Safety Indices and Their Application to Nuclear Waste Management Safety Assessment

J. W. Voss

April 1979

**Prepared for the U.S. Department of Energy
under Contract EY-76-C-06-1830**

**Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute**

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

The views, opinions and conclusions contained in this report are those of the contractor and do not necessarily represent those of the United States Government or the United States Department of Energy.

PACIFIC NORTHWEST LABORATORY
operated by
BATTELLE
for the
UNITED STATES DEPARTMENT OF ENERGY
Under Contract EY-76-C-06-1830

Printed in the United States of America
Available from
National Technical Information Service
United States Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151

Price: Printed Copy \$ ____*; Microfiche \$3.00

*Pages	NTIS
	Selling Price
001-025	\$4.00
026-050	\$4.50
051-075	\$5.25
076-100	\$6.00
101-125	\$6.50
126-150	\$7.25
151-175	\$8.00
176-200	\$9.00
201-225	\$9.25
226-250	\$9.50
251-275	\$10.75
276-300	\$11.00

3 3679 00049 1623

SAFETY INDICES AND THEIR
APPLICATION TO NUCLEAR WASTE
MANAGEMENT SAFETY
ASSESSMENT

J. W. Voss

April 1979

Prepared for
the U.S. Department of Energy
under Contract EY-76-C-06-1830

Pacific Northwest Laboratory
Richland, Washington 99352

SUMMARY

One preliminary safety study tool available for radioactive waste management assessment is the use of indices which may subjectively reflect the hazard or safety of a system, here called safety indices. In this study, thirteen indices have been examined to determine their potential applicability to operational waste management safety assessment. To present the indices, each is discussed as their authors presented them. Each tool is critiqued following its presentation. Two waste streams are presented on a normalized basis. These are the packaged spent fuel from the once-through fuel cycle and solidified high-level waste from a coprocessed UO_2 - PuO_2 fuel cycle. Seven of the indices are then calculated for a hypothetical surface storage scenario of the two wastes. The indices are examined on a consistent basis to identify any biases built into them, and to determine the sensitivities of each to various waste situations. The two waste streams are then compared on the basis of the indices to extend the understanding of the analysis techniques.

The results of the analysis fall into two categories, index evaluation and fuel cycle waste comparison. Only five of the indices are determined to be applicable to operational waste management safety assessment. The remainder are rejected either because they require very detailed input data; they are specifically designed for geologic isolation; they are extremely controversial in their application; or because they are particularly sensitive to a few specific radionuclides. The waste stream comparison yields three results: 1) the solidified high-level waste from the coprocessed UO_2 - PuO_2 fuel cycle may be potentially less hazardous than the packaged spent fuel from the once-through fuel cycle; 2) the removal of actinides, and especially plutonium, from spent fuel may reduce the potent hazard associated with the waste; and 3) after one million years of decay, the packaged spent fuel and solidified high-level waste are nearly the same on a hazard potential basis.

TABLE OF CONTENTS

SUMMARY	iii
INTRODUCTION	1
1.0 SAFETY INDICES	3
2.0 ALTERNATIVE FUEL CYCLES	35
2.1 SPENT FUEL DESCRIPTIONS	36
2.2 SOLIDIFIED HIGH-LEVEL WASTE DESCRIPTION	39
2.3 ANALYSIS SCENARIO	42
2.4 SUMMARY OF INPUT DATA	45
3.0 APPLICATIONS OF SAFETY INDICES	47
3.1 SAFETY INDICES OF THE ONCE-THROUGH FUEL CYCLE	47
3.2 SAFETY INDICES FOR THE COPROCESSED UO_2 - PuO_2 FUEL CYCLE	50
3.3 FUEL CYCLE COMPARISONS	52
4.0 CONCLUSIONS	61
4.1 SAFETY INDICES APPLICATION	61
4.2 FUEL CYCLE COMPARISONS	63
REFERENCES	Ref-1
APPENDIX A: GLOSSARY OF VARIABLES	A-1
APPENDIX B: TABULATION OF RCG VALUES FOR RADIONUCLIDES TO BE CONSIDERED	B-1
APPENDIX C: DECAY CONSTANTS OF RADIONUCLIDES IN SPENT FUEL	C-1

LIST OF FIGURES

1. Radioactivity Level in Various Waste Functions	5
2. Hazard Measure for LWR Fuel Cycle	8
3. Hazard Measures for LWR Wastes	15
4. Comparison of Waste and Fuel Hazard Measures	16
5. Waste Hazard Ratios for Different Values of d.	17
6. Hazard Index of PWR Waste as a Function of Decay Time	20
7. Hazards Available Index for Airborne Pathway: Radioactive Wastes in Underground Tank	21
8. An Estimated Behavior with Time of Typical PUREX Extraction Waste.	22
9. Relative Toxicity of Individual Fission Products in the Radioactive Waste from LWR Fuel; Burn-up 34,000 MWd/MTHM; Reprocessing after 150 Days of Cooling Time	26
10. Relative Toxicity Index of Individual Actinides in the High-Level Waste from LWR Fuel at the 1st Recycle. Burnup 34,000 MWd/MTHM; Reprocessing after 150 Days Cooling Time; Losses: 0.5% U and 0.5% Pu	27
11. Relative Toxicity of Nuclear Waste Over Time, Compared with that of Average Mineral Ores of Toxic Elements	28
12. Comparison of Hazards from High-Level Waste and from Uranium Ore- Processing Mill Tailings with Respect to Ingestion and Inhalation	31
13. Surface Storage Facility for Solidified High-Level Waste	43
14. Surface Storage Facility for Packaged Spent Fuel	44
15. Safety Indices of the Once-Through Fuel Cycle	49
16. Safety Indices of the Coprocessed UO_2 - PuO_2 Fuel Cycle	51
17. Radioactivity in Packaged Spent Fuel and Solidified High-Level Waste	53
18. Hazard Measure of Packaged Spent Fuel and Solidified High-Level Waste	54

19.	Modified Hazard Measure (HM2) of Packaged Spent Fuel and Solidified High-Level Waste	55
20.	Modified Hazard Measure (HM3) for Packaged Spent Fuel and Solidified High-Level Waste	56
21.	Potential Hazard Measure for Packaged Spent Fuel and Solidified High-Level Waste	57
22.	Hazard Index for Packaged Spent Fuel and Solidified High-Level Waste	59
23.	Relative Toxicity Index for Packaged Spent Fuel and Solidified High-Level Waste	60

LIST OF TABLES

1. Potential Hazard Measure of Several Significant Radionuclides Accumulated in High-Level Solid Waste by the Year 2020 . . .	18
2. Normalized Concentrations of Radioisotopes Dissolved from Devitrified High-Level Waste as they are Transported through the Environment	24
3. Long-Term Worldwide Waste Commitment	32
4. Summary of Safety Indices	33
5. Safety Indices and Their Formulation	34
6. Major Radionuclides in Spent Fuel	36
7. Performance Characteristics of Coprocessing	39
8. Major Radionuclides in Solidified High-Level Wastes	40
9. Summary of Input Data to Safety Indices	45
10. Radioactive Effluents from Coal-Fired Power Plants	46
11. Safety Indices of Once-Through Fuel Cycle	48
12. Safety Indices of Coprocessed $UO_2 - PuO_2$ Fuel Cycle	50

INTRODUCTION

This paper discusses the use of safety indices as a potential methodology for system safety assessment, explores the applications to nuclear waste management operations in two individual nuclear fuel cycles, and compares the two fuel cycles on the basis of their waste management analyses. This work was performed as a part of the Waste Management Safety Studies, sponsored by the Department of Energy. The objective of the study is to evaluate the safety of alternative waste management systems for viable nuclear fuel cycles up to the final disposal step; to develop the methodology necessary to permit the safety assessments; and to identify portions of systems that could benefit by safety design emphasis.

Safety assessment has the mission to assess and minimize the effects of potential accidents within some system. In its present state safety assessment may be segregated into three methodological categories. In increasing levels of effort, these are comparative, deterministic and probabilistic analysis. The comparative assessment technique contrasts two effects, one under examination and one as a baseline. The basis of the comparison may be subjective or numerical. The deterministic method examines the potential impacts of a system without regard to the frequency of that impact. A system may then have its safety judged on the basis of potential consequences. The probabilistic method couples the results of the deterministic method, or of a consequence analysis, with a calculation of the probability of an event occurring. When the entire spectrum of potential accidents has been considered, this technique is called risk assessment. The deterministic and probabilistic methods may be inputs to comparative assessments. All three methodologies contribute to the function of gaining perspectives about the safety of a system, and assessing and minimizing the potential effects of that system.

To perform a comprehensive safety assessment of the waste management operations of a nuclear fuel cycle requires detailed engineering designs of the facilities and accurate descriptions of the processing and handling procedures utilized within that fuel cycle. If sufficiently detailed information is available and with the proper effort, deterministic or probabilistic safety

assessment tools may be used for safety studies. However, the safety evaluation of alternative nuclear fuel cycles and their associated waste management operations restricts the analysis techniques which may be applied because of the large possible number of alternative fuel cycles and the relative scarcity of detailed design information within many fuel cycles. The inability to perform comprehensive safety assessments on waste management in alternative fuel cycles does not dispel the need to perform the analyses. The challenge is then to develop an assessment methodology which will allow a preliminary safety assessment to be performed in spite of the data limitations.

One assessment technique which has been proposed for the preliminary safety assessment of operational waste management systems is the use of indices which may subjectively reflect the hazard or safety of a system. These indices have been categorized under the single title of safety indices. Safety indices are comprised of many analysis formulas and techniques proposed by many researchers, each attempting to convey certain bits of information not included in previous techniques or perceived as being more important than data used in other analysis techniques.

It is the intent of this paper to present the various analyses which comprise safety indices, to critique the techniques, to apply the techniques to conceptual waste management operations, and to compare waste management operations for two nuclear fuel cycles. To present the various analysis techniques, each will be discussed as their authors presented them. A critique of each tool will follow its presentation. Waste management operations for two nuclear fuel cycles will be presented on a normalized basis, and each appropriate safety index will be applied. The techniques will be compared to identify any biases built into them, and finally, the safety index analysis results for the two fuel cycles will be compared.

1.0 SAFETY INDICES

Safety indices have been developed as comparative tools in safety assessment. Traditionally, safety indices have been used in place of a more comprehensive assessment method because of study limitations, either data or time. Safety indices may serve another purpose, to aid in the comparative safety assessment of alternative conceptual systems of which there is little available design information.

A review of the literature reveals many factors which have been included in the use of safety indices, as listed below:

- Quantity of radioactive material
- Decay constants
- Type of decay
- Energy of decay
- Physical form
- Chemical form
- Toxicity
- Time behavior
- Pathways to man
- Ingestion by man
- Public health and safety
- Probability of radioactive material in biosphere
- Comparison of hazards and risk to other materials
- Societal perception

Obviously, without the detailed results of a comprehensive safety analysis, safety indices could not accurately reflect all of these quantities. However, coupled with engineering experience and with design estimations, some of these parameters may be reflected with safety indices.

Two distinct types of safety indices have been developed. In broad terms, one analysis class introduces a single parameter to reflect the hazard or toxicity associated with the materials being discussed. The second, more controversial analysis method compares those hazards to some other material

which may be perceived as being "safe" or societally acceptable, or which exist in nature. In the following paragraphs, some of these safety indices are discussed, focusing on their methods, their potential utility and their weaknesses and limitation.

The determination of the amount of radioactive material in a waste form, Q , is a basic safety index. Q has the units of curies, or disintegrations per second. The use of Q as an indicator of hazard is widespread. Q is fundamental in any accident analysis since accident release magnitudes and quantities of radionuclides in nuclear wastes are expressed in curies (Smith, 1976). The total radioactivity also represents the maximum potential source term for any hypothetical release.

The total radioactivity, Q , in various phases of nuclear fuel cycles has been calculated in many studies, most recently using the computer code ORIGEN (Bell, 1973). The results of one such calculation is shown in Figure 1 (Winegardner and Jansen, 1974). Similar calculations may be performed for each alternative nuclear fuel cycle. As a safety analysis tool, this type of calculation may be used to show the quantities of radionuclides which are present at any time, thus the total potential source term. In addition, the time dependence of the existence of the radionuclides may be seen. If through some other analysis, the quantity of radioactive material below which control of the radionuclides is not necessary is defined, then the calculation of Q versus time will yield the time required for isolation. Regardless of the indirect implications, the use of Q in safety analysis is fundamental either as a direct measure or as a component of another calculation.

As a comprehensive tool, Q is lacking. While it is a basic input to nearly every other calculation, many important parameters necessary for safety analysis are not presented. These include the types and energies of the radioactive decays, the toxicity to man, the potential for release, the pathways to man, and other specific information detailing the nature of hazards posed by radioactive wastes.

The maximum permissible concentration or recommended concentration guide (RCG) and the maximum permissible intake (MPI) are both safety indices. The

FIGURE 1. Radioactivity Level in Various Waste Functions (Winegardner and Jansen, 1974)

RCG is defined for uncontrolled discharges of both soluble and insoluble radionuclides to either the air or the water (USNRC, 1976). The RCG is specified so that a "standard man" consuming both contaminated air and water will not exceed his annual dose limit, where the standard man consumes 7,300 m³/year of air and 0.8 m³/year of water.^(a) The MPI is defined as the product of RCG and annual consumption rate, as shown in Equations 1 and 2.

$$\text{MPI}_{\text{air}} = (7,300 \text{ m}^3/\text{yr}) (\text{RCG}_{\text{air}}) \quad (1)$$

$$\text{MPI}_{\text{water}} = (0.8 \text{ m}^3/\text{yr}) (\text{RCG}_{\text{water}}) \quad (2)$$

The MPI is defined for both soluble and insoluble species, since the RCG is defined for both. The units of RCG are microcuries per milliliter or curie per cubic meter as defined in the regulations (USNRC, 1976). The units of MPI are then curies per year.

The recommended concentration guide and maximum permissible annual intake, RCG and MPI, respectively, are poor parameters to use by themselves in discussing the safety of nuclear waste management. As stand-alone quantities, they may only indicate the relative toxicity of different radionuclides. However, they are of utility in safety assessment in two distinct areas. First, when compared for different radionuclides, RCG and MPI may correlate the relative toxicities of various species. Second, and more important, RCG and MPI will be fundamental inputs to other safety indices.

The most basic safety indices, Q and RCG, have been combined to form the hazard measure. The hazard measure of some radionuclide i, HM_i, is defined in Equation 3.

$$\text{HM}_i = Q_i / \text{RCG}_i \quad (3)$$

(a) The values for the air and water consumption of the standard or reference man are based on an air consumption of 830 l/hr and a water consumption of 2.2 l/day. Considerable variance may be found in the specification of air water consumption of the reference man. According to the ICRP (1975), the reference man consumes 1.1 m³/yr and the reference woman consumes 0.77 m³/yr of water. The annual air consumption for the reference man and woman ranges from 3,900 to 8,400 m³/yr and 3,100 to 7,700 m³/yr, respectively, depending on the level of activity of the individual.

