclide	9.60×10^3	4.45	3.93×10^2	1.77×10^{-4}
Thermal output W/Ci	3.39×10^{-5}	2.92×10^{-4}	1.52×10^{-3}	4.62×10^{-4}
W/g nuclide	3.26×10^{-1}	1.30×10^{-3}	8.41×10^{-1}	8.18×10^{-8}
Principal chemical form in DOG	H_2O	CO_2	Kr	$\text{I}_2, \text{CH}_3\text{I}$
Dilution factor by natural, nonradioactive isotope	2.2×10^7	1.3×10^3	~ 17	~ 1.3

^aFor processing of LWR fuel 1.5 y after discharge from reactor.

2. Parameters of the Fuel Reprocessing Plant

Since the nuclear fuel cycle facility at which H-3, I-129, C-14, and Kr-85 will appear is the fuel reprocessing plant, some disposal system descriptions will be based on the characteristics of this facility. The assumed fuel reprocessing plant characteristics used in this report are listed in Table B-2.

Table B-2. Reprocessing Plant Parameters

LWR Fuel Reprocessed, MTHM/y	1500
Operational Conditions	
Days per year	300
Hours per day	24
Air Sparge of Dissolution Process	
Flow rate, cfm	100
Total flow @ 20°C, L/y	1.22×10^9
Water ^a Volume (total for all reprocessing operations)	
Flow, m ³ /y	
max.	5.9×10^3
min.	3.02×10^3
Water in Waste Product, m ³ /y	
max.	3.88×10^3
min.	1.44×10^1
Water Recycle	90%
Water Concentration Ratio ^b (max.)	270:1

^aProcess dependent; volumes from processes assumed by [GRIMES].

^bVolume used in reprocessing operations/volume to be disposed of as liquid waste.

3. Quantities of the Airborne Wastes

The quantities of the airborne wastes that would be expected to appear at the fuel reprocessing plant characterized by Table B-2 are summarized in Table B-3. All estimates and descriptions that involve quantities of the airborne wastes in this report are derived from the values given in Table B-3.

Table B-3. Quantities of Airborne Wastes for 1500 MTHM/y Reprocessing Plant^a

	³ H	¹⁴ C	⁸⁵ Kr	¹²⁹ I
Nuclide Quantities, Ci/y				
At discharge	$1.2 \text{ E}6^b$	860	$1.38 \text{ E}7$	47
At 1.5-y cooling	$1.08 \text{ E}6^b$	860	$1.25 \text{ E}7$	47

^aQuantities will vary with fuel type, burnup, and cooling time. The entries in this table are reference values. The estimates in this report are based on a mixture of BWR and PWR fuels in the ratio of 1/2. The burnup of BWR fuel is assumed to be 27,500 MWd/MTHM and the burnup of PWR fuel is assumed to be 33,000 MWd/MTHM [CROFF].

^bIncludes quantities in hulls; the amount potentially airborne is 9.1×10^5 Ci/y at 1.5-y cooling.

APPENDIX C.

INFORMATION CATEGORIES FOR HLW CANISTERS

Information categories that are being recommended [ONWI-OCC] to characterize canisters for HLW are as follows:

Chemical composition

Physical properties:

- Dimensions, geometry, weight, density
- Buckling resistance
- Impact resistance

Flammability

Thermal properties:

- Thermal conductivity
- Specific heat
- Coefficient of thermal expansion
- Melting point
- Other temperature limits

Corrosion rate

Mechanical properties:

- Tensile strength
- Compressive strength
- Shear strength
- Young's modulus
- Shear modulus
- Bulk modulus
- Elongation
- Charpy impact
- Yield strength
- Poisson's ratio

APPENDIX D.

CURRENTLY FEASIBLE DISPOSAL SYSTEMS FOR H-3

Descriptions of H-3 disposal are given here with the intent of portraying how the job might be carried out using the technology at its current stage and applicable regulations. The descriptions consist of identification of the disposal systems that could be used and, wherever possible, quantitative representation of them. Although identification of systems for disposal of H-3 is readily possible, quantitative description is difficult because much of the technology is only at the conceptual or laboratory stage at this time. The descriptions given here are not offered as the only means of H-3 disposal; it is not proposed to limit the future options of organizations that will have the responsibility for this task. In fact, the descriptions will, in some places, point out the need for additional information and the inconvenience of applying some of the technologies.

1. Collection of H-3

Conditions for collection of H-3 are given in Table D-1 for three alternative processes: (1) collection of overheads from evaporation of solvent extraction raffinates, (2) voloxidation, and (3) thermal out-gassing.

2. Preparation of H-3 Waste Forms

The only packaging system for H-3 that has been shown to have a H-3 leak rate low enough to be acceptable to at least one disposal facility is the Mound package. The mixture of plaster and cement (3:1 by weight) in the Mound package is assumed to serve as the matrix for H-3 waste forms. Some of the conditions that would be used in preparation of this waste form are summarized in Table D-2. The retention of H-3 by the Mound package, however, is based not on the properties of the matrix but rather on the container materials and the nest of containers and coatings. There are two general types of H-3 wastes that would be incorporated into this plaster-cement matrix (Table D-2). One type is particulate solids, including molecular sieve pellets, hulls, and particulate zirconium tritide. The other type is tritiated water.

3. Conditions for Packaging of H-3 Waste Forms

Three different primary waste forms have been proposed for containment of tritium: tritiated water, zeolites, and zirconium tritide. All three forms are incorporated into cement, making concrete the final form for immobilization. The quantities to be processed into final form varies with the head-end processing method. For this study, maximum quantity levels for both dissolution and hulls have been assumed. The number of packages for the three waste forms using the Mound concept is presented in Table III-6. These packages have been adjusted by weight, volume, or curie content to comply with the transportation criteria, yet minimize the number of containers.

Table D-1. Conditions for Collection of H-3

1. Collection of Overheads from Evaporation of Solvent Extraction Raffinates

Feed: High Level Aqueous Waste

Temp.: 25–100°C

Press.: 1 atm

Atmosphere: Air

Reagents: None

Reaction or process: (a) evaporation, condensation, recycle of condensate

or (b) evaporation, condensation, recycle of condensate, isotope separation by catalyzed H_2 - H_2O exchange

Product of collection process:

(a) tritiated water, 7×10^5 Ci H-3 in 3900 m³ water/1500 MTHM by process a^a

or (b) tritiated water, 7×10^5 Ci H-3 in 86 m³ water/1500 MTHM by process b^a

2. Voloxidation

Feed: Chopped Fuel

Temp.: 550°C

Press.: 1 atm

Atmosphere: Air or O₂ (flowing)

Reagents: Air or O_2 , 13X molecular sieves to absorb THO

Reaction or process: $T \text{ (in fuel)} + O_2 \rightarrow T_2O$

$$\text{T}_2\text{O} + \text{molecular sieves} \rightarrow \text{absorbed T}_2\text{O}$$

Product of collection process: Intermediate: THO (7 x 10⁵ Ci in 14 m³ water/1500 MTHM)

Final: THO on 13X molecular sieves
(7 x 10⁵ Ci on 3 x 10⁶ g sieves/
1500 MTHM)

3. Thermal Outgassing

Feed: Chopped Fuel

Temp.: 1500°C

Press.: 1 atm (if He)

Atmosphere: He, He + H₂, or vacuum

Reagents: Flowing O_2 , catalyst for conversion of intermediate

T₂ to T₂O, molecular sieves to absorb T₂O or Zr

to react with intermediate T_2

Reaction or process: T (fuel) + T_2 (gas)

$$\text{T}_2 (\text{gas}) + \text{O}_2 \xrightarrow{\text{catalyst}} \text{T}_2\text{O}$$
$$\text{or } \text{T}_2 \text{ (gas)} + \text{Zr (solid)} \rightarrow \text{ZrT}_{1.5} \text{ (solid)}$$

Product of collec- Intermediate: T₂ (gas)

tion process: Final: THO on molecular sieves or
ZrT_{1.5}

^a[GRIMES] .

