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PRELIMINARY RISK ASSESSMENT METHOD DEVELOPMENT FOR CONFEDERATED TRIBES OF UMATILLA INDIAN RESERVATION - NUCLEAR WASTE REPOSITORY PROJECT

H-122097

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SUMMARY

In recent years, there has been a growing interest in determining and reducing risks in the broad areas of engineering design, operations of facilities and waste disposal. A reduction of risks and the consequences following systems failures or releases of contaminants is of concern to federal and state regulatory agencies, to the industries being regulated, and to financial and insurance institutions.

Risks are always present and inherent in most aspects of modern society. Man-made and natural catastrophes take their toll due to man's increasing population, modern technology, and the use of finite resources. Risks have increased in the areas of occupational and environmental hazards, and can significantly affect economic and societal benefits. Evaluating, understanding and controlling the risks from environmental hazards has recently become a major endeavor. No longer can these risks be simply ignored, nor neglected for fear that evaluating risk can only make matters worse and draw attention. Risks must be evaluated and dealt with in a systematic manner that recognizes economic, engineering, environmental and societal factors.

While there has been an unprecedented increase in public awareness of all kinds of environmental pollution during the last decade, no potential hazard seems to have evoked as much concern as have radioactive materials. The reason for focusing so much attention on one member of a wide range of pollutants is difficult to ascertain. There does, however, appear to be a certain "dread factor" associated with radioactivity and recognition in the case of radioactive components more than other types of pollutants that important issues in the risk assessment process remain unresolved.

Whatever the precise reasons for the increased attention, it appears that radiation hazards are in effect being explored as a paradigm of the impact of human activity on the biosphere and on social institutions. It has become essential that any solutions proposed for the problems of handling radioactive material be carefully analyzed for their effects on society, and a high level of cooperation is now necessary among the diverse areas of physical science; social science; and the humanities.

This report addresses the initial efforts by the CERT staff to identify and preliminarily characterize for the Confederated Tribes of the Umatilla Indian Reservation (CTUIR) the major elements necessary to develop a viable method or combination of methods to perform systematic risk assessments associated with a high level nuclear waste repository program at the Basalt Waste Isolation Project (BWIP) site on the Hanford Reservation.

The major elements comprising the development of a tribal risk assessment methodology as outlined in the introductory discussion in Section 1.0 of the report are: (1) characterization of scenarios for the potential release of contaminants; (2) characterization of the environmental dose to predesignated receptor locations principally by means of either atmospheric or hydrologic dispersion and transport of the contaminant from the point of release; (3) characterization of the human dose at specified receptor locations in terms of individual human health effects; and (4) conceptualization of a system for classifying and ranking those human health effects for each contaminant release scenario.

Although the potential exists within the proposed high-level nuclear waste repository program for significant releases of both radioactive and non-radioactive contaminants, major emphasis has been placed under the general guidelines for this study on the development of a risk assessment methodology that primarily addresses the potential radiological implications of possible contaminant releases.

However, the foregoing structured approach would be also readily adaptable to other types of hazardous contaminants as well.

Development of contaminant release scenarios is discussed in considerable detail in Section 2.0 for the two major categories of contaminant release to the CTUIR possessory and usage rights area: (1) both normal and accident transportation scenarios via rail, highway, and/or barge from shipment of high-level nuclear waste either through or adjacent to tribal lands, and, (2) scenarios arising from either normal or abnormal activities associated with the construction, operations, closure, and permanent storage of high-level nuclear waste in a mined geologic repository at the Hanford Site, Washington.

The preliminary identification and characterization of possible radioactive contaminants released to the natural environment via atmospheric, surface water and/or groundwater dispersion and transport as a consequence of the currently envisioned spectrum of activities associated with the proposed nuclear waste repository program is then presented in Section 3.0 of the report. A relatively comprehensive presentation of currently available computer-based atmospheric and hydrologic dispersion and transport modeling programs that could be modified and/or coupled sequentially to predictively compute either radioactive or non-radioactive environmental concentrations of a specific contaminant at prescribed distances from the source or point of release is also provided in Section 3.0.

A third major task effort is directed toward a review and preliminary evaluation of current predictive methods and/or computerized dosimetric models for the characterization of both external and internal radiation dose to humans and the related human health effects and risks in Section 4.0 of this report.

A preliminary methodology for classifying and ranking human health risks from predicted environmental concentrations of possible radionuclides arising from potential release scenarios within the high-level nuclear waste repository program is also introduced in Section 4.0 followed by a presentation of the salient conclusions and recommendations for this iterim report in Section 5.0.

The preliminary characterization of possible transportation release scenarios has identified on-reservation highway and rail accident scenarios as being a major class of scenarios that must be investigated in greater detail since current DOE program plans include high-level nuclear waste shipments by both trucks and trains through the CTUIR and its possessory and usage rights area. Therefore, preliminary mechanisms for the release of radioactive contaminants following a transportation accident scenario are generically developed in some detail on the basis of current spent fuel cask designs. The potential sequence of events for a radioactive release following a hypothetical accident is illustrated by utilization of fault- or event-tree analysis techniques. A method for computing total radioactive release fractions for the major radionuclides of concern from a cask containing nuclear reactor spent fuel also presented in Section 2.0 with accompanying preliminary results. The severity of the consequences for truck and rail accident scenarios is also discussed for shipping casks containing spent fuel high-level nuclear waste. The probabilities of transportation accidents involving highway, rail and/or barge shipments of high-level nuclear waste were examined on the basis of published data for both frequency and severity of transportation related accidents on a nationwide basis recognizing that similar, though. more localized, transportation statistical information is needed for the immediate area encompassing the CTUIR and should be developed for future, more definitive risk assessments.

Potential contaminant release scenarios directly related to the site-preparation construction, operation, closure, and long-term storage of high-level nuclear waste in a permanent, subsurface repository were preliminarily investigated on the basis of four generic classes of possible release scenarios: (1) uncertainties and potential omissions of significant consequence associated with characterization of the candidate repository site at Hanford, Washington; (2) potential disruptions due to natural system dynamics within the general area encompassing the candidate repository site; (3) potential disruptive release scenarios resulting from repository construction and operations; and (4) potential disruptive release scenarios induced by human activities other than repository construction and operation. Many different physical processes can affect the future behavior of an underground high-level nuclear waste repository. Detailed analysis and evaluation of these processes is necessary in order to develop pragmatic scenarios that could lead to significant releases of radioactive waste materials to the biosphere.

Selection of disruptive, release scenarios for detailed parametric characterization and subsequent release-risk assessment and detailed sequential processes or methods of analysis remain to be developed. In terms of the characterization of credible release scenarios for the high-level nuclear waste repository in basalt, the process will entail a minimum of four major procedural steps as follows:

(1) Develop a event of a comprehensive list of credible site-specific disruptive processes and events to the prescribed nominal or baseline conditions for the high-level nuclear waste repository.

(2) Adoption of selection criteria by which disruptive release scenarios can be systematically identified for more detailed analysis.

(3) Assessment of the occurrence probability and likely adversity of the consequences of potential disruptive release scenarios.

(4) Selection of those scenarios to be characterized sufficiently for use in a risk-consequence analysis.

Solicatation of expert opinion (the Delphi approach) affords an alternate approach to the identification, selection and classification of disruptive-release scenarios according to relative occurrence probabilities and could afford an effecting method for systematically reducing the almost monumental number of scenario possibilities for this category of potential scenario releases. Since more detailed characterization of two of the foregoing classes of repository release scenarios are associated with (1) the detailed characterization of the proposed Hanford Site and (2) a more definitive design of the repository, it is recognized that the tribal program, at least in the near-term, be directed towards development of those candidate repository release scenarios resulting from disruptive processes associated with regional and immediate siting area natural systems dynamics and/or potentially disruptive scenarios induced by human activities other than normal repository construction and operation.

As previously stated, an extensive compendium of computer-based mathematical models was compiled and evaluated as presented in Section 3.0. However, in order to perform detailed characterization of environmental concentrations resulting from both radioactive and non-radioactive contaminant dispersion and transport in atmospheric and hydrologic media, appropriate modifications to selected mathematical models and their associated computerized programs must be formally implemented in order to structure a sequential systematic methodology for inclusion in subsequent CTUIR environmental, health and safety risk assessments. Several specific examples of this general analytical approach are discussed in Sections 2.0, 3.0 and 4.0 of the report.

Since the primary federal regulatory standards governing the release of radioactive contaminants to the natural environment are promulgated in 40 CFR 190 and 40 CFR 191, for exnmple, on the basis of allowable radiation doses received by humans (whole body) and selected internal organs of the human body, the development of human dosimetric models is discussed in some detail in Section 4.0. The characterization of human dose as a possible consequences of the various activities comprising the high-level nuclear waste program is logically subdivided into mathematical models for the computation of external or whole body dose and the internal dose to specific organs or systems of the human body. Internal dose characterization is based on the most widely accepted models developed by the International Commission of Radiological Protection (ICRP). Utilizing these methods, outlined for the most part in ICRP 30, mathematical expressions are derived for the radioisotopes of major concern as a function of pathway or route of exposure (oral or ingestion), fraction of the ingested radioactive compound absorbed into the blood, retention in the pulmonary region, and target organs. Mathematical models for the evaluation of external radiation dose are also developed. Calculated examples for both internal and external doses are presented to demonstrate the methodology.

Characterization of human health effects is developed for both stochastic and non-stochastic effects although the current federal regulatory standards are based on stochastic human health effects only. However, provisions can be provided in the proposed mathematical models to compute only the stochastic health effects according to the prescriptions set forth in the regulatory standard; i.e., linear extrapolations to low radiation dose with translation or interpretation of the resultant calculated radiation dose in terms of early cancer fatalities. A preliminary system also is proposed for classifying and ranking potential human health effects on the basis of each potential release scenario. In order to make comparisons within and among various scenarios of possible contaminant release the health effects information is segregated into a probability/consequence (PXC) index. The PXC index is the cross-product of the probability of occurrence of a specific release scenario and the excess cases generated by each scenario. The PXC index allows a composite weighting of two factors such that a high probability/low consequence scenario can be ranked equally with a low probability/high consequence scenario. Therefore, PXC indices for each effect are developed to rank effects within a scenario and to make comparisons among arrays of scenarios.

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APPENDICES

1.0 INTRODUCTION

In recent years there has been a growing interest in determining and reducing risks in the broad areas of engineering design, operations of facilities and waste disposal. A reduction of risks and the consequences following systems failures or releases of pollutants is of concern to federal, state, and tribal regulatory agencies, the industries being regulated and to the financial and insurance institutions. Risks are always present and inherent in most aspects of modern society. Man-made and natural catastrophies take their toll due to man's increasing population, modern technology and use of finite resources. Risks have increased in the areas of occupational and environmental hazards, and can significantly affect economic and societal benefits. Evaluating, understanding and controlling the risks from environmental hazards has recently become a major endeavor. No longer can these risks simply be ignored, nor neglected for fear that evaluating risks can only make matters worse and draw attention. Risks should be evaluated and dealt with in a systematic manner that recognizes economic, engineering, environmental and societal factors.

One area that has received much attention and study is the risks associated with waste disposal. These have been wastes produced by almost all industries from mining and milling to chemical companies to nuclear and fossil-fueled power plants. These wastes are often produced in relatively large volumes and can be toxic or hazardous to both the natural and human environment.

Federal, state, tribal and local governmental regulatory discussions must address, of course, a wide range of possible pollutant effects. Human health concerns include genetic damage and neurological effects as well as cancer, and one must consider such adverse environmental impacts as ecosytem disruption; crop damage, and atmospheric impairment. At present, however, both regulatory agencies and the scientific community have progressed further in developing procedural guidelines for human health risk assessment than they have for environmental effects in terms of hazardous and/or nuclear waste disposal. Additionally, quantitative analytic techniques currently are most refined specifically for cancer assessment.

Health risk assessments are conducted by scientists, but they are not "classical science" in the strictest sense. For regulatory purposes, risk assessments represent a tool that can be used to analyze scientific evidence in order to evaluate the relationship between exposure to toxic or hazardous substances and the potential occurrence of disease. The risk assessment process involves, on one extreme, scientifically verifiable findings, and on the other extreme, judgements about the use of various kinds of scientific information.

No one should ever be misled into believing that results using present techniques have the status of incontrovertible scientific agreement. Despite its uncertainties, however risk assessment is presently the only tool we have for discriminating among environmental health problems. The central question both now and in the future that must be addressed is the extent to which risk assessment judgments can be made more consistent and more reflective of the state of scientific understanding.

There is no constant formula for conducting a risk assessment. Because this is an analytical tool, it can be argued that it must be tailored to the needs of the program in which it is used. Given the myriad of mandates within the various regulatory agencies previously cited, it is not surprising that there are a variety of reasons for performing risk assessments and an equal variety of methods used to conduct them.

A prior scoping study report prepared by the Council of Energy Resource Tribes (CERT) at the request of the Confederated Tribes of the Umatilla Indian Reservation (CTUIR) to evaluate the potential tribal roles and activities as an affected tribe under the Nuclear Waste Policy Act (NWPA) of 1982 identified the need to develop methods for risk assessment as they relate to possible future environmental impact assessments should a high level nuclear waste repository be located at a site on the Hanford Reservation (CERT 1984).

Therefore, a study was initiated by the CERT staff to provide the following information:

- An overview of current risk assessment methods that are used in environmental health, regulatory and public health toxicology, and radiological health.

- A preliminary evaluation of potential human exposures that could arise as a result of a nuclear waste repository program at Hanford Site, Washington.

- A review of current risk assessment methods that could be used to classify potential health and environmental hazards to the CTUIR that might evolve as a consequence of the foregoing nuclear waste repository program.

In the simpliest sense, population risks from toxic pollutants are a function of two measurable factors: hazard and exposure. To cause a risk, a substance has to be both toxic (present an intrinsic hazard), and be present in the human environment at some significant level (provide opportunity for human exposure). Risk assessments interpret the evidence on these two issues, judging whether or not an adverse effect will occur, and (if appropriate) making the necessary computations to estimate the extent of total effects. Risk assessments generally have one or more of the following four steps: (1) hazard identification, (2) dose-response assessments, (3) exposure assessments, and (4) risk characterization.

Hazard identification entails weighing the available evidence, usually in the form of an information data base, and determining whether a substance or group of substances exhibits a particular adverse health effect.

Once it is determined that a toxic substance is likely to cause a particular human effect, it potency must be determined by means of dose-response assessments; i.e., how strongly the toxic substance elicits a response at various levels of exposure or dose. Exposure assessment methodologies are then used to estimate the degree of human exposure to a toxic substance. The best method normally is direct measurement of monitoring of ambient conditions, but is often prohibitively expensive. In practice, one must usually rely on estimates of emissions or releases and limited monitoring information, combined with mathematical models that estimate resulting concentrations.

The degree of exposure of concern may vary from pollutant to pollutant. For many effects, the investigator may be primarily interested in lifetime exposures over the whole population; for others, the chief concern can be maximum levels of exposure to people near the emission source, or peak levels of short term exposure.

Finally, an estimate of the risk associated with the particular exposures in the situation or scenario being considered must be developed that reflects the intent of the basic regulatory standards that are applicable to a given class of activities or events. While the final calculations for risk characterization are oftentimes relatively straightforward (exposure times potency, or unit risk), the manner in which the information is presented is highly important. The final assessment should display all relevant information pertaining to the decision at hand, including such factors as the nature and weight of evidence for each steps of the process, the estimated uncertainty of the component parts, the distribution of risk across various sectors of the population, the assumptions contained within the estimates, and so forth.

Detailed review of the CTUIR scoping study report inevitably leads to risk assessment method development for two principal categories of health-related environmental impact assessment associated with the high level nuclear waste repository program as prescribed under the Nuclear Waste Repository Act (NWPA) that are of major importance to CTUIR; i.e., (1) transportation of high level nuclear wastes via highway, rail or barge through the CTUIR or its ceded lands to a proposed repository site located at Hanford Works, Washington, and (2) site preparation, construction, operation, closure and permanent storage of the high level nuclear waste at the proposed site location. Although various nuclear fuel cycle alternatives are currently being evaluated within the NWPA as graphically illustrated in figure 1-1, the two aforementioned major categories of environmental risk, must ultimately be assessed and mitigated.

Therefore, in terms of the development of risk assessment methodologies appropriate to the assessment of potential environmental impacts to the Umatilla Tribe as a consequence of a possible permanent nuclear waste repository being located at the Hanford site, the initial work effort by CERT has centered on the preliminary characterization of release scenarios, environmental concentrations, human dose determinations, and human health effects.

In more complex risk assessments such as those envisioned for this program, a system for classifying and ranking potential health risks from predicted environmental concentrations is also an important consideration.

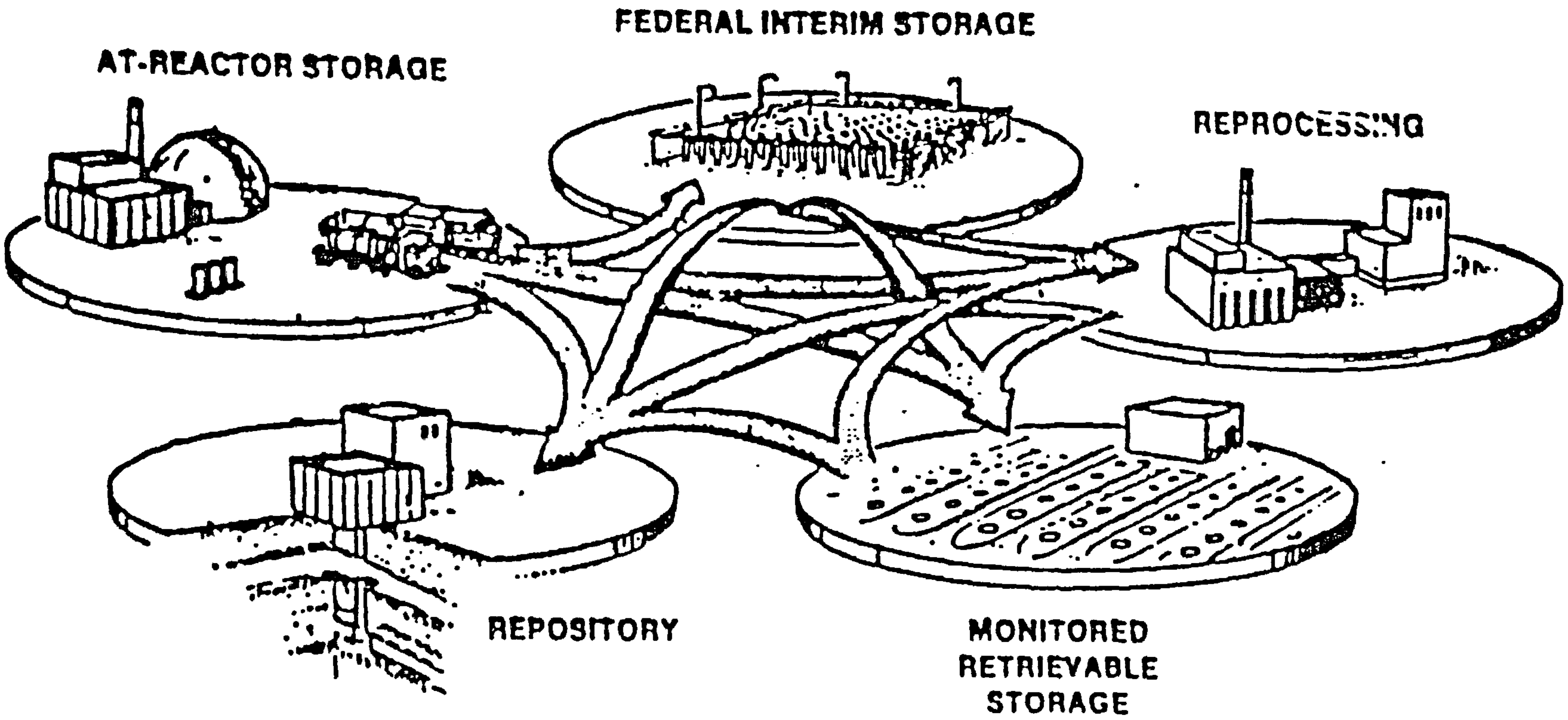


Figure 1-1. NWPA CURRENT NUCLEAR FUEL CYCLE ALTERNATIVES

The specific components of this risk assessment are briefly outlined below and discussed in more detail in subsequent sections of this report for both major categories of environmental impact assessment germane to the CTUIR high level nuclear waste program.

1.1 CHARACTERIZATION OF RELEASE SCENARIOS

The characterization of a release generally yields the following information:

- emission rate(s)
- environmental media (air, water, soil, food chain)
- frequency of occurrence
- character of the radioisotopes

The preliminary identification of possible release scenarios associated with a nuclear high level waste repository program at the Hanford site that could result in potentially significant environmental impacts to the CTUIR is presented in section 2.0 of this report.

1.2 CHARACTERIZATION OF ENVIRONMENTAL CONCENTRATIONS

All potentially applicable environmental pathways must be evaluated in order to predict contaminant concentrations for human or ecosystem receptors. This process involves the selection of appropriate models of environmental transport. It is important to note that models used to predict contaminant concentrations for human receptors may not be sufficiently sophisticated to be applicable to ecosystem receptors such as crops, forests, aquatic and terrestrial organisms. A more detailed discussion of the characterization of environmental concentrations is presented in Section 3.0.

1.3 CHARACTERIZATION OF HUMAN DOSE

Environmental concentrations are used to determine exposure and dose. Depending upon the route of exposure, internal and/or external dosimetry calculations must be carried out as outlined in Section 3.0 and 4.0. Since the basic nuclear radiation standards have been promulgated in terms of an allowable dose or dose rate to humans, these factors must ultimately be taken into account in dose calculations as discussed in greater detail in Section 4.0.

1.4 CHARACTERIZATION OF HUMAN HEALTH EFFECTS

Exposure to ionizing radiation conveys numerous human health risks including immediate and delayed effects. The probability of these adverse human health effects are typically calculated on the basis of epidemiological and animal test results. Generalized risk and excess case equations are developed in Section 4.0 for the life-cycle of a possible high level nuclear waste repository at the Hanford Site, Washington.

1.5 SYSTEM FOR CLASSIFYING AND RANKING POTENTIAL HEALTH RISKS FROM PREDICTED ENVIRONMENTAL CONCENTRATIONS

A possible system for classifying and ranking potential health risks from predicted environmental concentrations also is presented in Section 4.0 to illustrate the use of such systems in subsequent CTUIR environmental impacts assessments. Health consequences are classified by scenario. Ranking of health consequences within a scenario and among scenarios is accomplished by using a special index which is a function of the probability of a scenario occurring and the number of excess cases generated.

2.0 CHARACTERIZATION OF RELEASE SCENARIOS

The use of the Hanford location as a high level waste repository would result in numerous potential exposure scenarios due to both transport and storage of radioisotopes. While the operation of the repository will be designed to minimize risk, the potential sources of risk must be first identified and then subsequently evaluated to assess the probability of each risk-generating scenario, and the health risks it entails.

To develop risk scenarios and estimate risks, basic information is required on the nature of the hazardous materials which generate risk. The following categories of materials are expected to pose potential radiation risks to the population: defense waste sludge, and fission products and actinides from nuclear power plants and existing commercial high-level waste (CERT, 1984: Draft Environmental Assessment, 1984). The radioisotopes in these categories are listed in Appendix A. Their energies of emission, types of emissions and daughters are also presented in the appendix. Very short-lived radioisotopes are not included in Appendix A due to the presently proposed modes of interim storage of the commercial nuclear wastes prior to their ultimate disposal at a permanent subsurface repository location.

The proposed program cycle for ultimate disposal of spent nuclear fuel from commercial nuclear power plants - which constitutes the largest percentage of the high level waste planned for permanent storage in the underground repository - mandates a minimum storage interval of about six months to 1 year at the plant facilities complex after removal of the fuel from the reactor(s). Subsequently, current program planning for the permanent repository outlines additional storage intervals for the spent fuel at a Federal Interim Storage (FIS) facility and/or a Monitored Retrievable Storage (MRS) facility. Thus, for purposes of this preliminary analysis, only those radioisotopes with half-lives greater than 53 days are identified in Appendix A. These specific isotopes, all have greater than 0.78% of their initial activity remaining after one year. That is to state that the selected radioisotopes appearing in Appendix A have less than 7 half-lives in any one year radioactive decay period.

Ultimately, the potential human health impact of a radioactive release may depend upon the ability to respond quickly and appropriately to the emergency. Programs to provide emergency response services for incidents involving both nuclear facilities and radioactive material in transit are operated by federal, state, and local overnments in Oregon and Washington as previously cited in Section 6.6 of the CTUIR High-Level Nuclear Waste Scoping Study (CERT, 1984).

The Federal Emergency Management Agency (FEMA) is responsible for preparing a Federal Radiological Emergency Response Plan (FRERP). A "Master Plan" for commercial nuclear power plant accidents was published on December 23, 1980 (45 FR84910). Further development of the plan, scheduled for completion in 1984, entails certain revisions and expansions to incorporate provisions for responding to all types of peacetime radiological emergencies including transportation of radioactive material (RAM). Preliminary planning guidance was issued in April 1983, and further emergency planning assistance to state and other government authorities is under development.

FEMA also provides planning grants to state emergency services agencies to assist in development of state-wide emergency preparedness programs. Similar financial assistance to Indian tribes is authorized under FEMA's statutes, although to date such aid has been limited or non-existent.

Pursuant to its responsibilities in regulating nuclear activities, NRC requires operators of licensed facilities to prepare and receive approval of site-specific emergency response plans. These plans must be coordinated with state and/or local authorities to ensure that emergency notification, protective actions, and, if necessary, evacuation procedures can be carried out in a concerted manner.

NRC and DOT, in cooperation with other federal agencies, have developed a coordinated system for reporting and responding to transportation incidents involving radioactive materials.

Because of their proximity to locations of incidents and their traditional responsibilities for protecting public health and safety, state and local governments and Indian tribal governments have major roles in emergency preparedness and response activities. Incidents involving off-site impacts of nuclear activities such as the inadvertent release of radioactive materials from permanent or fixed nuclear facilities or transportation incidents usually require primary notification and emergency protective actions by state or local authorities.

Technical and on-site assistance is also available from Radiological Assistance Teams which are situated at several DOE regional centers including Hanford. These teams are dispatched to accident sites upon request and if utilized effectively, can reduce the magnitude and severity of human health risk for the range of both normal and accident release scenarios within the proposed nuclear waste repository program.

Some human health risks, both radiological and non-radiological in nature, may also be posed by various solvents, empty chemical containers, and unused decontamination solutions proposed for possible use at the Hanford nuclear repository site (DOE, Draft Environmental Assessment, 1984). Information will be needed on the nature and quantity of these materials as the underground repository design and operation becomes better defined during subsequent phases of the program.

2.1 TRANSPORTATION RELEASE SCENARIOS

Releases of radioactivity as a consequence of shipments of high level nuclear waste to an underground repository at the Hanford site can occur along designated transportation routes passing through various sections of the CTUIR and its ceded lands as previously cited in the CTUIR High Level Nuclear Waste Scoping Study report (CERT, 1984).

Potential release scenarios can be further subdivided into two broad classes: (1) releases due to a normal, orderly sequence of operations along the specific transportation artery extending from the interim storage facility site (nuclear power plant facilities complex, MRS, FIS, reprocessing plant, etc.) to the proposed permanent, underground repository location at Hanford, and, (2) release scenarios origins from accidents occuring on tribal lands during high level nuclear waste shipments along these same transportation routes.

2.1.1 Transportation Release Scenarios Normal Operational Sequence

Radioactive release scenarios arising from a normal or routine sequence of events during the shipments of nuclear wastes to the permanent repository are the easiest to define and characterize since these releases are primarily at time dependent functions of the direct radiation dose criteria utilized in the design of the shipping cask or containers.

The Department of Transportation (DOT) regulations, found in 49 C.F.R. Section 173.441, set radiation level limitations for any shipment of radioactive materials. Shipments of spent-fuel and other highly radioactive waste may be shipped in a truck trailer, assigned to the exclusive use of a shipper, if radiation levels do not exceed the following limits at any time during transportation:

(1) Two-hundred millirem per hour on the accessible external surface of the package, or 1000 millirem per hour if the following conditions are met:

  (a) the shipment is made in a closed transport vehicle;

  (b) provisions are made to secure the package so that its position within the transport vehicle remains fixed during transportation; and

  (c) there are no loading or unloading operations between the beginning and end of transportation.

(2) Two-hundred millirem per hour at any point on the outer surface of the transport vehicle, including the upper and lower surfaces. In the case of an open transport vehicle, the radiation limit is not to exceed 200 millirem per hour at any point on the vertical planes projected from the outer edges of the transport vehicle, on the upper surface of the load, and on the lower surface of the transport vehicle.

(3) Ten millirem per hour at any point two vertical meters (6.6 ft.) from the lateral surfaces of the transport vehicle, or in the case of an open transport vehicle, at any point two meters from the vertical planes projected from the outer edges of the conveyance, and

(4) Two millirem per hour in any normally occupied position in the car or transport vehicle. This last requirements <u>does</u> not apply to <u>private</u> motor carriers (those in the employ of the shipper) when the personnel are operating under a radiation protection program and wear radiation monitoring devices.

According to DOT rules found in 49 C.F.R. Section 173.443, the level of removable radioactive contamination on the external surfaces of radioactive packaging also "shall be kept as low as practicable," and at the very least, within maximum permissible limits established in the code section.

Further regulatory provisions established procedures for contamination control to be used according to the mode of transport. DOT regulations in 49 C.F.R. Section 177.843 set specific rules for controlling motor vehicle contamination.

Thus, determination of human radiation exposures from routine transportation scenarios can be established by relatively straight forward measurement and/or predictive modeling techniques once the origin and destination of a high level nuclear waste shipment and the mode of transportation (highway, rail or barge) are specified.

A flow chart of the RADTRAN II risk assessment computer model developed by Sandia National Laboratories to perform risk assessment of routine shipments of high level nuclear waste is shown in figure 2-1 and typifies the procedural steps commonly employed in current predictive models. The RADTRAN II model was utilized by DOE in the estimation of potential radiological risks to the region encompassing the CTUIR and its ceded lands as a consequence of routine high level nuclear waste shipments to a repository at Hanford, Washington.