Then the hazard measure of a collection of n radionuclides, HM , is defined in Equation 4.

$$HM = \sum_i HM_i = \sum_i \frac{Q_i}{RCG_i} \quad (4)$$

In both Equations 3 and 4, the RCG may be defined for soluble or insoluble species and for either air or water, depending on the needs of the analysis.

The units of HM is cubic meters of either air or water. The physical interpretation of HM is that it is the volume of air or water required to reduce the concentration of radionuclides to one RCG , i.e., the maximum permissible concentration. This is true for one radionuclide or a collection. The HM is not a constant over time. Just as Q is a time function, so the HM is a time dependent quantity. Thus, for a single radionuclide which decays to a stable nuclide, the HM will decay exponentially with the half-life of the species and will in the limit go to zero. Similar relations are true for decay chains such as the uranium series.

The hazard measure, HM , has been used extensively in gaining perspectives of waste management safety. Winegardner and Jansen (1974) calculated the HM for various portions of the light water reactor (LWR) fuel cycle, with full recycle as shown in Figure 2. Similar data may be developed for numerous other cases such as for specific radionuclides and for all radionuclides found in alternative nuclear fuel cycles as functions of time. The utility of HM is that it conveys both the quantity of radioactive material present in some analysis case as well as the radiotoxicity of the material. The HM is also very useful as an input to a more comprehensive calculation, and as the basis on which different cases may be compared. Various authors discuss the HM , and apply it to radioactive waste management in a variety of manners (McGrath, 1974; Hamstra, 1975; Smith, 1975; Haug, 1977; Karol and Post, 1977).

The hazard measure is one of the most widely used safety indices. In its most accepted use, the HM only demands the quantity of radioactive material to be considered as input from the nuclear fuel cycle analysis. While the physical

FIGURE 2. Hazard Measure for LWR Fuel Cycle
(Winegardner and Jansen, 1974)

interpretation of the HM is understood, that is the water or air to dilute Q to the RCG, the use of HM can yield misleading results. The real usefulness of the HM may be seen in comparing Figures 1 and 2. As shown in Figure 1, ore tailings are a negligible quantity in comparison to the total curies in high-level waste through at least 10^9 years of decay. However, in terms of HM, ore tailings are seen to exceed the HM of high-level radioactive waste after about 500 years of decay as shown in Figure 2. The information conveyed in this example is that in terms of total radiotoxicity, after about 500 years of decay, the high-level radioactive wastes may be less hazardous than mill tailings. Yet this conclusion does not give a complete picture. Factors not considered include the probability that either source will release radioactive material into the biosphere, the mobility of the radionuclides, the radiotoxicity of the nuclides, and the degree of local concentration of the radioactive sources, (e.g., a large inventory of radionuclides may be present in a repository whereas a lower total quantity of radionuclides may be present in each mill tail).

Several modified hazard measures have been proposed as safety indices. The first, here called HMI, attempts to account for man's exposure and consumption of radionuclides. HMI is defined below (Walsh, et al. 1977).

$$HMI_i = \sum_j Q_{ij}/Q_{Lij} \quad (5)$$

$$HMI = \sum_i HMI_i \quad (6)$$

where:

Q_{ij} = exposure, or dose to total-body, organ or tissue j
from all pathways from radionuclide i

Q_{Lij} = selected or established exposure limit for organ j
from radionuclide i

Q_{ij} may be further broken down as:

$$Q_{ij} = (Q_H + Q_G + Q_T + Q_W)_{ij} \quad (7)$$

where:

Q_H = exposure by inhalation

Q_G = exposure by ingestion

Q_T = exposure by transpiration

Q_W = exposure by wound entry

Each of these exposure modes may be broken down further to reflect the behavior of various species in the body.

The units of HMI are dependent upon those of its components. If the Q_{Lij} are comparable to the RCG, then HMI will be expressed in terms of volume of air or water. If Q_{Lij} are consumption limits, then HMI will have the units of time.

The modified hazard measures each offer improvements in regard to lending a comprehensive interpretation of a safety analysis. The HMI, presented in Equations 5, 6, and 7, accounts for specific pathways to and consumption modes of man for the various radionuclides. One problem using HMI is that man may not understand the biosphere well enough to calculate HMI without introducing a large uncertainty. In addition, the probability of a release occurring is not included in the analysis. Probably the most severe limitation in HMI is that the results of a comprehensive deterministic analysis are required as input. The reduction of the results of such an analysis to a single index is a drastic oversimplification.

A second modified hazard measure, HM2, is designed to account for radionuclides released to both air and water. The HM2 is defined in Equations 8 and 9 (McGrath, 1974).

$$HM2_i = Q_i [a_i/MPI_{wi} + b_i/MPI_{ai}] \quad (8)$$

$$HM2 = \sum_i HM2_i \quad (9)$$

where:

a_i, b_i = fractions of Q_i released in water and air, respectively.

The HM2 is designed to account for releases to both air and water. Thus, sufficient design information to determine the fractions of Q which might be released in the air or water must be determined. This determination may be for chronic or acute releases. Where sufficiently detailed design information is not available, a and b must be estimated, thus introducing uncertainties into the analysis.

A third modified hazard measure, HM3, is defined by Smith (1975), as shown in Equations 10 and 11.

$$HM3_i = \int_t^{t+d_i} Q_i(t')/MPI_i dt' \quad (10)$$

$$HM3 = \sum_i HM3_i \quad (11)$$

where:

d_i = the time period of concern over radionuclide i.

The term d_i , the time period of concern, is meant to account for the behavior of radionuclide i in the environment, as well as how man might perceive the presence of i in the biosphere. If it is felt that its effect will be on the order of generations, d_i might be 100 years; if over civilizations, d_i might be 5,000 years; if over the period of human existence, 5,000,000 years or longer.

The calculation of HM3 may be seen in a simple example. Assume that 1 Ci of ^{90}Sr is disposed of in the year 2000. The time period of concern for ^{90}Sr released to the environment is assumed to be 100 years. A failure is assumed to occur which releases soluble ^{90}Sr to the environment by a liquid pathway. Failure takes place at one of three times, at the years 2000, 2400, or 3000. For failure at 2000, t takes the value of zero in the integral of Equation 10, for 2400, $t = 400$, and for 3000, $t = 1000$. The term $t+d$ takes the value of 100, 500, and 1100 years, respectively. Recalling the half life of ^{90}Sr is 28.1 years, that the water intake of the standard man is $0.8 \text{ m}^3/\text{yr}$; and that the RCG_w of soluble ^{90}Sr is $3 \times 10^{-7} \text{ Ci/m}^3$ (USNRC, 1976), then HM3 may be determined as shown below.

- Failure Year = 2000, $t = 0$, $HM3 = 2 \times 10^8$
- Failure Year = 2400, $t = 400$, $HM3 = 8 \times 10^3$
- Failure Year = 3000, $t = 1000$, $HM3 = 3 \times 10^{-3}$

The measure HM3 introduces a term which reflects both societal perception and environment behavior, as shown in Equation 10. The time period of concern, while meant to lend a deeper meaning to HM3, presents a problem in its determination. No rigorous method exists for calculating such a value. Thus, estimation of this term is at best subjective.

It is significant that one application of the safety indices has been to determine the time period of concern for the presence of nuclear wastes in the environment. Hamstra (1975) used the HM to show that nuclear wastes from a U-Pu recycle fuel cycle need only be isolated from the biosphere for 300 to 500 years. His conclusion was founded on the comparison of $HM(t)$ for all reprocessing wastes, with uranium ore and mill tailings, as normalized to a consistent basis. Lyon (1976) used a similar argument to recommend the partition and transmutation of actinides resulting from the operation of a CANDU reactor. In both cases, the comparative base was a hazard perceived as being safe or existing. Selection of such a base is a subjective matter since the hazards from mill tailings and mined uranium ores are of some concern in this country, and thus are not universally viewed as being "safe".

A fourth modified hazard measure, called the potential hazard measure (PHM) is defined below (Gera and Jacobs 1972).

$$PHM_i = P_i \frac{Q_i}{MPI_i} \frac{1}{\lambda_i} \quad (12)$$

$$PHM = \sum_i PHM_i \quad (13)$$

where:

- PHM_i = potential hazard measure of radionuclide i
- P_i = probability of i being released to man
- λ_i = decay constant of i
- PHM = potential hazard measure of all radionuclides in a reference set

The PHM includes a term P_i , the probability of radionuclide i being released to man. Providing this term may be calculated, then the PHM is similar to the conventional definition of risk, that is the product of the probability of an event and the consequences of that event. Without detailed working knowledge of the system being analyzed, P_i may only be estimated, thus minimizing the utility of the calculation.

Considering for a moment the safety indices HM, HM2, HM3, and PHM, an interesting observation may be made. Recalling the definition of HM3; in Equation 10,

$$HM3_i = \int_t^{t+d_i} Q_i(t')/MPI_i dt'$$

and that $Q_i(t')$ may be expressed as:

$$Q_i(t') = Q_{oi} e^{-\lambda_i t'}$$

where:

$$Q_{oi} = \text{curies of } i \text{ present at } t = 0,$$

the $HM3_i$ may be rewritten as:

$$HM3_i = \frac{Q_{oi}}{MPI_i} \int_t^{t+d_i} e^{-\lambda_i t'} dt'$$

or

$$HM3_i = \frac{Q_{oi}}{(MPI_i)(\lambda_i)} \left[e^{-\lambda_i t} - e^{-\lambda_i (t + d_i)} \right]$$

When d_i , the period of concern, is greater than the half life of i , $0.1/2_i$, by a factor of three or more, then $HM3_i$ is approximately

$$HM3_i \cong Q_{oi} e^{-\lambda_i t} \left[\frac{1}{(MPI_i)(\lambda_i)} \right]$$

or

$$HM3_i = Q_i(t) \left[\frac{1}{(MPI_i)(\lambda_i)} \right]$$

If the probability of a radionuclide release, P_i , used in the calculation of PHM_i , is conservatively estimated at unity, then the $HM3_i$ and PHM_i are seen to be identical. In other words, $HM3_i$ may be shown to be a special case of PHM_i .

Simplifying even further, PHM_i , $HM3_i$, and $HM2_i$ may each be reduced to HM_i multiplied by a constant, as summarized below.

$$HM_i = Q_i / RCG_i$$

$$HM2_i = Q_i / MPI_i = \alpha (Q_i / RCG_i) = \alpha HM_i$$

where:

$$\alpha = 1 / (\text{annual consumption of air or water})$$

$$HM3_i \cong (Q_i / MPI_i)(\lambda_i)^{-1} = \beta (Q_i / RCG_i) = \beta HM_i$$

where:

$$\beta = 1 / [(\lambda_i)(\text{annual consumption of air or water})]$$

$$PHM_i = P_i Q_i / (MPI_i)(\lambda_i) = \xi (Q_i / RCG_i) = \xi HM_i$$

where:

$$\xi = P_i / [(\lambda_i)(\text{annual consumption of air or water})]$$

All of the modified hazard measures, have been used in nuclear waste management safety assessment. Smith and Kastenberg (1976) plotted the HM and the PHM of Equation 13 versus time for LWR wastes, as shown in Figure 3.

In Figures 4 and 5, Smith and Kastenber (1976) plotted HM3 for different radioactive waste components of the nuclear fuel cycle, and HM3 as a function of release time and period of concern, respectively. Gera and Jacobs (1972) calculated the PHM of solidified high-level waste expected from a U.S. nuclear economy, as shown in Table 1.

FIGURE 3. Hazard Measures for LWR Wastes (Smith and Kastenber, 1976)

Smith and Kastenbergl also plotted the HM3 of Equation 11 for LWR wastes, as well as showing the sensitivity of HM3 to variations in the time period of concern as shown in Figures 4 and 5, respectively. Gera and Jacobs (1972) calculated the potential hazard measure for all high level waste anticipated by the year 2020, as shown in Table 1.

FIGURE 4. Comparison of Waste and Fuel Hazard Measures (Smith and Kastenbergl, 1976)

FIGURE 5. Waste Hazard Ratios for Different Values of d
(Smith and Kastenber, 1976)

TABLE 1. Potential Hazard Measure of Several Significant Radionuclides Accumulated in High-Level Solid Waste by the Year 2020 (from Gera and Jacobs, 1972)

Nuclide	Quantity (Ci)	Potential Hazard Measure	
		Ingestion	Inhalation
<u>At Year 2020</u>			
⁹⁰ Sr	2.9 x 10 ¹⁰	3.6 x 10 ¹⁷	4.0 x 10 ¹⁷
¹³⁷ Cs	5.7 x 10 ¹⁰	2.0 x 10 ¹⁶	1.5 x 10 ¹⁶
²³⁸ Pu (a)	1.7 x 10 ⁸	5.3 x 10 ¹⁴	4.3 x 10 ¹⁸
²³⁹ Pu	8.4 x 10 ⁶	8.1 x 10 ¹⁵	7.4 x 10 ¹⁹
²⁴⁰ Pu (b)	1.3 x 10 ⁷	3.4 x 10 ¹⁵	3.1 x 10 ¹⁹
²⁴¹ Am (c)	7.9 x 10 ⁸	1.6 x 10 ¹⁶	3.3 x 10 ¹⁹
²⁴³ Am	2.7 x 10 ⁷	8.2 x 10 ¹⁵	2.0 x 10 ¹⁹
<u>After 300 Years Decay</u>			
⁹⁰ Sr	1.6 x 10 ⁷	2.0 x 10 ¹⁴	2.2 x 10 ¹⁴
¹³⁷ Cs	5.7 x 10 ⁷	2.0 x 10 ¹³	1.5 x 10 ¹³
²³⁸ Pu	1.6 x 10 ⁷	5.1 x 10 ¹³	4.1 x 10 ¹⁷
²³⁹ Pu	8.4 x 10 ⁶	8.1 x 10 ¹⁵	7.4 x 10 ¹⁹
²⁴⁰ Pu	1.3 x 10 ⁷	3.4 x 10 ¹⁵	3.1 x 10 ¹⁹
²⁴¹ Am	5.0 x 10 ⁸	1.1 x 10 ¹⁶	2.1 x 10 ¹⁹
²⁴³ Am	2.7 x 10 ⁷	8.2 x 10 ¹⁵	2.0 x 10 ¹⁹
<hr/>			
(a) Assumes all ²⁴² Cm decayed to ²³⁸ Pu.			
(b) Assumes all ²⁴⁴ Cm decayed to ²⁴⁰ Pu.			
(c) Assumes all ²⁴¹ Pu decayed to ²⁴¹ Am.			