Table D-2. Conditions for Preparation of H-3 Waste Forms

Matrix^a

Ratio of plaster-cement mixture to water: 3 kg mixture/1 L water

Ratio of plaster to cement: 1:3 by volume

Incorporation of Solids in Matrix (for Zeolites or ZrT_{1.5})

Ratio of incorporated solids to total solid: 0.09 by volume^b

Packages

DOT specification 6J^c

DOT specification 6L^c

DOT specification 6M^c

Mound^c

^aFrom [MERSHAD].

^bAssumed analogous to volumes of barium iodate incorporated into cement by ORNL.

^cDescribed in Part I.

The inner container is larger than a "2R" and was assumed to be a polyethylene-sealed container of 160 L capacity. This container is packaged inside a 6J (B)* drum. The package meets the Type B quantity[†] ($<5 \times 10^4$ Ci/pkg) and weight (<800 lb [363 kg], B configuration) of DOT and falls in the class B[‡] disposal level of NRC.

4. Conditions for Transportation of H-3 Waste Forms

Conditions for transportation of H-3 waste forms are relatively simple (Table D-3). All waste forms considered here are cement and all outer containers are 55-gal drums. These drums would be transported in the manner generally used for LLW. No shielding of the drums should be necessary. Contact handling methods should be permissible in loading, unloading, and emplacement. Shipment to a shallow-land burial ground is only one option in disposal of H-3. As mentioned above, the packagings considered here are the only ones currently available, and their limited volumes result in the handling of a rather large number of packages. One alternative to transportation

*B configuration for drum.

[†]Type B Quantity defined by 49 CFR.

[‡]Defined by 10 CFR 61 for shallow-land burial.

Table D-3. Conditions for Transportation of H-3 Wastes

Waste form:	Cement
Outer container:	55-gal drum
Possible Overpacks:	Insulated box (e.g., Super Tiger, ^a Poly-Panther, ^a N-55, ^b TRUPACT ^c)
Shielding required:	None
Handling method in loading, unloading, emplacement:	Contact
Preferred carrier vehicle:	Truck (exclusive use)

^aU.S. Ecology.

^bNuclear Packaging.

^cAlthough the TRUPACT was developed for TRU wastes, it could be used to carry airborne wastes in Type A packages.

of H-3 wastes is to devise an acceptable method of storage on the site of the fuel reprocessing plant. Tank storage of tritiated water has been suggested [NEA]. Storage in one of the solid forms discussed above is also an obvious option, but less suited to the concept of release of tritium after partial decay.

5. Final Emplacement of H-3 Waste Forms

The discussion given in Part I indicates that if H-3 wastes are to be collected, immobilized, and managed, several options exist for their final emplacement. At this time, however, only one of these options, SLB, is at the stage that permits compliance of technologies with regulations. The conditions for this emplacement mode are briefly summarized in Table D-4. Regulations that would govern onsite storage of H-3 wastes at a fuel reprocessing plant, an attractive option because of the relatively short half-life of H-3, have not been considered yet by regulatory agencies.

Table D-4. Conditions for Final Emplacement of H-3 Wastes

Emplacement mode: Shallow-land burial

Applicable regulations: 10 CFR 61; additional requirements may be developed by host states

Containers: 55-gal drums

Waste class (by 10 CFR 61): LLW, Class B^a

Container material: For Class B waste, waste form or container must maintain integrity for 300 y; container material must be determined; it seems likely that a cement waste form would qualify for this requirement.

^aH-3 could be emplaced as a Class A waste (maximum concentration, 40 Ci/m³). This practice would exclude it from the requirement of a container with 300-y lifetime, but would result in distributing it within a large volume of matrix. This large value of matrix would result in an impracticably large number of containers ($\sim 1 \times 10^5/1500$ MTHM) which, in turn, would require an impracticably large number of shipments ($\sim 1.7 \times 10^3/1500$ MTHM).

APPENDIX E.

CURRENTLY FEASIBLE DISPOSAL SYSTEMS FOR I-129

The descriptions of I-129 disposal given here represent disposal technology at its current stage, in compliance with applicable regulations. The disposal descriptions identify the technological systems that could be used, along with a few quantitative specifications that are possible. Although identification of disposal concepts for I-129 is relatively easy, quantitative specification is difficult since much of the technology is immature. The descriptions given here are not intended to limit the means of I-129 disposal or the future options of organizations that must carry it out.

1. Collection of I-129

Conditions for collection of I-129 are given in Table E-1 for three alternative processes: (1) mercurer, (2) iodox, and (3) silver sorbents.

2. Preparation of I-129 Waste Forms

Although several emplacement modes for I-129 have been suggested and discussed, only two modes promise to be applicable in the foreseeable future: shallow-land burial and geologic emplacement. Therefore, preparation of I-129 waste forms, summarized in Table E-2, is aimed at radionuclide concentrations consistent with these two modes of emplacement. Although prospects for ocean disposal are not favorable at this time, the waste form listed for geologic emplacement as HLW would also meet the concentration limits for ocean disposal as LLW (Section IV.F.2). Three waste forms are indicated in Fig. IV-1; $\text{Ba}(\text{IO}_3)_2 \cdot \text{H}_2\text{O}$ in cement, AgZI in cement, and PbZI without a matrix. The last waste form would be 1/8-in. dia pellets. Criteria for transportation and for the disposal modes considered here contain no explicit rule disallowing unagglomerated pellets as a waste form. The incorporation of PbZI in cement is ruled out because of its reaction with cement (Part I).

3. Conditions for Packaging of I-129 Waste Forms

Of the three different waste forms proposed for I-129, two utilize zeolites for capture. Moreover, two of the final waste forms incorporate cement into the system for immobilization, and the third is pellets. Optimization of these systems to disposal limits results in packages, presented in Table IV-10, that fit in a DOT-specification 6L packaging (55-gal drum).

The designs for each disposal mode meet the Type A quantity ($<2 \text{ Ci/pkg}$) of DOT and the Type C disposal level ($<8 \times 10^{-8} \text{ Ci/cc}$) of NRC, resulting in 2612 drums. Though the driving force behind the designs is the LLW criteria, disposal at a HLW site would permit up to a DOT Type B quantity. Thus, if the waste forms alone were to limit the curies per 6J container to the values given in Table IV-11, a much lower number of shipping containers (9-16) would result. The actual number of drums would probably be greater than this, perhaps up to twice the amount, depending on whether a 2R container is required and what its size is.