Preliminary estimates of transportation risks related to a repository operation at Hanford prepared by DOE for normal or routine operations are presented in table 2-1. The compilation of the regional and national radiological effects of routine high level nuclear waste shipments to a repository at Hanford of latent cancers over the operating lifespan of the repository and are categorized for the principal regional rail and highway corridors leading to the site as depicted in figure 2-2. The "northern route" includes effects to population groups along rail and highway routes from Coeur D'Alene, Idaho, to the potential repository site. Figure 2-2 illustrates that both major highway (U.S. No. 84) and rail transportation modes for the "southern route" pass through the Umatilla Reservation.

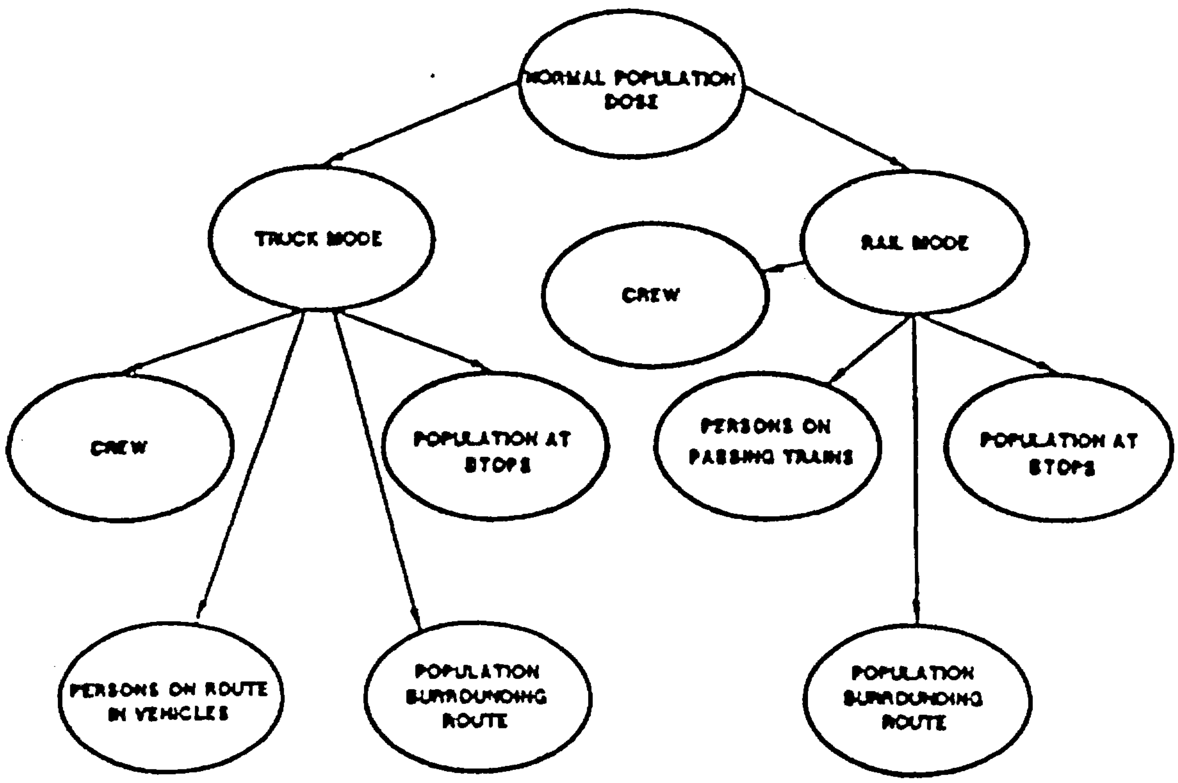


Figure 2-1. RADTRAN II COMPUTER MODEL FLOW CHART: ROUTINE HIGH LEVEL NUCLEAR WASTE SHIPMENTS

Table 2-1. RADIOLOGICAL EFFECTS OF ROUTINE HIGH LEVEL NUCLEAR WASTE SHIPMENTS TO A REPOSITORY AT HANFORD

Mode of shipping/exposure type	Latent cancers throughout waste receiving period		
	Regional all via northern route	Regional all via southern route	National
All via truck:			
Occupational	0.04	0.05	0.58
Nonoccupational	0.19	0.27	3.00
Total	0.23	0.32	3.58
All via rail:			
Occupational	0.0002	0.0003	0.003
Nonoccupatonal	1.14	1.72	18.000
Total	1.14	1.72	18.003

NOTE: Based on unit risk factors presented in the Transportation Appendix, Draft Environmental Assessment, Reference Repository Location, Hanford, Washington, U.S. Department of Energy, December 1984.

The radiological risk estimates derived from utilization of the RADTRAN II predictive model indicate that approximately 1.0 to 2.0 latent cancers could be induced to the regional population (including the CTUIR) along these transportation corridors during the operational phase of the repository.

It logically follows that the exposure levels defined and characterized for transportation scenarios involving a normal sequence of operations will represent the most probable series of events that will transpire during the majority of the shipments of high level nuclear waste to the repository should it be located at Hanford. Unfortunately, the current data base and methodologies are insufficient to provide more definitive risk assessments specific to the CTUIR at the present time.

Nevertheless, it must be emphasized, as a minimum, the CTUIR should be assured that external, direct whole body radiation doses to any tribal members on CTUIR lands as a result of any activities inherent in the normal shipment of high level nuclear waste to a repository located at Hanford do not exceed the EPA - prescribed basic radiation standards as set forth in 40 C.F.R. 191. The major ramifications of radiation standards as they relate to risk assessment method development will be discussed in greater detail in Section 4.0.

2.1.2 Transportation Accident Release Scenarios

Shipments of high level nuclear wastes to a repository represent that element of the national nuclear waste management system, as presently envisioned, that potentially affects the largest number of population centers across the nation. Thus, during the transportation phase of the nuclear repository program the most geographically diverse pathways exist for the accidental release of radioactive materials. Based on prior high level nuclear waste shipping experience and continuing studies of new shipping cask or container designs and related risks and safety, it is believed that the overall transportation risks to public health and safety and the environment are quite low. However, perturbations to the present statistical data base as a consequence of the anticipated major increase in both volume and frequency of high level nuclear waste shipments when the first permanent repository becomes operational must be factored into the development of risk assessment methodologies.

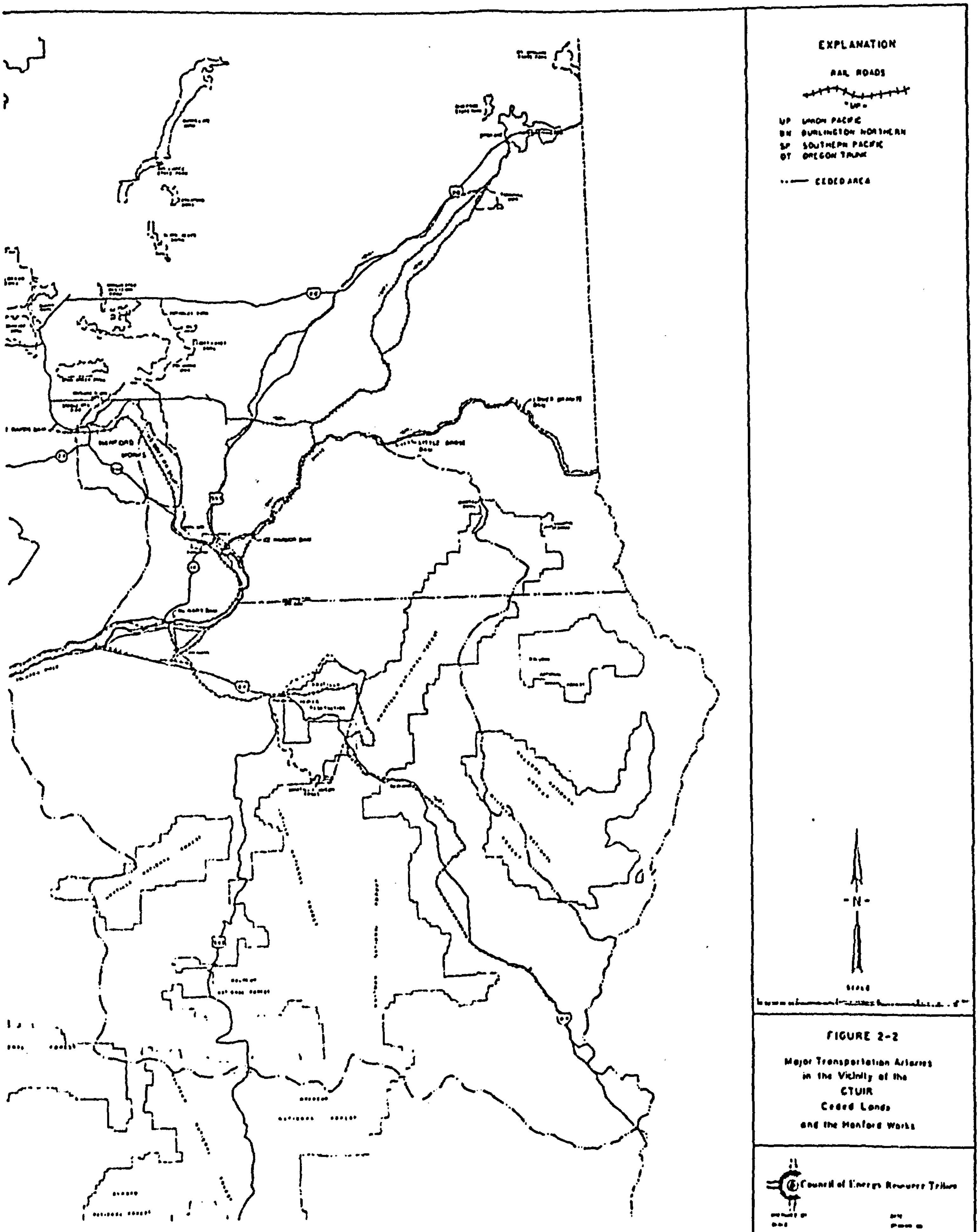
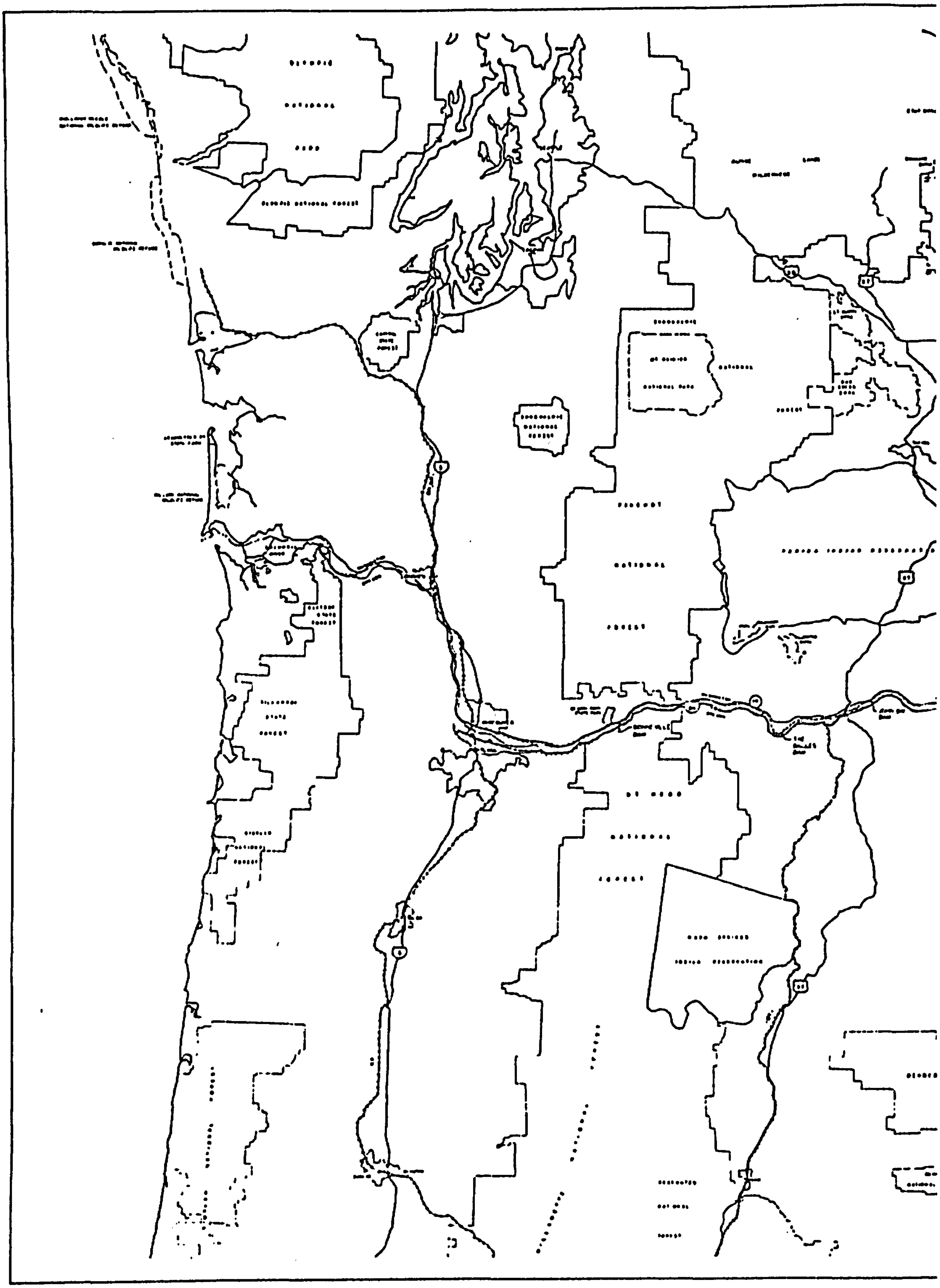


FIGURE 2-2

Major Transportation Arteries in the Vicinity of the CTUIR Ceded Lands and the Hanford Works



The radiological risk estimates derived from utilization of the RADTRAN II predictive model indicate that approximately 1.0 to 2.0 latent cancers could be induced to the regional population (including the CTUIR) along these transportation corridors during the operational phase of the repository.

It logically follows that the exposure levels defined and characterized for transportation scenarios involving a normal sequence of operations will represent the most probable series of events that will transpire during the majority of the shipments of high level nuclear waste to the repository should it be located at Hanford. Unfortunately, the current data base and methodologies are insufficient to provide more definitive risk assessments specific to the CTUIR at the present time.

Nevertheless, it must be emphasized, as a minimum, the CTUIR should be assured that external, direct whole body radiation doses to any tribal members on CTUIR lands as a result of any activities inherent in the normal shipment of high level nuclear waste to a repository located at Hanford do not exceed the EPA - prescribed basic radiation standards as set forth in 40 C.F.R. 191. The major ramifications of radiation standards as they relate to risk assessment method development will be discussed in greater detail in Section 4.0.

2.1.2 Transportation Accident Release Scenarios

Shipments of high level nuclear wastes to a repository represent that element of the national nuclear waste management system, as presently envisioned, that potentially affects the largest number of population centers across the nation. Thus, during the transportation phase of the nuclear repository program the most geographically diverse pathways exist for the accidental release of radioactive materials. Based on prior high level nuclear waste shipping experience and continuing studies of new shipping cask or container designs and related risks and safety, it is believed that the overall transportation risks to public health and safety and the environment are quite low. However, perturbations to the present statistical data base as a consequence of the anticipated major increase in both volume and frequency of high level nuclear waste shipments when the first permanent repository becomes operational must be factored into the development of risk assessment methodologies.

be postulated, but they have no practical meaning because of their very low probabilities. This presentation defines and examines a spectrum of credible, though highly unlikely, accident scenarios.

Although this presentation limits its conservatism to what is justified by experimental data and the judgments of recognized experts it willingly recognized the paucity of data available to support more realistic assumptions and the need to be conservative when data do not exist. However, this presentation will, for an interim period, provide a reasonable basis from which environmental consequences of transporting spent fuel can be calculated; data gathered in future studies must refine the analyses presented here.

Transportation accidents involving spent fuel have never resulted in a release of radioactive material to the environment, so it is very difficult to define a credible worst-case in which radioactive material is released. However, a majority opinion of recognized experts has concluded that a large breach (greater than 6.4 cm^2 (1.0 in ^{2} cross sectional area) is not credible to consider (Wilmot, 1981).

Mechanisms and Pathways

A radiological health problem exists when radionuclides are released from the spent fuel, somehow escape from the cask to the environment and eventually reach people. A number of mechanisms and pathways could allow a release to the environment. The way in which they are combined will be discussed in detail in a subsequent section that discusses fault tree discriptions of the spent fuel scenarios. This section specifically discusses only the mechanisms and pathways.

Accident scenario mechanisms and pathways can be placed into two categories as shown in figure 2-4, that correspond to two necessary steps for release to the environment. The radioactive material must first be released from the spent fuel to the cask cavity, and then this material must be released from the cavity to the environment. A third step that could also be considered separately is the release of radionuclides within the spent fuel rod itself, but since most of the release in this step occurs while the fuel is in reactor, it is combined with the first step. The first step is governed by characteristics of the spent fuel, and the second step is governed by characteristics of the spent fuel shipping cask.

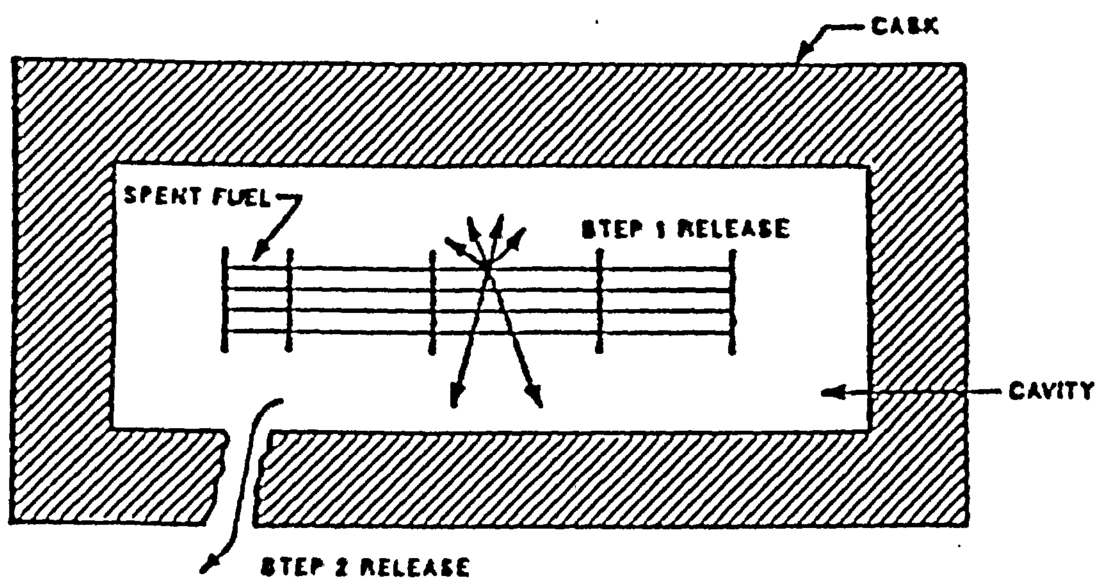


Figure 2-4. STEPS OF RELEASE IN A SPENT FUEL TRANSPORTATION ACCIDENT

Prior to proceeding, the terms used to discuss these events will be defined. The basic elements of a cask and a spent fuel assembly will be defined in general terms.

Four frequently used terms are rod, cladding, fuel-clad gap and assembly as typically illustrated in figures 2-5 and 2-6. A fuel rod or element is the smallest conponent of importance to transportation accidents. A fuel rod is simply a tube made of either zircaloy or stainless steel that contains reactor fuel pellets. The tubing itself is referred to as the cladding. The void space between the cladding and the fuel is referred to as the fuel-clad gap.

The rod contains, under pressure fission product gases generated in the operation of the nuclear reactor and an inert fill gas added during manufacture. A bundle of rods attached together by structural members is called a fuel assembly.

When the fissionable fuel in a reactor has been expended, it is removed. When this spent fuel is transported, it is carried in heavy casks, each holding one or more spent fuel assemblies. The capacity of a cask is commonly given in terms of the number of assemblies it can carry.

Several terms are used to describe a spent fuel cask as depicted in figure 2-7. In very general terms, a cask has two major components: a head and a body. The head is bolted to the body. At the union of the two components is at least one seal (and frequently two) that prevents release of coolant to the environment. The head and body are massive; they provide great strength and shielding from the radiation emitted from the spent fuel. Spent fuel is placed in the hollow core of the cask, called the cask cavity. Small components, referred to as valves and penetrations, are used for filling and draining the cask cavity.

A special term, "waterlogged rod," is used to describe a rod that has failed in a nuclear reactor. The term arises because water is drawn into the fuel-clad gap through a point of failure in a rod as a reactor is cycled through various power levels and shutdowns. From a safety standpoint, the waterlogged rod is different from an unfailed rod: it no longer is highly pressurized with fission gases and its original fill gas, it may have soluble fission products leached from the fuel-clad gap and it may be more brittle and susceptible to impact damage.

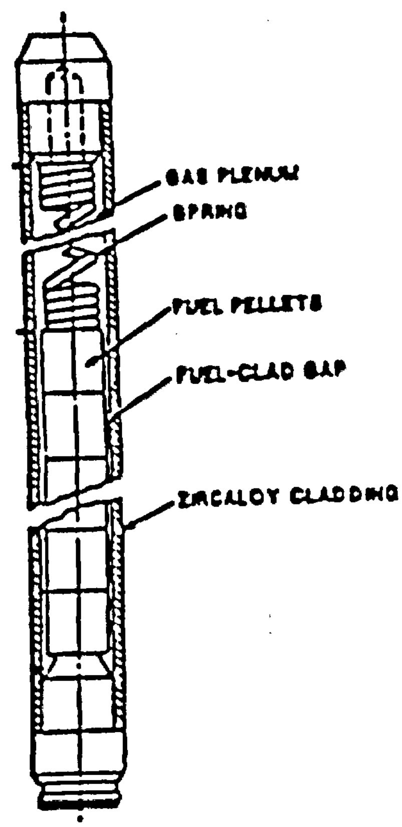


Figure 2-5. SCHEMATIC DIAGRAMS OF A FUEL ROD USED IN A PRESSURIZED-WATER REACTORS

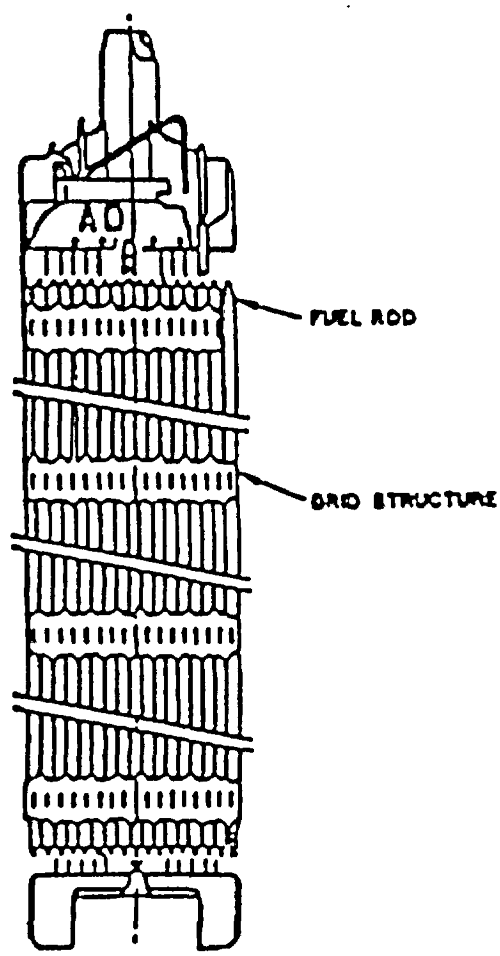


Figure 2-6. SCHEMATIC DIAGRAM OF A FUEL ASSEMBLY USED IN PRESSURIZED-WATER REACTORS

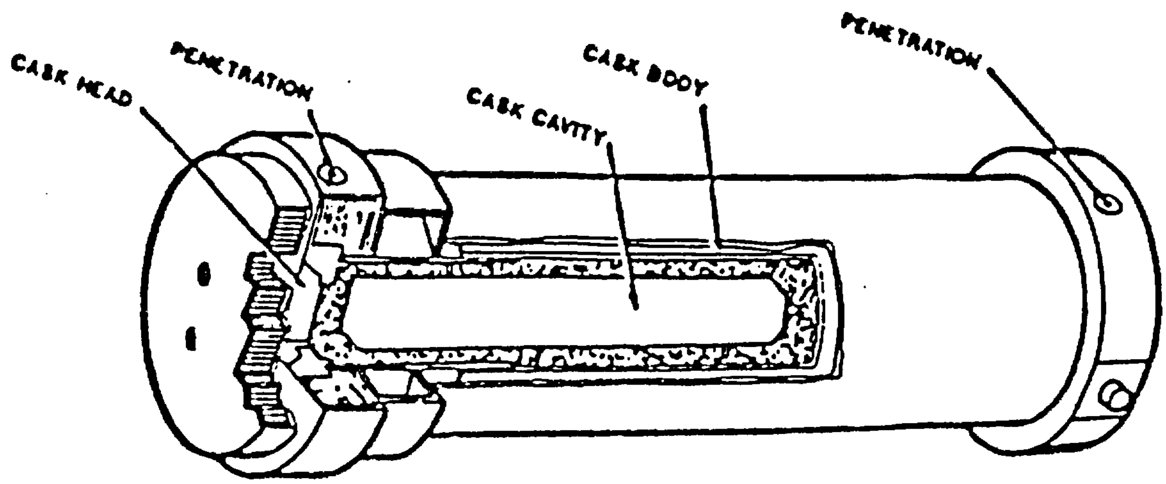


Figure 2-7. SPENT FUEL CASK SCHEMATIC

The rods can fail from a number of causes, the most important of which to this presentation is hydriding of the internal surfaces of the cladding. Hydriding will be discussed further in the subsequent discussion of the impact rupture mechanism.

The mechanisms and pathways will be discussed using these definitions.

Mechanisms for Release from the Spent Fuel Assembly to the Cask Cavity (Step 1)

The release from the spent fuel is dictated by mechanisms that are related to properties of the spent fuel. At least six mechanisms can be postulated: impact rupture, burst rupture, diffusion, leaching, oxidation and crud release. Each mechanism is distinct; however, diffusion, leaching and oxidation are important only after a rupture has occurred.

Impact rupture is a relatively easy mechanism to understand because it is simply the release of radioactive material by the mechnical disruption of the cladding and subsequent depressurization of the fuel rod. The mechanical force of an impact can cause a fuel rod to bend or otherwise deform and subsequently, when coupled with the venting of the fill gas and fission gases, can produce a driving force to actually release materials contained in the fuel-clad gap. Of course, the driving force must be sufficient to actually rupture the cladding.

Rupture of the cladding can occur more easily in a waterlogged rod that has failed due to hydriding. Hydriding occurs when hydrogen gas, which is liberated from impurity moisture in the fuel, can react with the zirconium in the fuel cladding to form zirconium hydrides. The reaction is referred to as hydriding. The hydriding causes a localized embrittlement of the cladding because zirconium alloys form hydride phases that have low ductility and through which cracks may easily propagate. The most common types of hydriding defects are pinholes in cladding blisters caused by hydriding and small cracks which have propagated through hydrided areas of cladding.

As a result of the embrittlement, hydrided cladding may be damaged more easily than unaffected rod cladding. The damage to cladding may be severe if the cladding has been grossly hydrided, but gross hydriding is an unusual occurrence. Waterlogged rods should occur no more often than once in 10,000 rods (Johnson, 1977), and not all of these rods fail as a result of hydriding. In addition, waterlogged rods normally occur in groups, that is, several failed rods occur in the same assembly. Thus, the number of assemblies containing failed, hydrided rods is smaller than the number of failed rods.

In hydrided, waterlogged rods, the mechanical force of impact is the only driving force available to release radioactive material because the rods are depressurized before the impact. However, should a failed rod self-seal, it may become slightly repressurized.

Burst rupture is an analogy to impact rupture. Burst rupture occurs in a severe thermal environment while impact rupture occurs in a severe impact environment. As a spent fuel rod is heated, pressures will increase inside a rod until it may burst. The effect is exacerbated by rapid rises in temperature and very high temperatures. As the rod is subjected to increased pressures, it mechanically deforms until it bursts, creating a hole that has a diameter approaching a few millimeters. The release of pressure through the hole will vent material from the spent fuel-clad gap, material which is in the vicintiy of the hole.

Once a spent fuel rod has been ruptured, vaporized fission products can diffuse in the fuel-clad gap and out the rupture opening. Very high temperatures increase the likelihood of this mechanism, referred to as the diffusion mechanism.

Fission products and spent fuel may be leached from the surface of the spent fuel pellets or from the fuel-clad gap if water can come into direct contact with the spent fuel. This mechanism, which requires that a rod must have been ruptured, is accelerated when the temperature of the water is elevated.

The fifth mechanism, oxidation, is normally expected to occur in the immediate area of a cladding breach and can be initiated on a large scale only after severe disruption of the rod cladding. If the spent fuel left bare by a rupture is exposed to severe thermal environments and flowing air or steam, the UO_2 may be oxidized to UO_{2+x} changing density and cracking macroscopically. The increased surface areas release significantly more fission products. Steam atmospheres are much more neutral toward UO_2 (Rhyne et al., 1979).

The crud-release mechanism is unique among the mechanisms because essentially no fission products are released and because the fuel cladding does not have to rupture for a release to occur. Some fission products may be present in solution in storage pools from rods previously ruptured in service and they may deposit with the crud in small concentrations. The mechanism simply involves the release of the corrosion products (crud) and some traces of fission products by impact, vibration and abrasion or a severe rapid thermal transient. As a result, crud would be expected to be dislodged in most severe transportation accidents. In addition to crud on the fuel rods, there is crud on the cask and structural components of the spent fuel assembly that can also be dislodged in an accident.