The hazard index, HI, shown in Equations 14 and 15, is a relative quantity. The concentration of radionuclides in a control volume, say a waste form, is divided by the RCG, thus yielding, the number of times the radionuclides would which it is contained, V, as shown in Equations 14 and 15.

$$HI_i = HM_i/V = \frac{Q_i/V}{RCG_i} \quad (14)$$

$$HI = \sum_i HI_i \quad (15)$$

Unlike HM, the HI is a dimensionless number.

The HI has been used to examine various aspects of radioactive waste management. Claiborne (1975) used the HI to examine the benefits of actinide removal from high-level wastes, as shown in Figure 6. The number of similar calculations which may utilize the HI are almost limitless. The HI may be calculated for specific radionuclides as well as for total inventories generated in any fuel cycle.

The hazard index, HI, shown in Equations 14 and 15, is a relative quantity. The concentration of radionuclides in a control volume, say a waste form, is divided by the RCG, thus yielding, the number of times the radionuclides would need to be diluted to achieve the RCG. However, as with the HM, the HI has distinct pieces of information missing, such as the actual movement of radionuclides through the environment and the effects of those species on man.

A hazards available index, HA, has been developed to account for the pathways over which radionuclides must travel to reach man, as shown below (following Bruns, 1976).

$$HA = \log_{10} (HI) + TF \quad (16)$$

where TF = transport factor

The transport factor is defined over the various environmental pathways as shown in Equation 17.

FIGURE 6. Hazard Index of PWR Waste as a Function of Decay Time (Claiborne, 1975)

$$TF = \log_{10} \left[TF_{R_1} + TF_{R_2} + \dots + TF_{R_m} \right] \quad (17)$$

where:

$R_1, \dots, R_m$ = routes to man by consecutive pathways

$$TF_{R_1} = \left[(P_a)(P_b) \dots (P_n) \right]_i \quad (18)$$

P_n = fraction of radioactive material transported by pathway a, b, . . . , n

The use of HA is demonstrated in Figure 7 for the case of radioactive material in an underground tank reaching man via one airborne pathway. To determine the total HA for this case, all pathways would need to be evaluated.

HAZARD AVAILABLE = TRANSPORT FACTORS + HAZARD INDEX

$$HA = TF + \log_{10} HI$$

$$HI = \sum_i \frac{Q_i/V}{RCG_{ai}}$$

$$TF = \log_{10} [(P_a)(P_b)(P_c)(P_d)]$$

WHERE P_a = FRACTION OF (Q) MOVED TO SURFACE BY DIFFUSION

P_b = FRACTION OF (P_a) (Q) MOVED TO AIR BY RESUSPENSION

P_c = FRACTION OF (P_b) (P_a) (Q) MOVED NEAR MAN, REDUCED BY DILUTION

P_d = FRACTION OF (P_c) (P_b) (P_a) (Q) REACHING MAN BY DEPOSITION

FIGURE 7. Hazards Available Index for Airborne Pathway: Radioactive Wastes in Underground Tank (after Bruns, 1976)

The use of the HA has been rather limited. This is caused by its relatively recent development and the complexity of the input data required. Bruns (1976) calculated the HA for "typical" PUREX waste and plotted it versus time. On the same figure, he plotted the HA for fallout as a comparative basis, as shown in Figure 8. Note that the HA uses the HI as a basic parameter.

FIGURE 8. An Estimated Behavior with Time of Typical PUREX Extraction Waste (Bruns, 1976)

The hazards available index, HA, presented in Equations 16, 17 and 18, is a comprehensive assessment of movement of radioactive material through the biosphere to man. It considers individual pathways as well as the bulk quantity of radioactive material present. Two major terms are missing. These are the probabilistic portion of the safety analysis, and the consequences portion. A more quantitative limitation of the HA index is the level of effort required to complete its calculation, provided the individual terms may themselves be determined.

Another safety index proposed is the isolation time, T (Voss and Post, 1976). This parameter's meaning is the minimum time which groundwater must travel during which the various radionuclides must be retarded by chemical exchange reactions to reduce their concentrations to one RCG, given a set of sorption characteristics, dilution parameters, and radioactive waste form leaching characteristics, as shown in Equation 19 for one radionuclide.

$$T_i = - \left(\frac{1}{\lambda I} \log_e \left[\text{RCG} \frac{V_f}{A} \frac{FD}{L} \right] \right)_i \quad (19)$$

where:

- I = ion exchange holdup factor ($1 + K_D$)
- V_f = volume flow rate of groundwater
- FD = factor of dilution
- A = available leaching area of waste form
- L = leach rate of waste form

Calculation of T_i assumes that the parameters are constant, independent of time. The dilution factor, D, is calculated as a volume ratio of the uncontaminated and contaminated water which mix during radionuclide transport to man. Note that a single T_i may not be readily calculated for a set of radionuclides.

The isolation time presented in Equation 19, has not been widely used. It was used to calculate the minimum time of groundwater transport to achieve RCG levels of dominant fission products in high-level waste, as shown in Table 2 (Voss and Post, 1976). Calculation of T requires detailed definition of the waste management systems as well as the biosphere.

**TABLE 2. Normalized Concentrations of Radioisotopes
Dissolved from Devitrified High-Level Waste
as they are Transported through the
Environment(a)**

^{90}Sr	Normalized Concentration ^(b)					Groundwater Transport Time (Years)
	^{154}Eu	^{106}Ru	^{147}Pm	^{144}Ce	^{137}Cs	
1.02E10	1.56E9	5.30E8	3.72E8	1.38E8	4.50E7	0
9.6E9	5.28E8	4.0E-2	4.74E5	3.20E-2	3.57E7	0.01
9.72E9	1.79E8	--	6.05E2	--	2.83E7	0.02
9.49E9	6.05E7	--	7.71E-1	--	2.25E7	0.03
9.26E9	2.05E7	--	--	--	1.79E7	0.04
9.04E9	6.94E6	--	--	--	1.42E7	0.05
8.83E9	2.35E6	--	--	--	1.13E7	0.06
8.62E9	7.95E5	--	--	--	8.93E6	0.07
8.41E9	2.69E5	--	--	--	7.09E6	0.08
8.21E9	9.12E4	--	--	--	5.63E6	0.09
8.02E9	3.09E4	--	--	--	4.46E6	0.10
6.30E9	6.11E-1	--	--	--	4.43E5	0.20
4.95E9	--	--	--	--	4.39E4	0.30
3.89E9	--	--	--	--	4.36E3	0.40
3.06E9	--	--	--	--	4.33E2	0.50
2.41E9	--	--	--	--	4.29E1	0.60
1.89E9	--	--	--	--	4.26	0.70
1.49E9	--	--	--	--	4.22E-1	0.80
1.17E9	--	--	--	--	--	0.90
9.19E8	--	--	--	--	--	1.00
8.28E7	--	--	--	--	--	2.00
7.46E6	--	--	--	--	--	3.00
6.72E5	--	--	--	--	--	4.00
6.06E4	--	--	--	--	--	5.00
5.46E3	--	--	--	--	--	6.00
4.92E2	--	--	--	--	--	7.00
3.99	--	--	--	--	--	8.00
3.60E-1	--	--	--	--	--	9.00
--	--	--	--	--	--	10.00
<u>Isolation Time (Years)</u>						
<9.0	<0.20	<0.01	<0.03	<0.01	<0.80	

a. Following Voss and Post, 1976.

b. Each calculated concentration is normalized to the RCG. Then when the normalized concentration equals unity, the concentration is equal to the RCG.

The isolation time is lacking in several areas. First, the calculation considers only radionuclide transport by groundwater, and uses a rather simplistic model for that pathway. Leaching is considered the only release mechanism. The actual interaction of the various species with man is not considered, nor is the probabilistic analysis considered. In that it was designed for use in waste disposal, it has limited application to waste management operations of alternate nuclear fuel cycles.

Several authors discuss a relative toxicity index (RTI) as a safety index (Hamstra, 1975; Haug, 1976; J. Cohen and Tonnessen, 1977; Haug, 1977; Rochlin, 1977). The RTI may be defined as the ratio of water required to dilute a quantity of waste to RCG levels and the quantity required to dilute the amount of uranium ore mined to generate the waste quantity. This is shown in Equation 20.

$$RTI = \frac{\sum_i (Q_i / RCG_i) \text{ waste}}{\sum_j (Q_j / RCG_j) \text{ equivalent uranium ore}} \quad (20)$$

Applied to a more general case the hazard measure associated with radioactive waste may be compared to the hazard measure of some other substances perceived as posing no excessive hazards to the public. This may include various naturally occurring elements (J. Cohen and Tonnessen, 1977). Note that for this calculation to make sense, the RTI must be redefined to reflect the concentrations of the various materials in their matrix as shown in Equation 21.

$$RTI = \frac{\sum_i (Q_i / RCG_i) \text{ waste}}{\sum_j (Q_j / RCG_j) \text{ element as comparative base}} \quad (21)$$

The denominator of Equation 21 may reflect some nonradioactive species as well as radioactive ones. A stable element or compound equivalent to RCG must be defined making this calculation possible.

The relative toxicity index has been used as a safety assessment tool for radioactive waste management. Haug (1977) plotted the RTI of fission products and of actinides in high-level wastes as a function of time with uranium ore as the comparative basis, as shown in Figures 9 and 10. J. Cohen and Tonnessen (1977)

plotted the RTI for LWR wastes, fast breeder reactor (FBR) wastes, and several natural occurring elements (Hg, Cr, Se, Pb, Cd, Ag, Ba, As) using 0.2% uranium ore as a basis, as shown in Figure 11. This type of comparison may be made for any number of additional examples. Depending on the scope of the safety analysis performed, hazards associated with radioactive material may be compared to many relevant quantities.

FIGURE 9. Relative Toxicity of Individual Fission Products in the Radioactive Waste from LWR Fuel; Burn-up 34,000 MWd/MTHM; Reprocessing after 150 Days of Cooling Time. (Haug, 1977)

FIGURE 10. Relative Toxicity Index of Individual Actinides in the High-Level Waste from LWR Fuel at the 1st Recycle. Burnup 34,000 MWd/MTHM; Reprocessing after 150 Days Cooling Time; Losses: 0.5% U and 0.5% Pu. (Hauq, 1977)

FIGURE 11. Relative Toxicity of Nuclear Waste Over Time, Compared with that of Average Mineral Ores of Toxic Elements. (J. Cohen and Tonnessen, 1977)

The relative toxicity indices discussed have several advantages. In calculating RTI, radioactive waste is compared to something either considered as not posing a hazard to man, or which is existing and to which man has no choice in accepting its hazard. However, in making the calculation, a safety analyst is pressed into making these philosophical judgments on his own, thus making

the analysis results subjective. In addition, the calculation does not consider that the radioactive waste and the reference material may differ in their availability to man.

Several authors carry the comparison of hazards from wastes and existing hazards into a more controversial arena. This is a calculation of the maximum number of cancer mortalities from both radioactive wastes and the material of comparative base over some length of time (J. Cohen, 1975; Bruns, 1976; B. Cohen, Feb. 1976; B. Cohen, Oct. 1976; B. Cohen, 1977; J. Cohen and Tonnessen, 1977).

A simplified explanation of one method used for these calculations follows for the case of ingestion of radionuclides. An initial calculation is to be made of the number of cancer deaths per year resulting from the ingestion of a particular radionuclide, i , in spent fuel resulting in a cancer in some body organ, j (B. Cohen, 1977). The number of lethal doses per year, D_{ij} , is:

$$D_{ij} = \frac{Q_i}{RCG_{ij}} \times \frac{d_{ij}}{C} \times r_j \times P \quad (22)$$

where:

- Q_i = quantity of i per unit mass of spent fuel (Ci/MT)
- RCG_{ij} = maximum permissible concentration of i for limiting body organ j (Ci/m³)
- d_{ij} = dose to organ j resulting from one year's consumption of water with 1 RCG of i (rem)
- C = annual consumption of water for standard man (0.8 m³/yr)
- r_j = risk of cancer death from dose to body organ j per rem (lethal doses/rem)
- P = mass of spent fuel produced annually (MT/yr)

To calculate the overall number of cancer deaths per year for all radionuclides in spent fuel and for cancers in all body organs, D , the following summation would be formed:

$$D = \sum_j \text{body organs} \sum_i \text{radionuclides} D_{ij} \quad (23)$$

To complete the comparative analysis, a similar calculation would be performed for the amount of uranium found in nature or for some other material.

The more controversial comparison of maximum cancer mortalities from radioactive wastes and from some comparative base has been used to demonstrate that high-level waste, and in particular plutonium, is a lessor hazard than others which man lives with. B. Cohen (1977) calculated that after less than 1000 years of decay, high-level radioactive waste has a lower cancer potential than does uranium mill tailings, when ingestion is considered. A similar calculation was made for inhalation, both as shown in Figure 12. J. Cohen (1975) showed that the toxic production rate of ^{239}Pu which goes to waste projected for the year 2000 is comparable to the toxic production rate of lead to waste in 1973, as shown in Table 3. He further argues that ^{239}Pu has a 24,000 year half-life while lead is stable and will persist indefinitely in the environment. Similar calculations may be made about more specific quantities relating to radioactive waste management.

The safety index of cancer mortalities is an example of a more sophisticated analysis, here a deterministic method, being simplified for use in a comparative analysis. While this index may be a maximum estimate of the impacts of nuclear waste, several difficulties exist in using such an index. Seldom are individual pathways examined in this technique. Instead, the radioactive material is assumed to be homogeneously dispersed throughout the environment. In addition to this enormous dilution, the linear dose hypothesis is assumed. More significantly, an implicit assumption to the method is that man-made and system changes have no influence on a system's ability to retain radioactive material (Campbell, McGrath and Cullingford, 1978). While many technical arguments have developed over the validity of the assumptions, polarization among the technical community without technical resolution seems to be the major discussion result. In the author's opinion the main problem with using the cancer mortality techniques is that the controversy surrounding the use detracts from the results, so that perspectives which may have been gained are distorted by the controversy.

FIGURE 12. Comparison of Hazards from High-Level Waste and from Uranium Ore-Processing Mill Tailings with Respect to Ingestion and Inhalation. (B. Cohen, 1977)

TABLE 3. Long-Term Worldwide Waste Commitment

	Plutonium-239 (Year-2000)	Lead (Year-1973)
Annual Waste Production (g/yr)	1×10^7 (a)	4×10^8 (c)
Approximate Lethal Dose (g) ^(b)	0.4	10.0
Toxic Production Rate $\frac{(\text{Lethal Doses})}{\text{Year}}$	3×10^7	4×10^7
Persistence (Half-Life)	2.4×10^4 Yr	∞

(a) U.S. Projected for Year 2000 = 0.2 MCi (ORNL-4451). U.S. Nuclear Capacity in Year 2000 = 1/3 World Capacity (UCID-16670).

(b) Via Ingestion of Soluble Material.³

(c) World Production (1973) - 4.1×10^3 Tons (U.S. Stat. Abs.)
Assume 10% is Lost to Waste.