Table E-1. Conditions for Collection of I-129

1. Iodox

Feed: Off-gas from shear, Voloxidizer, dissolver

Temp.: 25-40°C

Press.: 1 atm

Atmosphere: air

Reagents: 22-23 M HNO₃Reaction or Process: $\text{CH}_3\text{I} + 2\text{HNO}_3 = \text{CH}_3\text{NO}_3 + 1/2\text{I}_2 + 1/2\text{N}_2\text{O}_4 + \text{H}_2\text{O}$ Equipment: Gas-liquid scrubber column^aProduct: Ba(IO₃)₂·H₂O(solid)2. Mercurex

Feed: Off-gases from shear, Voloxidizer, dissolver

Temp.: Ambient

Press.: 1 atm

Atmosphere: air

Reagents: 0.2-0.4 M Hg(NO₃)₂, in 8-14 M HNO₃Reaction or Process: $6\text{I}_2 + 6\text{Hg}(\text{NO}_3)_2 + 6\text{H}_2\text{O} \rightarrow 5\text{HgI}_2 + \text{Hg}(\text{IO}_3)_2 + 12\text{HNO}_3$

or

(contd)

Table E-1. (contd)

Equipment:	Gas-liquid scrubber column ^a
End Product:	Ba(IO ₃) ₂ ·H ₂ O (solid)
3. <u>Adsorption on Silver Sorbents</u>	
Feed:	Off-gases from shear, Voloxidation, dissolver
Temp.:	150°C
Press.:	1 atm
Atmosphere:	air
Reagents:	Silver-exchanged Mordenite, AgZ
Reaction or Process:	Ag ₂ O(s) + I ₂ (g) + 2AgI(s) + 1/2O ₂ (g)
	Ag ₂ O(s) + H ₂ O + 2CH ₃ I(g) + 2AgI(s) + 2CH ₃ OH(g)
Equipment:	Column packed with AgZ
Product:	AgZI (solid)

^aEquipment promoting transfer of chemical species from a gas stream to a liquid stream in a columnar (usually cylindrical) arrangement and housing.

Table E-2. Conditions for Preparation of I-129 Waste Forms

Preparation for Shallow-Land Burial

Ba(IO₃)₂·H₂O solids,
incorporated into portland cement; 0.06 wt % of total, 0.08 Ci/m³.

AgZI pellets,
incorporated into portland cement; 0.2 wt % of total, 0.08 Ci/m³.

PbZI pellets, no matrix, 0.08 Ci/m³.

Preparation for Geologic Emplacement^a

Ba(IO₃)₂·H₂O solids,
incorporated into portland cement; 18 wt % of total, 26 Ci/m³.

AgZI pellets,
incorporated into portland cement; 34 wt % of total, 15 Ci/m³.

PbZI pellets, no matrix, 23 Ci/m³.

^aThe concentrations expressed here are classed as HLW by Draft 40 CFR 191, but would be classed as LLW for ocean disposal by LODC (IAEA) [IAEA-1980].

4. Conditions for Transportation of I-129 Waste Forms

Conditions for transportation of I-129 waste forms are given in Table E-3. Two of the waste forms considered here are cement, and one is loose zeolite pellets. All carrier containers are 55-gal drums, which would be transported in the manner generally used for LLW. If I-129 waste forms were managed as HLW, it seems likely that truck transportation would be preferred. The standard 40-ft truck trailer carries 60 drums. Shielding of the drums will be unnecessary. Contact handling methods will be permissible in loading, unloading, and emplacement. As mentioned above, the packagings considered here are the only ones currently available, and their limited volumes result in the handling of a large number of packages. The drums of I-129 waste would probably be carried in an overpack, such as the Super Tiger, that provides insulation from both fire and physical shocks. During transportation for the cement waste forms, thermal insulation would be required to prevent overpressurization by release of water from the cement in the event of fire.

Table E-3. Conditions for Transport of I-129 Waste Forms

	Cement Waste Forms		Pellet Zeolite Waste Forms	
	SLB	Geologic	SLB	Geologic
Outer Container	55-gal drum	55-gal drum	55-gal drum	55-gal drum
DOT Specification	6La	6Ja	6La	6Ja
Overpack Requirements ^b	Thermal, shock resistant	Thermal, shock resistant	Thermal, shock resistant	Thermal, shock resistant
Shielding Requirements	none	none	none	none
Handling in Loading, Unloading, Emplacement	Contact	Contact	Contact	Contact
Preferred Carrier Vehicle	Truck	Truck	Truck	Truck

^aThis packaging meets the requirements of transportation regulations (49 CFR), but is unlikely to meet requirements of emplacement criteria, which are as yet not completely defined or interpreted.

^bPossible overpacks are the Super Tiger (U.S. Ecology), Poly-Panther (U.S. Ecology), N-55 (Nuclear Packaging), and TRUPACT, an overpack developed for TRU wastes that could be applied to airborne wastes in Type A packages.

5. Conditions for Final Emplacement of I-129 Waste Forms

In the conditions for emplacement summarized in Table E-4, SLB and geologic emplacement are considered. Onsite storage, considered a possibility for H-3 and Kr-85, is unattractive for a nuclide with an extremely long half-life such as I-129. Both greater confinement disposal and ocean disposal were discussed as possibilities for disposal of I-129. The criteria for greater confinement disposal initially proposed by a DOE contractor [CARD-1981] are in the process of being revised. For ocean disposal, the criteria drafted by IAEA for the LODC [IAEA-1980] seem to be generally accepted by the OECD nations, but this mode of disposal is not now permitted by the U.S. government.

Table E-4. Conditions for Final Emplacement of I-129 Wastes

Shallow-Land Burial

Applicable criteria: 10 CFR 61; additional requirements may be developed by the host state

Waste class (by 10 CFR 61): LLW, Class C

Containers: 55-gal drums

Container material: For Class C waste, waste form or container must maintain integrity for 300 y; container material must be determined.

Concentration limit: 0.08 Ci/m^3

Intermediate-Depth Burial

Applicable criteria: Not available: expected for non-near surface disposal as 10 CFR 61 Part B

Geologic Emplacement

Applicable criteria: Draft 40 CFR 191,^a Draft 10 CFR 60,^b
ONWI-OOC

Waste class: HLW (by 40 CFR 191), at concentrations in excess of $1 \times 10^{-3} \text{ Ci/g}^a$

Containers: 55-gal drums^b

Container material: To be determined

Concentration limit: None

^aDraft 40 CFR 191 may not be applicable to separated I-129 wastes since it is addressed only to spent fuel and the wastes of the first or subsequent solvent extraction cycles of an FRP.

^bDefinitions of acceptable design and material for containers of HLW that is neither TRU, spent fuel, nor vitrified mixed fission products may be developed later.

APPENDIX F

CURRENTLY FEASIBLE DISPOSAL SYSTEMS FOR C-14

Descriptions of C-14 disposal summarized here represent presently conceived technologies in accord with presently available criteria and regulations. The conditions listed here describe technological systems for C-14 disposal in quantitative terms as far as possible. Although qualitative descriptions are readily expressed, quantitative description has a very limited basis in experience and must, therefore, be accepted as liable to change. The details laid out are not the only means of accomplishing C-14 disposal. Furthermore, they also point out the need for additional data if the disposal systems are to be specified more completely.

1. Collection of C-14

Conditions for collection of C-14 are given in Table F-1 for the five alternative processes indicated in Fig. V-1: (1) reaction with caustic slurry, $\text{Ca}(\text{OH})_2$, (2) reaction with caustic scrub, NaOH , (3) adsorption on molecular sieves, (4) adsorption in fluorocarbons, and (5) reaction with solid $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$.

2. Preparation of C-14 Waste Forms

Although several alternative collection processes are considered for C-14, its waste management scheme becomes very much simplified with respect to final waste forms. All of the collection processes result in only two intermediate waste forms, $\text{Ba}(\text{CO}_3)_2$ or $\text{Ca}(\text{CO}_3)_2$. Since these two intermediate products are, in turn, incorporated into cement, preparation of a final waste form is concentrated on only one type of matrix. Thus, the main variable to be decided in preparation of C-14 waste forms from the intermediate collection is the proportion of solid carbonate to cement. This proportion depends on the set of acceptance criteria to be met. Since the acceptance criteria are unique to each disposal mode and, perhaps, to each disposal site, the proportion will depend ultimately on which disposal mode has been selected. Since regulations are in place for SLB and also a substantial effort is under way to develop regulations and technology for geologic disposal, conditions for preparation of waste forms that would be acceptable for these two modes are given in Table F-2. Although ocean disposal shows little promise of application in the near future, it should be noted that the waste form developed by R&D work would also qualify for ocean disposal, at least with respect to the allowed concentration of C-14.