Another mechanism, the zirconium-water reaction has been postulated by Resnikoff (Resnikoff, 1979) as important. This reaction if credible for transportation accidents, would have three potentially hazardous results: hydrogen gas would be liberated, heat would be given off as the exothermic chemical reaction proceeds and the cladding could become embrittled. However this exothermic reaction does not even begin to become important in a water cooled nuclear reactor environment until temperatures over 980°C (1800°F) are reached and may become significant at temperatures above 1100°C (about 2000°F) (Lewis, 1977). A minimum limit of 980°C may be applicable to a transportation accident involving a water-cooled cask, but this temperature is presently considered to be about the maximum temperature the external surface of the cask might experience in an accident scenario. However, in a transient thermal environment currently considered credible for a transportation accident by DOE, the spent fuel temperature would not exceed the maximum surface temperature of the cask. Thus, the zirconium-water reaction is not currently deemed a credible mechanism for a transportation accident scenario.

The six credible release mechanisms can result in a release from the spent fuel to the cask cavity, but release to the environment can occur only if a pathway through the cask exits. Potential release pathways through the cask are described in the following text.

Pathways for Release from the Spent Fuel Cask Cavity to the Environment (Step 2)

Spent fuel casks have not been experimentally tested to failure, and consequently, experimental failure threshold values do not exist. As a result, the following subsection relies almost entirely on discussions that took place during the spent fuel scenario workshop in May 1980 (Wilmot et al, 1981).

A release from the cask cavity to the environment may occur if the cask is compromised in a several ways. Casks that have valves used to fill the cask cavity can be compromised if these valves are in some way damaged so as to destroy the valve, to cause it to stick open or to sever the valve piping leading to the cask body. The valves can be vulnerable to both heat and impact, but casks are designed to protect valves from such environments. Some casks do not have valves but have penetrations that allow access to the cask cavity. These penetrations may be more vulnerable to compromise than other parts of the cask even though they are protected. In spite of their vulnerability, because valves and penetrations present a very small target requiring a precisely positioned impact or a localized fire in order to fail, the probability of their failure would be expected to be less than the probability for failure of a cask closure seal.

Cask closure seals that prevent leakage between the cask head and body can be damaged so as to create a pathway from the cask cavity to the environment. Such damaged seals would not be expected to provide a pathway with a large cross section because only small sections of the seals would be likely to fail. The pathway could indeed act as a filter for particulate releases. The damage to cask seals may result from either impact or heat. For example, a closure seal could fail if head bolts yield sufficiently to create a release pathway. The head bolts could be deformed either mechanically or possibly by differential thermal expansion resulting from uneven heating of the cask head and body. Nonetheless, like valves and penetrations, seals are designed to minimize the likelihood of damage resulting from severe environments. This is the type of failure resulted in a cupful of coolant being lost after an 134 kph (84 mph) crash test performed by Sandia National Laboratories (Jefferson and Yoshimura, 1978).

Another release pathway is a small breach (cross sectional area less than 6.4 cm^2 (1 in^2) of the cask body or head. Such a breach would most likely be a fine crack that would allow only limited releases and would protract release times. This sort of a breach is very unlikely and was not evidenced in the severe tests conducted by Sandia National Laboratories (Finley et al, 1970).

Because the small breach is very unlikely and a large breach would be even more unlikely, a large breach presently is not considered credible. No pathway that has a cross sectional area greater than 6.4 cm^2 (1 in^2) is considered credible for the conditions expected in even the most severe transportation accidents.

Other potential pathways could exist but they are not addressed in detail in this preliminary report. For example, potential pathways might include those induced by human error and sabotage or terrorist activities.

Necessary Parameters for Generic Scenarios

A group of generic scenarios is difficult to formulate for a complex system of casks, spent fuel characteristics and modes of transportation. Variations in cask design as well as spent fuel characteristics must be accounted for in the formulation of a generic scenario. Casks are designed differently for different modes of transportation. Cask designs determine which pathways of release from the cask cavity to the environment are possible and which spent fuel release mechanisms can be initiated. The properties of the spent fuel also are important in determining which spent fuel release mechanisms to the cask cavity are possible as well as in determining the release source terms. The mode of transportation influences how much spent fuel can be carried in a cask due to weight restrictions and influences what the maximum accident environments would be if an accident should occur. This particular section will discuss the effects that cask design, fuel characteristics and mode of transportation have on generic scenarios.

Many cask designs are presently licensed for the transportation of spent fuel as presented in table 2-2. As can be seen from table 2-2, the characteristics of these casks differ markedly. Designs differ in weight, dimensions, shielding, coolant, number of spent fuel assemblies carried, decay heat removal capacity, type of closure seals and use of penetrations and valves. Casks of different design will, of course, respond differently to a specific transportation accident.

For example, if a cask is equipped with external valves, a valve might break open or otherwise be damaged so as to provide a direct path from the cask cavity to the environment. If the valves do not exist, as they do not in some cask designs, then this particular pathway is no longer possible.

As discussed earlier, at least one seal is used between the head and body of a cask to ensure the leak tightness of the cask cavity. The seal may be made from an elastomer or a metal. Both are adequate but have different failure thresholds. Particularly, for the thermal environment, the differences can be very large: 280°C for teflon and probably greater than 500°C for metal.

Table 2-2. LICENSED AND AVAILABLE SHIPPING CASKS FOR CURRENT - GENERATION SPENT FUEL (DOE, 1978)

Cask Designation	Number of Assemblies PWR[b]	BWR[b]	Approximate Loaded Cask Weight MT	(Ton)	Usual Transport Mode	Shielding Gamma	Neutron	Cavity Coolant	Maximum Heat Removal, kw
NFS-4 or NAC-1	1	2	23	(25)	Truck	Lead & Steel	Borated Water & Antifreeze	Water/ air[d]	11.5/ 7.5
NLI 1/2	1	2	22	(24)	Truck	Lead, Uranium & Steel	Water	Helium	10.6
TN-8	3	-	36	(40)	Truck[c]	Lead & Steel	Borated Solid Resin	Air	35.5
TN-9	-	7	35	(38)	Truck[c]	Lead & Steel	Borated Solid Resin	Air	24.5
IF-300	7	18	63	(70)	Rail[d]	Uranium & Steel	Water & Antifreeze	Water	76.0[e]
NLI 10/24	10	24	88	(97)	Rail	Lead & Steel	Water	Helium	97.0[f]
TN-12	12	32	97	(107)[g]	Rail	Steel	Borated Solid Resin	Air	135.0

[a]Cask Initials: NAC = Nuclear Assurance Corporation
NFS = Nuclear Fuel Services, Inc.
NLI = NL Industries (previously National Lead Company)
TN = Transnucleaire
IF = "Irradiated Fuel," symbol used by General Electric Company

[b]PWR (Pressurized Water Reactor) and BWR (Boiling Water Reactor).

[c]Overweight permit required. The TN-8 and TN-9 casks are also licensed for rail and water shipments.

[d]Truck shipment for short distances with overweight permit.

[e]Licensed decay heat load is 62 kw.

[f]Licensed decay heat load is 70 kw.

[g]With a cask body extension for abnormally long fuel, the loaded cask weight is 103 MT (113 tons).

The type of coolant used in the cask cavity determines whether some potential spent fuel mechanisms can be initiated. For example, the air in an air-cooled cask allows oxidation to be a potential release mechanism. The coolant in other casks minimized this possibility. The water in the water-cooled casks could leach the spent fuel rods should the cladding be breached in some manner. But, if a cask contains a coolant other than water, leaching is not a credible mechanism unless waterlogged rods, which already contain water from the reactor, are present in the cavity.

The design heat loading is critical in establishing the maximum fuel and cladding temperature that can be realized during accident conditions. Spent fuel in some casks may become hotter than the spent fuel in others if the casks are involved in fires that produce identical, external thermal environments.

As these examples show, specific design features are important in determining what can happen in an accident. Spent fuel characteristics can be equally important.

Characteristics of spent fuel can vary markedly depending on its age (time since it was removed from the reactor), burnup (original amount of radioactivity) and type (what reactor it was used in). Each of these characteristics influences the rate of decay-heat generation, which is critical to a generic scenario because many of the spent fuel release mechanisms are thermally initiated. As spent fuel ages, the rate of decay-heat generation decreases. As a result, older fuel may not generate sufficient heat to initiate such mechanisms as burst rupture. Low-burnup fuel does not generate as much heat as high-burnup fuel. Therefore, for a given fuel age, low-burnup fuel is less likely to be ruptured. The type of reactor that the spent fuel comes from also influences heat generation rate. Pressurized Water Reactor (PWR) spent fuel normally undergoes higher burnup than Boiling Water Reactor (BWR) spent fuel.

The burst-rupture mechanism is very sensitive to the type of fuel. Generally, PWR spent fuel is more likely to fail by this mechanism than BWR spent fuel. Experimental evidence also indicates that PWR fuel would be expected to rupture at temperatures about 100°C lower than temperatures at which BWR spent fuel would be expected to rupture.

The physical condition of the spent fuel can also determine whether some release mechanisms are possible. For example, a waterlogged rod that has failed by hydriding may be more likely to fail by the impact-rupture mechanism.

The mode by which spent fuel is shipped influences the worst credible environment that can be postulated for a scenario. The mode of transport determines the cask weight and capacity, and in addition, it determines the potential accident environments. For example, the rail mode is generally thought to provide the greatest source for fires because a train may be carrying an enormous supply of flammable materials with the spent fuel shipment. The impact created during the rail accident may be considerably different from that for a truck accident. A spent fuel cask carried on a railcar may be buffered by other railcars so that it will not receive a very large impact. On a truck, the buffering may not be as great.

Thus, many factors, including cask design, fuel characteristics and mode of travel, must be considered when developing a scenario. A generic scenario necessarily deals with the worst set of factors possible, and in some cases, the combination of factors used in developing a worst-case scenario may result in one that unrealistically predicts results greater than those predicted if specific cask design, fuel characteristics and mode are considered.

Accident Environments

Two questions normally arise about accident environments: (1) What are the extreme environments that can be produced by very severe transportation accidents? (2) Are these environments sufficient to cause failures that would allow a release to the environment?

In general, the damage resulting from these worst-case accident environments has not been thoroughly investigated. However, the effects of hypothetical test conditions (Title 49, Code of Federal Regulations, Part 173.398) have been examined in detail for each specific cask design as part of the licensing process before it can be used commercially. In performing these examinations, the vast majority of all real accident conditions have been analyzed as the discussions on accident probabilities will explain.

Most Severe Environments Produced in Transportation Accidents

The best estimate of the most severe environments can be obtained by reviewing historical transportation accidents. Such a review is available in at least four references (PNL, 1978; Clark et al, 1976; Dennis et al, 1978; Anderson and Peterson, 1978). The most severe thermal environment appears to be easier to quantify than the most severe impact environment. Two conclusions arise from a review of recent literature: (1) a maximum credible fire temperature would be about 1000°C and (2) a maximum credible fire duration would be one to several hours (2 hours is most often selected in constructing accident scenarios).

In the licensing process for spent fuel casks, the licensee must analyze the effects of a 1/2-hour fire at 800°C (1472°F). In addition, the effects on cask components such as valves, penetrations, and seals and the maximum temperatures that may result in the spent fuel rods must also be determined. Table 2-3 presents the results from analyses for several currently licensed casks as taken from their Safety Analysis Reports for Packaging (SARP). The hypothetical=fire analyses predict a range of temperatures for the various casks; an average value for the maximum fuel temperataure is calculated to be about 540°C (1000°F). A credible worst-case fire (2 hours at 1000°F) would result in considerably higher temperatures.

The test condition analyses use a maximum heat loading that would correspond to a full load of fuel that is 120 to 150 days old. If older fuel is shipped (10 years old spent fuel is currently be proposed by DOE), the decay heat generation rate is considerably reduced. Figure 2-8 shows the effect of increasing aged on heat generation rate. If 1-year-old fuel is shipped instead of 120-150-day fuel, the heating rate is halved; 2-year-old fuel results in 1/4 the heating rate of 150-day fuel. Therefore, the maximum fuel temperature would be lower for older spent fuel.

The thermal environment in a credible worst-case fire appears to be sufficient to cause a release from the spent fuel to the cavity and to create a failure pathway from the cask cavity to the environment. The release fractions, the mechanisms initiated and the pathways created are discussed in a subsequent section.

The maximum impact environment is not so well quantified; however, analyses of cask designs indicate that a credible impact environment can be postulated in which both the

Table 2-3. CASK ANALYSIS FOR HYPOTHETICAL FIRE TEST CONDITIONS

Cask	Coolant	Design Heat Removal Capacity (kw)	Spent Fuel Age (day)	Assemblies	Peak Fuel Temperature After Fire Test
NAC-1 & NRS-4	Water	115	120	1 PWR/2 BWR	510°C
NLI 1/2	Helium	10.6	150 PWR 120 PWR	1 PWR/2 BWR	594°C
TN 8/9	Air	35.5/24.5	150	3 PWR/7 BWR	525°C
NLI 10/24	Helium	70.0	150	10 PWR/24 BWR	533°C
[illegible]-300	Water/Air	76.0 62.0	120	7 PWR/18 BWR	858°C 518°C

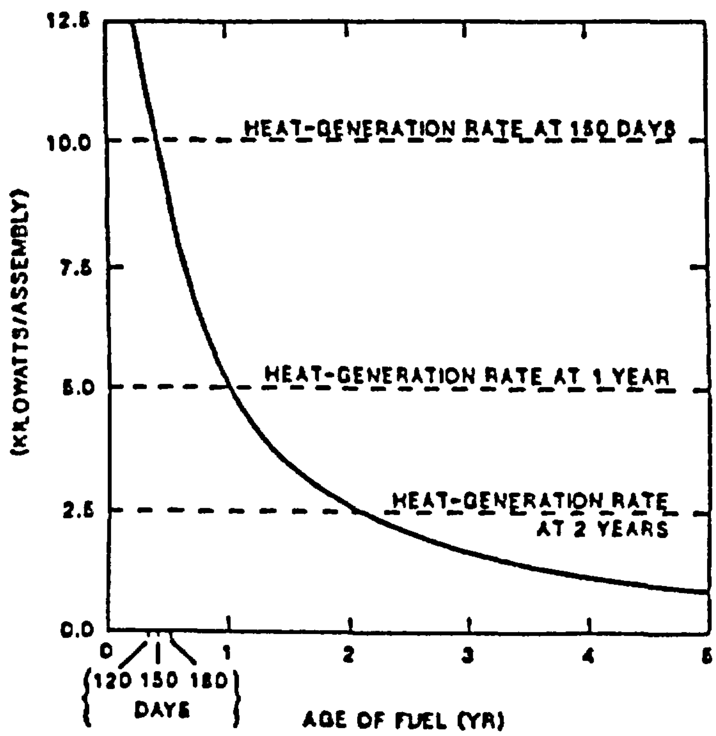


Figure 2-8. DECAY-HEAT GENERATION RATE FROM PWR SPENT FUEL

spent fuel and the cask will fail. The conclusion to be drawm from the foregoing discussion is that extreme conditions can be postulated that are credible (though very unlikely) and that could result in a release of radioactivity to the environment. These conditions are based primarily on analyses and need considerably more experimental substantiation.

Spent Fuel and Cask Failure Thresholds

The failure-mechanism thresholds for spent fuel are reasonably well defined as shown in table 2-4. The thermally activated mechanisms generally require temperatures in excess of 400°C (750°F), in order to release measurable quantities of radioactive material. As temperatures increase to the range of 600°C to 700°C each of these mechanisms except the zirconium-water reaction could be activated and could significantly contribute to release.

The impact thresholds have been analytically considered. No experimental data exist from real accidents because no accident has occurred to date that has involved a severe enough impact to be considered a worst case.

Data which do exist were generated by tests run at Sandia National Laboratories (SNL) and Oak Ridge National Laboratory (ORNL). The SNL tests simulated severe accident conditions for spent fuel casks (Jefferson and Yoshimura, 1978). In each test, the casks contained unirradiated fuel which was to have been used for the nuclear ship Savannah. The fuel did not fail in any of the tests, but in some tests, it was mechanically deformed or bowed. The regulatory drop test from 9 meters (30 feet) onto an unyielding target has been performed at ORNL using the same type of fuel. Some assemblies were bowed, but no cladding failed (Rhyme, et al 1979). However, the applicability of these test results to irradiated, commercial fuel is questionable.

Nevertheless unfailed spent fuel is quite rugged and capable of sustaining severe impact environments. The values listed in table 2-4 for impact rupture show a range of impact-failure limits for spent fuel. The range is wide because specific cask and impact geometries were used. Because the values are dependent on so many variables, they should be applied only to the exact configurations for which they are calculated in the various references that have been previously listed.

Table 2-4. FAILURE THRESHOLDS FOR RELEASE FROM SPENT FUEL TO CASK CAVITY

	Impact Rupture	Burst Rupture[a]	Leaching	Crud	Diffusion	Oxidation	Zirconium-Water
Thermal	NA	Incipient failure 565°C Expected failure 671°C	Occurs at room temperature but usually accelerated by elevated temperature (fuel must have been failed)	Undefined thermal shock probably not credible	400°C (fuel rods must have been failed	430°C	1000°C
Impact	<u>Side Impact</u> Predicted thresholds to fail spent fuel: 71 g to rupture 45 kph (28 mph) cask velocity Experimental thresholds to fail spent fuel: 122 g no failure <u>End Impact</u> Predicted thresholds to fail spent fuel: 38 g to bend 60 kph (41.4 mph) cask velocity Experimental thresholds to fail spent fuel 234 g to rupture	NA	NA	Probably less than regulatory test conditions	NA	NA	NA

[a]Burst rupture is a complicated mechanism that is dependent on many factors in addition to temperature such as heating rate, fuel age and fuel condition.

Table 2-5. BREACHING THRESHOLDS OF A CASK ALLOWING RELEASE FROM CASK CAVITY TO THE ENVIRONMENT

	Penetration or Valve Failure	Closure Seal Failure	Small Breach	Large Breach
Thermal	30 minutes, 1000°C fire valve seal failure	30 minutes, 1000°C fire (Teflon seal)	Unknown, but greater than for a seal failure	Not credible
Impact	Unknown, requires direct hit on a penetration or valve	Side Impact 64 kph (40 mph) cask velocity 131 kph (82 mph) locomotive velocity End Impact 77 kph (48 mph) cask velocity 131 kph (62 mph) cask velocity Side-Center 74 kph (46 mph) cask velocity Puncture	Unknown, but greater than for a seal failure	Not credible

The failure thresholds for spent fuel are such that releases from the spent fuel to the cask cavity could occur. The release of this material from the cavity to the environment is determined, in terms, by the failure thresholds for the cask. As shown in table 2-5, these thresholds are generally not known but are very dependent on cask design. If the accident environment is severe enough, i.e., capable of producing damage much more severe than the regulatory tests the casks can be analytically shown to fail. The maximum credible failure, however, is currently considered to be a small breach such as crack in the cask body or head (Wilmot, 1981).

Even though these failure thresholds are shown by analysis to be attained under very unlikely conditions, no release of radioactive materials from spent fuel shipments has occurred. Releases can be postulated in keeping with a worst-case analysis even though they would be very unlikely.

Fault Tree Description of Event Sequence

Fault-tree diagrams can and have been used to sequence and combine events that result in a release of radioactivity to the environment. A fault tree is a graphical representation of the relationship between specified events and an ultimate undesired event, which in this study is the release of radioactive material to the environment. The relationship between events can be very complex and the use of a fault tree is an attempt to reduce the confusion resulting from such complexity.

Before examining the fault trees drawn for the release of radioactive materials, the logic and the symbols used in the fault trees will be explained. The logic proposed for release of radioactive material from the spent fuel is shown in figure 2-9. The release to the environment is considered to occur in two phases, which are defined according to the sequence of the events during a transportation accident. The specific thermal and impact environments define the length of time the release remains in a phase. For example, when a transportation accident occurs there may be an immediate release of radioactive material to the cask-cavity coolant and a subsequent rapid release of the coolant through an impact-caused breach. Then additional delayed releases of radioactive material may occur from a prolonged fire that may accompany the accident. These delayed releases are the second phase while the immediate releases are considered the first phase. Consequently, two fault trees are drawn in order to present the phases separately.

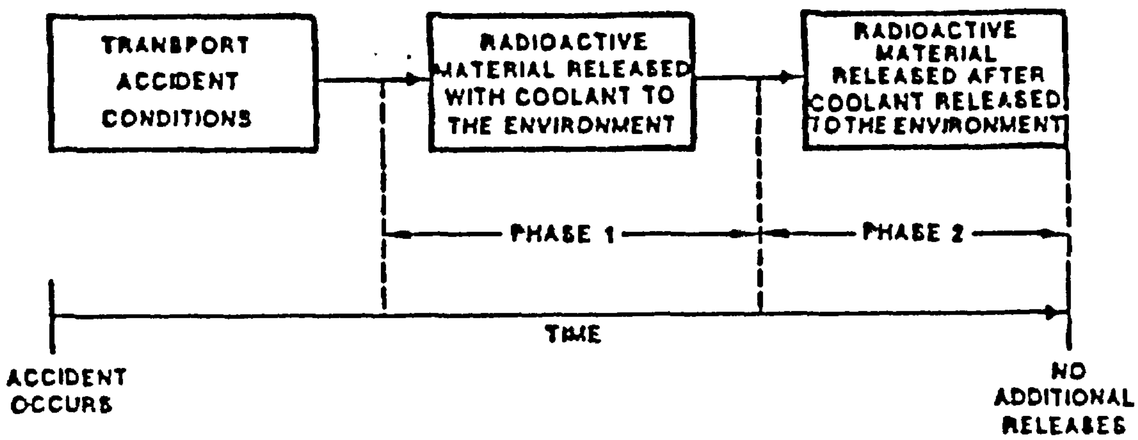


Figure 2-9. SPENT FUEL RELEASE LOGIC

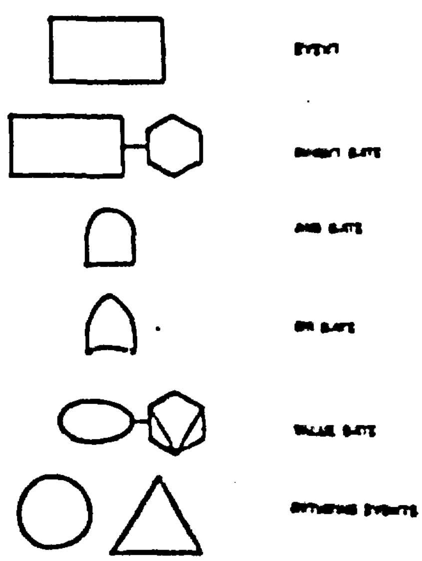


Figure 2-10. FAULT-TREE SYMBOLS

Seven symbols are used to construct fault trees in this presentation. They are shown and labelled in figure 2-10. The rectangle is referred to as an event symbol; within each rectangle, a particular event will be described. The hexagon with an attached rectangle is called an inhibit-gate; its function is to restrict the sequence of events in a fault tree until a particular criterion or limit has been satisfied. The purpose of this symbol and the others will become clearer as the fault trees are described.

And-gate and or-gate symbols have functions similar to that of the inhibit-gate except that more than one event controls the gate. The and-gate requires that two or more events must occur before the events in the fault tree can continue. The or-gate requires that at least one of multiple input events must occur before the events continue. The value-gate requires that a specific value or list of comparators is used to decide if the events can progress beyond the gate.

The circle and the triangle describe initiating events. There is no fundamental difference between them except that events with a characteristic or set of characteristics in common may be identified by using a common symbol.

In fault tree construction, each of these symbols is linked to show relationships and time sequence.

Figure 2-11 is the fault tree for Phase 1, the release of radioactive material to the environment along with the coolant. The rectangular box at the top of the fault tree describes the undesirable event, which is the release of radioactive material to the environment. Proceeding down the fault tree, one encounters a hexagon with an attached rectangular box. This inhibit gate requires that, in order for coolant to reach the environment, the cask must be oriented properly (e.g., the breach or crack must be below the level of the coolant). The next symbol encountered is an and-gate requiring that both events below it must occur for the event above it to occur: if radioactive material is to be released to the environment while in the coolant, radioactive material must be in the coolant, and a release pathway to the environment must exist.

The fault tree is said to branch at this point. Following the left hand branch, the inhibit-gate requires that coolant be in the cask cavity when radioactive material is released to the cavity. The next gate encountered is an or-gate requiring that in order to have a

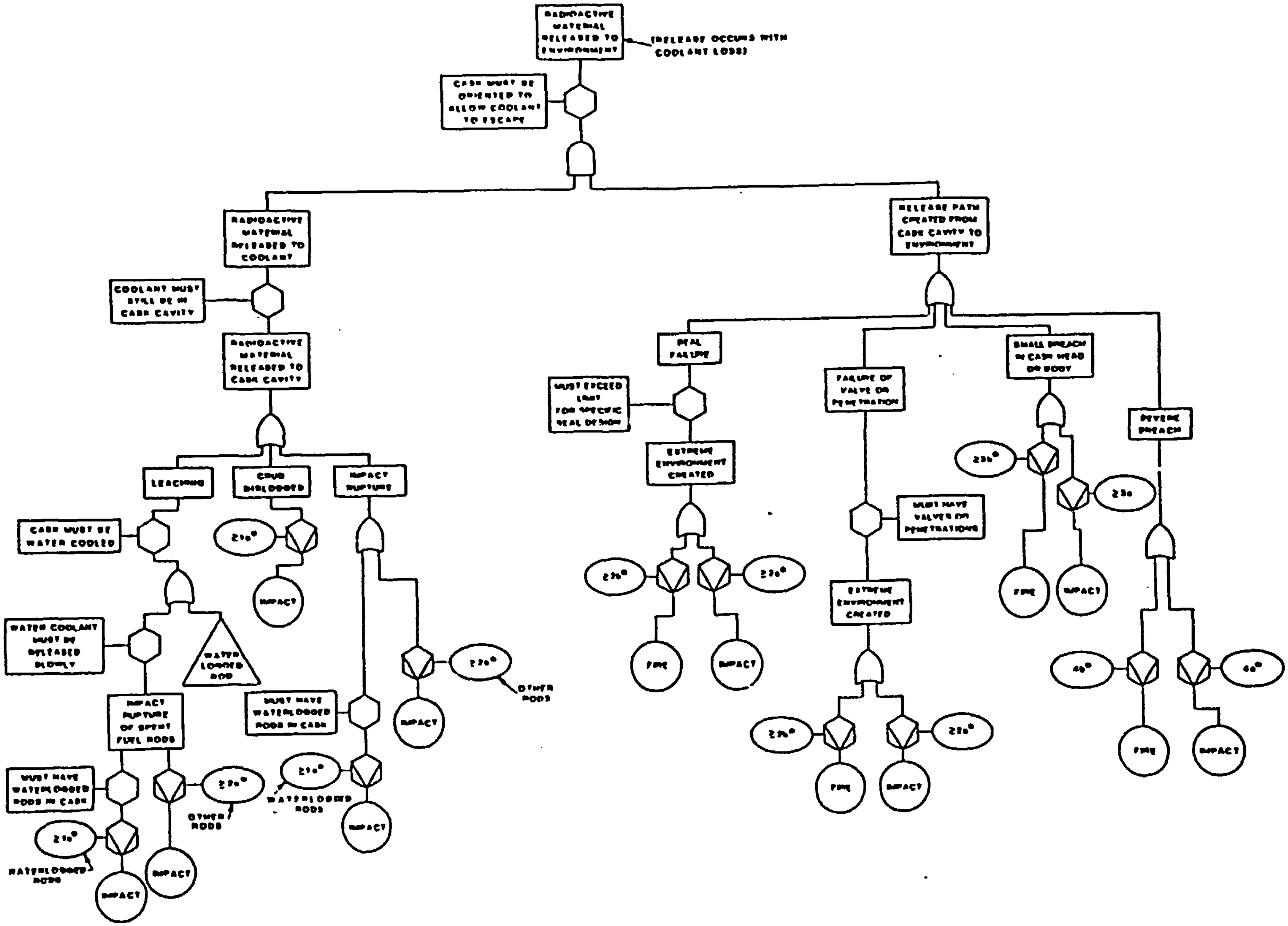


Figure 2-11. FAULT TREE FOR RELEASE OF RADIOACTIVE MATERIAL AND COOLANT TO THE ENVIRONMENT (PHASE 1)

release of radioactive material to the cask cavity only one of three mechanisms needs to be initiated: the leaching mechanism, the crud mechanism or the impact-rupture mechanism. As an example, the leaching-mechanism branch will be described. In order for the leaching mechanism to occur, the cask must be water-cooled. Furthermore, in order to initiate the leaching mechanism a spent fuel rod must be ruptured somehow. The rupture can occur as the result of the impact-rupture mechanism that has resulted from an accident that has caused damage greater than that resulting from the regulatory test conditions, as presented in table 2-6, or possibly as the result of impacting a waterlogged rod at less than regulatory conditions.

In an attempt to quantify the impacts necessary to initiate a release mechanism, a value-gate is placed above each initiating event. The value-gate gives a value that is described in table 2-6. Because of the uncertainties in quantifying impact environments, table 2-6 does not include absolute values. However, for each environment (fire and impact) four relative values, 1 through 4, are given. As the value increases the severity of accident conditions increases. These relative branches should be used only for comparing branches. The end of the first major branch in figure 2-11 has been reached. Now the second branch must be considered.

The right-hand branch will now be traced. In order to have a pathway, it is necessary that one of four events occurs: a seal must fail, a valve or penetration must fail, a small breach must be created or a large breach must be created. Following the branch for a failed penetration, one sees that a penetration must exist if a penetration is to fail. A penetration may fail by either an impact or a fire, but, in accordance with the value-gates, the conditions generated must be at least as severe as the regulatory-test conditions. Similar branches exist for each of the other cask-failure mechanisms.

Figure 2-12 represents the second phase of the release which is the delayed release of radioactive material after the coolant is released. Perusal of this fault tree, one discovers that the pathway must be large enough to allow radioactive material to escape. A release occurs only if two branches of the fault tree are satisfied: radioactive material must be released to a coolant-free cask cavity, and the radioactive material must have a pathway from the cavity to the environment. The second branch is essentially identical to the second branch described in figure 2-11 and will not be discussed again.