To summarize, a variety of methods have been proposed for use as safety indices. Each method introduces or exempts information from its calculation not found in other methods. To display this, all of the methods introduced are shown in Table 4, along with an indication of the information included in the technique. In addition, all of the indices, and their formulations, are shown on Table 5. A glossary of all of the variables presented in this section and their meaning is in the Appendix.

TABLE 4. Summary of Safety Indices

Index	Radionuclide Content(a)	Regulatory Definition(b)	Standard Consumption(c)	Release Scenario(d)	Pathway Analysis(e)	Societal Hazard Potential Perception(f)	Probabilistic Release Calculation(g)	Waste Properties			Environmental Scenario(h)	Identify Comparative Basis(i)	Dose Response Calculation(j)
								Decay Constant	Specific Volume	Leach Rate			
Q	X												
RCG		X											
MPI		X	X										
HM	X	X											
HM1		X	X	X	X								
HM2	X	X	X	X									
HM3	X	X	X			X							
PHM	X	X	X				X	X					
HI	X	X							X				
HA	X	X		X	X				X				
T _i		X		X	X			X		X	X		
RTI	X	X										X	
D	X	X	X									X	X

- NOTES: (a) This is the calculation of radionuclides as a function of time.
 (b) This is the RCG limits as established by USNRC in 10CFR20, App. B, or by ICRP.
 (c) The water and air consumption by a standard man are 0.8 and 7,300 m³/yr, respectively.
 (d) A release scenario implies the calculation of a radionuclide source term, the phase of the release, either water or air, specific radiochemical data, etc.
 (e) This is the calculation of the behavior of the radionuclides in the environment.
 (f) This is the calculation of the time period of concern for a release.
 (g) This is the calculation of the probability of a release occurring.
 (h) This is the specification of the environment into which radionuclides are released.
 (i) This is the establishment of a comparative basis.
 (j) This is the determination of the dose-response model, eg., the linear dose-response model.

TABLE 5. Safety Indices and Their Formulation

<u>INDEX TITLE</u>	<u>DEFINITION</u>
TOTAL RADIOACTIVITY	$Q = \sum_{i=1}^n Q_i(t)$
RECOMMENDED CONCENTRATION GUIDE	RCG, AS IN IOCFR 20
MAXIMUM PERMISSIBLE INTAKE	$MPI_{air,water} = (RCG) \text{ times (annual consumption)}_{air,water}$
HAZARD MEASURE	$HM = \sum_{i=1}^n Q_i / RCG_i$
MODIFIED HAZARD MEASURE 1	$HM1 = \sum_{i=1}^n \sum_{j=1}^m Q_{ij} / Q_{Lij}$
MODIFIED HAZARD MEASURE 2	$HM2 = \sum_{i=1}^n Q_i [a_i / MPI_{wi} + b_i / MPI_{ai}]$
MODIFIED HAZARD MEASURE 3	$HM3 = \sum_{i=1}^n \int_t^{t+d} Q_i(t') / MPI_i dt'$
POTENTIAL HAZARD MEASURE	$PHM = \sum_{i=1}^n P_i \frac{Q_i}{MPI_i} \frac{1}{\lambda_i}$
HAZARD INDEX	$HI = \sum_{i=1}^n \frac{Q_i / V_i}{RCG_i}$
HAZARDS AVAILABLE INDEX	$HA = \log_{10} (HI) + TF$
ISOLATION TIME	$T_i = - \left(\frac{1}{\lambda} \log_e \left[RCG \frac{V_f FD}{A L} \right] \right)_i$
RELATIVE TOXICITY INDEX	$RTI = HM_{WASTE} / HM_{COMPARATIVE \text{ BASE}}$
CANCER DEATHS	$D = \sum_{i=1}^n \sum_{j=1}^m \frac{Q_i}{RCG_{ij}} \frac{d_{ij}}{C} r_j P$

2.0 ALTERNATIVE FUEL CYCLES

In the following sections, the spent fuel discharged as waste from the once-through fuel cycle and the solidified high-level waste discharged from a coprocessed UO_2 - PuO_2 fuel cycle are described in sufficient detail to apply the various safety indices. The facilities and management procedures described in this section for spent fuel and solidified high-level waste are conceptual. They are introduced only for the application of the various methodologies. As shown in Table 4, the data about the waste forms required to utilize all of the safety indices are:

- Radionuclide Content (Q)
- Regulatory Definition (RCG)
- Standard Consumption (MPI)
- Release Scenario
- Pathway Analysis
- Societal Hazard Potential Perception
- Probabilistic Release Calculations
- Decay Constant
- Specific Volume
- Leach Rate
- Environmental Scenario
- Comparative Basis
- Dose Response Calculation

A common reference point is required within the two fuel cycles to assure equity in the analysis. The spent fuel and solidified high-level waste are the waste streams containing the majority of the radionuclides within the once-through and coprocessed UO_2 - PuO_2 fuel cycles, respectively. The normalization basis will be the energy generated, or $\text{GW(e)-yr.}^{(a)}$ When a scenario is necessary to complete the analysis, the two wastes will be assumed to be in interim storage in a surface storage facility, following the RSSF (ARHCO, 1977). The spent fuel is described within Section 2.1, the solidified high-level waste in Section 2.2, and the surface storage facility and one postulated failure scenario in Section 2.3.

(a) The unit energy normalization must be used with care, as it may inherently bias the analysis when accumulated inventories are involved.

2.1 SPENT FUEL DESCRIPTIONS

Spent fuel is assumed to be discharged from a typical Light Water Reactor (LWR). The radionuclide content of spent fuel, considering only major or long-lived quantities of radiosotopes, is tabulated in Table 6 for decay periods up to one million years, and as normalized to GW(e)-yr generated, calculated with the computer program ORIGEN (Bell, 1973) for typical LWR operating conditions.

TABLE 6. Major Radionuclides in Spent Fuel

Isotope	Ci/GW(e)-yr After Discharge from Reactor						
	5E-1 yr	1E+1 yr	1E+2 yr	1E+3 yr	1E+4 yr	1E+5 yr	1E+6 yr
⁷⁹ Se	1.3E+1	1.3E+1	1.3E+1	1.3E+1	1.2E+1	4.6	3.1E-4
⁸⁵ Kr	3.6E+5	1.9E+5	6.1E+2	--	--	--	--
⁸⁷ Rb	6.5E-4	6.5E-4	6.5E-4	6.5E-4	6.5E-4	6.5E-4	6.5E-4
⁸⁹ Sr	2.1E+6	--	--	--	--	--	--
⁹⁰ Sr	2.6E+6	2.0E+6	2.2E+5	5.0E-5	--	--	--
⁹⁰ Y	2.6E+6	2.0E+6	2.2E+5	5.0E-5	--	--	--
⁹¹ Y	3.6E+6	--	--	--	--	--	--
⁹³ Zr	6.5E+1	6.5E+1	6.5E+1	6.5E+1	6.5E+1	6.1E+1	4.2E+1
⁹⁵ Zr	6.5E+6	--	--	--	--	--	--
^{93m} Nb	6.5	2.9E+1	6.5E+1	6.5E+1	6.5E+1	6.1E+1	4.2E+1
^{95m} Nb	1.4E+5	--	--	--	--	--	--
⁹⁵ Nb	1.3E+7	--	--	--	--	--	--
⁹⁹ Tc	5.0E+2	5.0E+2	5.0E+2	5.0E+2	4.6E+2	3.5E+2	1.9E+1
¹⁰³ Ru	1.6E+6	--	--	--	--	--	--
¹⁰⁶ Ru	1.3E+7	1.3E+4	--	--	--	--	--
^{103m} Rh	1.6E+6	--	--	--	--	--	--
¹⁰⁶ Rh	1.3E+7	1.3E+4	--	--	--	--	--
¹⁰⁷ Pd	3.8	3.8	3.8	3.8	3.8	3.7	3.4
¹²⁶ Sn	1.8E+1	1.8E+1	1.8E+1	1.8E+1	1.7E+1	9.2	1.8E-2
¹²⁵ Sb	2.6E+5	2.0E+4	1.9E-6	--	--	--	--

TABLE 6. Major Radionuclides in Spent Fuel (Cont'd)

Isotope	Ci/GW(e)-yr After Discharge from Reactor						
	5E-1 yr	1E+1 yr	1E+2 yr	1E+3 yr	1E+4 yr	1E+5 yr	1E+6 yr
^{126m} Sb	1.8E+1	1.8E+1	1.8E+1	1.8E+1	1.7E+1	9.2	1.8E-2
¹²⁶ Sb	1.8E+1	1.8E+1	1.8E+1	1.8E+1	1.7E+1	9.2	1.8E-2
^{125m} Te	1.1E+5	8.4E+3	7.6E-7	--	--	--	--
^{127m} Te	1.6E+5	1.3E-5	--	--	--	--	--
¹²⁷ Te	1.6E+5	1.3E-5	--	--	--	--	--
¹²⁹ I	1.3	1.3	1.3	1.3	1.3	1.3	1.3
¹³⁴ Cs	6.5E+6	2.2E+5	--	--	--	--	--
¹³⁵ Cs	1.0E+1	1.0E+1	1.0E+1	1.0E+1	1.0E+1	9.9	8.0
¹³⁷ Cs	3.6E+6	2.9E+6	3.6E+5	3.4E-4	--	--	--
^{137m} Ba	3.4E+6	2.7E+6	3.4E+5	3.1E-4	--	--	--
¹⁴¹ Ce	9.2E+5	--	--	--	--	--	--
¹⁴⁴ Ce	2.3E+7	3.1E+3	--	--	--	--	--
¹⁴⁴ Pr	2.3E+7	3.1E+3	--	--	--	--	--
¹⁴⁷ Pm	3.4E+7	2.4E+5	1.1E-5	--	--	--	--
^{148m} Pm	6.5E+4	--	--	--	--	--	--
¹⁵¹ Sm	4.6E+4	4.2E+4	2.0E+4	1.5E+1	--	--	--
¹⁵⁴ Eu	2.2E+5	1.4E+5	2.9E+3	--	--	--	--
¹⁵⁵ Eu	2.2E+5	4.6E+3	--	--	--	--	--
^{166m} Ho	2.1E-2	2.1E-2	1.9E-2	1.1E-2	6.5E-5	--	--
²²⁷ Th	--	--	9.9E-4	1.3E-2	1.3E-1	8.0E-1	9.9E-1
²²⁸ Th	4.2E-2	5.0E-1	2.4E-1	4.2E-5	5.0E-6	6.5E-5	6.5E-4
²³⁰ Th	--	--	8.4E-3	2.3E-1	2.5	1.6	1.4E+1
²³¹ Th	6.1E-1	6.1E-1	6.1E-1	6.5E-1	7.3E-1	9.9E-1	1.0
²³⁴ Th	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1
²³¹ Pa	6.9E-5	2.0E-4	1.4E-3	1.3E-2	1.3E-1	8.4E-1	1.0
²³³ Pa	1.2E+1	1.2E+1	1.6E+1	3.6E+1	4.2E+1	4.2E+1	3.1E+1

TABLE 6. Major Radionuclides in Spent Fuel (Cont'd)

Isotope	Ci/GW(e)-yr After Discharge from Reactor						
	5E-1 yr	1E+1 yr	1E+2 yr	1E+3 yr	1E+4 yr	1E+5 yr	1E+6 yr
^{234m} Pa	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1
²³⁴ Pa	1.2E-2	1.2E-2	1.2E-2	1.2E-2	1.2E-2	1.2E-2	1.2E-2
²³² U	1.6E-1	5.3E-1	2.4E-1	4.2E-5	--	--	--
²³³ U	8.8E-4	1.5E-3	6.9E-3	1.1E-1	1.7	1.5E+1	3.3E+1
²³⁴ U	5.7E-1	2.8	1.7E+1	3.1E+1	3.0E+1	2.6E+1	1.3E+1
²³⁵ U	6.1E-1	6.1E-1	6.1E-1	6.5E-1	7.3E-1	9.9E-1	1.0
²³⁶ U	8.4	8.4	8.4	9.2	1.2E+1	1.3E+1	1.3E+1
²³⁷ U	1.1E+2	6.5E+1	9.9E-1	1.6E-4	7.3E-5	3.8E-8	--
²³⁸ U	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1
²³⁷ Np	1.2E+1	1.2E+1	1.6E+1	3.6E+1	4.2E+1	4.2E+1	3.1E+1
²³⁹ Np	5.3E+2	5.3E+2	5.3E+2	5.0E+2	2.2E+2	6.1E-2	4.6E-6
²³⁸ Pu	8.0E+4	7.6E+4	3.8E+4	4.2E+1	--	--	--
²³⁹ Pu	1.1E+4	1.1E+4	1.1E+4	1.1E+4	8.4E+3	6.5E+2	4.6E-6
²⁴⁰ Pu	1.7E+4	1.7E+4	1.7E+4	1.6E+4	6.1E+3	6.1E-1	5.0E-8
²⁴¹ Pu	4.2E+6	2.6E+6	3.8E+4	6.5	3.0	1.6E-3	--
²⁴² Pu	6.1E+1	6.1E+1	6.1E+1	6.1E+1	6.1E+1	5.3E+1	9.9
²⁴¹ Am	7.6E+3	6.1E+4	1.3E+5	3.2E+4	3.0	1.6E-3	--
^{242m} Am	4.2E+2	3.8E+2	2.6E+2	4.2	--	--	--
²⁴² Am	4.2E+2	3.8E+2	2.6E+2	4.2	--	--	--
²⁴³ Am	5.3E+2	5.3E+2	5.3E+2	5.0E+2	2.2E+2	6.1E-2	4.6E-6
²⁴² Cm	6.5E+5	3.2E+2	2.1E+2	3.6	--	--	--
²⁴³ Cm	1.5E+2	1.2E+2	1.8E+1	6.1E-8	--	--	--
²⁴⁴ Cm	5.0E+4	3.4E+4	1.1E+3	--	--	--	--
²⁴⁵ Cm	6.9	6.9	6.9	6.5	3.0	1.6E-3	--
²⁴⁶ Cm	1.3	1.3	1.3	1.1	3.0E-1	5.3E-7	--
TOTAL	1.3E+8	1.3E+7	1.4E+6	6.2E+4	1.6E+4	1.7E+3	6.8E+2

Integral to the use of safety indices are the RCG values for the radionuclides to be considered. These values are listed in Appendix B (USNRC, 1976). The decay constants of each of the radionuclides found in Table 5 is listed in Appendix C. To calculate several of the safety indices, the specific volume of the spent fuel is required. A typical canister containing spent fuel contains 0.50 MTHM. The canister has an estimated volume of 0.46 m^3 . Also, the quantity of spent fuel discharged is 38.2 MTHM/GW(e)-yr. Thus, the specific volume of packaged spent fuel is $35.1 \text{ m}^3/\text{GW(e)-yr}$. This information coupled with the failure scenario data listed in Section 2.3 is sufficient to apply the assessment methodologies.