3. Conditions for Packaging of C-14 Waste Forms

The CaCO_3 and BaCO_3 waste forms proposed for carbon-14 would be immobilized in a concrete matrix and placed in a container for final disposal. Adaptation of these systems to disposal limits results in packages, as presented in Table V-10, that fit in a DOT Spec. 6L package (55-gal drum).

The designs for these disposal modes meet the Type A quantity ($<60 \text{ Ci/pkg}$) of DOT and the Type C disposal level ($<8 \times 10^{-6} \text{ Ci/cc}$) of NRC, resulting in ~478 drums from a 1500 MTHM/y fuel reprocessing plant. This seems to be a reasonable number of drums.

Table F-1. Conditions for Collection of C-14

1. Reaction with Caustic Slurry, Ca(OH)_2

Feed: Off-gases from shear, Voloxidizer, scrubber

Temp.: Ambient

Press.: 1 atm

Atmosphere: air

Reagents: Ca(OH)_2 slurryReaction or Process: $\text{CO}_2 + \text{Ca(OH)}_2 \rightarrow \text{CaCO}_3 + \text{H}_2\text{O}$

Equipment: Stirred cell contactor

Product: Intermediate, none

Final, CaCO_3 (solid)2. Reaction with Caustic Scrub, NaOH

Feed: Off-gases from shear, Voloxidizer, scrubber

Temp.: Ambient

Press.: 1 atm

Atmosphere: air

Reagents: NaOH solution

Reaction or Process: $\text{CO}_2 + 2\text{NaOH(aq)} \rightarrow \text{H}_2\text{O} + \text{Na}_2\text{CO}_3(\text{aq})$ $\text{Na}_2\text{CO}_3(\text{aq}) + \text{Ca(OH)}_2 \rightarrow 2\text{NaOH(aq)} + \text{CaCO}_3(\text{solid})$

Equipment: Packed scrubbing column for 1st reaction;

Mixing tank for 2nd reaction

Product: Intermediate, Na_2CO_3 (aq.)Final, CaCO_3 (solid)

(contd)

Table F-1. (contd)

3. Reaction with Molecular Sieves

Feed: Off-gases from shear, Volcxidizer, scrubber

Temp.: { (1) adsorption at ambient temperature and above
1 atm pressure and desorption accomplished by
decreasing the pressure
Press.: { or (2) adsorption at cryogenic temperatures and desorption
accomplished by warming the bed to ambient temperature

Atmosphere: air

Reagents: 5A molecular sieve pellets

Reaction or Process: $\text{CO}_2(\text{g}) \xrightarrow{\text{adsorption}} \text{CO}_2(\text{adsorbed})$ (a)

$\text{CO}_2(\text{adsorbed}) + \text{air} \xrightarrow{\text{desorption}} \text{CO}_2(\text{g})$ (b)

$\text{CO}_2(\text{g}) + \text{Ca}(\text{OH})_2 \rightarrow \text{CaCO}_3 + \text{H}_2\text{O}$ (c)

Equipment: Tower packed with molecular sieves for reactions a and b;

Stirred cell contactor for reaction c

Product: Intermediate, CO_2 (adsorbed)

Final, CaCO_3 (solid)

4. Absorption in Fluorocarbons

Feed: Off-gases from shear, Voloxidizer, scrubber

Temp.: 50° to -32°C along the column length

Press.: 6-20 atm for absorption: 1 atm for desorption

Atmosphere: air

Reagents: Fluorocarbon liquid

Reaction or Process: $\text{CO}_2(\text{g}) \xrightarrow[6-20 \text{ atm}]{\text{Fluorocarbon}} \text{CO}_2(\text{solution})$ (a)

$\text{CO}_2(\text{solution}) \xrightarrow{1 \text{ atm}} \text{CO}_2(\text{g})$ (b)

$\text{CO}_2(\text{g}) + \text{Ca}(\text{OH})_2(\text{slurry}) \rightarrow$
 $\text{CaCO}_3(\text{solid}) + \text{H}_2\text{O}$ (c)

(contd)

Table F-1. (contd)

Equipment:	Gas-liquid absorption tower for reactions a and b; Stirred cell contactor for reaction c
Product:	Intermediate, CO_2 (solution in fluorocarbon), CO_2 (g) Final, CaCO_3 (solid)
5. Reaction with $\text{Ba(OH)}_2 \cdot 8\text{H}_2\text{O}$	
Feed:	Off-gases from shear, Voloxidizer, scrubber
Temp.:	Depends on CO_2 concentration and relative humidity, generally in the range of 20 to 35°C
Press.:	1 atm
Atmosphere:	air
Reagents:	Solid $\text{Ba(OH)}_2 \cdot 8\text{H}_2\text{O}$
Reaction or Process:	$\text{CO}_2(\text{g}) + \text{Ba(OH)}_2 \cdot 8\text{H}_2\text{O}(\text{solid}) +$ $\text{BaCO}_3(\text{solid}) + 9\text{H}_2\text{O}(\text{g})$
Equipment:	Gas-solid reaction tower
Product:	Intermediate, none Final, BaCO_3 (solid)

Table F-2. Conditions for Preparation of C-14 Waste Forms

Preparation for Shallow-Land Burial

CaCO_3 solids,
incorporated into portland cement; 0.84 wt % of total, 8 Ci/m³.

BaCO_3 solids,
incorporated into portland cement; 1.7 wt % of total, 8 Ci/m³.

Preparation for Geologic Emplacement^a

CaCO_3 solids,
incorporated into portland cement; 12 wt % of total, 1.2×10^2 Ci/m³.

BaCO_3 solids,
incorporated into portland cement; 18 wt % of total, 9.7×10^1 Ci/m³.

^aThe concentrations expressed here are classed as HLW by Draft 40 CFR 191, but would be classed as LLW for ocean disposal by LODC (IAEA) [IAEA-1980].

4. Conditions for Transportation of C-14 Waste Forms

Table F-3 summarizes conditions for transportation of C-14 waste forms. All waste forms considered here are monolithic concrete shapes in drums. Since truck shipment is the mode of transportation recommended for LLW, and since these drums of C-14 can be handled in the same manner as LLW because of absence of penetrating radiation, it is reasonable to recommend truck shipment for C-14 wastes. Contact handling of the C-14 packages, permissible because of the absence of penetrating radiation, should lend simplicity to loading, unloading, and emplacement operations. To avoid overpressurization of drums by water vaporizing from heated concrete in a fire, the drums should be carried in an overpack that affords thermal insulation.

Table F-3. Conditions for Transport of C-14 Waste Forms

	CaCO ₃		BaCO ₃	
	SLB	Geologic	SLB	Geologic
Outer Container	55-gal drum	55-gal drum	55-gal drum	55-gal drum
DOT Specification	6L ^a	6J ^a	6L ^a	6J ^a
Overpack Requirements ^b	Thermal, shock resistant	Thermal, shock resistant	Thermal, shock resistant	Thermal, shock resistant
Shielding Requirements	none	none	none	none
Handling in Loading, Unloading, Emplacement	Contact	Contact	Contact	Contact
Preferred Carrier Vehicle	Truck	Truck	Truck	Truck

^aThis packaging meets the requirements of transportation regulations (49 CFR), but is unlikely to meet requirements of emplacement criteria, which is as yet not completely defined or interpreted.

^bPossible overpacks are the Super Tiger (U.S. Ecology), Poly-Panther (U.S. Ecology), N-55 (Nuclear Packaging), and TRUPACT, an overpack developed by DOE for TRU wastes, that could be applied to airborne wastes in Type A packages.