Table 2-6. ACCIDENT-ENVIRONMENT CATEGORIES

Impact	Fire	
1a	1b	Environment that produces conditions less severe than regulatory conditions
2a	2b	Environment that produces conditions somewhat greater than or equal to the regulatory conditions
3a	3b	Environment that produces credible conditions much greater than the regulatory conditions
4a	4b	Not credible

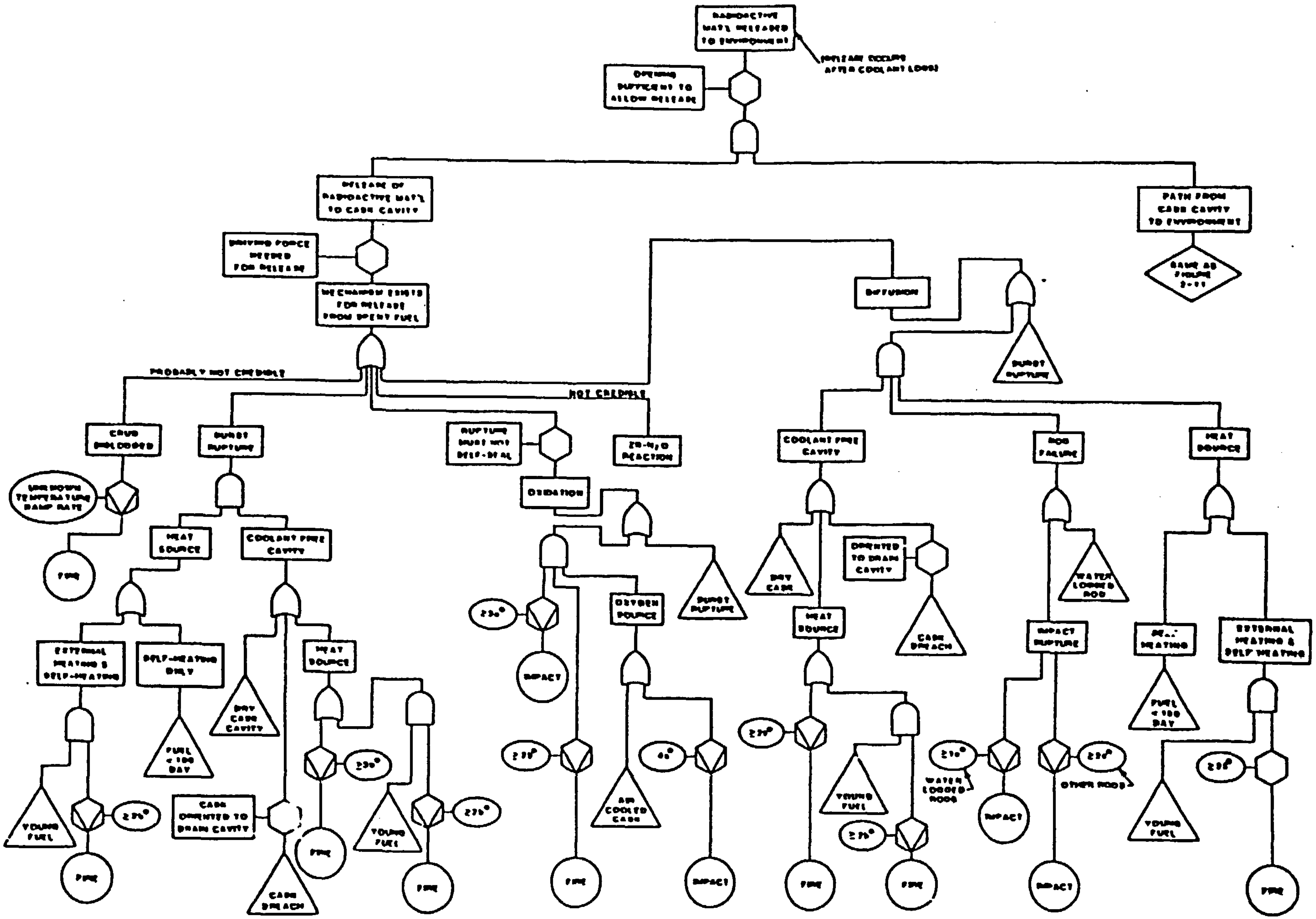


Figure 2-12. FAULT TREE FOR RADIOACTIVE MATERIAL RELEASED AFTER COOLANT RELEASED TO THE ENVIRONMENT (PHASE 2)

Following the first branch, it is seen that there must be a driving force to get the radioactive material out of the spent fuel and into the cavity. A mechanism for release must exist and can be any of four: crud dislodged, burst rupture, oxidation or diffusion. Each of the four can independently or in combination result in a pathway of release to the cask cavity fromthe spent fuel. Examining the diffusion branch as an example, it is seen that diffusion is a possible mechanism if a rod has failed due to the burst-rupture mechanism or if a rod has ruptured, if a severe thermal environment (sufficient to vaporize cesium) exists, and if the cavity is free of coolant. The rod rupture can occur in either of two ways to satisfy the requirement for the diffusion mechanism. The other branches, heat source and coolant-free cask, can be followed in an analogous manner. Each branch ends with an initiating event that must have a severity as indicated by the value-gate. This discussion completes the fault tree for the second phase in release to the environment.

By combining the two fault trees described in figures 2-11 and 2-12 a complete fault tree is generated. The complete fault tree can be used to define a series of accident scenarios. The scenarios can be created by following different branches of the or-gates, remembering that for an or-gate to be satisfied only one input event is required. However, even though only one is required, several could occur. As a result, the worst scenario might be envisioned as the one in which all events leading into or-gates occur.

Table 2-7 summarizes the mechanisms and pathways for two representative worst case transportation accident scenarios-one involving a water-cooled spent fuel shipping cask (Scenario 1) and another employing an air-cooled cask (Scenario 2). Scenario 1 considers all of the six spent fuel release mechanisms that are currently deemed technically credible for water-cooled casks and considers a seal failure and a small breach as the pathways for release from the cask cavity to the environment. Similarly, Scenario 2 considers all mechanisms of release that are credible for an air-cooled cask. The oxidation mechanism is included even though severe impact must have occurred and a replenished oxygen supply must be available. If the fuel that has been grossly failed in the reactor is shipped in an air-cooled cask, the oxidation mechanism could possibly take place in phase 1 before the air coolant escapes, but somewhat arbitrarily, it is assumed that oxidation occurs in phase 2 as shown in table 2-7. In either case, whether the event occurs in phase 1 of phase 2 is irrelevant, since the radioactive release factor is unaffected.

Table 2-7. SUMMARY OF REPRESENTATIVE TRANSPORTATION ACCIDENT RELEASE MECHANISM AND PATHWAYS FOR WATER-COOLED AND AIR COOLED SPENT FUEL SHIPPING CASK SCENARIOS

Procedural	Mechanisms for release to the cask cavity (Step 1)						Pathways for release from the cask cavity to the environment (Step 2)			
Phase	1				2					
Scenario	Crud	Impact Rupture	Leaching	Burst Rupture	Diffusion	Oxidation	Value of Penetration	Seal	Small Breach	Severe Breach.
1	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes	Yes	Not credible
2	Yes	Yes	No	Yes	No	Yes	Yes	Yes	Yes	Not credible

Radioactive Release Fractions

Once the events and their sequence have been determined as representatively illustrated in table 2-7, the fractions of the materials that are released must be defined. The release fractions are unique for each event. Thus release fractions for events that release radioactive material from the spent fuel to the cask cavity will be developed initially in Step 1 followed by an analysis of material release fractions in Step 2 that are emitted from the cask cavity to the environment.

If spent fuel somehow is damaged and its cladding fails, an important question to be answered is: What fraction of radioactive material is released from the rod to the cask cavity? In this section, the release fractions are estimated for a single rod. When explicitly needed for calculating release fractions, the inventory in a single rod is assumed to be that for a typical PWR rod. Because the release fractions for spent fuel are a function of release mechanisms, each potential mechanism is considered separately. The first three discussed are phase I mechanisms and the last three are Phase 2. A summary of the Phase 1 and 2 release fractions for spent fuel release mechanisms (Step 1) is presented in table 2-8.

All of the noble gases that are in the fuel-clad gap are assumed to be released in both representative accident scenarios. Since no more than 20 percent of the noble gases generated during reactor operation would be expected in the gap of most current spent fuel on the basis of recent LWR design and operating experience (Acey and Voglewede, 1980), the release fraction from the rod is 20% of the total inventory of the noble gases generated in the spent fuel. The remaining radionuclides are assumed to be released as particulates that would be swept out as the rod depressurizes after being ruptured. The fraction of particulates released is based upon the fraction experimentally determined by workers at ORNL for the burst rupture mechanism (Lorenz et. al., 1980). Impact rupture can be expected to generate more particles than were present in the spent fuel before impact and available for release during burst rupture. But, since the gas pressure in the rods that sweeps out particulate material is less for impact rupture, and, since the release pathway would be expected to be more restricted by cladding deformation, the release fractions were arbitrarily reduced to 10 percent and used for impact rupture as shown in table 2-8.

Table 2-8. SUMMARY OF PHASE 1 AND PHASE 2 RELEASE FRACTIONS FOR SPENT FUEL RELEASE MECHANISMS (STEP 1)

Radionuclide	Impact Rupture	Phase 1 Leaching Waterlogged	Phase 1 Leaching Other	Crud	Burst* Rupture	Phase 2 Diffusion (steam)	Oxidation (air)
Noble Gases	0.2	-	-	-	0.2	0.02	0.02
Cs 134	2×10^{-6}	0.003	2×10^{-4}	-	0.004	5×10^{-4}	0.03
Cs 137	2×10^{-6}	0.003	2×10^{-4}	-	0.004	5×10^{-4}	0.03
I 129	2×10^{-6}	0.003	2×10^{-4}	-	0.007	4×10^{-3}	0.08
Sr 90	2×10^{-6}	2×10^{-4}	3×10^{-5}	-	2×10^{-5}	-	-
Ru 106	2×10^{-6}	-	-	-	2×10^{-5}	-	8×10^{-4}
Actinides	2×10^{-6}	2×10^{-4}	2×10^{-5}	-	2×10^{-5}	-	-
Co 60	-	-	-	0.25	-	-	-

* Diffusion or oxidation releases must be added to burst releases when the burst rupture mechanism occurs.

Two sets of release fractions for the leaching mechanism are presented in table 2-8. The first set is to be applied to waterlogged rods and the second is to be applied to all other rods when the leaching occurs over a relatively short time. The values given for waterlogged rods are considerably greater than for the values given for shorter time periods because the waterlogged rod is assumed to have been leaching for a very long time before an accident. The maximum fraction of cesium that can be leached from spent fuel as reported by the NRC (U.S. Nuclear Regulatory Commission, 1979) is utilized for the waterlogged-rod release fraction as shown in table 2-8. The iodine values are assumed to be the same. The values for strontium and the actinides are derived from Battelle Pacific Northwest Laboratories studies assuming a leaching period of one week (BMI/PNL, 1978). However, in most accidents very little time probably will be available for leaching. If a few rods are ruptured by an impact, the impact may have been severe enough to have caused a breach in the cask that would allow the coolant to escape. As the coolant level decreased, fewer rods would be exposed to the coolant. The leak rate from the cask then would determine the duration of leaching, which in this case might only be about an hour.

The foregoing values developed by BMI/PNL are very conservative estimates of the leached fractions because they are based on tests performed on fuel fragments from high-burnup fuel that were in free contact with a flowing leachant.

Since, the amount of crud released in an accident has not been determined experimentally no data exist to support most of the following assumptions. The consensus of most contemporary experts is that about 25 percent of the crud that plates on the spent fuel assemblies as a consequence of reactor operation and contaminates the cask cavity surfaces is loosely adhering. The remaining 75 percent of the crud adheres tightly and requires abrasion and chemical treatment for removable.

The predictions of release fractions for burst rupture and diffusion rely heavily on prior work conducted at Oak Ridge National Laboratory (ORNL) where a series of experiments was conducted to quantify and characterize fission product releases under conditions postulated for severe transportation accidents involving spent fuel. The release fractions are calculated using ORNL data and empirical equations. A basic assumption for release fractions for the burst rupture mechanism is that once the rod bursts, all of the fission gases in the fuel-clad gap are assumed to be released.

Once the rod has burst, the diffusion mechanism may proceed if steam is present (Scenario 1) or the oxidation mechanism may proceed if dry air is present (Scenario 2). As a result this presentation as summarized in table 2-8, assumes that any rod that fails by burst rupture is also affected by either the diffusion or oxidation mechanism.

Thus, the diffusion mechanism considered in the two scenarios represented in the analysis is only for a steam atmosphere. Diffusion in an air atmosphere (Scenario 2) is considered to be oxidation.

Very little characterization of the oxidation mechanism has been performed. Oxidation is restricted to areas where the bare fuel is exposed. In order for oxidation to occur, rod failure is assumed to expose fuel surfaces along the fuel-clad gap. The release fractions for cesium and iodine are markedly increased when spent fuel is heated in dry air instead of steam, and significant releases of ruthenium occur. The oxidation mechanism results in expansion of the spent fuel and can seal ruptures or holes. Unless the clad is stripped from the spent fuel, the oxidation mechanism may be self-limiting.

In general, there are very few data to support assumptions about how much a cask restricts the release of radioactive materials once they have reached the cask cavity. Past conservative assumptions were simply to treat the cask as though it did not restrict release at all. In other words, the cask was assumed not even to exist once an accident had occured.

This presentation bases its release fractions from the cavity to the environment on the collection judgment of a prior workshop conducted to analyze transportation accidents (Wilmot, 1980).

Therefore, the data on the most important release fractions from the spent fuel cask cavity to the environment as a consequence of possible transportation accidents presented in tables 2-9 and 2-10, is supported by the experience of cask designers, transporters, material scientists and other technical investigators familiar with cask properties.

Table 2-9. RELEASE FRACTIONS FROM CASK CAVITY TO ENVIRONMENT (Step 2)

Radionuclide	Water-Cooled Cask: In Water (Phase 1)	Water-Cooled Cask: After Water Released (Phase 2)	Gas-Cooled Cask: In Gas Coolant (Phase 1)	Gas-Cooled Cask: After Coolant Released (Phase 2)
Noble Gases	1.0	0.5	0.6	0.5
Volatiles (CsI)	1.0	0.05	0.05	0.05
Particulates (Sr, Actinides, Co)	0.5*	0.05	0.05	0.05

* Value of 1.0 used for leaching.

Table 2-10. FRACTIONS IMPORTANT IN PATHWAY TO PEOPLE

Radionuclide	Fraction 10 um (aerodynamic diameter)	Fraction Aerosolized: Release with Water: Fire	Fraction Aerosolized: Release with Water: No Fire	Fraction Aerosolized: Release with Gas: Fire	Fraction Aerosolized: Release with Gas: No Fire
Noble Gases	1.0	1.0	1.0	1.0	1.0
Volatiles (Cs,I,Ru)	1.0	1.0	0/0.1*	1.0	1.0
Particulates (Sr,Actinides,Co)	0.05**	1.0	0.1	1.0	1.0

* Iodine

** Value of 1.0 used for leaching Sr and actinides

Before they reach the environment, the radionuclides released from the spent fuel must pass through small passages that in most cases will be relatively cool. As a result, there are many places where the radionucldes can condense, settle, plate out or be filtered out before reaching an exist hole to the natural environment.

All noble gases released while water is in a cask cavity were assumed to be released while about 50 percent would be released after the water escaped as shown in table 2-9. For gas-cooled casks that are maintained at greater than atmospheric pressure Rhyne and his coworkers predict that 60 percent of the gases in the cavity would be dischareged when the gaseous coolant was initially released (Rhyne et al, 1979). Even though not all casks are pressurized, a value of 60 percent has been selected. Once a cask has been initially depressurized, release fractions for the noble gases would be considerably less. However, if a fire heats the gases in a cask, 50 percent release of the gases in the cavity can be predicted. Thus, a value of 50 percent was selected for the noble gas release fraction once a water-cooled (Scenario 1) or gas-cooled cask (Scenario 2) has had its coolant discharged.

The volatiles are released on a molecular scale and would not be expected to occur as particles larger than respirable size. They would be easily suspended in water coolant, where they would be expected to remain suspended much longer than in a gas coolant; volatiles released to a gaseous cask atmosphere would be much more likely to plate out on cask walls. For these reasons, 100 percent of the volatiles released to a water coolant are assumed to be released to the environment. A much smaller value of 5 percent was assumed for the release to a gaseous cask atmosphere.

The release fractions for particulates correspond to those for the volatiles except for the conditions where a water-cooled cask still has its coolant in the cavity. Since the volatiles are very small particles, they would be expected to remain suspended longer in water than the particulates, which have sizes that could be considerably greater. The oather values for volatiles and particulates were assumed to be the same.

Therefore, the release fractions for the Phase 1 and Phase 2 spent fuel release mechanisms (Step 1), previously presented in table 2-8, would be multiplied by those fractions shown in table 2-9 and others to calculate an overall total release fractions.

Table 2-11. FRACTION OF FAILED RODS - SPENT FUEL SHIPPING CASKS - SCENARIOS 1 AND 2

Scenario	Mechanism	Fraction of Rods Failed
1	Crud	1.0*
	Leaching	0.1
	Impact Rupture	0.1
	Burst Rupture	0.9
	Diffusion	1.0
2	Crud	1.0*
	Impact Rupture	0.1
	Burst Rupture	0.9
	Oxidation	1.0

* No rods actually need to fail, but all of the crud inventory is assumed to be available for release.

This value is believed to be a maximum since it has been reported by Johnson of PNL that 1×10^{-4} would be the expected fraction of rod failure (Johnson, 1977). A value of 0.001 is supported by undocumented studies produced by the United Kingdom Energy Agency. Ten percent of the unfailed rods were arbitrarily assumed to fail during impact in those scenarios that are deemed to be severe enough. As a result 0.1 is the fraction of rods leached in Scenario 1 which is severe enough to cause impact failure of the rod.

The burst-rupture mechanism is sensitive to the type of fuel. Calculations indicate that PWR rods are much more likely to fail than BWR rods, and it was shown to be possible that in extreme environments all PWR rods would fail by the burst-rupture mechanism. Even though it is probably just as likely that a BWR assembly would be shipped as it is that a PWR assembly would be, all rods that had not been failed previously by some other mechanism are assumed to fail by the burst rupture mechanism, in keeping with the NWPA worst-case philosophy. Consequently, a value of 0.9 is evidenced in table 2-11 for Scenarios 1 and 2.

The diffusion and oxidation mechanisms can only occur if rods have been previously failed by another mechanism. As a result, all the rods that have been failed in either Phase 1 or Phase 2 can be subjected to one or the other of these mechanisms. Thus, the fraction for oxidation or diffusion for both Scenarios 1 and 2 is 1.0 as shown in table 2-11.

Therefore, all the component fractions necessary to calculate total release fractions have been estimated. The calculational flow diagram shown in figure 2-13 demonstrates the various interrelationships of the components necessary to determine the total radioactive release fractions for credible spent fuel cask transportation accident scenarios with a postulated release to the natural atmospheric environment. At the top of the diagram are two parrallel rows of boxes representing component release fractions. One row represents Phase 1, and the other represents Phase 2. Otherwise, the rows are identical; only one row needs to be examined closely. The first box on the left represents the release fraction from the spent fuel to the cask cavity, and the second the fraction of rods failed. The product of these two boxes represents the release fraction from Step 1. Their product multiplied times the third box, which represents the release in Step 2 from the cask cavity to the environment, is the total release fraction from Phase 1. The sum of the box for Phase 1 and of the corresponding box for Phase 2

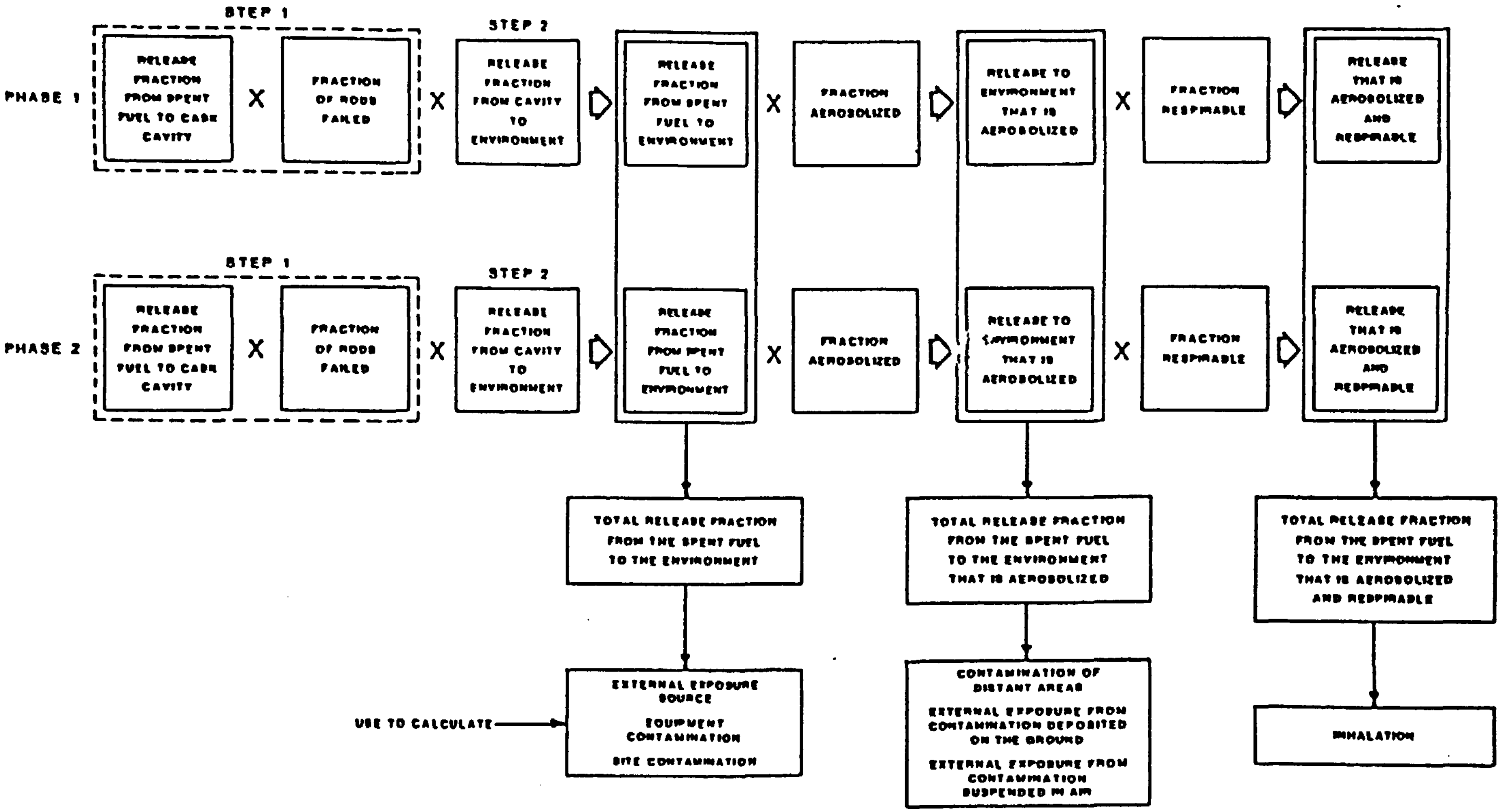


Figure 2-13. FLOW DIAGRAM FOR CALCULATING TOTAL RELEASE FRACTIONS TO THE ATMOSPHERE FROM SPENT FUEL SHIPPING CASK ACCIDENT SCENARIOS

is the total release fraction of radioactive material from a particular scenario. The sum is the total fraction of all radioactive material released to the environment. It is the released fraction of the total inventory of radioactive material in the cask, including particulates, gases and volatiles whether or not they are aerosolized or respirable. These release fractions will be assumed to be applicable to all spent fuel regardless of time in the reactor, time out of the nuclear reactor (age) or reactor type. The one exception might be the age of the fuel. If the spent fuel to be considered is much older than one year, the likelihood of the burst rupture mechanism is greatly diminished. The actual age necessary to preclude this mechanism is quite dependent on the cask in which it is being carried. However, based upon the decay heat generation rate curves for LWR spent fuel previously presented in figure 2-8, it appears reasonable that if the spent fuel is over two years old the burst rupture mechanism could not occur unless a hotter and longer fire than the maximum credible fire postulated for both Scenarios 1 and 2 in this presentation occurs.

In order to determine the amount of radioactivity escaping from a generic scenario like those postulated here, the total release fraction is multiplied by the entire cask inventory of radioactive material. This particular total release would be used to determine such information as the external exposure source that will be encountered by emergency-response teams and the levels of site and equipment contamination.

More useful information can be obtained by multiplying this fraction by the fraction aerosolized. In figure 2-13, the product is represented by the third box from the right. The sum of the boxes for Phase 1 and Phase 2 is the total release fraction from the spent fuel to the environment that is aerosolized. If this fraction is multiplied by the total cask inventory, the amount released and aerosolized is calculated. This amount is useful for determining levels of contamination distant from the site, calculating exposure to the public from ground contamination and calculating external exposure to the public from the passing cloud.

A final and very useful number can be calculated by multiplying this last factor by the fraction of material that is respirable (10 micron aerodynamic diameter). When the sum of these factors for Phase 1 and Phase 2 is calculated and multiplied by the cask inventory, the result is the amount of release that can be inhaled by people (first box on the right in figure 2-13).

In summary, the flow diagram depicted as figure 2-13 shows the computational procedure for the three sets of radioactive material release fractions that are useful in predicting the atmospheric environmental consequences of transportation accident scenarios involving both water-cooled and air-cooled spent fuel shipping casks. The numerical results of the foregoing computational procedure are, in turn, presented in table 2-12 for the two representative cask accident scenarios developed in this section of the report.

Probability of Transportation Accident Release Scenarios

Intuitively, one would expect that the transportation accident release scenarios postulated for a water-cooled spent fuel cask (Scenario 1) and an air-cooled spent fuel cask (Scenario 2) would have a relatively low probability of occurrence. However, predictive quantification of the accident probabilities for the foregoing scenarios, or any scenario for that matter requires a reasonably accurate estimate of both the annual number of shipments and the total annual transport kilometers or miles of spent fuel shipments to the proposed permanent repository located at Hanford, Washington from each of the various locations of the eighty or more commercial nuclear power generating stations throughout the United States.

The historical estimates of spent fuel accident frequencies in terms of accidents per kilometer or accidents/mile shown in table 2-13 for both highway (truck) and rail (rail car) can then be multiplied by the total annual shipment kilometers or miles from each commercial nuclear power plant site to the permanent repository to determine the total number of accidents per year (Wilmot et al, 1983).

Current probability estimates of accident frequencies via highway or rail for spent fuel shipments or any other form of high level nuclear waste, presently are hampered by the paucity of actual accident data as exemplified by a summary of RAM (radioactive material) transportation accidents involving Type B Packages for the period from 1971 to 1982 shown in table 2-14 (Wolfe, 1984).

Additional uncertainties in the utilization of the spent fuel shipment accident rates for highway and rail transportation modes, previously presented in table 2-13, are probably due to the large anticipated increase in the total number of high level nuclear waste shipments (including commercial spent fuel) to the first permanent repository beginning

Table 2-12. SUMMARY OF RADIOACTIVE MATERIAL RELEASES FRACTIONS FOR WATER-COOLED (SCENARIO 1) AND AIR-COOLED (SCENARIO 2) SPENT FUEL CASK-TRANSPORTATION ACCIDENT SCENARIOS FOR ENVIRONMENTAL CONSEQUENCES ANALYSIS

Release Factor	Material Fraction Released from Spent Fuel to Environment		Material Fraction Aerosolized Released from Spent Fuel to Environment		Material Fraction Aerosolized, Respirable Material Released from Spent Fuel to Environment	
Radionuclide	Scenario 1	Scenario 2	Scenario 1	Scenario 2	Scenario 1	Scenario 2
Co 60	1×10^{-1}	1×10^{-2}	1×10^{-1}	1×10^{-2}	6×10^{-3}	6×10^{-4}
Nobel Gases	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-2}	1×10^{-1}	1×10^{-1}
Cs 134	3×10^{-4}	1×10^{-3}	3×10^{-4}	1×10^{-3}	3×10^{-4}	3×10^{-4}
Cs 137	3×10^{-4}	1×10^{-3}	3×10^{-4}	1×10^{-3}	3×10^{-4}	3×10^{-4}
I 129	4×10^{-4}	4×10^{-3}	4×10^{-4}	4×10^{-3}	4×10^{-4}	5×10^{-4}
Sr 90	4×10^{-6}	9×10^{-7}	4×10^{-6}	9×10^{-7}	3×10^{-6}	5×10^{-8}
Ru 106	1×10^{-6}	4×10^{-5}	1×10^{-6}	4×10^{-5}	6×10^{-8}	2×10^{-6}
Actinides	3×10^{-6}	9×10^{-7}	3×10^{-6}	9×10^{-7}	2×10^{-6}	5×10^{-8}

Table 2-13. ESTIMATED SPENT FUEL CASK SHIPMENT ACCIDENT FREQUENCIES FOR TRUCK AND RAIL TRANSPORTATION AS A FUNCTION OF POPULATION DISTRIBUTION

Accident Rate	Urban	Suburban	Rural
Truck (accidents/km)	4.7×10^{-6} (0.9)	8.1×10^{-7} (17.2)	4.0×10^{-8} (81.9)
Rail (accidents/railcar/km)	1.5×10^{-5} (1.1)	1.9×10^{-6} (15.7)	1.0×10^{-7} (83.2)

*1. Rural population corresponds to 6 people/km^2 (mean density)

2. Suburban population corresponds to 719 people/km^2 (mean density)

3. Urban population corresponds to 3,861 people/km^2 (mean density)

4. Numbers in parentheses denote percentage of travel time in the population zone.

Table 2-14. SUMMARY OF RAM TRANSPORTATION ACCIDENTS INVOLVING TYPE B PACKAGES (1971-1982)

Date of Acc/Inc	Mode	Description of Package Involved	RAM Involved	Package Size (lb)	No. Pkg Shipped	Accident Conditions
07/10/71	Highway	Lead container	Co-60	700	1	Collision
12/08/71	Highway	Cask, spent fuel	Fissile	49,000	1	Truck left road and cask was thrown off
03/10/74	Highway	Container	Ra-192	66	1	Trailer involved
03/29/74	Rail	Cask, spent fuel	RAM LS .	220,000	1	Derailment
08/09/75	Highway	Cask	U-235, U-238, Pu-239	16,000	1	Trailer ran off road and overturned
05/06/77	Highway	Camera, radiography	Ir-192		1	Pickup truck accident
08/11/77	Highway	Camera radiography	Ir-192		1	Truck in accident with gasoline truck
10/03/77	Highway	Radiography source	Ir-192		1	One-vehicle accident
02/09/78	Highway	Cask,	Spent fuel		1	Truck buckled from cask weight
04/10/78	Highway	Camera, radiography	Ir-192	50	1	One-vehicle accident
07/26/78	Highway	Steel cask, lead lined	Cs-137		2	Jeep overturned
08/13/78	Highway	Cask, spent fuel		23,000	1	Empty cask broke through trailer bed
08/27/85	Highway	Camera radiography	Ir-192		1	Pickup truck accident
09/11/78	Highway	Camera, radiography	Ir-192		1	Truck overturned
09/15/78	Highway	Cask	Ir-192		1	Truck overturned
11/28/78	Highway		Ra		1	Truck overturned
01/10/79	Highway	Cylinder	Ir-192	6,800	5	Vehicle rear-ended truck
01/14/80	Highway	Cask, teletherapy source	UF_6		1	Semi struck truck
01/31/85	Highway	Cask	RAM LSA		2	Semi jackknifed
07/21/80	Highway	Exposure device, magnaflux	IR-192		1	Truck collided with car
08/22/80	Highway	Cylinder, 30B	UF_6, fissile		5	Truck forced off highway
09/06/80	Rail	Cylinder, 30B	UF_6, fissile	5,000	8	Train wreck
09/29/80	Rail	Radiography source	Sr-90, Y-90		3	Rail accident
06/09/81	Highway	Source, shielded	Am-241/Be		1	Pickup accident
09/02/81	Highway	Source	. -192			Vehicle in crash
11/03/82	Highway	Cask	. pty	28,000	2	Truck overturned, cask was thrown off
				TOTAL	45	

in the year 1996. This incongruency is vividly illustrated by comparing a prior data summary of United States spent fuel shipments for the years 1964 through 1979, shown in table 2-15 (Wilmot, 1981), with the projections for spent fuel shipments over the 26-year program envisioned for the first repository as presented in table 2-16.