2.2 SOLIDIFIED HIGH-LEVEL WASTE DESCRIPTION

The coprocessed $\text{UO}_2 - \text{PuO}_2$ has been proposed as an alternative nuclear fuel cycle offering built-in proliferation resistance. One assessment of the solvent extraction process in coprocessing yields the separation efficiencies, shown in Table 7 (Pobereskin et.al., 1977). Based then on these process efficiencies, and on radioisotopic tabulations generated with ORIGEN (Bell, 1973), the radionuclide content of solidified high-level waste is shown in Table 8, as normalized to GW(e)-yr. The final bit of information required is the specific volume of solidified high-level waste, which is $3 \text{ m}^3/\text{GW(e)-yr}$ (USERDA, 1976).

TABLE 7. Performance Characteristics of Coprocessing

Element	Distribution of Constituents (%)		
	Feed	Product	HLW(a)
U	100	99.99	<0.01
Pu	100	99.9	<0.1
Np	100	95	5
Am	100	<0.1	99.9
Cm	100	<0.1	99.9
Ru	100	1.2	98.7
Zr	100	1.6	98.5
Nd	100	0.1	99.9
Ce	100	0.1	99.8
All Others ^(b)	100	1	99

(a) HLW - High-level waste.

(b) Separation efficiency based on estimate of process performance, except I, of which only 0.5% goes to waste.
(Pobereskin, et. al., 1977.)

TABLE 8. Major Radionuclides in Solidified High-Level Wastes

Isotope	Ci/GW(e)-yr After Discharge from Reactor						
	5E-1 yr	1E+1 yr	1E+2 yr	1E+3 yr	1E+4 yr	1E+5 yr	1E+6 yr
⁷⁹ Se	1.3E+1	1.3E+1	1.3E+1	1.3E+1	1.2E+1	4.5	3.0E-4
⁸⁷ Rb	6.1E-4	6.1E-4	6.1E-4	6.1E-4	6.1E-4	6.1E-4	6.1E-4
⁸⁹ Sr	2.0E+6	--	--	--	--	--	--
⁹⁰ Sr	2.3E+6	1.9E+6	2.0E+5	4.5E-5	--	--	--
⁹⁰ Y	2.3E+6	1.9E+6	2.0E+5	4.5E-5	--	--	--
⁹¹ Y	3.5E+6	--	--	--	--	--	--
⁹³ Zr	6.0E+1	6.0E+1	6.0E+1	6.0E+1	6.0E+1	5.6E+1	3.8E+1
⁹⁵ Zr	6.4E+6	--	--	--	--	--	--
^{93m} Nb	6.4	2.8E+1	6.0E+1	6.0E+1	6.0E+1	5.6E+1	3.8E+1
^{95m} Nb	1.4E+5	--	--	--	--	--	--
⁹⁵ Nb	1.2E+7	--	--	--	--	--	--
⁹⁹ Tc	4.9E+2	4.9E+2	4.9E+2	4.9E+2	4.5E+2	3.5E+2	1.9E+1
¹⁰³ Ru	1.7E+6	--	--	--	--	--	--
¹⁰⁶ Ru	1.4E+7	1.4E+4	--	--	--	--	--
^{103m} Rh	1.7E+6	--	--	--	--	--	--
¹⁰⁶ Rh	1.4E+7	1.4E+4	--	--	--	--	--
¹⁰⁷ Pd	4.5	4.5	4.5	4.5	4.5	4.5	4.2
¹²⁶ Sn	2.1E+1	2.1E+1	2.1E+1	2.0E+1	1.9E+1	1.0E+1	2.0E-2
¹²⁵ Sb	2.9E+5	2.2E+4	2.0E-6	--	--	--	--
^{126m} Sb	2.1E+1	2.1E+1	2.1E+1	2.0E+1	1.9E+1	1.0E+1	2.0E-2
¹²⁶ Sb	2.0E+1	2.0E+1	2.0E+1	2.0E+1	1.9E+1	1.0E+1	2.0E-2
^{125m} Te	1.2E+5	9.1E+3	8.7E-7	--	--	--	--
^{127m} Te	1.7E+5	1.4E-5	--	--	--	--	--
¹²⁷ Te	1.7E+5	1.4E-5	--	--	--	--	--
¹²⁹ I	6.7E-3	6.7E-3	6.7E-3	6.7E-3	6.7E-3	6.7E-3	6.5E-3
¹³⁴ Cs	6.4E+6	2.2E+5	--	--	--	--	--
¹³⁵ C2	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	9.5
¹³⁷ Cs	3.6E+6	2.8E+6	3.6E+5	3.3E-4	--	--	--
^{137m} Ba	3.4E+6	2.7E+6	3.3E+5	3.1E-4	--	--	--
¹⁴¹ Ce	9.1E+5	--	--	--	--	--	--
¹⁴⁴ Ce	2.2E+7	3.0E+3	--	--	--	--	--
¹⁴⁴ Pr	2.2E+7	3.0E+3	--	--	--	--	--
¹⁴⁷ Pm	3.4E+6	2.4E+5	1.1E-5	--	--	--	--
^{148m} Pm	6.8E+4	--	--	--	--	--	--
¹⁵¹ Sm	4.5E+4	4.2E+4	2.1E+4	1.6E+1	--	--	--
¹⁵⁴ Eu	2.4E+5	1.6E+5	3.2E+3	--	--	--	--

TABLE 8. Major Radionuclides in Solidified High-Level Wastes (Cont'd)

Isotope	Ci/GW(e)-yr After Discharge from Reactor						
	5E-1 yr	1E+1 yr	1E+2 yr	1E+3 yr	1E+4 yr	1E+5 yr	1E+6 yr
¹⁵⁵ Eu	2.3E+5	4.9E+3	--	--	--	--	--
^{166m} Ho	2.5E-2	2.5E-2	2.4E-2	1.4E-2	7.9E-5	--	--
²²⁷ Th	2.5E-5	1.4E-4	1.2E-3	1.3E-2	1.2E-1	8.7E-1	1.1
²²⁸ Th	1.2E-1	7.6E-1	3.8E-1	6.4E-5	6.4E-6	8.3E-5	8.7E-4
²³⁰ Th	3.7E-4	1.7E-3	3.1E-2	6.8E-1	7.2	4.5E+1	1.9E+1
²³¹ Th	6.1E-1	6.1E-1	6.1E-1	6.1E-1	7.2E-1	1.1	1.1
²³⁴ Th	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1
²³¹ Pa	3.8E-4	5.3E-4	1.7E-3	1.3E-2	1.3E-1	8.7E-1	1.1
²³³ Pa	1.5E+1	1.6E+1	2.1E+1	5.7E+1	6.8E+1	6.4E+1	4.9E+1
^{234m} Pa	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1	1.2E+1
²³⁴ Pa	1.2E-2	1.2E-2	1.2E-2	1.2E-2	1.2E-2	1.2E-2	1.2E-2
²³² U	3.2E-5	8.4E-5	3.7E-5	6.5E-9	--	--	--
²³³ U	1.4E-7	2.1E-7	9.2E-7	1.8E-5	2.7E-4	2.3E-3	5.3E-3
²³⁴ U	1.2E-3	1.8E-3	5.3E-3	9.2E-3	8.8E-3	7.3E-3	1.7E-3
²³⁵ U	6.1E-5	6.1E-5	6.1E-5	6.1E-5	7.3E-5	1.1E-4	1.1E-4
²³⁶ U	9.9E-4	9.9E-4	1.0E-3	1.1E-5	1.5E-3	1.8E-3	1.8E-3
²³⁷ U	1.8E-2	1.1E-2	1.6E-4	1.6E-7	7.6E-8	--	--
²³⁸ U	1.2E-3	1.2E-3	1.2E-3	1.2E-3	1.2E-3	1.2E-3	1.2E-3
²³⁷ Np	7.6E-1	7.8E-1	1.1	2.9	3.4	3.2	2.5
²³⁹ Np	9.0E+1	9.0E+1	9.0E+1	8.2E+1	3.6E+1	1.1E-4	3.2E-6
²³⁸ Pu	2.1E+2	2.0E+2	9.9E+1	1.3E-1	--	--	--
²³⁹ Pu	1.4E+1	1.4E+1	1.4E+1	1.3E+1	1.1E+1	8.4E-1	6.5E-8
²⁴⁰ Pu	2.8E+1	2.8E+1	2.8E+1	2.6E+1	1.0E+1	1.0E-3	--
²⁴¹ Pu	6.9E+3	4.6E+3	6.5E+1	6.5E-2	3.0E-2	1.6E-5	--
²⁴² Pu	1.5E-1	1.5E-1	1.5E-1	1.5E-1	1.5E-1	1.3E-1	2.4E-2
²⁴¹ Am	1.6E+4	1.8E+4	2.0E+4	4.7E+3	2.6E-3	--	--
^{242m} Am	1.9E+3	1.8E+3	1.2E+3	1.9E+1	--	--	--
²⁴² Am	1.9E+3	1.8E+3	1.2E+3	1.9E+1	--	--	--
²⁴³ Am	1.8E+3	1.8E+3	1.8E+3	1.6E+3	7.3E+2	2.1E-1	6.5E-5
²⁴² Cm	1.8E+6	1.5E+3	9.6E+2	1.6E+1	--	--	--
²⁴³ Cm	3.8E+2	3.1E+2	4.2E+1	1.5E-7	--	--	--
²⁴⁴ Cm	2.8E+5	1.9E+5	6.1E+3	--	--	--	--
²⁴⁵ Cm	6.9E+1	6.9E+1	6.9E+1	6.5E+1	3.0E+1	1.6E-2	--
²⁴⁶ Cm	1.3E+1	1.3E+1	1.3E+1	1.1E+1	3.0	5.3E-6	--
Total	1.2E+8	1.0E+7	1.1E+6	7.3E+3	1.6E+3	6.5E+2	2.1E+2

2.3 ANALYSIS SCENARIO

To calculate some of the safety indices, a failure scenario must be defined, along with the environment in which the failure occurs. For this analysis, it is assumed that the packaged spent fuel from 1 GW(e)-yr energy generation in the once-through fuel cycle and the solidified high-level waste from 1 GW(e)-yr energy generation in the coprocessed UO_2 - PuO_2 fuel cycle are in storage in a surface facility, as shown in Figure 13 for solidified high-level waste (Smith and Kastenber, 1976) and Figure 14 for packaged spent fuel (Rockwell, 1977).

The hypothesized failure is a blockage of the storage units air passages with dust, airborne vegetation, small animals and associated debris so as to restrict the natural convective cooling (ARHCO, 1977). This incident is assumed to occur at a waste age of 0.5 year. This hypothesized loss of flow accident is assumed to cause an overpressure of the waste container, and a subsequent airborne release of 1×10^{-9} of the contained radionuclides. This event is assumed to have a probability of occurrence of $10^{-7}/\text{yr.}^{(a)}$

In the once-through fuel cycle, each packaged spent fuel canister has a volume of 0.5 m^3 . Recalling that the specific volume of packaged spent fuel is $35.1 \text{ m}^3 \text{ GW(e)-yr}$, then 70 surface storage facilities are required to store the packaged spent fuel from 1 GW(e)-yr of energy generation. Similarly, in the coprocessed UO_2 - PuO_2 fuel cycle, solidified high-level waste containers each contains 0.22 m^3 , and the specific volume of this waste form is $3 \text{ m}^3/\text{GW(e)-yr}$. Thus, from 1 GW(e)-yr of energy generation, 14 surface storage facilities are required.

(a) When a safety index is calculated for this hypothesized failure scenario, the release from that event is followed from the time of release, 0.5 years out-of-reactor, to one million years.

FIGURE 13. Surface Storage Facility for Solidified High-Level Waste (Smith and Kastenber, 1976)

FIGURE 14. Surface Storage Facility for Packaged Spent Fuel (Rockwell Hanford Co., 1977)

2.4 SUMMARY OF INPUT DATA

The input data to be used in the safety indices' calculation are summarized in Table 9. The nature of the scenario, and the limitations on this analysis suggests the omission of three indices, HMI, HA, and T_i . The first two, HMI and HA, require an environmental transport calculation to be performed. The isolation time, T_i , was designed for utilization in a deep geologic isolation or shallow land burial calculation. The calculation of cancer events, D, for fuel cycle evaluation will not be performed due to the extremely controversial nature and the complexity of the calculation.

The omission of these four indices leaves two bits of data yet to be introduced. The societal hazard potential perception, or the time period of concern for the release of radionuclides from spent fuel or high-level waste to the environment is assumed to be 100,000 years (Willrich, 1977), needed to calculate HM3. One possible comparative basis needed to calculate RTI is the radionuclides emitted from a coal-fired electric generating plant, as tabulated in Table 10 (McBride et.al., 1977).

TABLE 9. Summary of Input Data to Safety Indices

Information	Input Data
Radionuclide Content	
Once-Through Fuel Cycle	Table 6
Coprocessed UO_2PuO_2 Fuel Cycle	Table 8
Regulatory Definition (RCG)	Appendix B
Standard Consumption	0.8 m^3 water/yr and $7,300 \text{ m}^3$ air/yr
Release Scenario	Surface storage facility containing packaged spent fuel and solidified high-level waste from 1 GW(e)-yr generation has natural correction blockage causing destructive overpressure of waste packages and airborne release of 1×10^{-9} of inventory.
Pathway Analysis	Not specified
Societal Hazard Potential Perception	100,000 years
Probabilistic Release Calculation	Estimated 10^{-7} /yr
Decay Constants	Appendix C
Specific Volume	
Once-Through Fuel Cycle	$35.1 \text{ m}^3/\text{GW(e)-yr}$
Coprocessed UO_2PuO_2 Fuel Cycle	$3 \text{ m}^3/\text{GW(e)-yr}$
Leach Rate	Not specified
Environmental Scenario	Not specified
Comparative Basis	Radionuclide emission and impact of coal-fired power generation, Table 10
Dose Response Calculation	Not specified

TABLE 10. Radioactive Effluents from Coal-Fired Power Plants^(a)

Isotope	(Ci/GW(e)-yr)
<u>²³⁸U Decay Chain</u>	
²³⁸ U	1.1E-2
²³⁴ Th	1.1E-2
^{234m} Pa	1.1E-2
²³⁴ U	1.1E-2
²³⁰ Th	1.1E-2
²²⁶ Ra	1.1E-2
²¹⁸ Po	1.1E-2
²¹⁴ Pb	1.1E-2
²¹⁴ Bi	1.1E-2
²¹⁴ Po	1.1E-2
²¹⁰ Pb	1.1E-2
²¹⁰ Bi	1.1E-2
²¹⁰ Po	1.1E-2
<u>²³⁵U Decay Chain</u>	
²³⁵ U	5.0E-4
²³¹ Th	5.0E-4
²³¹ Pa	5.0E-4
²²⁷ Ac	5.0E-4
²²⁷ Th	5.0E-4
²²³ Ra	5.0E-4
²¹⁹ Rn	5.0E-4
²¹¹ Pb	5.0E-4
²¹¹ Bi	5.0E-4
²⁰⁷ Tl	5.0E-4
<u>²³²Th Decay Chain</u>	
²³² Th	7.1E-3
²²⁸ Ra	7.1E-3
²²⁸ Ac	7.1E-3
²²⁸ Th	7.1E-3
²²⁴ Ra	7.1E-3
²¹² Pb	7.1E-3
²¹² Bi	7.1E-3
²⁰⁸ Tl	2.6E-3
<u>Radon Releases</u>	
²²⁰ Rn	5.7E-1
²²² Rn	1.1E-0

(a) A capacity factor of 0.7 is assumed for the coal-fired power plant. (McBride, et. al., 1977)

3.0 APPLICATIONS OF SAFETY INDICES

The safety indices have been calculated for the two waste streams specified in Section 2. In Section 3.1, the analysis results are presented for the once-through fuel cycle, and in Section 3.2, the coprocessed UO_2 - PuO_2 fuel cycle. The results are discussed independently to recognize the differences between the various indices. In Section 3.3, the two fuel cycles are compared on the basis of the analysis results to further examine the methodologies, and to identify any differences between the two fuel cycles.