5. Final Emplacement of C-14 Waste Forms

At this time, disposal of separated C-14 wastes is being considered in regulations and policies being developed for both SLB and geologic emplacement. Conditions are summarized in Table F-4 for final emplacement by both these modes. As in the case of I-129, the long half-life of C-14 would rule out consideration of onsite disposal at the fuel-reprocessing plant. The amount of decay during the lifetime of the plant would be too small to offer any advantage. Comments on conditions for disposal of C-14 by greater confinement disposal are withheld because of the premature status of the GCD concept and criteria proposed for it. Similarly, no conditions are given for ocean disposal because of the present uncertainty on what procedures, if any, may be used by U.S. waste consignors.

Table F-4. Conditions for Final Emplacement of C-14 Wastes

Shallow-Land Burial

Applicable criteria: 10 CFR 61; additional requirements may be developed by host state

Waste class (by 10 CFR 61): LLW, Class C

Containers: 55-gal drums

Container material: For Class C wastes, waste form or container must maintain integrity for 300 y; material must be determined.

Concentration limit: 8 Ci/m^3

Geologic Emplacement

Applicable criteria: Draft 40 CFR 191^a, Draft 10 CFR 60,^b
ONWI-OOC

Waste class: HLW (by 40 CFR 191^a), at concentrations in excess
of $8 \times 10^{-6} \text{ Ci/g}$

Containers: 55-gal drums^b

Container material: to be determined

Concentration limit: none

^a Draft 40 CFR 191 may not be applicable to separated C-14 wastes since it is addressed only to spent fuel and to the wastes of the first or subsequent solvent extraction cycles of an FRP.

^b Definitions of acceptable design and material for containers of HLW that is neither TRU, spent fuel, nor vitrified mixed fission products may be developed later.

APPENDIX G.

CURRENTLY FEASIBLE DISPOSAL SYSTEMS FOR Kr-85

The lack of provision for Kr-85 in programs developing final emplacement of radioactive wastes forces the conclusion that disposal of this nuclide from fuel reprocessing cannot be carried out now. The disposal systems described, therefore, are choices that would require the least effort in planning and the least amendment to regulations, if disposal of Kr-85 were required at this time. There has not been sufficient correlation between laboratory developments on Kr-85 control and planning of final emplacement, making difficult a firm decision on what packaging and emplacement mode should be specified. Nevertheless, conditions of the parts of a disposal system that could be put into action are given in the following sections, including collection, waste form preparation, packaging, transportation, and final emplacement.

1. Collection of Kr-85

Conditions for collection of Kr-85 are briefly summarized in Table G-1 for the three alternative processes included in the flowsheet of Fig. VI-1: (1) liquid absorption, (2) cryogenic distillation, and (3) absorption on solids. Selecting some reference set of process conditions from the many publications that have reported laboratory experiments in these three processes is difficult. We have found it convenient, for description of process conditions, to quote the conditions suggested in process concepts [WAGGONER, BLANCETT, BROWN-1978]. This course is followed with the understanding that selection of the best of alternative processes is not one of the objectives of this compliance analysis. It should be noted, however, that the evaluation by Waggoner [WAGGONER] made the following conclusions concerning the collection processes for Kr-85: (1) cryogenic distillation is the only demonstrated route to date, (2) fluorocarbon absorption may offer economic and technical advantages if fully developed and demonstrated, and (3) adsorption on mordenite has progressed only as far as bench-scale demonstration and is estimated to be more costly than the other two processes.

2. Preparation of Kr-85 Waste Forms

The product of each of the three alternative processes for collection of Kr-85 is gaseous krypton, which can be used to prepare any of the three final waste forms listed in Table G-2: compressed gas, sputtered metal, and zeolite.

3. Packaging of Kr-85 Waste Forms

The three waste forms of Kr-85 considered in this work are assumed to be packaged as follows: The compressed-gas waste form would be packaged in metal cylinders. No DOT-specification packagings would be applicable except for an NRC-certified packaging that involves a modified DOT-Specification 3A cylinder. Zeolites, compressed by HIP processing into monolithic blocks, are assumed to be packaged in DOT-Specification packagings 6L, 6J, or 6M. These packagings each involve a 55-gal drum as the outer container. Sputtered-metal waste form is deposited as a layer on the inner surface of sections of metal pipe. These pipe sections would be capped and placed in DOT-Specification packaging 6J, which provides a 55-gal drum envelope.

Table G-1. Conditions for Collection of Kr-85

1. Liquid Absorption

Feed: product of pretreatment of off-gases from shear, Voloxidizer, dissolver

Temp: absorption -40°C

Press.: absorption 108 psia

Atmosphere: air

Reagents: CCl_2F_2 (Refrigerant-12)

Reaction or Process: $\text{Kr}(\text{gas}) \xrightarrow{\text{absorp.}} \text{Kr}(\text{soln})$

$\text{Kr}(\text{soln}) \xrightarrow{\text{stripping}} \text{Kr}(\text{gas})$

$\text{Kr}(\text{gas}) \xrightarrow[\text{13X molecular sieves}]{\text{CCl}_2\text{F}_2 \text{ removal}} \text{Kr}(\text{gas})$

$\text{Kr}(\text{gas}) \xrightarrow[\text{-148}^{\circ}\text{C}]{\text{concentr.}} \text{Kr}(\text{gas})$

Equipment: Gas-liquid contact column and feed compressor;
cold traps for Kr separation

Product: Intermediate, none

Final, $\text{Kr}(\text{gas})$

2. Cryogenic Distillation

Feed: product of pretreatment of off-gases from shear, Voloxidation, dissolution

Temp: condensation, -184°C ; N_2 stripping, -111°C ; product recovery, -93°C

Press.: condensation, 2.9 atm; stripping, 2.8 atm; product recovery, 2.7 atm

Atmosphere: N_2

Reagents: Liquid N_2

(contd)

Table G-1. (contd)

Equipment: feed compressor, columns for condensation, stripping, recovery

Product: Intermediate, none
Final, Kr(gas)

3. Solid Adsorption

Feed: product of pretreatment of off-gases from shear, Voloxidizer, dissolver

Temp: Xe adsorption, 25°C; Kr adsorption, -80°C; Kr concentration, -80°C

Press.: Xe adsorption, 1 atm; Kr adsorption, ~1 atm; Kr concentration, ~1 atm

Atmosphere: -

Reagents: Ag mordenite, H mordenite

Equipment: gas-solid columns

Product: Intermediate, none
Final, Kr(gas)

Table G-2. Conditions for Preparation of Kr-85 Waste Forms

1. Compressed Gas

Procedure: (1) gas condensed then expanded into final container, or
(2) gas mechanically compressed into final container

Pressure: 500-2000 psi

Loading: 30 Ci/cc (500 psi)-120 Ci/cc (2000 psi)

2. Sputtered Metal

Sputtered material: 79 wt % Ni, 21 wt % La

Deposit thickness: 1 cm

Loading: ≤ 22 Ci/cc

3. Zeolites

Adsorbent: Zeolite 5A

Procedures: compacted to monolith (1.4 g/cc suggested)

Loading: ≤ 3.8 Ci/cc

Gas cylinders at 500 psi would require ~100 cylinders, and sputtered metal in DOT-Specification 6J packaging would require 1.2×10^4 drums. Zeolites, if loaded to the capacity of the drum, would require a minimum of 1.9×10^3 drums, depending on requirements for a 2R container and its size.