Table 2-15 indicates that an average shipment rate of about 300 assemblies per year was experienced from 1964-1979.

Recent projections by DOE and NRC predict a total mileage for transportation of spent fuel of 300,000 to 400,000 kilometers per year until such time as away-from-reactor storage sites; e.g., FIS, MRS, and the first permanent repository, are made available for storing spent fuel that can no longer be stored at reactor sites. These mileage projections are consistent with the average annual number of spent fuel shipments as shown in table 2-15, that were utilized to develop the transportation accident frequency data presented in table 2-13. This very limited accident frequency data base has been utilized by DOE as the basis for both radiological and non-radiological risk assessment of transportation accident scenarios in their recent Draft Environmental Assessment (DOE, 1984) notwithstanding their own projected recognition of an average number of spent fuel shipments from both PWRs and BWRs of 6,180 if all shipments were made by truck or an average of 800 spent fuel shipments per year assuming all rail shipments as shown in table 2-17. Therefore, it may be concluded that once the first repository operation is initiated the spent fuel shipment rate would be increased by at least a factor of 20 for an all-highway transportation mode. Thus, it must be emphasized that projections of future spent fuel accident occurrences must be based upon a continuing review of shipping experience, coupled with more definitive projected shipping volumes, which depends upon government policies, and the actual locations of away-from-reactor interim storage sites and the permanent repository site(s).

Probabilities and Accident Rates for Spent Fuel Transportation Accident Scenarios of Varying Severity

The spent fuel cask accident frequency rates for truck and rail transportation modes previously presented in table 2-13 are for accident of all severities within three distinct population zones (urban, suburban or rural) along a prescribed route. Because there have been no transportation accidents involving spent fuel as severe, for example, as the worst case accident scenarios developed in this section of the report for a water-cooled spent

Table 2-15. SUMMARY OF SPENT FUEL SHIPMENTS IN THE UNITED STATES (1964-1979 INCLUSIVE)

Year	Assemblies Shipped
1964	264
1965	39
1966	165
1967	269
1968	67
1969	217
1970	180
1971	561
1972	93
1973	267
1974	137
1975	492
1976	712
1977	675
1978	512
1979	147
Total	4797

Average Shipment ≈ 300 assemblies per year

Table 2-16. ESTIMATED REQUIREMENTS FOR SPENT FUEL SHIPMENTS TO THE FIRST REPOSITORY (1996 - 2022)

Origin[b]	All shipments by rail	All shipments by truck
	SPENT FUEL	
Indiana	354	1,473
Ohio	993	8,241
Michigan	1,087	7,947
Texas	764	5,310
New Jersey	2,458	20,860
New York	1,106	9,495
Massachusetts	2,152	14,601
Minnesota	642	5,620
Iowa	384	3,309
Illinois	3,662	26,501
Wisconsin	400	3,167
Tennessee	2,071	16,367
North Carolina	1,634	13,162
Georgia	773	6,115
Florida	605	4,315
Virginia	445	3,113
Louisiana	1,011	8,177
Kansas	122	852
Southern California	841	5,875
Northern California	530	4,264
Washington	431	3,465
Total	22,465	173,229

[a]Source: Neuhauser et al (1984)
[b]Representative locations

Table 2-17. PROJECTED SPENT FUEL SHIPMENTS FROM WATER REACTORS TO THE FIRST NATIONAL REPOSITORY

Shipments Parameter	Pressurized Water Reactor	Boiling Water Reactor	Total of Both Reactor Types
Spent Fuel Assemblies:			
Total Number	102,550	141,300	243,850
Average Number/Yr	3,660	5,050	8,710
Quantity of Uranium:			
Total Metric Tons	45,240	24,760	70,000
(Tons)	(49,760)	(27,240)	(77,000)
Average Metric Tons	1,620	880	2,500
(Tons)/year	(1,780)	(970)	(2,700)
Truck Shipments (if all shipments are by truck):[a]			
Total Shipments	102,550	70,650	173,200
Average Shipments/Yr	3,660	2,520	6,180
Rail Shipments (if all shipments are by rail):[b]			
Total Shipments	14,650	7,850	22,500
Average Shipments/Yr	520	280	800

a Assumed truck cask capacity is 1 pressurized water reactor or 2 boiling water reactor fuel assemblies.

b Assumed rail cask capacity is 7 pressurized water reactor or 18 boiling water reactor fuel assemblies.

fuel shipping cask (Scenario 1) and an air-cooled spent fuel shipping cask (Scenario 2), the probability or fraction of accidents at least as severe as the conditions in the foregoing scenarios must be estimated. McClure has estimated 0.1 percent for both truck and rail as the fraction of accidents involving impact that are more severe than the regulatory test conditions as set forth in 49 CFR 173.398 (McClure, 1981). His estimate for the fraction of accidents involving fire that are more severe than the regulatory test conditons is about 0.2 percent for rail and 0.1 percent for truck. It is important to remember that the two foregoing scenarios are more severe than the regulatory test conditions: although the regulations due allow limited release of radioactive material, the scenarios in this report postulate gross release.

Figure 2-14 and 2-15 provide the basis for the preceding discussion. The cumulative probability that an accident will produce a velocity change less than a specified change is plotted against vehicle velocity change in figure 2-14. The regulatory test condition, which is to drop a cask from 9 meters (30 feet) onto an unyielding target, is represented by a vertical line drawn from the 48 kph (20 mph) velocity change. The regulations specify velocity change of the package, but the velocity changes in the figure are vehicle velocity changes, not equivalent package velocity. Since insufficient data exist to perform a detailed analysis of the velocity reduction resulting from crushing a vehicle, it is conservatively assumed in current analyses of transportation accident scenarios that the impact velocity of the vehicle is the same as the impact velocity of the cask. To understand why this assumption is conservative, it is interesting to note that in various tests performed at Sandia National Laboratory (SNL), when a vehicle hit a very hard surface head on at 98 kph (61 mph), the resulting cask impact was about 43 kph (27 mph) (Huerta, 1978). In another test, where the vehicle velocity change was 135 kph (84 mph), the resulting cask impact velocity was approximately 99 kph (62 mph).

The regulatory tests also require that the impact be made on an essentially unyielding target. Since no absolutely unyielding targets occur in nature, a derating scheme must be used to estimate equivalent velocity changes for unyielding surfaces which are labeled as moderately hard and relatively soft in figure 2-14. It can be shown, for example, that a 63 kph (39 mph) velocity change on a hard target would produce damage equivalent to the 48.2 kph (30 mph) velocity change on an unyielding target.

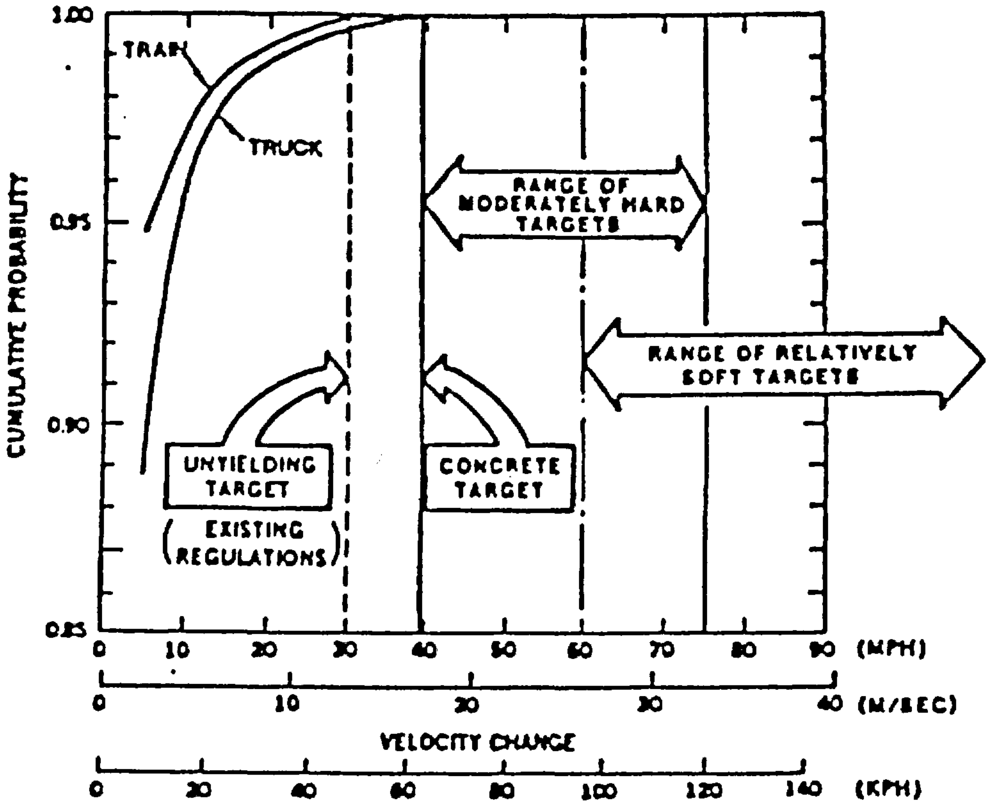


Figure 2-14. PROBABILITY THAT THE VELOCITY CHANGE IN A TRANSPORTATION ACCIDENT WILL BE LESS THAN A SPECIFIED VELOCITY CHANGE

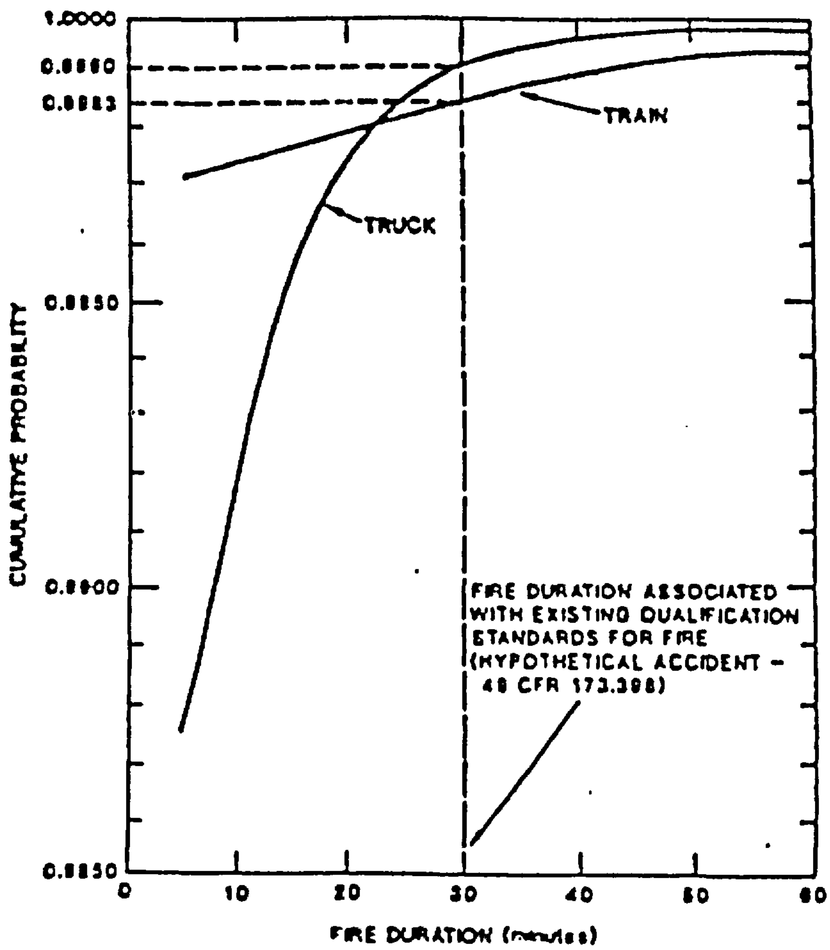


Figure 2-15. PROBABILITY THAT THE DURATION OF A FIRE IN A TRANSPORTATION ACCIDENT WILL BE LESS THAN A GIVEN DURATION

Since in a real accident environment (e.g. collision with a concrete abutment) at least a 63 kph (39 mph) velocity change would be required to exceed the equivalent regulatory test conditions.

A similar representation for the fire environment is shown in figure 2-15. Cumulative probability curves are shown for truck and rail, and the regulatory test condition is shown as the vertical line drawn from the point denoting a fire of 30-minute duration. The inherent conservatism implicit with the preceding line of demarcation is that the regulation requires complete immersion in the flame. Because the accident scenarios are more severe than the regulatory test conditions,.figure 2-15 is interpreted to mean that the accident scenarios are more severe than 99.8 percent of all train accidents and 99.9 percent of all truck accidents. This percentage is even more conservative because the accident scenarios postulate a fire with a flame at a temperature of 1000°C (1832°F).

Because most of the transportation accident scenarios will consider both fire and impact, the probabilities of both occurrences must be combined. Since the probabilties of fire and impact are assumed to be independent parameters, the fractions of accidents less severe than the scenarios involving both fire and impact are approximately 99.9998 for rail and 99.9999 for truck. The precision of such numbers can rightly be questioned because of the lack of data, especially for the more severe classes of accidents. However, at this stage in the development of risk assessment methodologies, the order of magnitude of the foregoing probabilities is deemed more important; it is in the general range of 1 in one million. That is, in every one million accidents of all severities, 1 or 2 accidents, as severe as the scenarios involving both impact and fire could be expected.

Various federal agencies; i.e. DOE, DOT, and NRC, responsbile for various aspects of the high level nuclear waste repository program have further refined the classification of transportation accident severities as illustrated typically in table 2-18 for an impact environment. The data presented in table 2-18 are referenced to the spent fuel release mechanisms associated with potential spent fuel cask accident scenarios with varying degrees of severity. Table 2-18 shows that the worst case water-cooled cask accident scenario (Scenario 1 in this presentation) is classified as a severity category V accident and the air-cooled cask accident scenario (Scenario 2 in this presentation) is classified as a severity category VI accident under the existing NRC accident severity classification format. A schematic diagram generally relating both puncture and impact speed and fire duration to the foregoing presented in figure 2-16. Additionally, a summary of the accident severity fractions as a function of the three previously defined population zones for each of the eight transportation accident severity categories is outlined in table 2-19. As evidenced from tables 2-16 and 2-17, the severity of the accident increases as the number of possible release mechanisms (or the number of branches of the fault tree) increases.

In order to associate a value for a release fraction with a specific accident scenario data similar to that previously derived for tables 2-8 through 2-12 are used. Thus, for a given accident scenario, one should find the corresponding values summarized in table 2-19. For example, Co-60 in table 2-19 has a release fraction of 0.012 for Accident Severity Category III. For more severe accident categories, the release fractions must be summed. For example, the release fraction for cesium (Cs) is 2×10^{-4} for accident severity Category V in table 2-19, which is the sum of Cs-134 and Cs-137 release fractions ($1 \times 10^{-8} + 2 \times 10^{-4} = 2 \times 10^{-4}$).

Potential Barge (Waterway) Transportation Accident Scenarios

It has been suggested that a viable means of transportation spent fuel from nuclear power plants to away-from-reactor storage sites or to permanent repositories would be to use barges on the navigable waterways encompassing the United States. Preliminary reviews have been made of the feasibility of this alternative by examining the location of reactor sites as projected to 1985 and their proximity to navigable waterways (U.S. NRC, 1977).

This analysis revealed that approximately 74 percent of the projected 1985 nuclear generating capacity will be sited within 80 kilometers (50 miles) of navigable waterways (including the ocean).

In terms of a possible high level nuclear waste repository location at the Hanford, Washington site the Columbia River waterway system would represent a possibly viable mode of transportation. However, barge shipments to the Hanford site repository location have not actually been seriously evaluated to date.

Table 2-19. SPENT FUEL CASK ACCIDENT SCENARIO RELEASE FRACTIONS FOR TRUCK AND RAIL TRANSPORT AS A FUNCTION OF ACCIDENT SEVERITY CATEGORY

	Transportation Accident Severity Category					
	1	2	3	4	5	6
Severity Fraction						
Truck						
Urban	0.604	0.395	3.8×10^{-4}	3.8×10^{-7}	2.5×10^{-7}	1.3×10^{-7}
Suburban	0.602	0.394	4.0×10^{-3}	4.0×10^{-6}	3.0×10^{-6}	2.0×10^{-6}
Rural	0.603	0.394	3.0×10^{-3}	3.0×10^{-6}	5.0×10^{-6}	7.0×10^{-6}
Rail						
Urban	0.624	0.375	3.8×10^{-4}	3.8×10^{-7}	2.5×10^{-7}	1.3×10^{-7}
Suburban	0.622	0.374	3.8×10^{-3}	4.0×10^{-6}	3.0×10^{-6}	2.0×10^{-6}
Rural	0.623	0.374	3.0×10^{-3}	3.0×10^{-6}	5.0×10^{-6}	7.0×10^{-6}
Release Fraction						
Co60	0	0	0.012	0.012	0.012	0.012
Kr	0	0	0	0.01	0.1	0.11
Cs	0	0	0	1×10^{-8}	2×10^{-4}	2.8×10^{-4}
Eu, Sr, Pu	0	0	0	1×10^{-8}	5×10^{-8}	5×10^{-8}
Ru	0	0	0	1×10^{-8}	1×10^{-6}	4.2×10^{-5}
Aerosol Fraction						
Co60	0	0	1	1	1	1
Kr	0	0	0	1	1	1
Cs	0	0	0	1	1	1
Sr, Ru, Pu, Eu	0	0	0	1	1	1
Respirable Aerosol						
Co60	0	0	0.05	0.05	0.05	0.05
Kr	0	0	0	1	1	1
Cs	0	0	0	0.05	1	1
Sr, Ru, Pu, Eu	0	0	0	0.05	0.05	0.05

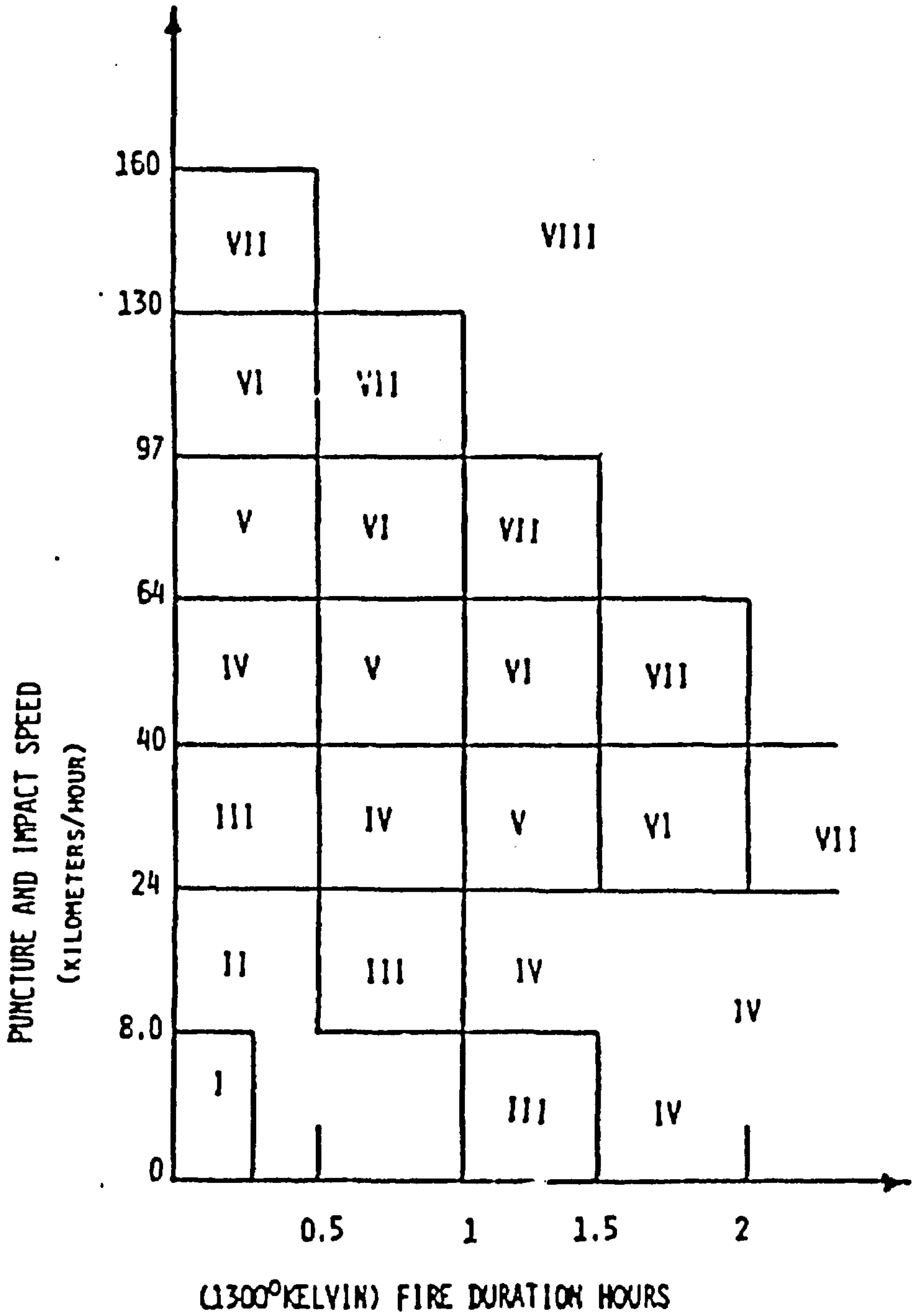


Figure 2-16. ACCIDENT SEVERITY CATEGORY CLASSIFICATION SCHEME FOR TRUCK AND RAIL TRANSPORT

For example, records for the calendar year 1973 for domestic waterborne traffic show a total of 6.67×10^{11} ton-miles. Precise data are not available to indicate what fraction of those ton-miles was barge traffic; however, a reasonable estimate seems to be 1.73×10^{11} ton-miles of barge traffic. According to the U. S. Coast Guards annual statistics of casualties, there was an estimated 1395 barge accidents in 1973, of which about 60 percent involved cargo bargoes.

However, the available data cannot be analyzed in the same way as the data for rail or truck transport. On the basis of available data, it is estimated that the average net cargo weight of a typical barge is about 1200 tons, leading to a total number of barge miles of about 1.44×10^{8}. This yields an accident rate of about 6 accidents per million barge kilometers.

Very little data are available on the severity of accidents involving barges. Since barges travel only a few miles per hour, the velocity of impacts in accidents is small. However, because of the large mass of the vehicle and cargo, large forces could be encountered by packages, for instance, spent fuel casks aboard barges. A forward barge could impact on a bridge pier and suffer crushing forces as other barges are pushed into it. A coastal or river ship also could knife into a barge. Fires could result in either case. An extreme accident, i.e., an extreme impact plus a long fire, is considered to be of such low probability that it is in not considered a design-basis accident. The likelihood of a long fire in barge accidents is small because of the availability of water at all times. Also, since casks could be kept cool sprays or submergence in water, there is compensation for loss of mechanical cooling.

The likelihood of cargo damage occurring in barge accidents is estimated to be much less than in the cases of truck or rail accidents on the basis of the accident severity breakdown for ship and barge shown in table 2-20.

If a cask were accidentally dropped into water during barge transport, it is unlikely that it would be adversely affected unless the water was very deep. Most fuel is loaded into casks under water, so immersion would have no immediate effects. The water would remove the heat, so overheating would not occur. Each cask is required by NRC regulations (10 CFR § 71.32 (b)) to be designed to withstand an external pressure equal to the water pressure at a depth of 15 m (50 ft.), and most designs will withstand external pressure at much greater depths. If a cask seal were to fail due to excessive

Table 2-20. FRACTIONAL OCCURRENCES* FOR SHIP AND BARGE ACCIDENTS BY SEVERITY CATEGORY AND POPULATION DENSITY ZONE

Accident Severity Category**	Fractional Occurrences**	Accident Severity Category	Fractional Occurrences (this assessment)	Fractional Occurrences According to population density zone: Low	Medium	High
minor - 2	.897	I	.897	0	.5	.5
minor - 3	.0794	II	.0798	0	.5	.5
moderate - 2	.00044					
moderate - 3	.00113	III	.00113	0	.9	.1
moderate - 4	.0186	IV	.0186	0	.9	.1
severe - 2	.0000052	V	.0000052	.1	.9	0
severe - 3	.000072	VI	.000072	.1	.9	0
severe - 4	.000195	VII	.000195	.1	.9	0
extra severe - 1	.000013	VIII	.000013	.1	.9	0

* Overall accident rate = 6.06×10^{-6} accidents/kilometer

** NUREG-0056, September, 1976

pressure in deep water, only the small amount of radioactivity in the cask coolant and gases from perforated elements in the cask cavity would be likely to be released. Even if the cask shielding were ruptured as a result of excessive pressure, the direct radiation would be shielded by the water. About 10 m of water, which is the depth of most storage pools, would be ample shielding for radiation, even from fully exposed fuel elements.

A previous study has concluded that the pressure seals on a spent fuel cask that is dropped into the ocean might begin to fail at a depth of 200 meters (360 feet) a typical depth at the edge of the continental shelf, and release contaminated coolant. (Heaberlin et al, 1976). The fuel elements, which contain most of the radioactive material, provide excellent containment. In an operating reactor, the fuel elements are under the water at elevated temperatures and at pressures on the order of 1000 to 2000 psi. Thus exposure to water pressures at depths of 600 to 1200 m should have no substantial effect on the fuel elements themselves. The study concluded that they would not fail until they reached a depth of approximately 3000 meters. Once they failed, the fuel pins wold release fission products into the ocean, but these would be dispersed into such a large volume of the ocean that the concentrations would be very small. Certain nuclides such as cesium and plutonium could be reconcentrated through the food chain to fish and invertebrates that could be eaten by man; but, as pointed out in the study, the possiblities of a single person consuming large quantities of seafood, all of which was harvested from the immediate vicinity of the release, is very remote, especially since most seafood is harvested in areas over the continental shelves.

In virtually all cases, except those in which the cask was submerged to extreme depths, recovery would be possible with normal salvage equipment. If the cask and elements could not be recovered, corrosion could open limited numbers of weld areas within about 2000 years with possible localized failures occurring sooner. However, by that time most of the radioactivity would have decayed. Subsequent release would be gradual, and the total amount of radioactivity released at any one time and over the total period would be relatively small. Considering the extremely low probability of occurrence, the major reduction in radioactivity due to radioactive decay, and the dilution that would be available, there would be little environmental impact from single events of this kind.

Should a shipment be accidentally dropped during transfer to a barge, the main effect will likely be limited to that of rather severe damage to the barge. It is possible that a fuel cask could penetrate the barge decks and fall into the relatively shallow water of the breakwater basin. As previously discussed, there would be at most only minor radiological consequences, unless diffculty is experienced in recovering the cask or casks after a rather severe cask breach due to impact and/or fire prior to the immersion in water.

Waterborne traffic spends a very small fraction of its travel in high-population-density regions. The highest traffic density will probably occur in the port areas and, as a result, be associated with lower speed. .Categories VI, VII, and VIII accidents probably require relatively large forces, a long-term fire, or an explosion, which are more likely to occur in open water. Categories III through V are more likely to be the result of a lower speed collision in a dock area, either with another vessel or a pier. The population density of dock areas of most cities was considered to be representative of a medium-population zone. Hence, Class III-V accidents are assumed to occur in a medium-population zone. Categories I and II accidents are not likely to involve another vessel, since they are very minor in nature. Hence, they are considered to occur either in open waters or while securely moored. These assumptions are reflected in table 2-20.