3.1 SAFETY INDICES OF THE ONCE-THROUGH FUEL CYCLE

The results on the once-through fuel cycle assessment using the safety indices are summarized in Table 11. The quantity of radionuclides, $Q[\text{Ci/GW(e)-yr}]$, is taken from Table 6. The hazard measure, $\text{HM}[\text{m}^3 \text{ water/GW(e)-yr}]$, is calculated by dividing the quantity of each radionuclide by the RCG value listed in Appendix B. The modified hazard index, HM2, is calculated for the postulated release of radionuclides from the loss of coolant accident in the surface storage of packaged spent fuel, 1×10^{-9} of inventory of radionuclides resulting from the generation of 1 GW(e)-yr. This inventory is divided by the maximum permissible intake, MPI, or the product of the RCG values in air found in Appendix B and the standard man's annual intake, $7,300 \text{ m}^3$. The second modified hazard measure, HM3, is calculated by integrating the quantity of radionuclides, released in the postulated loss of coolant scenario divided by the maximum permissible intake, MPI. The lower limit of integration, t , (as shown in Equation 10) was assigned the value of each time shown in Table 11, while the upper limit, $t + d$, was found by adding 10^5 years, the time period of concern, to the lower limit. This assignment may be interpreted as examining the release of 1×10^{-9} of the inventory from 0.5 yr to 10^6 yr following reactor discharge. The potential hazard measure, PHM, was calculated as the product of the probability of the loss of flow accident, $1 \times 10^{-7}/\text{yr}$, and the resulting release of radionuclides divided by the product of the maximum permissible intake and the decay constant of each radionuclide, MPI and λ . The hazard index, HI, was calculated by dividing the hazard measure of spent fuel in water by the specific

volume of spent fuel or $35.1 \text{ m}^3/\text{GW(e)-yr}$. Finally, the relative toxicity index was calculated as the quotient of the hazard measure in air of the improbable release of radionuclides from the loss of coolant accident in the storage of packaged spent fuel and the chronic release of radionuclides from power generation in burning coal.

TABLE 11. Safety Indices of Once-Through Fuel Cycle^(a)

Index	Time After Discharge from Reactor(yr)						
	5E-1 yr	1E+1 yr	1E+2 yr	1E+3 yr	1E+4 yr	1E+5 yr	1E+6 yr
Q(Ci)	1.3E+8	1.3E+7	1.4E+6	6.2E+4	1.6E+4	1.7E+3	6.8E+2
HM(m^3)	2.8E+13	7.9E+12	9.3E+11	1.4E+10	3.0E+9	2.0E+8	5.0E+7
HM2 ^(b)	5.2E+5	4.0E+5	2.3E+5	8.4E+4	3.3E+4	1.7E+3	9.5E+1
HM3(yr)	1.2E+9	1.3E+9	1.3E+9	1.2E+9	7.9E+8	6.8E+7	8.2E+6
PHM ^(b)	4.9E+2	4.9E+2	4.9E+2	4.9E+2	4.5E+2	3.8E+2	3.6E+2
HI ^(b)	8.0E+11	2.3E+11	2.6E+10	4.0E+8	8.5E+7	5.7E+6	1.4E+6
RTI ^(b)	1.9E-2	1.5E-2	8.9E-3	3.1E-3	1.2E-3	6.2E-5	3.4E-6

(a) All indices normalized to the unit energy generated, GW(e)-yr.

(b) Dimensionless numbers.

Each of the indices has been normalized and plotted on Figure 15, so that they may be compared. Examination of Figure 15 shows several interesting characteristics of the various indices. The indices Q, HM, and HI all behave similarly in time, i.e., both curves have the same shape and decrease six orders-of-magnitude during the decay period. The other indices, PHM, HM3, HM2 and RTI, are seen to be less sensitive to the decay of radionuclides. This difference in the relative behavior is attributed to two factors. The PHM, HM3, HM2 and RTI are all reflective of the RCG values in air. As seen in Appendix B, the RCG_{air} values for the actinides, especially ^{230}Th and ^{231}Pa in the ^{238}U and ^{235}U decay series, respectively, are much smaller than for β - or γ -emitters, thus causing the presence of α -emitters to dominate the calculations. The second reason for the time behavior discrepancy, in the cases of PHM and HM3, is that the indices are calculated as quotients of the decay constants.

Again, the presence of the long-lived actinides forces the behavior of the indices to be especially sensitive to the decay of only a selected few species over the time frame considered.

FIGURE 15. Safety Indices of the Once-Through Fuel Cycle

Based on the presentation of the various safety indices in Figure 15, it is concluded that the indices PHM and HM3 are not suitable for application to the safety assessment of operational waste management systems, as the calculations are likely to be dominated by a few particularly long-lived species which do not reflect the immediate chronic or accidental hazard associated with the system, and also because relatively little information is gained for the short term safety assessment. These two particular indices are probably much more suitable for safety assessments of radioactive waste disposal operations which typically consider periods over geologic time. Further, it is concluded that the other indices are adequate for operational waste management systems safety assessment. The indices HM and HI are perceived as being slightly more suitable than Q, as the radiotoxicity of nuclides being considered is reflected in their computation.

3.2 SAFETY INDICES FOR THE COPROCESSED UO_2 - PuO_2 FUEL CYCLE

The results of the coprocessed UO_2 - PuO_2 fuel cycle assessment using the safety indices are summarized in Table 12. The methodologies for individual calculations are adequately discussed in Section 3.1.

Each of the indices presented in Table 12 has been normalized and sketched in Figure 16. As discussed earlier, Q, HM, and HI are much more sensitive to the total radioactivity of the waste stream than are HM2, RTI, HM3 and PHM. Examination of Figure 16 reveals no additional conclusions regarding the potential applicability of the safety indices than those discussed in Section 3.1.

TABLE 12. Safety Indices of Coprocessed UO_2 - PuO_2 Fuel Cycle^(a)

Index	Time After Discharge from Reactor(yr)						
	5E-1 yr	1E+1 yr	1E+2 yr	1E+3 yr	1E+4 yr	1E+5 yr	1E+6 yr
Q(Ci)	1.2E+8	1.0E+7	1.1E+6	7.3E+3	1.6E+3	6.5E+2	2.1E+2
HM(m ³)	2.7E+13	7.5E+12	8.1E+11	1.6E+9	2.3E+8	4.2E+7	1.9E+7
HM2 ^(b)	2.7E+5	1.1E+5	2.0E+4	4.5E+3	5.9E+2	8.7E+1	4.0E+1
HM3(yr)	2.6E+7	2.6E+7	2.4E+7	1.6E+7	7.9E+6	5.7E+6	2.7E+6
PHM ^(b)	3.0	3.0	3.0	2.9	2.3	2.3	1.5
HI ^(b)	9.0E+12	2.5E+12	2.7E+11	5.3E+8	7.7E+7	1.4E+7	6.3E+6
RTI ^(b)	1.0E-2	4.1E-3	7.4E-4	1.7E-4	2.2E-5	3.2E-6	1.4E-6

(a) All indices normalized to the unit energy generated, GW(e)-yr.

(b) Dimensionless numbers.

FIGURE 16. Safety Indices of the Coprocessed $UO_2 - PuO_2$ Fuel Cycle

3.3 FUEL CYCLE COMPARISONS

The safety indices have been calculated for the packaged spent fuel and solidified high-level wastes of the once-through and coprocessed $\text{UO}_2 - \text{PuO}_2$ fuel cycles, respectively. The indices have to be viewed separately for each fuel cycle. In this section, the fuel cycles are compared on the basis of the indices to further study their potential applicability to operational waste management systems assessment and to identify differences between the fuel cycles.

In Figure 17, plotted are the number of curies found in packaged spent fuel in the once-through fuel cycle and solidified high-level waste in the coprocessed $\text{UO}_2 - \text{PuO}_2$ from the generation of 1 GW(e)-yr of energy. As shown, solidified high-level waste has a slightly lower total radionuclide inventory than does packaged spent fuel. The first 500-600 years of decay represents the short-lived fission products, especially ^{90}Sr , ^{137}Cs and their daughters. The slightly lower activity level during the first 1,000 years of decay for solidified high-level wastes is representative of the coprocessing steps, i.e., some fission products, due to separations efficiencies, either remain in $\text{UO}_2 - \text{PuO}_2$ products as contaminants or are found in other coprocessing plant wastes. The difference found in the time period from 10^3 years to 10^6 years is attributed primarily to the removal of plutonium from spent fuel in coprocessing causing the high-level waste activity to be less than spent fuel. The plutonium isotopes of fairly long half-lives, ^{239}Pu , ^{240}Pu and ^{242}Pu represent a substantial fraction of the radionuclides in spent fuel for the decay intervals considered. Thus, the numerical influence of the removal is obvious.

The hazard measure, HM, for the two wastes is plotted in Figure 18. As with Q, the HM for solidified high-level waste from the coprocessed $\text{UO}_2 - \text{PuO}_2$ fuel cycle is lower than for the packaged spent fuel from the once-through fuel cycle. The hazard measure for decay times less than 600 years represents the short-lived fission products, mostly ^{137}Cs and ^{90}Sr . The hazard measure for decay periods greater than 600 years represents the actinides in the wastes. The largest variance between the two fuel cycles occurs at 10^4 years of decay, a difference of slightly more than a factor of 10. This can be attributed

directly to the removal of ^{239}Pu and ^{240}Pu from the waste in the coprocessed $\text{UO}_2 - \text{PuO}_2$ fuel cycle, as these isotopes with half-lives of 24,400 and 6,540 years, respectively, dominate the HM calculation for the packaged spent fuel.

FIGURE 17. Radioactivity in Packaged Spent Fuel and Solidified High-Level Waste

FIGURE 18. Hazard Measure of Packaged Spent Fuel and Solidified High-Level Waste

The modified hazard measure, HM2, for the packaged spent fuel and solidified high-level waste is plotted in Figure 19. As with Q and HM, the HM2 of solidified high-level waste of the coprocessed UO₂ - PuO₂ fuel cycle is lower than the HM2 of spent fuel from the once-through fuel cycle when considered on a unit energy generation basis. The magnitude of the difference, however, is much larger for HM2 than Q or HM. This larger discrepancy, which ranges from a factor of 1.9 to 560, is directly attributable to the removal of uranium

and plutonium from solidified high-level waste in the coprocessed $\text{UO}_2 - \text{PuO}_2$ fuel cycle. As noted previously, the RCG values in air for alpha-emitters, specifically the plutonium isotopes (except ^{241}Pu), are much lower than for the fission products, sometimes 10^6 lower for α -emitters than for fission products. This tends to magnify the impact of removal of uranium and plutonium from the solidified high-level wastes. At decay periods approaching 10^6 years, the large variance in the HM2 plots of the two waste streams begins to disappear. This occurs as the transuranics in the package spent fuel and solidified high-level waste decay and the long-lived actinides in the ^{235}U and ^{238}U decay series, specifically ^{231}Pa and ^{230}Th , approach equilibrium and dominate the HM2 calculation.

FIGURE 19. Modified Hazard Measure (HM2) of Packaged Spent Fuel and Solidified High-Level Waste

The second modified hazard measure, HM3, is plotted in Figure 20 for the packaged spent fuel and solidified high-level wastes. As observed in the previous indices, the HM3 of packaged spent fuel is larger than the solidified high-level wastes. The difference in the two waste streams, for decay periods less than 10^5 years, is about a factor of 100. This difference approximately reflects the separation efficiency of plutonium removal in the coprocessing of spent fuel, which is 99.9%. Noteworthy is that the decay of fission products is not reflected in the HM3 index. This occurs as the HM3 is proportional to the quotient of the total radioactivity and the decay constants of the radionuclides in the wastes. The presence of the actinides, some of which are typified by extremely small decay constants (see Appendix C), tend to dominate the calculation of HM3, whereas the most abundant fission products are typified by relatively large specific activities, with half-lives usually less than 100 years. As with HM2, the HM3 curves tend to converge for decay times approaching 10^6 years, as caused by the equilibration of the long-lived uranium decay series' daughters ^{231}Pa and ^{230}Th , and by the presence of long-lived transuranic ^{237}Np .

FIGURE 20. Modified Hazard Measure (HM3) for Packaged Spent Fuel and Solidified High-Level Waste

The potential hazard measure, PHM, of spent fuel from the once-through fuel cycle and of solidified high-level waste from the coprocessed $\text{UO}_2 - \text{PuO}_2$ fuel cycle is plotted in Figure 21, on the basis of the energy generated, 1 GW(e)-yr. As observed previously, the PHM for solidified high-level waste is roughly a factor of 100 less than the packaged spent fuel, reflecting the separations efficiencies for the actinides from spent fuel in coprocessing. As with the modified hazard measure HM3, the decay of the short-lived fission products is not reflected in the PHM. The calculation of PHM is particularly sensitive to the presence of those relatively abundant, long-lived actinides, as the PHM is inversely proportional to the decay constants. Closer examination of the PHM calculation for packaged spent fuel shows that presence of ^{238}U , a relatively innocuous α -emitter with a very long half-life, forces the PHM to converge to its decay. At 10^6 years of decay, the PHM_i for ^{238}U is 340 while for the total packaged spent fuel waste stream, it is 360, thus demonstrating this numerical dominance.