4. Conditions for Transportation of Kr-85 Waste Forms

Krypton-85 waste forms can be packaged in containers that meet DOT regulations, as suggested by the information in Tables VI-2, VI-3, and VI-6, even though the destination of Kr-85 wastes is indeterminate at this time. It has been suggested that onsite disposal may be the best strategy for disposal of Kr-85, but shielding would be needed if such wastes were to be transported from the fuel reprocessing site. Shielding and cooling are not required for disposal of any of the other airborne wastes. To satisfy this need, two courses are suggested: (1) the shielding would be contained within individual packages so that the radiation level at the outer surface is <1000 mrem/h* or ≤ 200 mrem/h for contact handling; and (2) no shielding would be contained within the packages, but the packages would be transported in a shielded overpack. In this later case, for loading, unloading, and emplacement operations, equipment must be available to permit remote handling if the overpack is removed at the destination and reused.

*Limit for exclusive-use vehicle.

One consequence of shielding is that the standard 40-ft truck trailer can carry fewer packages. Conditions for transport of Kr-85 are summarized in Table G-3.

Table G-3. Conditions for Transportation of Kr-85
Produced by Fuel Reprocessing

Packaging: As prescribed by Table VI-2.

Type of Transportation: Truck, exclusive use, closed.

Shielding: Limits dose at surface of any package to ≤ 200 mrem/h.

Limits dose at surface of vehicle to ≤ 200 mrem/h.

Limits dose at any point 2 m from vertical planes
projected by outer surface of vehicle to ≤ 10 mrem/h.

Limits dose at any occupied position in vehicle to ≤ 2 mrem/h.

Release Limits in Accidents: Any material released must be in
gaseous state only.

<u>Waste Form</u>	<u>Permissible Type A Package Inventory for Normal Form,^a Ci</u>
Compressed (Gas)	5
Uncompressed (Sputtered Metal or Zeolite)	10^3

^aLarger package inventories would require Type B packaging.

5. Final Emplacement of Kr-85 Waste Forms

Since in the development of SLB regulations, consideration of any possibility of emplacement of Kr-85 from reprocessing was indefinitely postponed, it is not possible to describe what would be required for shallow land burial of Kr-85. Regarding the possibility of geologic emplacement of Kr-85, there are no direct references to Kr-85 in developing regulations, and EPA draft criteria suggest that geologic emplacement is applicable only to isotopes with half-lives > 20 y, where separated isotopes are concerned. Therefore, any specifications for geologic emplacement of Kr-85 would be neither possible nor appropriate in the light of current policy.

For ocean disposal, the criteria of the IAEA, although they do not refer directly to Kr-85, can be used to infer some conditions that would be applied if Kr-85 were disposed of by this method. These conditions are (1) the concentration of Kr-85 in the waste form must be $\leq 10^{-4}$ Ci/Mg, (2) the

waste form must be solid, (3) the specific gravity of the waste form must be >1.2 , (4) the container must retain its contents during descent to depths of 4000 m or more, and (5) the package must contain no buoyant material that, if returned to the surface, would present a radioactive hazard or interference with the legitimate uses of the sea.

For onsite disposal, there are few guidelines that can be used to develop specifications at this time. The only source of such guidelines seems to be 10 CFR 50, Appendix F, which gives prescriptions for management of radioactive wastes other than HLW at the sites of fuel reprocessing plants. From this source, it is concluded that Kr-85 stored onsite must be removed by the time the fuel reprocessing plant is decommissioned. If the removal mechanism is release to the atmosphere after partial decay during onsite storage, (e.g., as described by [DOE/ET-0028]), then it is assumed that this operation must be carried out in accord with the concentration guidelines of 10 CFR 20 and the release limit of $50,000 \text{ Ci/GW}_{\text{e-y}}$ of 40 CFR 190.

REFERENCES

ALTOMARE

P. M. Altomare, M. Barbier, N. Lord, and D. Nainan, Assessment of Waste Management of Volatile Radionuclides, prepared for the U.S. Environmental Protection Agency by the Mitre Corporation, Report ORP/CSD 79-2 (May 1979).

ANS 16.1

Measurement of the Leachability of Solidified Low-Level Radioactive Wastes, American Nuclear Society Draft Standard, ANSI/ANS-16.1 (April 1981).

ANSI

American National Standard Administrative Guide for Packaging and Transporting Radioactive Materials, American National Standards Institute, Inc. Report ANSI-N14.10.1-1973 (September 14, 1973).

BARNES

M. W. Barnes, B. E. Sheetz, L. D. Wakely, S. D. Atkinson, and D. M. Roy, Stability of I and Sr Radiophases in Cement Matrices, Scientific Basis for Nuclear Waste Management, Proc. of the Materials Research Society Annual Meeting, November 1981, S. V. Topp, Ed., North-Holland Publishing Co. Inc., New York, Vol. 6 (1982).

BIBLER-1977

N. E. Bibler and E. G. Orebaugh, Radiolytic Gas Production from Tritiated Waste Forms, E. I. du Pont de Nemours and Co., Savannah River Laboratory Report DP-1459 (July 1977).

BIBLER-1978

N. E. Bibler and M. L. Hyder, Radiolytic Gas Production from Concrete Containing Savannah River Plant Waste, E. I. du Pont de Nemours and Co., Savannah River Laboratory Report DP-1464 (January 1978).

BLANCETT

A. L. Blancett, Engineering Design of Integrated Hot Off-Gas Facility, E. I. du Pont de Nemours and Co., Savannah River Laboratory Report DP-1628 (March 1982).

BRAY

L. A. Bray, L. L. Burger, L. G. Morgan, and D. L. Baldwin, Thermal Release of Volatile Fission Products from Irradiated Nuclear Fuel, Hanford Engineering Development Laboratory, Pacific Northwest Laboratory Report PNL-4488, in publication.

BROWN-1978

R. A. Brown, D. A. Knecht, and T. R. Thomas, Reference Facility Description for the Recovery of Iodine, Carbon, and Krypton from Gaseous Wastes, Idaho Chemical Programs Operations Office, Allied Chemical Corporation Report ICP-1126 (April 1978).

BROWN-1980

R. A. Brown, Discussion of paper by E. Warnecke and S. Ahner, Air-Cooled Krypton-85 Storage Facility with Natural Convection, Management of Gaseous Wastes from Nuclear Facilities, International Atomic Energy Agency, Vienna, IAEA Report IAEA-SM-245/11, pp. 645-659 (December 1980).

BROWN-1983A

R. A. Brown, J. D. Christian, and T. R. Thomas, Airborne Radionuclide Waste Management, Exxon Nuclear Idaho Co., Inc. Report ENICO-1132 (June 1983).

BROWN-1983B

R. A. Brown, J. D. Christian, and T. R. Thomas, Airborne Radionuclide Waste Management Reference Document, Exxon Nuclear Idaho Co., Inc. Report ENICO-1133 (July 1983).

BURGER-1983A

L. L. Burger, Battelle Pacific Northwest Laboratories, private communication to L. E. Trevorrow, August 1983.

BURGER-1983B

L. L. Burger and R. D. Scheele, The Status of Radioiodine Control for Nuclear Fuel Reprocessing Plants, Pacific Northwest Laboratory Report PNL-4689 (July 1983).

BUSH

R. P. Bush, Carbon-14 Waste Management, United Kingdom Atomic Energy Authority, Harwell, Report AERE-R 10543 (April 1983).

CARD-1981

D. H. Card, P. H. Hunter, J. A. Adam, and R. B. White, Criteria for Greater Confinement, Ford, Bacon & Davis Utah, Inc., U.S. Dept. of Energy Report DOE/ID/01570--T20 (also NVO-234) (May 1981).