RADTRAN II Risk Assessment Computer Model

The RADTRAN II computer model described in conjunction with radiological risk assessment involving normal or routine shipments of high level nuclear waste section 2.1.1 can also be utilized to evaluate radiological risk from transportation accident release scenarios as illustrated by the flow chart in figure 2-17. Although figure 2-17 implies that RADTRAN II accomodates atmospheric dispersion to the natural environment from the point of contaminant release from a transportation accident scenario such is not the case. Airborne material disperses from the accident site as a function of the prevailing meteorological conditions. Generally, these conditions can be described in terms of time-integrated atmospheric dilution factors (Curies-sec/m^3) as a function of area within an isopleth contour on which it applies. In RADTRAN II the user must specify a set of integrated concentration values and corresponding areas which have been computed assuming a totally reflective lower boundary. The code then calculates a set of airborne concentration and deposition contours out to a maximum area of $10^9 m^2$. Thus, in most practical situations the analyst must utilize an atmospheric dispersion model to develop the contaminant dispersion characteristics of the contaminant release in any event. However, the RADTRAN II model provides a very effective method for quantifying the release of specific radionuclides to the environment (source term) once

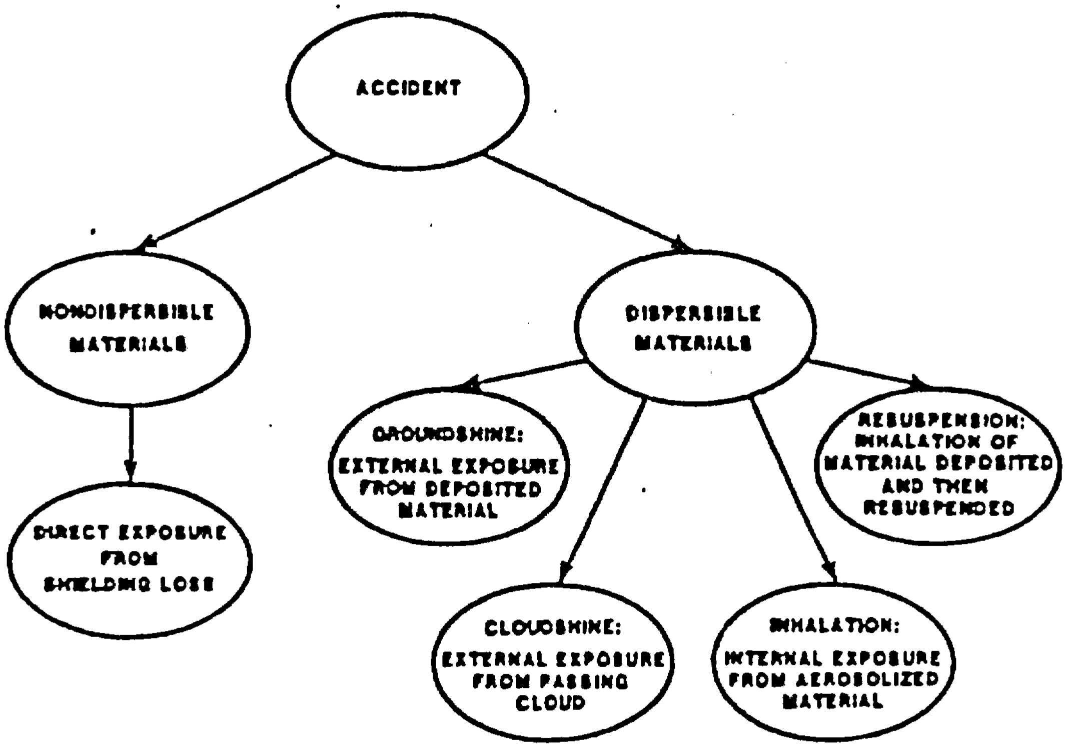


Figure 2-17. RADTRAN II COMPUTER MODEL FLOW CHART: TRANSPORTATION ACCIDENT RELEASE-ATMOSPHERIC DISPERSION

the mechanisms for contaminant release in an accident scenarios have been established by means of fault tree analysis as previously described. RADTRAN II also has the capability to provide an estimate of human health effect from a transportation accident release to the atmosphere which will be discussed in greater detail in a subsequent section of the report. As previously mentioned, RADTRAN II will not accommodate the analysis of a water immersion accident scenario. Since many of the proposed transportation routes for high level nuclear waste shipments pass along major waterways and barge shipments still remain a possibility, this omission in the code must be considered a major deficiency in terms of the CTUIR program to develop risk assessment methodologies for evaluation of transportation accident scenarios involving high level nuclear waste shipments through tribal lands.

2.2 RELEASE SCENARIOS FOR HIGH LEVEL NUCLEAR WASTE REPOSITORY OPERATION

2.2.1 Synopsis of Geologic Repository Background Information

A number of means of disposal of solidified high-level nuclear wastes have been proposed over the last 25-30 years. The most developed of these is emplacement in mined cavities, called repositories, deep in the earth. Because such repositories are capable of performing so well in the opinion of DOE that decided in 1981 that mined repositories should receive primary emphasis in the national program, although some research on other technologies is continuing. The passage of NWPA in 1982 by Congress affirmed DOE's decision.

Basalts beneath the Hanford Site near Richland, Washington are being studied by DOE and its contractors for the first terminal repository for high-level nuclear waste. The Hanford candidate site merits consideration because (1) it is federally owend and dedicated to nuclear purposes and (2) geologic studies completed to date indicate that the site is underlain by a thick sequence of basalt flows, several of which have dense, thick interoirs with low porosities and permeability.

The nearly horizontal strategraphic layering of basalt flows in the candidate site area acts as an important hydrologic control on the site's high level nuclear waste isolation capability. Interbedded volcaniclastic sediments and flow-top breccias in the upper two of the three formations comprising the Columbia River Basalt Group in the Pasco Basin commonly contain significant quantities of groundwater and are considered to be aquifers. Deeper Columbia River basalts contain relatively less abundant groundwater, but significant water-bearing zones locally may be present. The proposed underground facility currently is anticipated to consist of five drilled shafts accessing underground chambers at an expected depth of about 1,000 meters (3,000 ft).

The lavas underlying the candidate site comprise part of the Columbia Plateau flood-basalt province as shown in figure 2-18. The province has an area of approximately 200,000 km^2 and is estimated to contain on the order of 200,000 km^3 of tholeiitic basalts. Individual flows commonly are laterally extensive and may range upwards of 100 meters (300 ft) in thickness. The Pasco Basin, in the south-central part of the Columbia Plateau, as shown in figure 2-18, occupies about 5,180 km^2 (2,000 mi^2) and contains the DOE Hanford Site. Columbia River basalts within the Pasco Basin are at least 1,460 meters (4,800 ft) thick and in most of the basin are overlain by glacio-fluvial, fluvial-lacustrine, and aeolian sediments. Volcanic sediments locally are interbedded between basalt flows, particularly in the upper part of the basalt section as illustrated in figure 2-19.

The Cold Creek syncline is located in the southern and southwestern part of the Pasco Basin and contains the candidate repository site as shown previously in figure 2-18. The syncline is a topographic and structural basin that is bounded by the Umtanum Ridge-Gable Mountain anticline to the north and by the Yakima Ridge-Rattlesnake Mountain anticline to the south. Two subtle depressions are present along the northwest trending hinge line of the syncline: the Cold Creek Valley depression and the Wye Barricade depression. The candidate site is located within the Cold Creek Valley depression, where the Columbia River basalts are within a few degrees of horizontal.

The Columbia River Basalt Group is the youngest assemblage of tholeiitic flood basalts known. It has been dated radiometrically as ranging from 6 to 16.5 million years old (Watkins and Bakse, 1974, McKee et al, 1977), but more than 99% of the basalt was erupted during a 2.5 to 3 million year interval beginning approximately 16 million years ago (Swanson and Wright, 1978). The basalts were erupted from vents, now exposed as north-trending dikes, in the southeastern part of the Columbia Plateau. The Columbia

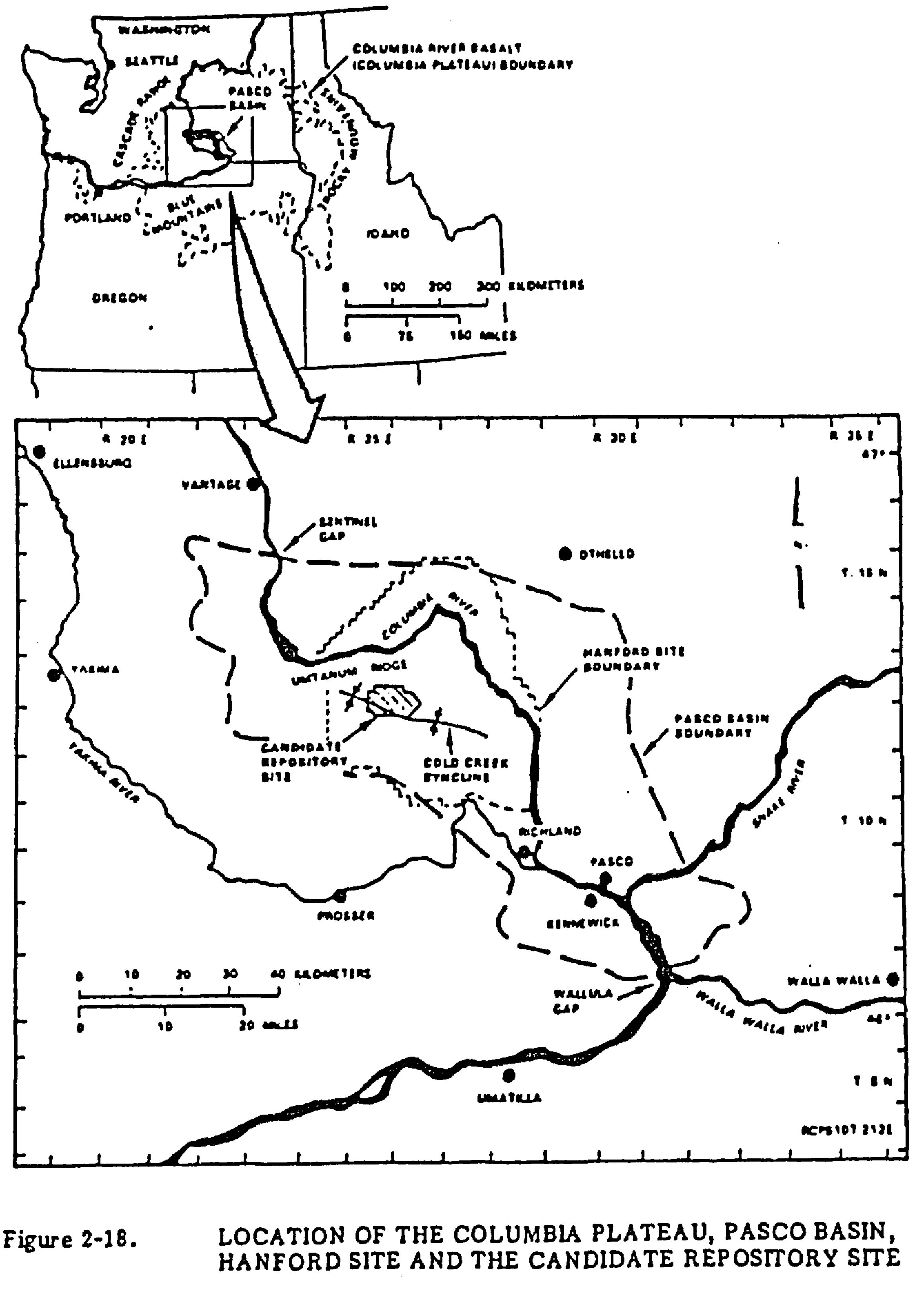


Figure 2-18. LOCATION OF THE COLUMBIA PLATEAU, PASCO BASIN, HANFORD SITE AND THE CANDIDATE REPOSITORY SITE

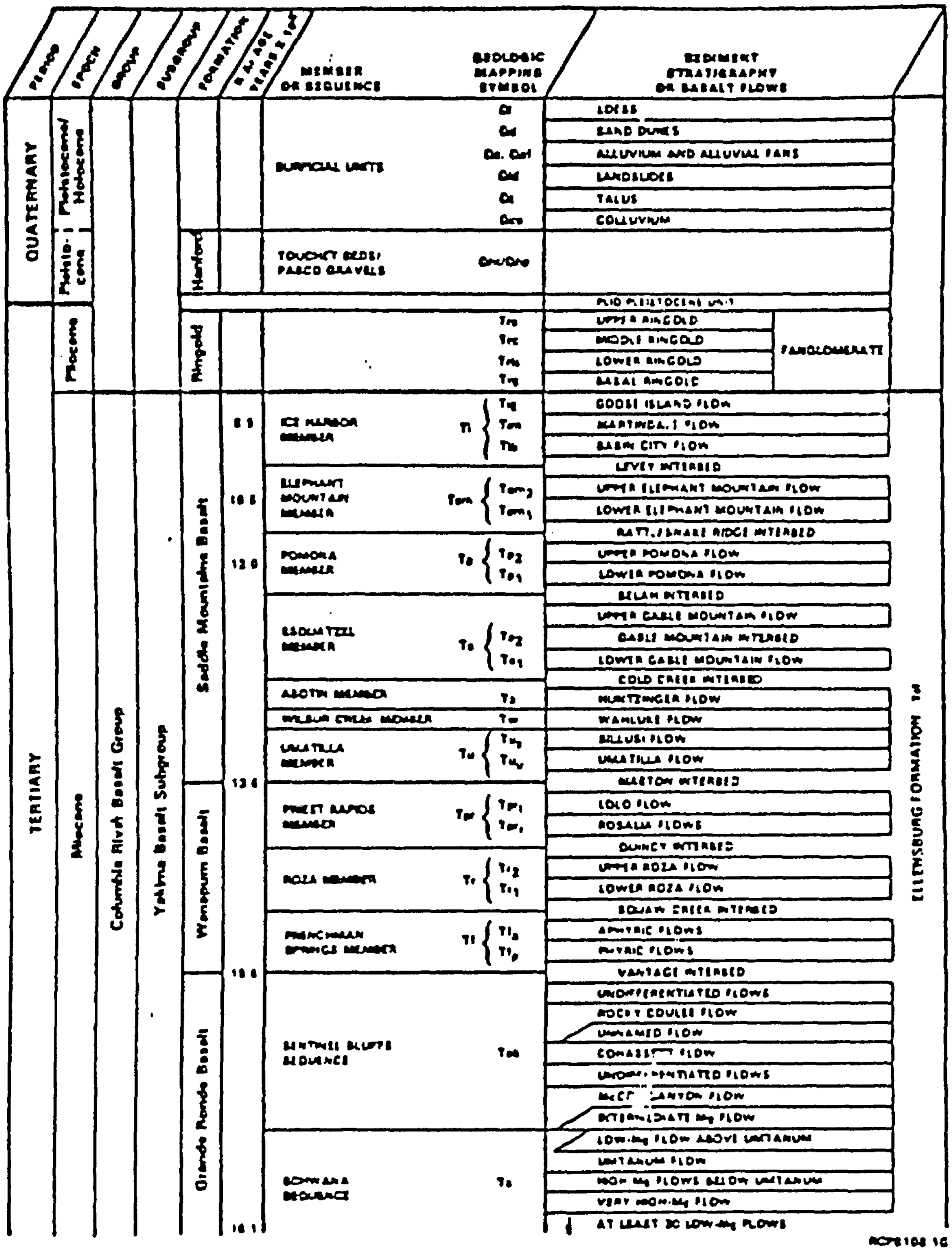


Figure 2-19. STRATIGRAPHY OF THE PASCO BASIN

River basalts have been subdivided into five formations, three of which are present in the Pasco Basin. The two oldest formations, Imnaha and the Picture Gorge basalts, are present at the surface only at the southeastern and southern margins, respectively, of the Columbia Plateau. The three younger formations, the Grande Ronde, Wanapum and Saddle Mountains, are present within the Pasco Basin as shown in figure 2-19.

In the Pasco Basin, as elsewhere in the Columbia Plateau, Grande Ronde basalts are the most voluminous and areally extensive formation of the group. Although their thickness varies as a consequence of the buried topography onto which it was erupted and subsequently eroded, it is known to exceed 1,000 meters (3,000 ft) in the Pasco Basin. The formation probably consists of hundreds to thousands of individual flows. Within the Cold Creek syncline the more than 1,000 meters of Grande Ronde basalts consist of at least 50 flows that average from 4 to 150 meters thick. The top of the Grande Ronde basalts typically is distinguished by a zone of weathering or a thin bed of volcaniclasitc sediment. Grande Ronde basalts are exposed at the margins of the Pasco Basin in the Sentinel Gap. Wallula Gap, and Umtanum Ridge areas previously depicted in figure 2-18.

The Grande Ronde basalts conformably are overlain by flows of the Saddle Mountains Basalt, the youngest formation of the Columbia River Basalt Group. The Wanapum Formation is the second-most voluminous of the formations of the Columbia River Basalt Group. Wanapum basalts define the surface of much of the Columbia Plateau. Compared to the underlying Grande Ronde flows, Wanapum basalts have a relatively high ferrous oxide (FeO) and titanium dioxide (TiO_2) content. Saddle Mountain basalts comprising less than 1% of the Columbia River Basalt Group, are characterized by the greatest chemical petrographic, and paleomagnetic variability of any formation of the Columbia River Basalt Group. Additionally, volcaniclastic sediments of the Ellensburg Formation commonly are interbedded with Saddle Mountain flows, contract to their lessor abundance in the underlying basalt formations Saddle Mountain basalts contain a number of major water-bearing horizons. Wanapum and Saddle Mountains basalts, within the Cold Creek syncline, are composed of as many as 20 flows, with a total thickness of about 700 meters (2300 ft) overlying the Columbia River basalts in the Cold Creek syncline are up to 220 meters (720 ft) of fluvial-lacustrine sediments.

2.2.1.1 Intraflow Features of the Candidate Basalts Several geologic features of the basalt candidate repository site influence groundwater flow and, hence, control the most likely means of radionuclide transport to the accessible environment. Principal among these are features, of primary geologic origin internal to the individual basalt flows. Effects of intraflow characteristics on candidate horizon suitability are important considerations for repository construction and for assessment of waste isolation performance.

The most significant differences among candidate flows are depth, thickness of intraflow zones, and the character and predictability of intraflow structures, especially flow tops.

Tops of Columbia Plateau flood basalts typically consist of a vesiculated and/or flow-brecciated crust. The crustal zones commonly grade downward into a vesicular zone that, in turn, grades into the massive, nonvesiculated interior of the flow. This dense flow interior generally consists of two parts, basal colonnade and central entablature, that are distinquished principally by differing cooling joint patterns and petrographic features. A basal colonnade consists of relatively well-formed, hexagonal columns bounded by cooling joints and is generally less fractured than the overlying entablature. The basal part of the colonnade commonly is a thin (about 0.5 meters) zone of fractural, glassy basalt. In some flows, however, the basal part of the colonnade may consist of a thick, pillowed zone occupying as much as half of the total flow thickness. The central entablature, overlying the basal colonnade, typically is comprised of irregularly to regularly jointed basalt broken by cooling joints into smaller columns than those that characterize the colonnade.

Although all potential candidate flows have not yet been identified, thickness of the principal intraflow features of candidate repository-construction horizons preliminary identified are briefly summarized in table 2-21. The nature of the internal characteristics of the candidate flows currently is known primarily from outcrops and drill core observations. However, because the internal structures of plateau flood basalts commonly change laterally, large scale subsurface explorations within the candidate repository site are planned to reduce predictive uncertainty of intraflow characteristics.

Umtanum Flow

Within the Hanford Site the Umtanum flow is near the top of the Schwana sequence (below the Sentinel Bluffs sequence) of the Grande Ronde Basalt. The flow is the deepest candidate flow and is about 900 meters (2,950 ft) below mean sea level (MSL) within the

Table 2-21. THICKNESS OF FLOW TOPS AND INTERIORS OF THREE OF THE CANDIDATE REPOSITORY HORIZONS THUS FAR IDENTIFIED[a]

Description	Borehole RRL-2 (m)	Borehole RRL-6 (m)	Borehole RRL-14 (m)	(Mean Value) (m)
Cohassett				
Flow top	1.8	3.0	10.4	4.9
Flow interior above vesicular zone	29.3	22.3	23.5	25.0
Laterally extensive vesicular zone within the flow interior	3.7	4.6	3.0	3.8
Flow interior below vesicular zone	45.1	46.6	35.9	42.5
McCoy Canyon				
Flow top	5.5	12.2	11.7	9.8
Flow interior[b]	34.8	30.5	33.2	32.8
Umtanum				
Flow top	45.1	28.5	20.9	31.5
Flow interior	25.5	41.7	39.6	35.6

[a] Thickness of interiors of the Cohassett, McCoy Canyon, and Umtanum flows within the candidate repository site. Data for boreholes RRL-2, -6, and -14 depict discrete measurements. Mean values are also given.

[b] Discontinuous zones of vesiculation occur within the interior of the McCoy Canyon flow.

candidate repository site. Within the Hanford Site, the flow thins markedly to the west and northeast of the candidate repository site and is relatively consistent in overall thickness within the central Pasco Basin. Thinning of the flow to the north of the Hanford Site is believed to reflect constructional mechanisms of flow emplacement, rather than the presence of a structural topographic feature at the time of emplacement.

Within the candidate repository site, total flow thickness is on the order of 70 meters (230 ft).

McCoy Canyon Flow

The McCoy Canyon flow is near the base of the Sentinel Bluffs sequence. Within the candidate repository site, the McCoy Canyon flow overlies the Umtanum flow. The flow is approximately 850 meters (2,790 ft) below MSL within the candidate repository site, or about 150 meters (490 ft) beneath the Cohasset flow. Total flow thickness is approximately 45 meters (150 ft) in the candidate repository siting area. The flow thins from northwest to southeast across the Hanford Site, diminishing in thickness from greater than 50 meters (165 ft) to less than 23 meters (75 ft). The thickness variation is believed to have been controlled by paleotopography of the surface upon which the flow was emplaced. The major controlling factor of the paleosurface may have been either structural deformation after emplacement of the Umtanum flow, or constructional, resulting directly from emplacement of the Umtanum flow.

Cohasset Flow

The Cohasset basalt flow is approximately 700 meters (2,300 ft) below MSL within the candidate repository site. The flow stratigraphically is near the center of the Sentinel Bluffs sequence, uppermost of the Grande Ronde basalt sequences. The flow is approximately 75 meters (250 ft) thick within the candidate repository site and is known to thin toward the southeastern edge of the Hanford Site to less than 50 meters (165 ft). Such thinning is thought to result from the mechanics of flow emplacement, rather than from thinning due to structural control of paleotopography. Within the interior of the Cohasset flow (about 30 meters from the flow top), an isolated zone characterized by scattered vesicles averaging 0.5 cm to several centimeters in diameters aids in stratigraphic identification of the flow. Flow brecciation is not present in this vesiculated zone and cooling joints pass through it unabated.

The flow-top portion of the Cohasset flow is relatively thin and apparently is more laterally consistent in thickness than is the flow top of the Umtanum flow. The interior zone of the Cohasset flow within the candidate repository site appears everywhere to be greater than 63 meters (210 ft) compared to 30 meters and 25 meters for the interiors of the McCoy Canyon and Umtanum flows, respectively. Tiering of colonnades and entablatures of the interior of the Cohasset flow is common. Fracture abundance within the interior portion of the Cohasset flow is similar to that of the McCoy Canyon and Umtanum flows, but because of colonnade-entablature tiering, it is less predictable than in the Umtanum flow. Fracture width within the interior of the Cohasset flow appears, on the basis of limited, current data, to exceed that of the Umtanum flow.

Structure

Columbia River basalts in the western and the central parts of the Columbia Plateau have been folded into asymmentrical, west- to northwest-trending linear anticlines separated by broad, interviewing synclines. This portion of the plateau has been termed the Yakima Fold Belt subprovince. Known faults associated with the anticlinal fold axes were probably developed at about the same time as the folding. Distribution and thickness variations of basalt flows in the Pasco Basin and its bounding anticlines suggest that the Pasco Basin and the Cold Creek syncline were actively subsidiary by at least the late Grande Ronde time period relative to the Saddle Mountains and Rattlesnake Mountain anticlines. Rates of uplift and subsidence during the period of the late Grande Ronde time period through the Saddle Mountains time period are estimated on the basis of structural and stratigraphic mapping to be less than 40 million years. The steeply dipping flows exposed on anticlinal limbs contain the most extensive tectonic brecciation and faulting, with relatively intact basalt present in the intervening broad synclines.

Tectonic breccia zones have, however, also been identified from drill cores of the Columbia River basalts within the Cold Creek syncline locally exhibiting disclike fracturing. This suggests suggesting that the basalts are anisotripically stressed, with significantly (by a factor of two) greater horizontal than vertical stress. A rate of tectonic shortening across the Pasco Basin of about 0.4 to 0.04 millimeters per year in a north-south direction is suggested by seismic and geodetic data.

2.2.2 Preliminary Repository Characteristics

The function of a high-level nuclear waste repository at the proposed location on the Hanford Site would be to isolate the waste from the bioshpere for a time interval of at least 10,000 years so that it does not present a significant hazard to public health and safety. As currently envisioned, the repository will be designed for a specific waste disposal capacity of spent fuel or its equivalent as high-level waste, not to exceed 70,000 metric tons (77,000 short tons) of heavy metal prior to the startup of a second repository.

The present geologic repository conceptual designs at Hanford Site, shown as a generalized schematic in figure 2-20, has evolved since 1982 (RKE/PR, 1983). Design (RKE/PB, 1983) predates the adoption of the 70,000 metric ton (77,000 short ton) design capacity, which was based on an ultimate capacity to dispose of 47,400 metric tons (52,140 tons) of spent fuel and commercial high level equivalent. The basic components of the repository include the surface facilities, the access shafts and the repository horizon consisting of the shaft pillar area, the shafts, and underground drifts.

Most of the surface facilities described in the 1982 Conceptual Design (RKE/PB, 1983), as illustrated in figure 2-21, would be located in the central process area. The central process area would include all access to the surface and subsurface facilities. A general layout plan is shown in figure 2-22.

Figure 2-23 illustrates the generic, sequential emplacement process for spent fuel or commercial high-level waste as envisioned for the 1982 Conceptual Design. The shipping cask arrives at the repository by truck or rail and is moved into the surface waste handling facility shown conceptually in figure 2-24.

The waste container would be removed from the shipping cask and then moved to the primary hot cell to be inspected and repaired if required. The containers would be loaded into the waste transport shaft cage and transported down the shaft to the shaft pillar area, shown in figure 2-25. Waste containers would be transported from the shaft pillar area via the main entries to the emplacement boreholes. The containers would be left in the emplacement borehole for retrieval, if necessary, since the present repository design criteria provide the option of waste retrieval for up to 50 years after the initial

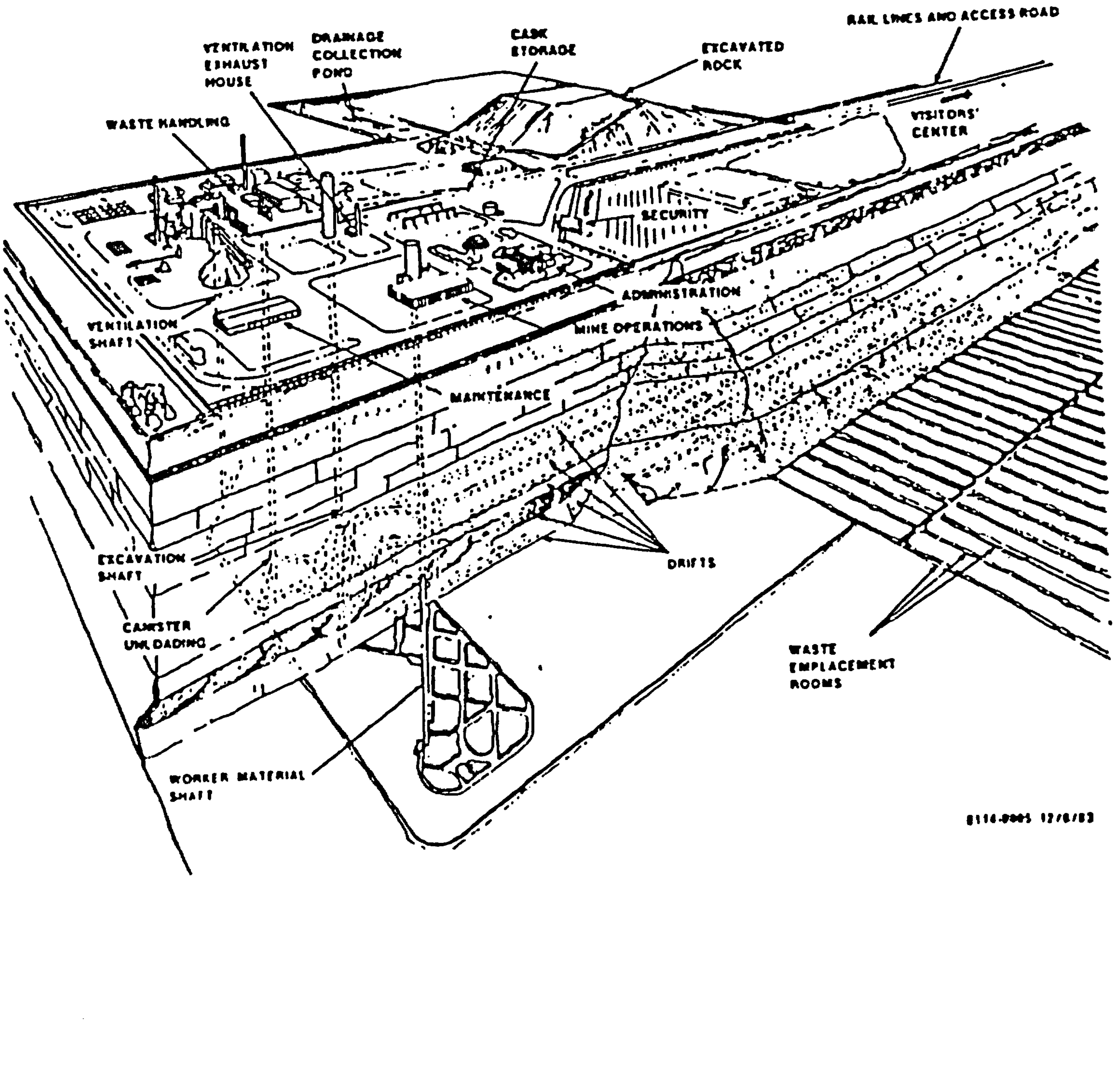


Figure 2-20. GEOLOGIC REPOSITORY-GENERAL SCHEMATIC OF SURFACE AND UNDERGROUND FACILITIES

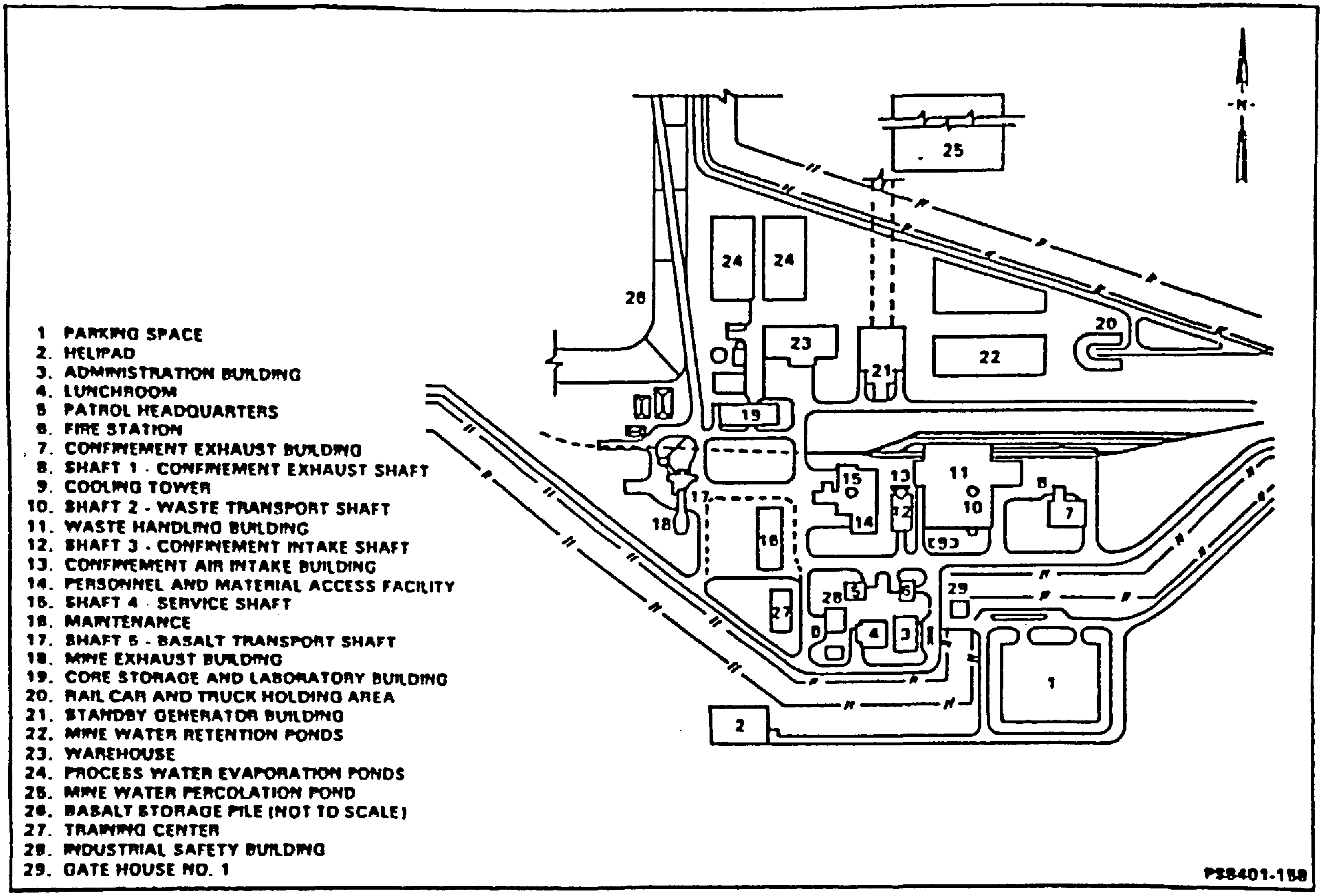


Figure 2-21. THE 1982 CONCEPTUAL DESIGN SURFACE FACILITIES PLAN—CENTRAL PROCESS AREA (RKE/PB, 1983)