FIGURE 21. Potential Hazard Measure for Packaged Spent Fuel and Solidified High-Level Waste

The hazard index in water, HI, for packaged spent fuel and for solidified high-level wastes is plotted in Figure 22. The pattern established with the previous indices, that is, the packaged spent fuel having a higher index value than solidified high-level waste, is broken in the consideration of HI. The meaning of HI permits consistency in the comparison of spent fuel and solidified high-level waste. Recall that the HI is proportional to the curie abundance per unit volume of the waste form. Thus a higher value of HI

FIGURE 22. Hazard Index for Packaged Spent Fuel and Solidified High-Level Waste

indicates that a more compact and localized waste form exists than for a lower value. From an operational waste management point of view, opposing conclusions may be derived on the basis of this information. A highly concentrated waste form is advantageous in that the total volume of waste, or number of containers, along with the total container-miles of required transportation, and total storage and disposal volumes are all lower than for a less concentrated waste form. The antithesis of these positive aspects is that each waste container, for a highly concentrated waste, has a higher radionuclide content, heat loading and radiation dose level than containers of a less concentrated waste form. This means that higher design constraints are placed on the operational waste management system, and that in the case of an unlikely accident while handling, transporting or storing the highly concentrated waste form, a higher absolute release may result, assuming differences in waste form properties are not considered. Further analysis of the HI plot of Figure 22 shows that the decay of the short-lived fission products is reflected. One interesting property of the HI may be seen in comparing Figures 15, 16 and 18. Figures 15 and 16 indicate that within the once-through and coprocessed $\text{UO}_2 - \text{PuO}_2$ fuel cycles, the normalized values of HM and HI for the packaged spent fuel and solidified high-level waste are superimposed though they are unique for each fuel cycle. Figure 18 shows that the HM of packaged spent fuel is consistently higher than for solidified high-level waste. The explanation for this seemingly anomalous behavior lies in the definition of HI. The HI curve for packaged spent fuel is the superposition of the quotient of the HM curve of Figure 18 and the specific volume of that waste form, $35.1 \text{ m}^3/\text{GW(e)-yr}$. Similarly, the HI curve for solidified high-level waste is the superposition of the quotient of its HM curve and specific volume, $3 \text{ m}^3/\text{GW(e)-yr}$. Thus, while the relative behavior of HM and HI are the same, the relative magnitude of the HI curve for packaged spent fuel has been lowered by more than a factor of 10 than the solidified high-level waste.

The plot of the relative toxicity index, RTI, for packaged spent fuel and solidified high-level waste is shown on Figure 23. Reviewing the calculation of RTI, it is the quotient of the hazard measure in air of the accidental airborne release of radionuclides resulting from the loss of coolant accident in the storage of packaged spent fuel and solidified high-level waste from 1 GW(e)-yr of energy generation, and the chronic release of actinides in the generation of 1 GW(e)-yr at a coal-fired power plant. It is noteworthy that the hazard measure of the accidental release scenario with a probability of 1×10^{-7} /yr is substantially less than the chronic release of radionuclides in coal-fired power generation by magnitudes ranging from 10^{-2} to 10^{-6} . As would be anticipated from earlier results, the RTI of solidified high-level waste is substantially lower than the packaged spent fuel, especially in the time frame of 10^2 to 10^5 years of decay. This results because of the removal of plutonium from the spent fuel in coprocessing, and because of the more restrictive RCG air values for α -emitters for fission products. For reasons discussed earlier, the RTI of packaged spent fuel and solidified high-level waste converge at decay periods approaching 10^6 years.

FIGURE 23. Relative Toxicity Index for Packaged Spent Fuel and Solidified High-Level Waste

4.0 CONCLUSIONS

On the basis of the safety indices' presentations in Section 3, the following conclusions are made regarding the application of safety indices:

- Safety indices are useful tools for operational waste management system safety assessment.
- For conceptual fuel cycles and their associated radioactive waste management systems with only generic design information available, HM1, HA, T_i and D are not suitable as safety assessment tools.
- RCG, by itself, is inappropriate as a safety assessment tool.
- HM3 and PHM are not suitable as safety assessment tools when long-lived α -emitters need to be considered within the system.
- Q, HM, HM2, HI and RTI are the most appropriate safety assessment tools whether discussing airborne or water-pathway releases.

The following conclusions were made regarding the comparison of packaged spent fuel from the once-through fuel cycle and solidified high-level waste from the coprocessed UO_2 - PuO_2 fuel cycle:

- The solidified high-level waste from the coprocessed UO_2 - PuO_2 fuel cycle may be potentially less hazardous than the packaged spent fuel from the once-through fuel cycle.
- The removal of actinides, and especially plutonium, from spent fuel may potentially reduce the hazard associated with the waste.
- After 10^6 years of decay, packaged spent fuel and solidified high-level waste are nearly the same on a hazard potential basis.

The conclusions are discussed in some depth in the following sections.

4.1 SAFETY INDICES APPLICATION

One purpose of this study has been to assess the suitability of the various safety indices as operational waste management system safety assessment tools. It is concluded that, in general, the safety indices are quite appropriate as

safety assessment tools. In the analysis presentations in Section 3, significant information about the conceptual waste management systems resulted. The more hazardous waste stream, characteristics of the potential hazard and some important operational information resulted.

Four of the safety indices were found to be inappropriate as safety assessment methodologies for conceptual waste management systems with only generic design information available. HM1, HA and T_1 require information that is extremely sensitive to the environmental, geologic and demographic conditions found at the site of the waste management operations. These three indices were originally conceived for application to shallow or deep geologic storage or isolation of radioactive wastes, and provided sufficient information is available, are most useful to the scenarios. The cancer mortality index, D, may be applied to the safety analysis of operational systems. However, by its very nature, it is a most controversial calculation. It is concluded that the controversy over its mere presentation would detract from its potential applicability. Also, the presentation of the cancer mortality potential of a system can be easily misinterpreted as an actual consequential result rather than a potential. It is concluded that the RCG values of the various radionuclides encountered in a fuel cycle is inappropriate as a waste management safety assessment methodology.

It is concluded that HM3 and PHM are not suitable as safety assessment tools when long-lived α -emitters are considered within waste management systems. This is based on the numerical dominance of HM3 and PHM by any relatively abundant, long-lived species which results as both are inversely proportional to the decay constants of the radionuclides being considered. Further, the RCG air values for α -emitters are much more restrictive than β - and γ -emitters. As long-lived actinides are typically α -emitters, their presence in a system is substantially magnified, causing, in some cases, the presence of relatively non-toxic isotopes (^{238}U or ^{235}U) to dominate the calculations.

Finally, it is concluded that Q, HM, HM2, and HI and RTI are the most appropriate of the safety index methodologies as operational waste management safety assessment tools. These indices all accurately reflect the presence

of all abundant species as well as their relative hazard potential. Significant information resulted from the application of these tools which aids in the absolute and comparative assessments of conceptual waste management systems.

4.2 FUEL CYCLE COMPARISONS

The packaged spent fuel from the once-through fuel cycle and the solidified high-level wastes from the coprocessed UO_2 - PuO_2 fuel cycle were compared on a unit energy-generated basis using the safety indices. The fuel cycle comparison is performed to further investigate the applicability of safety indices to operational waste management system safety assessment.

The most apparent analysis conclusion is that the solidified high-level waste from the coprocessed UO_2 - PuO_2 fuel cycle may be potentially less hazardous material than the packaged spent fuel from the once-through fuel cycle. This conclusion is brought on solely by the lower total radionuclide content in high-level waste than spent fuel, per unit energy. The conclusion could be easily disputed. For example, a number of chemical and mechanical operations precede the generation of solidified high-level waste from spent fuel. Each operation has a risk associated with it. It is possible that the total hazard associated solidified high-level wastes may be greater than for spent fuel if the entire system is considered. This, then, is an indication of the present formulation and application of the safety indices.

Another possible conclusion of the analysis is that the removal of actinides, and especially plutonium, from spent fuel may potentially reduce the associated hazard. This type of analysis has typically been used to support partitioning and transmutation of actinides from spent fuel and high-level waste (BNWL-1900, May 1974, Section 9). The conclusion here may be misleading though. Burkholder, et.al., (1976), investigated a hypothetical release of radionuclides from solidified high-level waste into groundwater. The subsequent dose calculation showed that plutonium becomes a dominant dose contributor only at times greater than one million years out-of-reactor. Further, the Reactor Safety Study (WASH-1400) presented dose calculations for a hypothetical core meltdown accident which indicated that only in isolated cases is plutonium a significant dose

contributer (here in the 50-year mineral bone dose). The safety indices cannot reflect this behavior and impact of radionuclides in the biosphere without an accompanying comprehensive analysis.

On the basis of the analysis, it is concluded that the solidified high-level waste and the packaged spent fuel have a converging hazard potential for decay periods approaching 10^6 years. This behavior results as most radioactive species decay to stable elements, leaving mostly the long-lived actinides ^{235}U and ^{238}U , and the species of their decay series.

REFERENCES

- Atlantic Richfield Hanford Company and Kaiser Engineers, Retrievable Surface Storage Facility Conceptual System Design Description. ARH-LD-140, REV, March 1977.
- Bell, M. J., ORIGEN - The ORNL Isotope Generation and Depletion Code. ORNL-4628, Oak Ridge National Laboratory, Oak Ridge, TN, May 1973.
- Bell, M. J. and Dillion, R. S., The Long-Term Hazard of Radioactive Wastes Produced by the Enriched Uranium, ^{238}U -Pu and ^{233}U -Th Fuel Cycles. ORNL-TM-3548, Oak Ridge National Laboratory, Oak Ridge, TN, November 1971.
- Bruns, L. E., "Hazards from Toxic Substances." from Proceedings of the Symposium on Waste Management, Tucson, AZ, October 1976.
- Burkholder, H. C., Cloninger, M. O., Baker, D. A. and Jansen, G., "Incentives for Partitioning High-Level Waste." Nuclear Technology. 31:202-217, November, 1976.
- Campbell, J. E.; McGrath, P. E. and Cullingford, M. C., "Development of Risk Assessment Methodology Applicable to Radioactive Waste Isolation." Probabilistic Analysis of Nuclear Reactor Safety, proceedings of an American Nuclear Society topical meeting, Newport, Beach, CA, May 1978.
- Claiborne, H. C., Effect of Actinide Removal on the Long-Term Hazard of High-Level Waste. ORNL-TM-4724, Oak Ridge National Laboratory, Oak Ridge, TN, January 1975.
- Cohen, B. L., "Environmental Hazards in High-Level Radioactive Waste Disposal." IEEE Transactions on Nuclear Science. NS-23(1), February 1976.
- Cohen, B. L., "Strong Radioactive Wastes Need Not be a Problem." Nuclear Engineering International. pg. 38, October 1976.
- Cohen, B. L., "High-Level Radioactive Waste from Light Water Reactors." Reviews of Modern Physics. 49(1), January 1977.
- Cohen, J. J., "A Systems Analysis Approach to Nuclear Waste Management." Proceedings, of the Symposium on Waste Management at Tucson, AZ, March 1975.
- Cohen, J. J. and Tonnessen, K. A., Survey of Naturally Occurring Hazardous Materials in Deep Geologic Formations: A Perspective on the Relative Hazard of Deep Burial of Nuclear Wastes. UCRL-52199, January 1977.
- Fleischman, R. M. and Liikala, R. C., Isotopic Composition and Radiological Properties of Uranium in Selected Fuel Cycles. BNWL-SA-5375, Battelle, Pacific Northwest Laboratories, Richland, WA, April 1975.

Gera, F. and Jacobs, D. G., Considerations in Long-Term Management of High-Level Radioactive Wastes. ORNL-4762, Oak Ridge National Laboratories, Oak Ridge, TN, February 1972.

Hamstra, J., "Radiotoxic Hazard Measure for Buried Solidified Radioactive Wastes." Nuclear Safety. 16(2), March-April 1975.

Haug, H. O., Some Aspects and Long Term Problems of High-Level and Actinide-Contaminated Spent Fuel Reprocessing Wastes from the U-Pu and Th-U Fuel Cycles. IAEA-SM-207/51, 1976.

Haug, H. O., Some Aspects and Long Term Problems of High-Level and Actinide-Contaminated Spent Fuel Reprocessing Wastes from the Nuclear Fuel Cycle. ORNL-tr-4302, translation of KFK-2022, 1977.

International Commission on Radiological Protection Task Group; Snyder, W. S.; Cook, J. J.; Nasset, E. S.; Karhausen, L. R.; Howells, G. P. and Tipton, I. H., Report of the Task Group on Reference Man. Pergamon Press, Oxford, 1975.

Jansen, G., Merrill, E. T. and Dawson, G. W., "Chemical Toxicities of High-Level Wastes Generated Through the Year 2000." in Advanced Waste Management Studies Progress Report, April-May-June, 1974. BNWL-B-223-11, pg. 32, Battelle, Pacific Northwest Laboratories, Richland WA, July 1974.

Karol, M. and Post, R. G., Comprehensive Operational Waste Classification System. University of Arizona Engineering Experiment Station Report No. 48, January 1977.

Lyons, R. B., Reduction of Long-Term Potential Hazard of Spent Fuel from CANDU Reactors by Heavy Element Recycles. AECL-5395, Atomic Energy of Canada Limited, Pinawa, Manitoba, July 1976.

McBride, J. P., Moore, R. E.; Witherspoon, J. P. and Blanco, R. E., Radiological Impact of Airborne Effluents of Coal-Fired and Nuclear Power Plants. ORNL-5315, Oak Ridge National Laboratory, Oak Ridge, TN, August 1977.

McGrath, P. E., Radioactive Waste Management Potentials and Hazards from a Risk Point of View. KFK-1992, 1974.

McSweeney, T. I., et. al., An Assessment of the Risk of Transporting Plutonium Oxide and Liquid Plutonium Nitrate by Truck. BNWL-1846, Battelle, Pacific Northwest Laboratories, Richland, WA, August 1975.

Pobereskin, M., Kok, K. and Madia, W. J., Coprocessing - Evaluation of Chemical Reprocessing Without Plutonium Separation. BMI-EQAC-1, Battelle, Columbus Laboratories, Columbus, OH, October 1977.

Rochlin, G. I., "Nuclear Waste Disposal: Two Social Criteria." Science, pg. 23, January 7, 1977.

Rockwell Hanford Company, Spent Unreprocessed Fuel Facility Engineering Studies. RHO-LD-1, October 1977.

Schneider, K. J. and Platt, A. M., eds, High-Level Radioactive Waste Management Alternatives. Section 9: Transmutation, Volume 4, BNWL-1900, Battelle, Pacific Northwest Laboratories, Richland, WA, May 1974.

Smith, C. F., Risk Assessments for the Disposal of High-Level Radioactive Wastes. PhD Thesis at UCLA, 1975.

Smith, C. F. and Kastenber, W. E., "On Risk Assessment of High-Level Radioactive Waste Disposal." Nuclear Engineering and Design. 39:293-333, 1976.

USERDA, Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle. ERDA-76-43, May 1976.