CARD-1983

D. H. Card, Ford, Bacon & Davis Utah, Inc., personal communication to L. E. Trevorrow, March 1983.

CHRISTENSEN

A. B. Christensen, J. A. Del Debbio, D. A. Knecht, J. E. Tanner, and S. C. Cossel, The Immobilization of Krypton-85 in Zeolite 5A and Porous Glass, Exxon Nuclear Idaho Co., Inc. Report ENICO-1102 (December 1981).

CRANDALL

D. B. Crandall, Alternative Waste Disposal Concepts, An Interim Technical Assessment, Bechtel National, Inc., San Francisco, CA, Battelle Office of Nuclear Waste Isolation Report ONWI-65 (November 1979).

CROFF-1980

A. G. Croff and C. W. Alexander, Decay Characteristics of Once-Through LWR and LMFBR Spent Fuels, High Level Wastes, and Fuel-Assembly Structural Material Wastes, Oak Ridge National Laboratory Report ORNL/TM-7431 (November 1980).

CROFF-1983

A. G. Croff and R. T. Jubin, Oak Ridge National Laboratory, Status and Strategy of the U.S. Commercial Airborne Waste Management Program, IAEA-CN-43/458, presented at the Int. Conf. on Radioactive Waste Management, IAEA, Seattle, WA, 16-20 May 1983.

DOE/EIS-0046F

Management of Commercially Generated Radioactive Waste, Vol. 1., Section 4.2.2, Reprocessing Cycle, U.S. Dept. of Energy Report DOE/EIS-0046F (October 1980).

DOE/ET-0028

Technology for Commercial Radioactive Waste Management, Vol. 3, Section 5.6, Krypton Storage, U.S. Dept. of Energy Report DOE/ET-0028, pp. 5.6.1-5.6.23 (May 1979).

DOE/NE-0017-1

Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, U.S. Dept. of Energy Report DOE/NE-0017-1 (October 1982).

DOE ORDER 5480.1, CHPT. III

Safety Requirements for the Packaging of Fissile and Other Radioactive Materials, Change 3, Department of Energy Order 5480.1, Chapter III (May 1981).

DOE ORDER 5480.1, CHPT. XI

Requirements for Radiation Protection, Change 6, Department of Energy Order 5480.1, Chapter XI (August 1981).

DOE ORDER 5820

Radioactive Waste Management, Department of Energy Order 5820, Draft (July 1983).

EDLING

D. A. Edling and J. F. Griffin, Certification of ERDA Contractors Packaging with Respect to Compliance with DOT Specification 7A Performance Requirements, Monsanto Research Corporation, Mound Laboratory Report MLM-2228 (June 1975).

EIS

Draft Environmental Impact Statement on 10 CFR 61 "Licensing Requirements for Land Disposal of Radioactive Waste", U.S. Nuclear Regulatory Commission Report NUREG-0782, Vol. 1 (September 1981).

EMERSON

E. L. Emerson, The Information Data Base at the Transportation Technology Center, Proc. Symp. on Waste Management, Tucson, AZ, March 8-11, 1981, R. G. Post and M. E. Wacks, Eds., Arizona Board of Regents, Vol. 1, pp. 367-374 (1982).

EPA

Radiological Impact Caused by Emissions of Radionuclides into Air in the United States, U.S. Environmental Protection Agency Report EPA-520/7-79-006 (August 1979).

FEDREG 10218

Requirements for Transportation of Radioactive Materials, U. S. Dept. of Transportation, Federal Register 48(48), 10218-10247 (March 10, 1983).

FEDREG 48234

Packaging of Radioactive Materials for Transportation and Transportation of Radioactive Material Under Certain Conditions: Compatibility with IAEA Regulations, Federal U.S. Nuclear Regulatory Commission, Proposed Rules, 10 CFR, Part 71, Federal Register 44(161), 48234 (August 17, 1979).

FEDREG 58196

Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes, U.S. Environmental Protection Agency, Proposed Rules, Code of Federal Regulations, Title 40, Part 191, Federal Register 47(250) (December 29, 1982).

FINNEY

B. C. Finney, R. E. Blanco, R. C. Dahlman, G. S. Hill, F. G. Kitts, R. E. Moore, and J. P. Witherspoon, Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle--Reprocessing Light-Water Reactor Fuel, Oak Ridge National Laboratory Report ORNL/NUREG/TM-6 (January 1977).

GIBBS

G. E. Gibbs, Management of High Specific Activity Tritiated Liquid Wastes, Tritium Waste Control: October 1981-March 1982, P. H. Lamberger and M. L. Rogers, Eds., Monsanto Research Corp., Mound Laboratory Report MLM-2969, pp. 11-16 (August 1982).

GRIMES

W. R. Grimes, E. C. Hampson, D. J. Larkins, J. O. Skolrud, and R. W. Benjamin, An Evaluation of Retention and Disposal Options for Tritium in Fuel Reprocessing, Oak Ridge National Laboratory Report ORNL/TM-8261 (August 1982).

HUNTER

P. H. Hunter, The Criteria and Technical Concept for Demonstrating Greater Confinement Disposal of Radioactive Wastes at Arid Western Sites, Proc. Third Annual Information Meeting DOE Low-Level Waste Management Program, New Orleans, LA, November 4-6, 1981, D. E. Large et al., Compilers, Oak Ridge National Laboratory Report ORNL/NFW-81/34 (December 1981).

IAEA-1980

The IAEA Revised Definition and Recommendations of 1978 Concerning Radioactive Wastes and Other Radioactive Matter Referred to in Annexes I and II to the Convention, INFCIRC/205/Add.1/Rev. 1 (August 1978); also available as Appendix B to A. N. Hagen, An Analysis of International Issues Associated with Ocean Disposal of Low-Level Radioactive Waste, prepared for the U.S. Environmental Protection Agency by MITRE Corp., MITRE Corp. Report MTR-80W159 (May 1980).

IAEA-1982

Site Investigations for Repositories for Solid Radioactive Wastes in Shallow Ground, Technical Series No. 216, International Atomic Energy Agency, Vienna (1982).

JEFFERSON

R. M. Jefferson and L. L. Bonzon, Available Packaging and Transportation Systems, Proc. Int. Symp. on the Management of Wastes from the LWR Fuel Cycle, Denver, CO, July 11-16, 1976, U.S. Dept. of Energy Report CONF-760701, p. 448 (1976).

KLETT

R. D. Klett, Ed., Krypton-85 Disposal Program Conceptual Design Phase: Final Report, Sandia National Laboratories Report SAND81-1957 (November 1981).

KNOX

J. B. Knox and K. R. Peterson, Estimates of Dose to Northern Hemisphere Population Groups from ^{85}Kr Emitted by a Single Nuclear Fuel Reprocessing Plant, Nucl. Saf. 13(2), 130-135 (1972).

LARGE

D. E. Large, R. S. Lowrie, L. E. Stratton, and D. G. Jacobs, Compilers, Proceedings of the Third Annual Information Meeting DOE Low-Level Waste Management Program, New Orleans, LA, November 4-6, 1981, Oak Ridge National Laboratory Report ORNL/NFW-81/34 (December 1981).

LITTLE

D. K. Little, R. S. Eby, J. L. Norton, J. L. Patton, R. M. Schultz, and J. M. Varagona, Noble Gas Removal and Concentration by Combining Fluorocarbon Absorption and Adsorption Technologies, Proc. of the 17th DOE Nuclear Air Cleaning Conf., Denver, CO, August 2-5, 1982, pp. 694-717 (February 1983).

MCKAY

H. A. C. McKay, Background Considerations in the Immobilization of Volatile Radionuclides, Management of Gaseous Wastes From Nuclear Facilities, International Atomic Energy Agency, Vienna, IAEA Report IAEA-SM-245/8, pp. 59-80 (December 1980).