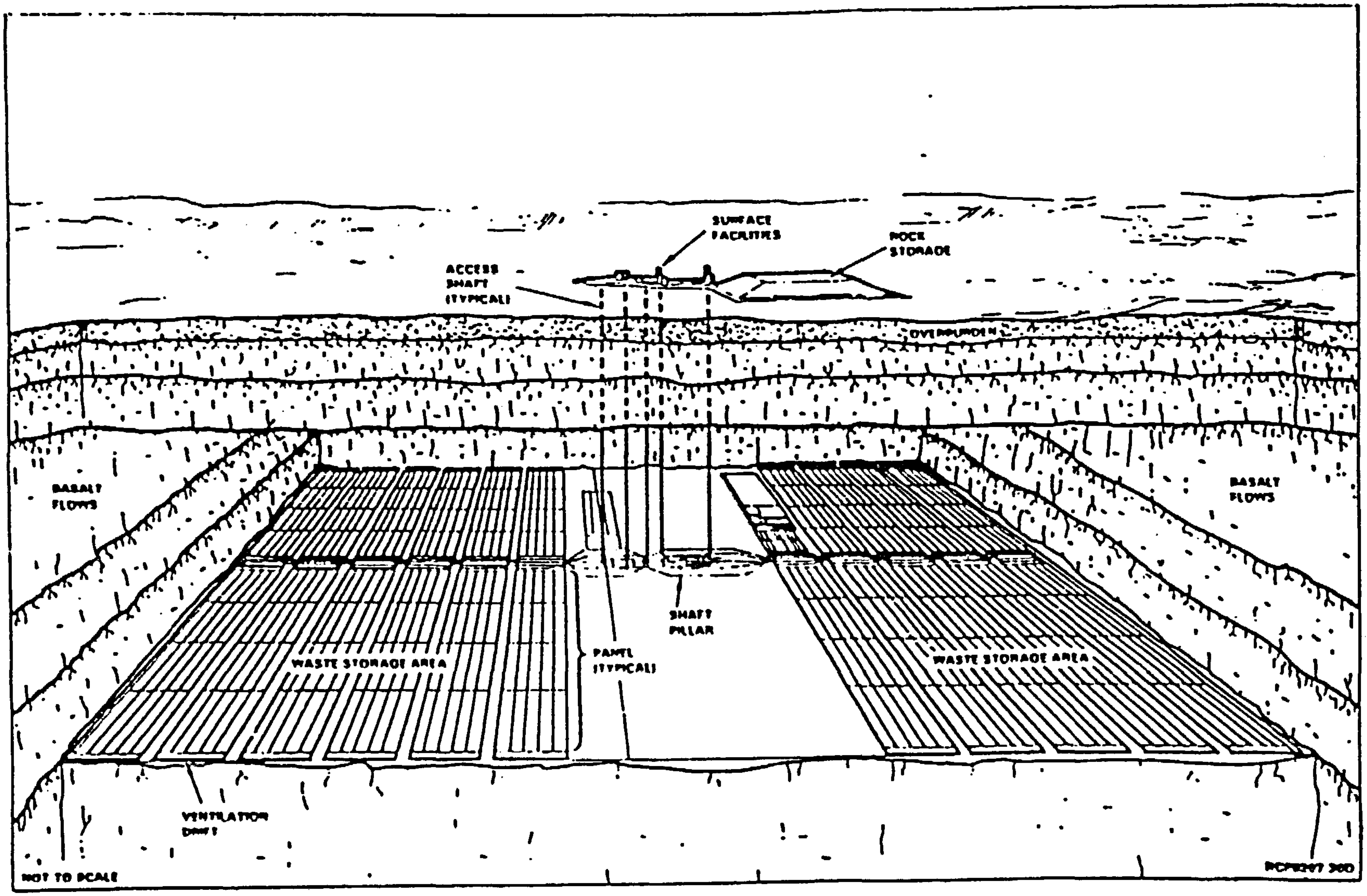


Figure 2-22. GENERAL SCHEMATIC OF REFERENCE REPOSITORY UNDERGROUND LAYOUT-1982 CONCEPTUAL DESIGN (RKE/PB, 1983)

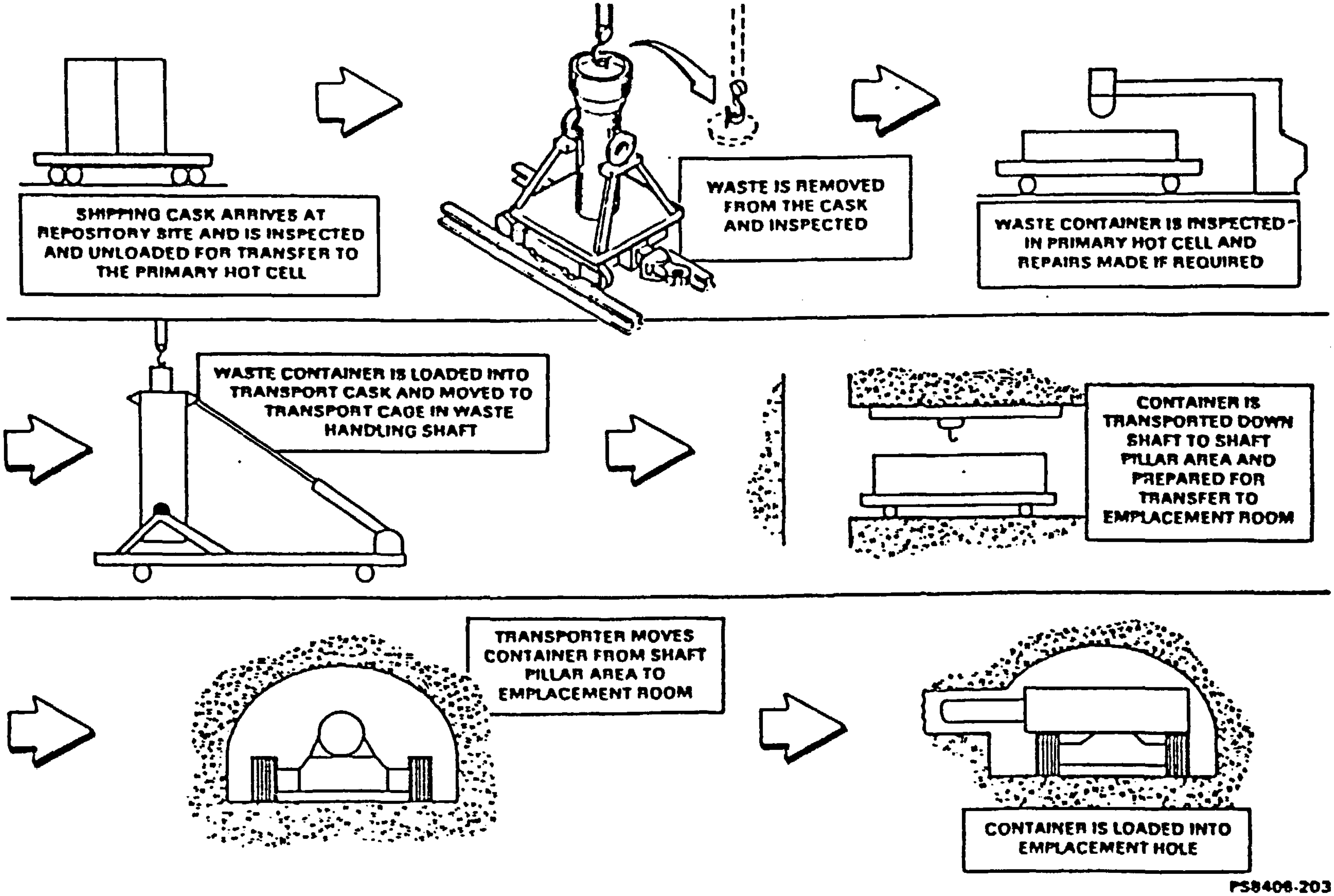


Figure 2-23. THE 1982 CONCEPTUAL DESIGN GENERALIZED EMPLACEMENT PROCESS (RKE/PB, 1983)

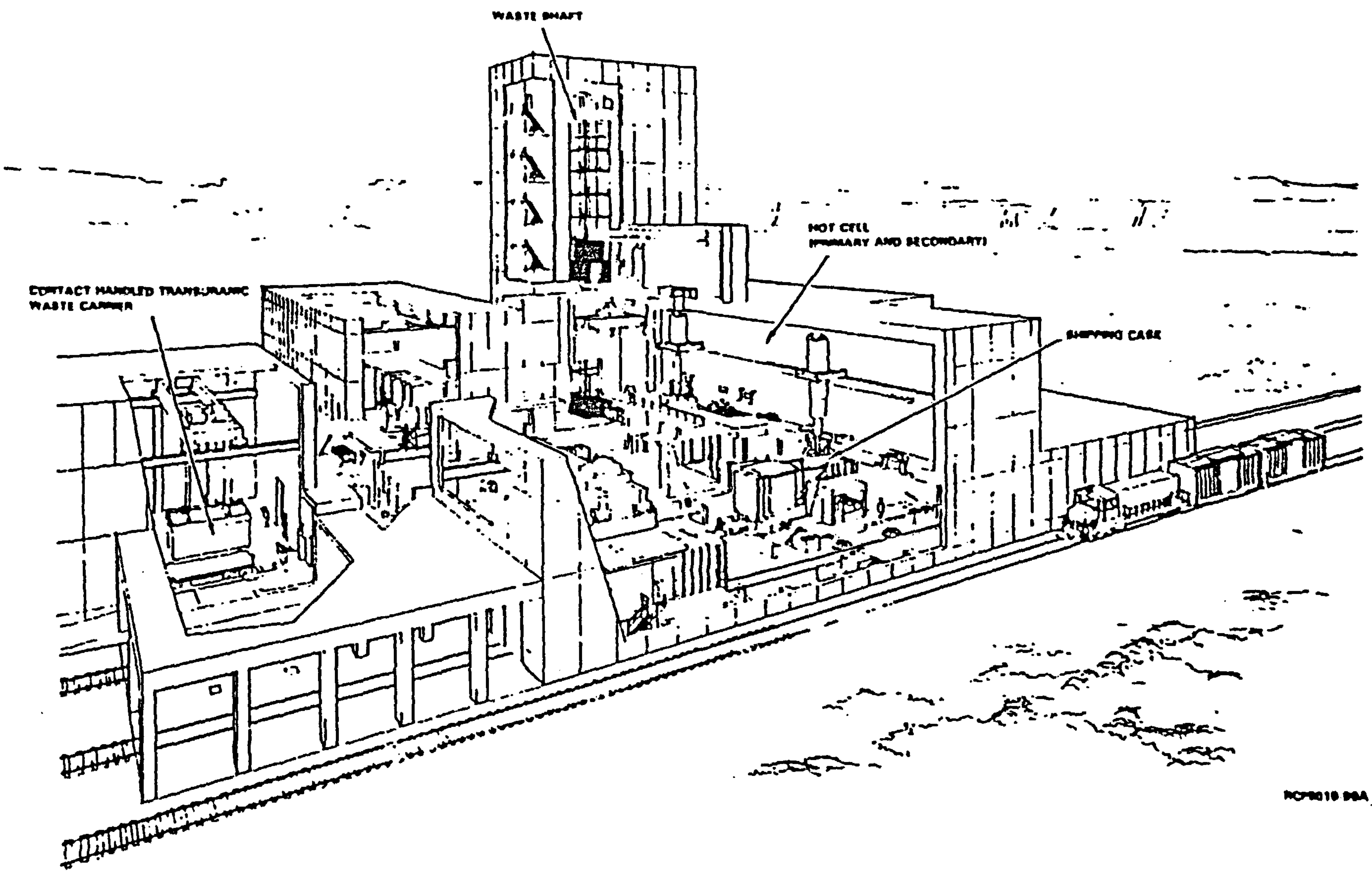


Figure 2-24. THE 1982 CONCEPTUAL DESIGN WASTE HANDLING FACILITY (RKE/PB, 1983)

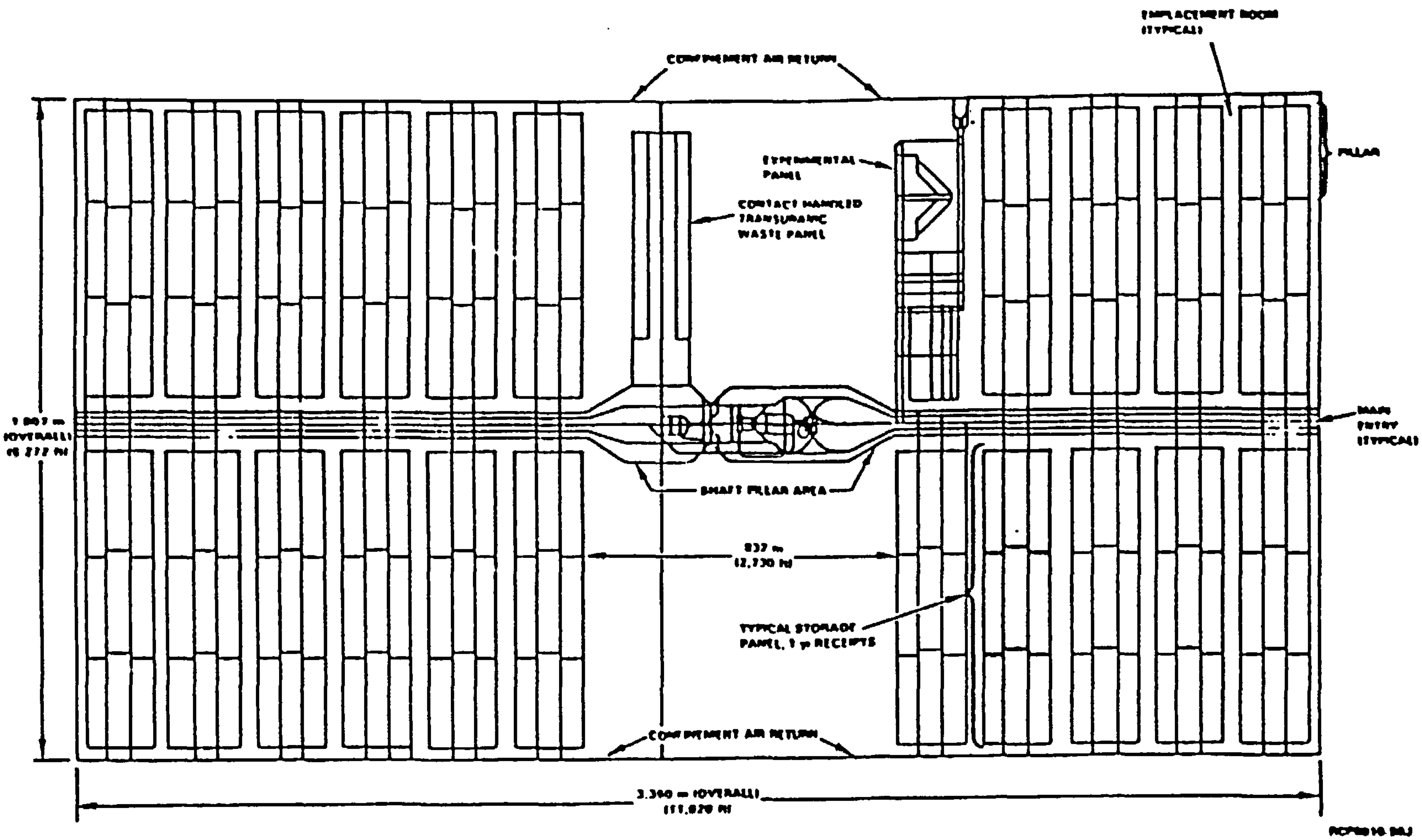


Figure 2-25. THE 1982 CONCEPTUAL DESIGN UNDERGROUND FACILITIES LAYOUT (RKE/PB, 1983)

emplacement (DOE, Draft Environmental Assessment, 1984). Following the retrieval period, the emplacement holes would be packed and the underground openings backfilled with an engineered fill material of low permeability.

Table 2-22 presents the waste emplacement features for the current design concept (DOE, Draft Environmental Assessment, 1984) as compared to the 1982 Conceptual Design (RKE/PB, 1983). The current container data and packing requirements resulted in part from the Alternate Waste Package Study (Westinghouse, 1984). The emplacement borehole length and number of containers per emplacement borehole were evaluated in the Waste Emplacement Optimization Study (RKE/PB, 1984) and are included in the current concept available at the time of this study activity. The emplacement borehole diameter is derived from container and packing requirements. The values shown in Table 2-22 consider symmetrical storage (i.e., two storage boreholes directly across from each other) for both the 1982 Conceptual and the Repository Underground Layout Study (RKE/PB, 1984). The current, short borehole emplacement concept provides greater confidence for emplacement and retrieval as compared to multiple container storage per borehole.

Table 2-23 compares the ventilation requirements for the 1982 Conceptual Design and the more recent Repository Underground Layout Study. Other than the decrease in ambient rock temperature, the requirements have not changed significantly. The increase in total waste storage capacity, and the increase in the required emplacement room length due to single-container storage contributes significantly to the total ventilation capacity requirements, however.

The physical dimensions for the Repository Underground Layout Study (RKE/PB, 1984), on which the current design concept is based are presented in table 2-24. The repository length-to-width ratio has changed slightly from the 1982 Conceptual Design, but the areal extent has increased linearly with the increase in storage capacity-by approximately 50 percent. The more recent emplacement concept with single container storage and less pitch between boreholes has yielded at 470 percent increase in the length of emplacement room per container stored as shown in table 2-22. However, this is partially offset by a pillar-width reduction from 65 meters (212 feet) to 30 meters (100 feet) as shown in table 2-24. Additionally, the 50 percent increase in storage capacity has been the primary factor in the 2.50 percent increase in ventilation air requirements as also illustrated in table 2-24.

Table 2-22. DESIGN COMPARISONS OF WASTE EMPLACEMENT FEATURES[a]

Description	1982 Conceptual Design	Current Design Concept
Container heat generation (pressurized water reactor), W	1,650	2,020
Container OD, cm (in.)	41.7 (16.4)	50.3 (19.8)
Container length, cm (in.)	411 (162)	411 (162)
Packing method[b]	Pneumatic	Prepackaged sections
Packing thickness, cm (in.)	15.2 (6)	15.2 (6)
Emplacement borehole dia. cm (in.)	76.2 (30)	89.0 (35)
Emplacement borehole pitch (center-to-center spacing), m (ft)	18.3 (60)	6.7 (22)
Emplacement borehole length, m (ft)	61 (200)	6.1 (20)
Containers of pressurized water reactor spent fuel/emplacement borehole	13	1
Meter of emplacement room/ container	0.7	3.3

[a] DOE, Draft Environmental Assessment, 1984.
[b] Packing is a mixture of bentonite and crushed basalt.

Table 2-23. DESIGN COMPARISONS OF VENTILATION REQUIRMENTS[a]

Description	1982 Conceptual Design	Current Design Concept
Maximum number of shafts	5	9
Maximum allowable shaft ID, m (ft)	No limit defined	3.7 (12)
Main airway (drift) maximum velocity, m/min (ft/min)	457 (1,500)	457 (1,500)
Service shaft maximum velocity, m/min (ft/min)	610 (2,000)	610 (2,000)
Ventilation shafts maximum velocity, m/min (ft/min)	1,067 (3,500)	1,220 (4,000)
Ambient rock temperature, °C (°F)	57 (134)	52 (125)
Maximum air volume and phase, m^3/min (ft^3/min)	Backfilling, 11,980 (4.2×10^5)	Operations, 30,300 (1.07×10^6)
Room cooldown time for backfilling, days	53	90

a DOE, Draft Environmental Assessment, 1984.

Table 2-24. REPOSITORY DESIGN COMPARISONS OF OVERALL PHYSICAL DIMENSIONS AND SHAFT ARRANGEMENTS[a]

Description	1982 Conceptual Design	Current Design Concept
Overall underground dimensions, m (ft)	1,610 x 3,360 (5,270 x 11,020)	1,930 x 4,150 (6,330 x 13,600)
Areal extent, ha (acres)	540 (1,334)	800 (1,978)
Total length—emplacement rooms, m (ft)	27,600 (90,600)	156,100 (512,000)
Size—emplacement rooms, m (ft)	3.1 x 6.1 (10 x 20)	3.1 x 7.0 (10 x 23)
Pillar width, m (ft)	65 (212)	31 (100)
NUMBER AND DIAMETER OF SHAFTS		
Waste handling	One 3.7-m (12-ft) ID	One 3.7-m (12-ft) ID
Service/mine intake	One 4.9-m (16-ft) ID	One 3.7-m (12-ft) ID
Basalt hoisting/ mine exhaust	One 4.3-m (14-ft) ID	One 3.7-m (12-ft) ID
Mine intake	Not applicable	One 3.1-m (10-ft) ID
Mine exhaust	Not applicable	One 3.1-m (10-ft) ID
Confinement intake	One 3.7-m (12-ft) ID	Two 3.7-m (12-ft) ID
Confinement exhaust	One 3.4-m (11-ft) ID	Two 3.7-m (12-ft) ID

a DOE, Draft Environmental Assessment, 1984.

Although the number of shafts has increased from five in the 1982. Conceptual Design to a total of nine in the current preliminary design, the cross-sectional area has increased only 40 percent with the advent of the more recent shaft maximum inner diameter of 3.7 meters (12 feet) as presented in table 2-24.

2.2.3 Preliminary Repository Waste Package Characteritics

The waste package constitutes the primary containment mechanism for the high-level radioactive waste during transfer and emplacement operations at the repository. In addition, the waste package provides post-emplacement containment following fixed, permanent storage since it is one subsystem comprising a total system of multiple barriers controlling the release of radionuclides to the accessible environment. The other two principal subsystems included in the current radionuclide containment system are the repository seals and the basalt host rock. The waste package consists of three major components: the waste form, container, and packing as graphically illustrated in figure 2-26.

The waste form may be either unprocessed spent fuel or canisters of vitrifed high-level nuclear waste. The container will be a hermetically sealed (welded) metal vessel containing the waste form. The current packing concept is a mixture of crushed basalt and clay used to fill the space between the container and the basalt host rock surface surrounding the container in the emplacement hole. The waste package is emplaced in short horizontal boreholes that extend from the wall of emplacement rooms of the repository as shown in figure 2-27. The conceptual emplacement sequence for the packing and the container is shown if figure 2-28.

2.2.3.1 Current Waste Package Conceptual Designs Of the primary requirements or design criteria for the waste forms is that it must be resistant to both groundwater and airborne dispersal thereby enhancing the waste package barrier containment function during waste package fabrication, transfer emplacement, and long-term (10,000 years or more) burial.

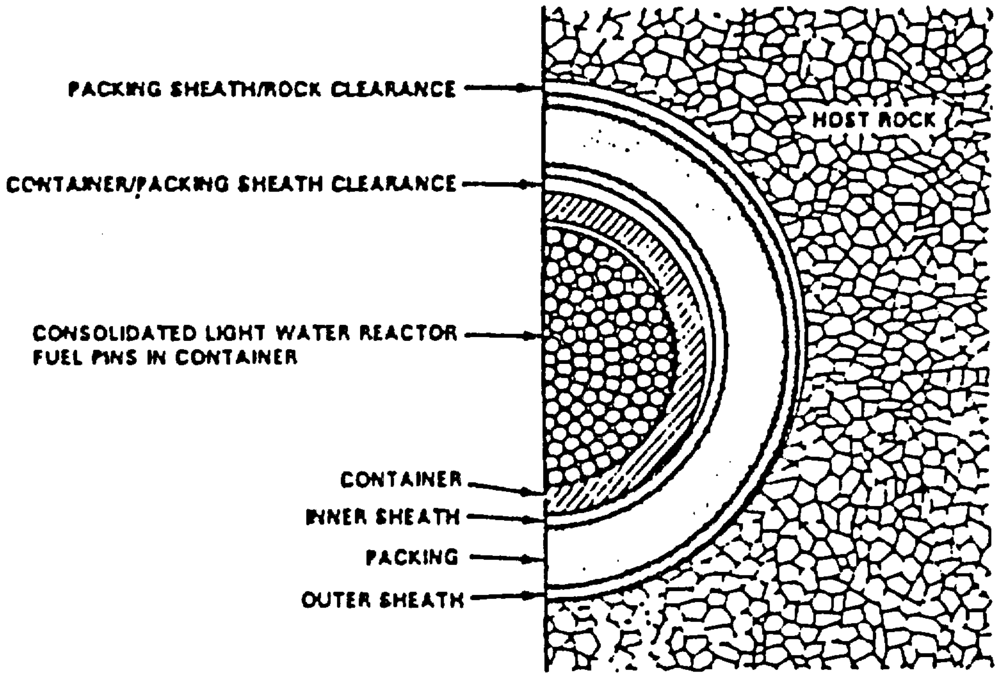


Figure 2-26. TYPICAL CROSS SECTION OF WASTE PACKAGES CONSOLIDATED LIGHT WATER REACTOR FUEL PINS

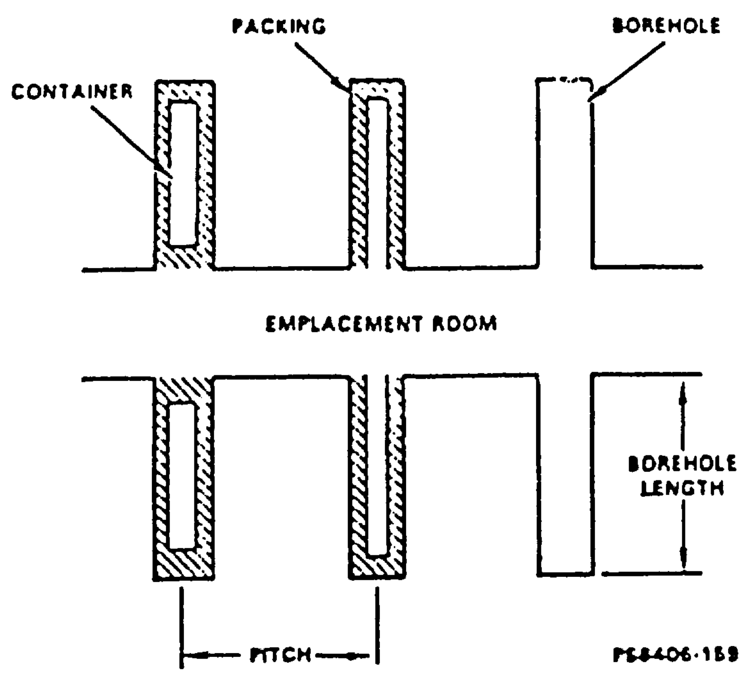


Figure 2-27. PRELIMINARY PLAN VIEW FOR REPOSITORY WASTE EMPLACEMENT

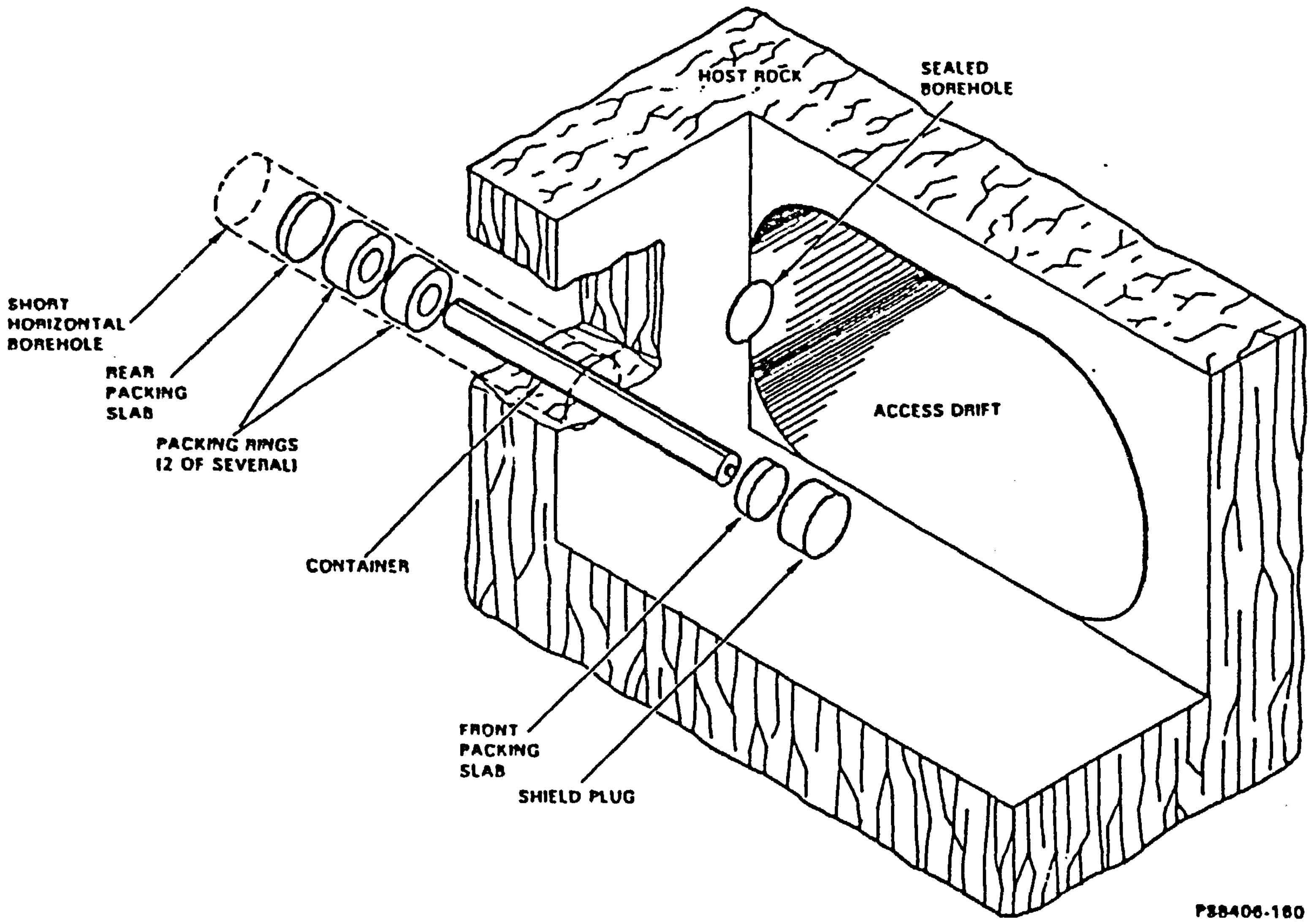


Figure 2-28. CONCEPTUAL SCHEMATIC OF WASTE EMPLACEMENT SEQUENCE

Current design dimensions and weights for both spent fuel and commercial high-level nuclear waste are presented in table 2-25.