USNRC, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants. WASH-1400, October 1975.

USNRC, Reactor Safety Study: An Assessment of Accidental Risks in U.S. Commercial Nuclear Power Plants. Appendix VI, "Calculation of Reactor Accident Consequences," Section 13, "Calculated Results with Consequence Model," WASH-1400, October 1975.

USNRC, "Standards for Protection Against Radiation." Code of Federal Regulations-Energy. Title 10, Chapter 1, Part 20, 1976.

Voss, J. W. and Post, R. G., Dissolution of High-Level Nuclear Waste Solids. University of Arizona Engineering Experiment Station Report No. 29, 1976.

Walsh, P. J.; Kilough, G. G.; Paizyck, D. C.; Rohwer, P. S.; Rupp, E. M.; Whitfield, B. L.; Booth, R. S. and Reridon, R. J., CUMEX-A Cumulative Hazard Index for Assessing Limiting Exposures to Environment Pollutants. ORNL-5263, Oak Ridge National Laboratory, Oak Ridge, TN, April 1977.

Willrich, M.; Lester, R. K.; Greenberg, S. C.; Mitchell, H. C. and Wolker, D. A., Radioactive Waste Management and Regulation. The Free Press. New York, 1977.

Winegardner, W. K. and Jansen, G., "Toxicity of Various Waste Fractions." in Advanced Waste Management Studies Progress Report, April-May-June, 1974, pg. 26, BNWL-B-0223-11, Battelle, Pacific Northwest Laboratories, Richland, WA, July 1974.

APPENDIX A

Glossary of Variables

<u>Variable</u>	<u>Page Where Defined</u>	<u>Definition</u>
a_i	10	fraction of radioisotope i released to water
A	23	available leaching area of a waste
b_i	10	fraction of radioisotope i released to air
C	29	annual consumption of water for standard man
d_i	11	time period of concern for the presence of radioisotope i in the environment
d_{ij}	29	dose to organ j resulting from one year's consumption of water with 1 RCG of radioisotope i
D	29	overall cancer deaths for all radioisotopes for cancers in all body organs
D_{ij}	29	cancer deaths for radioisotopes i for cancer in body organ j
FD	23	factor of dilution
HA	19	hazards available index
HI	19	hazard index for all radioisotopes
HI_i	19	hazard index for radioisotope i
HM	7	hazard measure for all radioisotopes
HM_i	6	hazard measure for radioisotope i
$HM1$	9	first modified hazard measure for all radioisotopes
$HM1_i$	9	first modified hazard measure for radioisotope i
$HM2$	10	second modified hazard measure for all radioisotopes
$HM2_i$	10	second modified hazard measure for radioisotope i
$HM3$	11	third modified hazard measure for all radioisotopes

Glossary of Variables (Cont'd)

<u>Variable</u>	<u>Page Where Defined</u>	<u>Definition</u>
$HM3_i$	11	third modified hazard measure for radioisotope i
I_i	23	ion exchange holdup factor for radioisotope i
KD_i	23	exchange constant for sorption of radioisotope i
L	23	leach rate of a waste
MPI_i	4	maximum permissible intake of radioisotope i
$MPI_{air,i}$	6	maximum permissible intake of radioisotope i in the air
$MPI_{water,i}$	6	maximum permissible intake of radioisotope i in the water
P	29	mass of spent fuel produced annually
P_i	12	probability of radioisotope i being released to man
$P_{i,n}$	21	fraction of radioisotope i transported to man by pathway n
PHM	12	potential hazard measure of all radioisotopes
PHM_i	12	potential hazard measure of radioisotope i
Q	4	radioactivity of all radioisotopes
Q_i	6	radioactivity of radioisotopes
Q_{ij}	9	exposure or dose to total-body, organ or tissue j from all pathways from radioisotope i
Q_{Gij}	9	exposure or dose to total-body, organ or tissue j from radioisotope i by ingestion pathway
Q_{hij}	9	exposure or dose to total-body, organ or tissue j from radioisotope i by inhalation pathway

Glossary of Variables (Cont'd)

<u>Variable</u>	<u>Page Where Defined</u>	<u>Definition</u>
Q_{Lij}	9	selected or established exposure limit for organ j from radioisotope i
Q_{Tij}	9	exposure or dose to total-body, organ or tissue j from radioisotope i by transportation pathway
Q_{Wij}	9	exposure or dose to total-body, organ or tissue j from radioisotope i by wound entry pathway
Q_{oi}	13	radioactivity of radioisotope i at $t = 0$
r_j	29	risk of cancer death from dose to body organ j per rem
RCG_i	4	recommended concentration guide for radioisotope i in either air or water
RCG_{ij}	29	maximum permissible concentration of radioisotope i for limiting body organ j
$RCG_{air,i}$	6	recommended concentration guide for radioisotope i in air
$RCG_{water,i}$	6	recommended concentration guide for radioisotope i in water
RTI	25	relative toxicity index for all radioisotopes
t	11	time
t'	13	dummy variable in integration
T_i	23	isolation time for radioisotope i
TF	19	transport factor
TF_{Ri}	21	transport factor for R_i route to man
V	19	volume in which waste is entrained
V_f	23	volume flowrate of groundwater
α	14	dummy variable
β	14	dummy variable
δ	14	dummy variable
λ_i	12	decay constant of radioisotope i
$\theta^{1/2}_i$	14	half-life of radioisotope i

APPENDIX B

Tabulation of RCG Values for Radionuclides to be Considered

Isotope	RCG _{Air} (Ci/m ³)	RCG _{Water} (Ci/m ³)	Isotope	RCG _{Air} (Ci/m ³)	RCG _{Water} (Ci/m ³)
⁷⁹ Se	1E-10	3E-6	¹²⁷ Te	3E-8	2E-4
⁸⁵ Kr	3E-7	N/A	¹²⁹ I	2E-11	6E-8
⁸⁷ Rb	2E-8	1E-4	¹³⁴ Cs	4E-10	9E-6
⁸⁹ Sr	3E-10	3E-6	¹³⁵ Cs	3E-9	1E-4
⁹⁰ Sr	3E-11	3E-7	¹³⁷ Cs	5E-10	2E-5
⁹⁰ Y	3E-9	2E-5	^{137m} Ba	3E-8	3E-6
⁹¹ Y	1E-9	3E-5	¹⁴¹ Ce	5E-9	9E-5
⁹³ Zr	4E-9	8E-4	¹⁴⁴ Ce	2E-10	1E-5
⁹⁵ Zr	1E-9	6E-5	¹⁴⁴ Pr	3E-8	3E-6
^{93m} Nb	4E-9	4E-4	¹⁴⁷ Pm	2E-9	2E-4
^{95m} Nb	1E-10	3E-6	^{148m} Pm	1E-10	3E-6
⁹⁵ Nb	2E-8	1E-4	¹⁵¹ Sm	2E-9	4E-4
⁹⁹ Te	2E-9	2E-4	¹⁵⁴ Eu	1E-10	2E-5
¹⁰³ Ru	3E-9	8E-5	¹⁵⁵ Eu	3E-9	2E-4
¹⁰⁶ Ru	2E-10	1E-5	^{166m} Ho	6E-9	3E-5
^{103m} Rh	2E-6	1E-2	²²⁷ Th	6E-12	2E-5
¹⁰⁶ Rh	1E-10	3E-6	²²⁸ Th	2E-13	7E-6
¹⁰⁷ Pd	1E-10	3E-6	²³⁰ Th	8E-14	2E-6
¹²⁶ Sn	1E-10	3E-6	²³¹ Th	4E-8	2E-4
¹²⁵ Sb	9E-10	1E-4	²³⁴ Th	1E-9	2E-5
^{126m} Sb	3E-8	3E-6	²³¹ Pa	4E-14	9E-7
¹²⁶ Sb	1E-10	1E-4	²³³ Pa	6E-9	1E-4
^{125m} Te	4E-9	1E-4	^{234m} Pa	3E-8	3E-6
^{127m} Te	1E-9	5E-5	²³⁴ Pa	1E-10	3E-6

(a) For soluble species, unless insoluble is more restrictive.

Tabulation of RCG Values for Radionuclides to be Considered (Cont'd)

<u>Isotope</u>	<u>RCG_{Air}(Ci/m³)</u>	<u>RCG_{Water}(Ci/m³)</u>	<u>Isotope</u>	<u>RCG_{Air}(Ci/m³)</u>	<u>RCG_{Water}(Ci/m³)</u>
²³² U	9E-13	3E-5	²⁴¹ Pu	3E-12	2E-4
²³³ U	4E-12	3E-5	²⁴² Pu	6E-14	5E-6
²³⁴ U	4E-12	3E-5	²⁴¹ Am	2E-13	4E-6
²³⁵ U	4E-12	3E-5	^{242m} Am	2E-13	4E-6
²³⁶ U	4E-12	3E-5	²⁴² Am	1E-9	1E-4
²³⁷ U	1E-10	3E-6	²⁴³ Am	2E-13	4E-6
²³⁸ U	3E-12	4E-5	²⁴² Cm	4E-12	2E-5
²³⁷ Np	1E-13	3E-6	²⁴³ Cm	2E-13	5E-6
²³⁹ Np	2E-8	1E-4	²⁴⁴ Cm	3E-13	7E-6
²³⁸ Pu	7E-14	5E-6	²⁴⁵ Cm	2E-13	4E-6
²³⁹ Pu	6E-14	5E-6	²⁴⁶ Cm	2E-13	4E-6
²⁴⁰ Pu	6E-14	5E-6			

APPENDIX C

Decay Constants of Radionuclides in Spent Fuel

<u>Isotope</u>	<u>Decay Constant (yr⁻¹)</u>	<u>Isotope</u>	<u>Decay Constant (yr⁻¹)</u>	<u>Isotope</u>	<u>Decay Constant (yr⁻¹)</u>	<u>Isotope</u>	<u>Decay Constant (yr⁻¹)</u>
⁷⁹ Se	1.1E-5	¹²⁶ Sn	6.9E-6	¹⁵⁴ Eu	8.1E-2	²³⁸ U	1.6E-10
⁸⁵ Kr	6.5E-2	¹²⁵ Sb	2.5E-1	¹⁵⁵ Eu	1.4E-1	²³⁷ Np	3.2E-7
⁸⁷ Rb	1.5E-11	^{126m} Sb	1.9E+4	^{166m} Ho	5.8E-4	²³⁹ Np	1.1E+2
⁸⁹ Sr	5.0	¹²⁶ Sb	2.0E+1	²²⁷ Th	1.4E+1	²³⁸ Pu	7.9E-3
⁹⁰ Sr	2.4E-2	^{125m} Te	4.4	²²⁸ Th	3.6E-1	²³⁹ Pu	2.8E-5
⁹⁰ Y	9.5E+1	^{127m} Te	2.3	²³⁰ Th	9.0E-6	²⁴⁰ Pu	1.1E-4
⁹¹ Y	4.3	¹²⁷ Te	6.5E+2	²³¹ Th	2.4E+2	²⁴¹ Pu	4.6E-2
⁹³ Zr	7.3E-7	¹²⁹ I	4.4E-8	²³⁴ Th	1.0E+1	²⁴² Pu	1.8E-6
⁹⁵ Zr	3.9	¹³⁴ Cs	3.4E-1	²³¹ Pa	2.1E-5	²⁴¹ Am	1.6E-3
^{93m} Nb	5.8E-2	¹³⁵ Cs	3.0E-7	²³³ Pa	9.4	^{242m} Am	4.6E-3
^{95m} Nb	7.0E+1	¹³⁷ Cs	2.3E-2	^{234m} Pa	3.1E+5	²⁴² Am	3.8E+2
⁹⁵ Nb	7.2	^{137m} Ba	1.4E+5	²³⁴ Pa	9.1E+2	²⁴³ Am	9.4E-5
⁹⁹ Tc	4.6E-7	¹⁴¹ Ce	7.8	²³² U	9.6E-3	²⁴² Cm	1.6
¹⁰³ Ru	6.4	¹⁴⁴ Ce	8.9E-1	²³³ U	4.4E-6	²⁴³ Cm	2.5E-2
¹⁰⁶ Ru	6.9E-1	¹⁴⁴ Pr	2.1E+4	²³⁴ U	2.8E-6	²⁴⁴ Cm	3.9E-2
^{103m} Rh	6.5E+3	¹⁴⁷ Pm	2.6E-1	²³⁵ U	9.8E-10	²⁴⁵ Cm	8.2E-5
¹⁰⁶ Rh	2.8E+3	^{148m} Pm	6.1	²³⁶ U	3.0E-8	²⁴⁶ Cm	1.5E-4
¹⁰⁷ Pd	1.1E-7	¹⁵¹ Sm	7.5E-3	²³⁷ U	3.7E+1		

DISTRIBUTION

No. of
Copies

No. of
Copies

OFFSITE

	A. A. Churm Chicago Patent Group DOE Chicago Operations Office 9800 South Cass Avenue Argonne, IL 60439		R. J. Feinberg Knolls Atomic Power Laboratory P.O. Box 1072 Schenectady, NY 12301
334	DOE Technical Information Center		S. C. Thomac Department of Pharmacology and Toxicology Dartmouth Medical School Hanover, NH 03755
	M. L. Wheeler Los Alamos Scientific Laboratory P.O. Box 1663 Mail Stop 490 Los Alamos, NM 87545		H. C. Burkholder Office of Nuclear Waste Isolation Battelle Memorial Institute 505 King Ave. Columbus, OH 43201
	J. E. Hooper FMC Corporation Engineered Systems Division 328 Brokaw Road Santa Clara, CA 95022		K. J. Schneider c/o IAEA Kärntner Ring 11 P.O. Box 645 A-1011 Vienna, Austria
	R. K. Weatherwax California Energy Commission Mail Stop #19 Sacramento, CA 92506		B. Rawles Office of Nuclear Waste Isolation Battelle Memorial Institute 505 King Avenue Columbus, OH 43201
	R. G. Post University of Arizona Department of Nuclear Engineering Tucson, AZ 85721		
	G. L. Woodruff University of Washington Department of Nuclear Engineering Seattle, WA 98105		
	Prof. ir. D.G.H. Latzko University of Technology Rotterdamseweg 139a P.O. Box 5055 2600 GB Delft, Holland		

ONSITE

2	<u>DOE Richland Operations Office</u>
	M. W. Shupe J. A. Fernandez
2	<u>Rockwell Hanford Company</u>
	R. E. Isaacson D. Wood

No. of
Copies

84 Pacific Northwest Laboratory

A. Brandstetter
N. M. Burleigh
E. A. Eschbach
C. A. Geffen
J. Greenborg
R. J. Hall
P. J. Pelto
A. M. Platt
R. E. Rhoads
J. V. Robinson
J. W. Voss (62)
E. C. Watson
L. D. Williams
W. K. Winegardner (3)
Publishing Coordination (2)
Technical Information (5)
Ma