MCMANUS

G. J. McManus, F. A. Duce, S. J. Fernandez, and L. P. Murphy, A Model of Iodine-129 Process Distribution in a Nuclear Fuel Reprocessing Plant, Exxon Nuclear Idaho Co., Inc. Report ENICO-1108 (April 1982).

MERSHAD

E. A. Mershad, W. W. Thomasson, and J. J. Dauby, Tritiated Liquid Waste Packaging, Monsanto Research Corporation, Mound Laboratory Report MLM-2440 (August 1977).

MOLECKE

M. A. Molecke, Gas Generation from Transuranic Waste Degradation: Data Summary and Interpretation, Sandia National Laboratories Report SAND79-1245 (December 1979).

MRC

Tritium Waste Control: April-September 1977, Monsanto Research Corporation, Mound Laboratory Report MLM-2484 (January 1978).

NEA

Radiological Significance and Management of Tritium, Carbon-14, Krypton-85, and Iodine-129, Nuclear Energy Agency, Organization for Economic Cooperation and Development, report by an NEA Group of Experts, OECD, Paris, France (1980).

NORMAN

C. Norman, U.S. Considers Ocean Dumping of Radwastes, Science 215, 1217-1219 (March 5, 1982).

NWN

Nuclear Waste News, Low-Level Radioactive Waste Disposal Container Introduced by NUS Corp., p. 200 (October 6, 1983).

ONWI-OCC

Interim Product Specifications and Data Requirements for Disposal of Defense High-Level Waste Forms in Salt Geologic Repositories, Battelle Project Management Division, Office of Nuclear Waste Isolation Report ONWI-OOC (December 1982).

PALMS

J. M. Palms, V. R. Veluri, and F. W. Boone, The Environmental Impact of I-129 Released by a Nuclear Fuel-Reprocessing Plant, Nucl. Saf. 16(5), 593-602 (1975).

PARSONS

Design Documentation - Krypton Encapsulation Preconceptual Design, The Ralph M. Parsons Company Report 6154-3-1, Pasadena, CA (July 1981).

PENZHORN

R. D. Penzhorn, P. Schuster, H. E. Noppel, and L. M. Hellwig, Long-Term Storage of Krypton-85 in Zeolites, Management of Gaseous Wastes from Nuclear Facilities, IAEA, Vienna, IAEA Report IAEA-SM-245/10, pp. 291-300 (December 1980).

PHILLIPS

J. Phillips, F. Feizollahi, R. Martineit, W. Bell, and R. Stouky, A Waste Inventory Report for Reactor and Fuel-Fabrication Facility Wastes, prepared by NUS Corporation for the U.S. Dept. of Energy and its contractor, Battelle Memorial Institute, Project Management Division, Office of Nuclear Waste Isolation Report ONWI-20 (March 1978).

PROUT

W. E. Prout, W. S. Durant, A. G. Evans, M. R. Buckner, and W. V. Wright, Management of Radioactive Waste Gases from the Nuclear Fuel Cycle, Volume II, Evaluation of Storage/Disposal Alternatives, Savannah River Laboratory, USNRC Report NUREG/CR-1546, Vol. II.

RN

Radwaste News 4(1) (January 4, 1983).

SCHEELE

R. D. Scheele and L. L. Burger, Selection of a Carbon-14 Fixation Form, Pacific Northwest Laboratory Report PNL-4447 (September 1982).

SIMON

Summary of the Discussions of Technical Panel I: Gaseous Wastes, in Radioactive Waste Management and Disposal, Proc. of the First European Community Conf., Luxembourg, May 20-23, 1980, R. Simon and S. Orlowski, Eds., Harwood Academic Publishers, New York (1980).

SMITH

M. J. S. Smith, Discussion of Technical Panel I: Gaseous Wastes, in Radioactive Waste Management and Disposal, Proc. of the First European Community Conf., Luxembourg, May 20-23, 1980, R. Simon and S. Orlowski, Eds., Harwood Academic Publishers, New York, p. 255 (1980).

TADLOCK

W. E. Tadlock and G. C. Abell, Radiation Chemistry Studies of Tritium Fixation Packages, in Tritium Waste Control: April-September 1977, Monsanto Research Corp., Mound Laboratory Report MLM-2484 (January 1977).

TILL

J. E. Till, E. L. Etnier, and H. R. Meyer, Methodologies for Calculating the Radiation Dose to Man from Environmental Releases of Tritium, Nucl. Saf. 22(2), 205-213 (1981).

TINGEY-1982A

G. L. Tingey, E. D. McClanahan, J. M. Lytle, N. R. Gorden, and R. W. Knoll, Krypton-85 Storage in Sputter-Deposited Amorphous Metals, Pacific Northwest Laboratory Report PNL-SA-10536 (June 1982).

TINGEY-1982B

G. L. Tingey et al., Krypton Implantation, Nuclear Waste Management Semi-annual Progress Report, October 1981-March 1982, T. D. Chikalla and J. A. Powell, Compilers, Battelle Pacific Northwest Laboratories Report PNL-4250-1 (June 1982).

VAN DALEN

A. Van Dalen, L. H. Von, and B. Verkerk, Study on the Possibility of Sea-Disposal of Krypton-85, Management of Gaseous Wastes from Nuclear Facilities, International Atomic Energy Agency, Vienna, IAEA Report IAEA-SM-245/1, pp. 597-613 (December 1980).

VELURI

V. R. Veluri, F. W. Boone, and J. M. Palms, The Environmental Impact of ¹⁴C Released by a Nuclear Fuel-Reprocessing Plant, Nucl. Safety 17(5), 580-590 (1976).

WAGGONER

R. C. Waggoner, Technical and Economic Evaluation of Processes for Krypton-85 Recovery from Power Fuel Reprocessing Plant Off-Gas, I. E. du Pont de Nemours and Co., Savannah River Laboratory Report DP-1637 (August 1982).

WARNECKE

E. Warnecke and S. Ahner, Air-Cooled Krypton-85 Storage Facility with Natural Convection, Management of Gaseous Wastes from Nuclear Facilities, International Atomic Energy Agency, Vienna, IAEA Report IAEA-SM-245/11, pp. 645-659 (December 1980).

WILLIAMS

J. D. Williams, Hittman Nuclear & Development Corp., Identification of Possible High Integrity Containers for Nuclear Waste Disposal, presented at Materials Research Society Annual Meeting, Boston, Massachusetts.

10 CFR 20

Standards for Protection Against Radiation, Code of Federal Regulations, Title 10, Part 20.

10 CFR 50 App F

Domestic Licensing of Production and Utilization Facilities, Appendix F, Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities, U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, Part 50.

10 CFR 60

Disposal of High-Level Radioactive Wastes in Geologic Repositories Technical Criteria, U.S. Nuclear Regulatory Commission, Proposed Rules, Code of Federal Regulations, Title 10, Part 60, Federal Register 48(120), 28194-28229 (June 21, 1983).

10 CFR 61

10 CFR Parts 2, 19, 20, 21, 30, 40, 51, 61, 70, 73, and 170, Licensing Requirements for Land Disposal of Radioactive Waste, U.S. Nuclear Regulatory Commission, Final Rule, October 1982.

10 CFR 71

Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions, U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, Part 71.

40 CFR 141

National Interim Primary Drinking Water Regulations, U.S. Environmental Protection Agency, Code of Federal Regulations, Title 40, Part 141.

40 CFR 190

Environmental Radiation Protection Standards for Nuclear Power Stations, U.S. Environmental Protection Agency, Code of Federal Regulations, Title 40, Part 190.

40 CFR 191

Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes, U.S. Environmental Protection Agency, Code of Federal Regulations, Title 10, Part 191, Federal Register 47(250), 58196-58206 (December 29, 1982).

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