Several potential problem areas arise when one considers the packaging of spent fuel rods. For example, intact spent fuel rod assemblies present a relatively straightforward approach to the design of a waste package. However, complexities to waste package design occur when failed rod assemblies are confronted. The proposed rod consolidation of fuel rods removed from their assemblies offers a possible solution to this potential problem, however. Consolidated spent fuel compacted rods are placed in waste containers approximately 50 centimeters (20 inches) in outer diameter by 411 centimeters (162 inches) long based upon current, preliminary design concepts as presented in table 3-25. However, there are a wide variety of spent fuel characteristics; e.g., sizes and number of fuel rods per assembly, heat and fission porduct activity generation rates, fuel burnup and age, etc., that must be considered prior to the final, approved waste package containter design.

Proposed high-level commercial reprocessing waste will consist of vitrified waste, presently in the form of a borosilicate glass, enclosed in a stainless steel canister. The present preliminary design concept, previously shown in table 2-25, specifies a cylindrical canister, 45.6 centimeters (18.0 inches) outside diameter by 325 centimeters (128 inches) long.

As previously inferred, the container currently is required to contain the waste form under the regulatory mandate within 10CFR60 (NRC, 1983) a minimum of 300 to 1,000 years. Thus, the aforementioned container internal diameters are essentially sized to accept the rods from four pressurized water reactor (PRW) assemblies or nine boiling water reactor (BWR) assemblies. In any event, the container design(s) must be able to accomodate dimensionally the largest expected number and size of fuel rods. Additionally, the internal length of the reference spent fuel container must also be designed to accept the longest anticipated fuel rods (including an acceptable clearance between the container head and fuel rod ends) for nuclear power reactors over a minimum time interval of about 25 years extending from the initiation of repository operations.

Tatle 2-25. CURRENT REPOSITORY WASTE PACKAGE DESIGN CHARACTERISTICS (DOE, DRAFT ENVIRONMENTAL ASSESSMENT, 1984)

Parameter	Spent Fuel	Commercial high-level waste
Container OD, cm (in.)	50.3 (19.8)	45.6 (18.0)
Required borehole dia, cm (in.)	89 (35)	84 (33.1)
Waste container overall length, cm (in.)	411 (162)	325 (128)
Loaded weight of container, t (tons[a])	7.6 (8.4)	2.7 (3)

[a] Weight is measured in metric tons (tons).

The container thickness will be based upon the structural requirements necessary to withstand hydrostatic pressure, additional thickness to allow for a 1,000 year corrosion depth, and any supplementary thickness that might be required to reduce groundwater radiolysis to acceptable levels, if necessary.

The current material for reference container construction is low-carbon steel. However, alternate, corrosion-resistant materials are being evaluated, including Fe9Cr1Mo (low alloy stee) and Cupronickel 90-10.

The packing has no pre-emplacement functions. After emplacement, the packing is required to provide an additional containment barrier to either preclude or limit groundwater intrusion to the container and to reduce radionuclide release and transport from the waste container by controlling groundwater Eh and pH. Thus, the packing material provides a low permeability environment between the waste package and the surrounding walls of the basalt host rock.

As previously shown in figure 2-28, the packing material in current designs is preformed in an annular shape and is emplaced first in a relatively short, horizontal borehole, then followed by subsequent emplacement of the waste container. The current reference packing material is 75 percent crushed basalt and 25 percent bentonite clay and has a minimum thickness of 15.2 centimeters (6 inches).

2.2.4 Development of Release Scenarios for High-Level Nuclear Waste Repositories

Many different physical processes can affect the future behavior of an underground high-high-level nuclear waste repository. Detailed analysis and evaluation of these processes is necessary in order to develop pragmatic scenarios that could lead to significant releases of radioactive waste materials to the biosphere.

For purposes of selecting disruptive, release scenarios for detailed parametric characterization and subsequent release-risk assessment, a sequential process or method of analysis must be developed for the repository program. In terms of the characterization of credible release scenarios for the high-level nuclear waste repository in basalt, the process will entail a minimum of four major procedural steps as follows:

1. Develop a comprehensive list of credible site-specific disruptive processes and events to the prescribed nominal or baseline conditions for the high-level nuclear waste repository.

2. Adopt selection criteria by which disruptive release scenarios can be systematically identified for more detailed analysis.

3. Assess the occurrence probability and likely adversity of the consequences of potential disruptive release scenarios.

4. Select scenarios to be characterized sufficiently for use in a risk-consequence analysis.

Analysis of potential disruptions, changes, or differences from nominal repository design conditions encompasses four general classes of possible release scenarios:

1. Uncertainties and potential omissions of significant consequence associated with characterization of the candidate repository site.

2. Potential disruptions due to natural system dynamics within the general area encompassing the candidate repository site.

3. Potential disruptive release scenarios resulting from repository construction and operations.

4. Potential disruptive release scenarios induced by human activities other than repository construction and operation.

2.2.4.1 Site Characterization and Uncertainties No candidate site can be characterized to 100 percent certainty. Reduction of all geologic and hydrologic characterization uncertainties to exceeding low values by extensive subsurface exploration in the basalt host rock will be unquestionably costly and may intrinsically reduce host rock isolation capability during the test program. The categorization of potential repository site characterization omissions and uncertainties, as used in this report, is in terms of probability of occurrence and potential adversity of consequences. While it is recognized that site characterization omissions and uncertainties do not represent changes or disruptions to actual site conditions, they have been included in this development of potential release scenarios because they constitute conditions that could be omitted in the site characterization planning process.

2.2.4.2 Natural Systems Dynamics Assessment of natural phenomena is largely dependent or work in the earth sciences. The various tasks entail both a description of the physical phenomena or process and an evaluation of its likelihood of occurring in the future. Such work is at the very forefront of current research in the geologic sciences. A generic list of natural phenomena that could possibly lead to disruptive, radioactive release scenarios for any geologic, high-level nuclear waste repository is presented in table 2-26. Although prediction of many events that occur on geologic time scales is beyond current capabilities, trends and ranges in behavior can often be determined.

2.2.4.3 Repository-Induced Disruptions Another major category of processes affecting waste disposal in a subsurface high-level nuclear waste repository is the waste- and repository-induced phenomena. As previously discussed, the waste emits considerable amounts of radiation and heat. The presence of the mined excavation affects the surrounding host rock mechanically and could modify the groundwater hydrology. The most localized effects involve interactions between the waste package and the immediately surrounding rock. The behavior of the back-filled mine and connecting shafts might also be modified. The largest-scale effects are principally those due to the effect of heat from the buried waste on the surrounding rock masses. Thus, the thermal output of the waste is important in several ways: it could lead to expansion of the rock mass leading to fracturing, it could locally increase rock permeability, and it could result in convection in groundwater. A generic list of waste and repository-induced phenomena is presented in table 2-27.

Given knowledge of potential disruptions induced by construction or operations of the repository and analysis of the likely resultant consequences, the repository will be designed, if economically feasible, to mitigate such consequences. Repository-induced disruptions of the containment system cannot be categorized according to occurrence probability evaluations based upon extrapolating past occurrences of geologic events or processes into the future. Such decisions must be made on the basis of best-judgment consensus of occurrences probability as indicated by past construction and engineering design experience. Performance of underground facilities under various conditions has

Table 2-26. GENERIC LIST OF NATURAL PHENOMENA THAT COULD LEAD TO POTENTIALLY DISRUPTIVE RELEASE SCENARIOS FOR A SUBSURFACE HIGH-LEVEL NUCLEAR WASTE REPOSITORY

Climatic fluctuations
Glaciation
Denudation and stream erosion
Magmatic activity
Extrusive
Intrusive
Epeirogenic displacement
Igneous emplacement
Isostacy
Orogenic diastrophism
Near-field faulting
Far-field faulting
Diapirism
Diagenesis
Static fracturing
Surficial fissuring
Impact fracturing
Hydraulic fracturing
Dissolutioning
Sedimentation
Flooding
Undetected features
Faults, shear zones
Breccia pipes
Lava tubes
Gas or brine pockets
Meteorites

Table 2-27. GENERIC LIST OF WASTE AND REPOSITORY-INDUCED PHENOMENA THAT COULD LEAD TO POTENTIALLY DISRUPTIVE RELEASE SCENARIOS FOR A SUBSURFACE HIGH-LEVEL NUCLEAR WASTE REPOSITORY

- Thermal effects
  - Differential elastic response
  - Nonelastic response
  - Fluid pressure changes
  - Local fluid migration
  - Canister migration
  - Convection
- Chemical effects
  - Geochemical alterations
  - Corrosion
  - Waste package-geology interactions
  - Gas generation
  - Seal-rock interactions
- Mechanical effects
  - Change in local state of stress
  - Readjustment of rock along joints
  - Local fracturing
  - Canister movement
    - Subsidence
  - Radiation effects
    - Material property changes
    - Radiolysis
    - Criticality
    - Decay product gas generation
    - Stored energy
  - Modification of hydrologic regime

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been considered by Barton (1982), Trent (1982), Owen (1982), Kuesel (1982), and Vortman (1982a, 1982b). Other waste-induced potential failures have been considered by Wallace and others (1980, 1982).

2.2.4.4 Disruptions Resulting from Human Activities Independent of Repository Construction and Operation The major category of processes and events affecting waste disposal in a subsurface, high-level nuclear waste repository that is the least amenable to scientific and engineering analysis relates to human-induced phenomena. Predictions of future activities by man are by their very nature entirely speculative. Specific investigations conducted to date in this area are quite limited.

Nevertheless, identification and further categorization of credible events and processes resulting from human activities independent of repository construction and operation are governed principally by four general guidelines established by NRC (NRC, 1982). The foregoing NRC guidelines are discussed in greater detail in a later section of this report. Given these four guidelines, human intrusion would be either prohibited during the time that the consequences of such intrusion would be relatively significant, or a future generation would be prepared to accept the risks of intentional intrusion. A generic list of human-induced phenomena is given in table 2-28. It should be noted that table 2-28 includes possible disruptions from human activities both dependent and independent of repository construction and operation.

2.2.5 Development of Repository Release Scenario Methodology

Given the four classes or categories of possible repository release scenarios outlined in Sections 2.2.2 through 2.2.4, a methodology must be developed to evaluate events, conditions, and processes (previously listed in a generic context in tables 27, 28, and 29 that are of practical concern for the permanent repository should it be located at the proposed Hanford Site. The basis for such a methodology must be, of necessity, the federal regulatory guidelines, standards, rules, and procedures for management and disposal of spent nuclear fuel, high-level, and transuranic radioactive wastes. The final standards (40 CFR 191) were promulgated by the U. S. Environmental Protection Agency (EPA) on September 1985. The Nuclear Regulatory Commission (NRC) previously published its final rule for 10 CFR Part 60 on June 21, 1983, establishing technical

Table 2-28. GENERIC LIST OF HUMAN-INDUCED PHENOMENA THAT COULD LEAD TO POTENTIALLY DISRUPTIVE RELEASE SCENARIOS FOR A SUBSURFACE HIGH-LEVEL NUCLEAR WASTE REPOSITORY

- Improper design or operation
  - Shaft seal failure
  - Improper waste emplacement
- Undetected past intrusion
  - Undiscovered boreholes
  - Mine shafts
- Inadvertent future intrusion
  - Archaeological exhumation
  - Weapons testing
  - Non-nuclear waste disposal
  - Resource mining (salt, mineral, hydrocarbon, geothermal)
  - Storage of hydrocarbons, compressed air, or hot water
- Intentional intrusion
  - War
  - Sabotage
  - Waste recovery
- Perturbation of groundwater system
  - Irrigation
  - Reservoirs
  - Intentional artificial groundwater recharge or withdrawal
  - Chemical liquid waste disposal
- Biosphere alteration
  - Establishment of population center
  - Climate modification

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criteria for disposal of high-level radioactive wastes in geologic repositories as required by the Nuclear Waste Policy Act of 1982. These criteria address siting, design and performance of a geologic repository and the design and performance of the package which contains the waste within the repository. Also included are criteria for monitoring and testing programs, performance confirmation, quality assurance, and personnel training and certification.

Subpart A of the recently enacted environmental standards (40 CFR 191) limits radiation exposures to members of the public from waste emplacement and storage operations at DOE disposal facilities that are not regulated by the NRC.

Subpart B established several different types of requirements for the disposal of these materials. The primary standards for disposal are long-term containment requirements that limit projected releases of radioactivity to the accessible environment for 10,000 years after permanent disposal in the geologic repository. These release limits were promulgated by the EPA to insure that risks to future generations from disposal of high-level radioactive wastes will be no greater than the risks that would have existed if the ore used to create the wastes had not been originally mined.

Prior to adoption of the final rule for 40 CFR Part 191 in September 1985, occurrence probabilities of potential disruptions had been considered in the following context of regulatory guidelines and standards:

- Reasonably Foreseenable Releases are relatively small releases, including releases from human intrusion, that have more than 1 chance in 100 of occurring within 10,000 years.

- Very Unlikely Releases are releases of moderate size, mostly from disruptive geologic phenomena, that have less than 1 chance in 100 and more than 1 chance in 10,000 of occurring within 10,000 years.

- Extremely Unlikely Releases are large releases, such as those resulting from meteorite impact of igneous intrusion, that have less than 1 chance in 10,000 of occurring in 10,000 years.

Maximum allowable release limits proposed by the EPA prior to the final rulemaking in September 1985 for the Very Unlikely occurrence probability category, were 10 times larger than those for the Reasonabley Forseeable category.· Release limits for the Extremely Unlikely category were never defined by the EPA. In each instance, the proposed maximum allowable releases were defined at a controlled area/accessible environment boundary that was defined at 10 km from the outermost limits of emplaced waste. NRC regulatory guidelines (NRC, 1983) have grouped disruption scenarios into two categories: "anticipated" and "unanticipated" events, processes, and conditions.

Unanticipated disruptions have been defined by the NRC in 10 CFR Part 60 (NRC, 1983) as those disruptions judged unlikely to occur during the time that performance must be evaluated, but that nevertheless, are sufficiently credible to warrant consideration. For natural phenomena, those disruptions have been defined to include processes and events for which there is no evidence of occurrence during the Quaternary period of geologic time, i.e., up to 1.6 million years before the present, or if evidenced during Quaternary time, are unlikely to occur during the waste isolation period of primary concern (10,000 years).

Anticipated disruptions, by comparison, have been defined as those disruptions that are expected to have credible potential for occurrence during the waste isolation period. For natural phenomena, in general, such expectations are based upon occurrence and occurrence frequency during Quaternary time at or near the candidate repository site.

Identification of such phenomena, as is evident from their definition, requires extensive scientific knowledge and judgment. The occurrence probabilities of such phenomena typically are not subject to exact quantification. In addition, use of the terminology "anticipated" and "unanticipated" implicitly bounds concerns regarding disruptions resulting from human activities.

Adoption of the final rule for 40 CFR 191 by EPA has altered the basis for evaluating occurrence probabilities of potential disruptions by eliminating the terms "reasonably foreseenable" and "very unlikely" releases. Instead, the permissible probabilities for two different levels of cumulative releases (over 10,000 years after disposal) are now incorporated into the containment requirements. Containment requirements for disposal systems for spent nuclear fuel, high-level, or transuranic radioactive wastes under section 191.13 of the final rule must be designed to provide a reasonable expectation, based upon performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall:

(1) Have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to table 2-29 (Table 1, Appendix A, 40 CFR191); and

(2) Have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to table 2-29.

Thus the numerical probabilities associated with the two foregoing release categories have been increased by an order of magnitude to reflect further assessments of the uncertainties associated with projecting the probabilities of geologic events such as fault movement.

The final rule for 40 CFR 191 also cleary indicates that comprehensive performance assessments, including estimates of the probabilities of various potential releases whenever meaningful estimates are practicable, are needed to determine compliance with containment requirements. Part 191.12 of the final rule defines "performance assessment" as an analysis that: (1) identifies the processes and events that might affect the disposl system; (2) examines the effects of these processes and events on the performance of the disposal system; and (3) estimates the cumulative releases of radionuclides, considering the associated uncertainties, caused by all significant processes and events with the additional intention of incorporating these estimates into an overall probability distribution of cumulative release, if possible.

In addition to the aforementioned changes related to the development of occurrence probabilities for potentially disruptive release scenarios, the final rule for 40 CFR Part 191 altered the concept of maximum allowable releases at a controlled area/accessible environment boundary by redefining a "controlled area" as a surface location, to be identified by passive institutional controls, that encompasses no more than 100 square kilometers and extends horizontally no more than five kilometers in any direction from the outer boundary of the original location of the radioactive wastes in a disposal system as well as the subsurface underlying such a surface location. Section 191.12 of the final rule defines "assessible environment" as the atmosphere, land surfaces, surface waters, oceans and all the lithosphere that is beyond the controlled area.

Table 2-29. RELEASE LIMITS FOR CONTAINMENT REQUIREMENTS-NUCLEAR WASTE REPOSITORY-40 CFR, PART 191, APPENDIX A, FINAL RULE, SEPTEMBER 19, 1985[a]

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

2.2.5.1 Generic Adverse Conditions Identified by NRC The following repository site conditions have been considered by the NRC to be potentially adverse to acceptable performance of any nuclear waste repository during the period of performance assessment, and therefore, of interest to disruption scenario identification and selection in this report:

- Failure of constructed surface-water impoundments that could cause flooding of the repository operation area

- Human-induced perturbation of the groundwater-flow system

- Changes in the regional groundwater-flow system due to surface-water impoundments resulting from natural events and processes

- Quaternary-age structural deformation affecting the regional groundwater-flow system

- Changes in physcial hydrological properties such as hydraulic gradient, velocity, storage coefficient, conductivity, potentiometric levels, and discharge points that might accelerate radionuclide migration

- Change in climatic conditions, with consequent change in groundwater conditions, that accelerate radionuclide migration

- Change in groundwater chemistry, increasing waste-form solubility or reactivity and decreasing sorption

- Historically recorded seismicity with severity and proximity sufficient to significantly affect the repository

- Evidence, based on correlations of seismicity to structural features, that either the frequency or magnitude or earthquakes may increase

- Evidence of igneous activity of Quaternary age

- Evidence of extreme erosion of Quaternary age

- Potential for occurrence of resources within the geologic setting of the repository site with value greater than the average value of resources for areas of equal size elsewhere

- Evidence of subsurface mining of resources within the repository site or evidence of drilling for any purpose within the site

- Geologic or groundwater conditions that could require extra-ordinary engineering measures in design and construction of the facility or in sealing or boreholes and shafts

- Geomechanical conditions that do not permit design of stable underground openings for the duration of operations through permanent closure of the facility.

2.2.5.2 <u>Chronology of Disruptive Release Scenario Identification for High-Level Nuclear Waste Repositories</u> The initial step in evaluating long-term performance of a repository; i.e., identification of events, processes and conditions, is limited mostly by the imagination of the investigator. The number of specific combinations and permutations of possible disruptions that may be considered is enormous. Deficiencies in the process of generating scenarios can translate directly into deficiencies in dependent analytical processes, however. Therefore, extreme care must be taken to assure that the phenomena considered adequately define the potentials for radionuclide releases. The challenge in considering various disruptive release scenarios is to identify and describe a relatively small number of scenarios that are demonstrably representative of consequences from the full range of credible disruptions.

An examination of work to date shows that the difficulty of choosing scenarios and evaluating their probabilities profoundly affects the choice of methods for repository safety analysis. The majority of studies that were reviewed select scenarios without using any formal procedure. The selection may be directed toward choosing the most likely case, toward defining a worst case in order to bound the consequences, or toward spanning a range of scenarios including both the more likely ones and relatively unlikely scenarios with greater consequences. Defining the most likely scenario can be either trivial (and therefore uninformative) or extremely difficult. For example, assuming a repository in basalt one would expect some release; a realistic description of such releases would require detailed models of the degradation of the waste package and the flow of water through fractural rocks. These processes currently are not well understood and as a result the most likely scenario is difficult to specify.

The worst-case approach is often adopted in order to avoid such problems. It has been followed most consistently and thoroughly in the study of a repository in granite by the Swedish Nuclear Fuel Safety Project (KBS, 1978). The attempt here is to bound the possible consequences under "credible" circumstances; that is, one ignores theoretically possible but highly unlikely events such as meteorite strikes. KDS uses repository design and worst-case data to effectively eliminate several categories of scenarios from consideration. For example, pathways to the biosphere via failed repository seals are removed by bentonite-containing backfill which swells on contact with water and would seal any gap. The exceedingly small transit time to the biosphere utilized in the safety analysis (unwarranted by the field data) would probably bound the effects of such events as geologic faulting or fracturing. In turn, the repository would be sited in such a way as to provide little reason for human intrusion (other than entry into the repository itself). In this manner, the range of scenarios is narrowed to consider only a single worst credible situation.

One study which deals with a relatively broad range of scenarios is by Giuffre et al (1980), who analyze groundwater transport through repositories in salt. A total of 34 scenarios were selected by another group at Lawrence Livermore Laboratory. Scenarios were selected by considering all of the various flow paths that might be present (boreholes, shafts, clay partings, fault zones, breccia pipes, etc.) and by modeling all relevant combinations.

The definition of scenarios is always somewhat arbitrary. Typically, it is not certain which events and processes will affect repository behavior. For example one might not know whether thermal expansion of the rocks around the repository will fracture overlying strata. One would then have two possibilities: fractured rocks and unfractured rocks. In a more complicated sequence of events, movement along a fault might occur before and after fracturing occurs. Other possibilities exist, of course, in vast numbers. Each combination of circumstances which might determine repository behavior is what constitutes a scenario. Therefore, a scenario may be described by equations and parameters which may or may not be known accurately. Circumstances requiring different mathematical descriptions are distinct scenarios.

The long time interval (10,000 years or more) associated with a permanent geologic repository is probably the most perplexing problem confronting practical definition of disruptive release scenarios. The possibilities that events could occur at vastly different times is one source of the dilemma. For example, one can describe a scenario in which a borehole seal fails after a specified time such as 1,000 years. Most informal scenario selection uses this approach; however, the choice of time is arbitrary. Radioactive decay and other gradual processes usually cause the consequences of a disruptive event to depend on its time of occurrence. For example, the consequences of a meteorite impact which fractured the rocks around a repository and created new pathways for groundwater would depend on the integrity of the waste package at the time of impact and the amount, if any, of radioactive waste migration which had occurred. In studies aimed at placing an upper bound on consequences this difficulty may be avoided by assuming the event occurs at the worst credible time which is usually, but not always, the earliest time. An alternative method is to perform the calculations more than once, assuming diferent times for the event. But neither technique provides a realistic estimate of the danger or risk, which would require taking account of the probability of occurrence.

Yet another approach is to define occurrence of an event at time, t, as a single scenario, with t an unknown parameter describing the time of occurrence. A method has been developed to calculate expected values of consequences when scenarios are defined in this way (Ross and Koplick, 1978). This method is limited to cases where changes in the geology occur only as discrete events and the events are Markov, i.e., the probability of an event depends only on the current situation and not the past history.

Fault- and event-tree analysis, described and utilized to develop release scenarios for high-level nuclear waste transportation scenarios in a earlier section of this report, has also been used to identify repository scenarios in several studies that lead to loss of containment (Logan and Berbano, 1978; Pertozze et al, 1977; Hill and Grimwood, 1978; Bingham and Barr, 1979; Schneider and Platt, 1974; d'Alexandro and Bonn, 1980).

When fault or event trees are used to analyze repository behavior, they must treat both continuing processes and discrete events. Processes are included in trees by describing the effect of the process on repository behavior. Erosion for example, may be treated by a separate calculation computing the time when the buried waste will be uncovered. The event of uncovering would appear in the tree. However, fault and event trees are not directly applicable to analyzying the processes themselves or their interactions.

One of the most plausible applications of fault-tree analysis to waste repository studies is to compute scenario probabilities. However, up to this time, it has been rarely possible to devise meaningful estimates of the probabilities of the events and processes that occur in the tree. Often expert opinion is used to provide a "best estimate" of the probability of an event. If a scenario consists of many events whose probability has been so estimated, the reliability of any computed scenario probability is questionable.

Fault and event trees have been valuable primarily as a means of organizing the thinking of the scenario analyst. Through the construction of a tree, the analyst hopes to ensure completeness by avoiding the omission of important phenomena which might contribute repository failure. The tree structure is especially useful in determining which sequences of phenomena are most worthy of detailed analysis.

Probably the most extensive use of these methods related to repository assessment is found in the draft environmental impact statement for the Waste Isolation Pilot Plant (WIPP, 1979). Five representative scenarios are chosen from a list compiled from previously generated lists and event-tree analysis. In all, 94 scenarios are identified, of which four result in the direct transfer of wastes to the surface; the remainder introduce the wastes into an aquifer overlying the repository. The complete list of scenarios and how they are derived are discussed by Bingham and Barr (1979). The scenarios are ranked in importance by assigning relative probabilites to events using expert opinion.

On the whole, attempts to apply fault and event trees to repository safety assessment have met with limited successes. While these methods are quite useful as a means of categorizing scenarios the difficulties in obtaining reliable probability data are a formidable obstacle to the use of fault and event trees for quantificative purposes.

The fault-tree or event-tree approach is based on events which either occur or do not occur. It is therefore inappropriate for analyzing processes which occur continuously at a finite rate. Systematic approaches to the description of scenarios involving such processes are under development at Sandia National Laboratory, Pacific Northwest Laboratory, and Lawrence Livermore Laboratory (Campbell et al, 1978, 1982; Stottlemeyer et al, 1980; Breenborg et al, 1978; Lee et al, 1978; Wallace et al, 1980). These approaches use simulation techniques to describe the effects of continuous processes on a repository.

For the most part, these simulation methods use ordinary differential equations to describe the evolution over time of a set of variables describing a scenario. In the PNL work, for example, these variables may also take random, discrete jumps, due to certain discrete events. The equations are solved numerically. The values given by the solution of the simulation equations appear as parameters in the subsequent equations describing the consequences of the scenarios.

Sandia's contribution to the identification and study of release scenarios is the numerical simulation analysis of releases induced by perturbations introduced by the repository itself. The method relies on modeling processes by differential equations which are first order in time. Whether this method will be useful tool in understanding the long-term dynamics of geologic and hydrologic systems remains to be confirmed.

The long-term data used to estimate scenario probabilities must be derived primarily from the sciences of geology, climatology and archaeology. The difficulties in obtaining these estimates are considerable. Archaeology has never been a predictive science. Climatology and geology have only recently developed and utilized predictive analytical techniques. However, for the most part, the data and techniques are such that estimates are more qualitative than quantitative.

The one type of scenario for which a probability can be calculated from observed data without heroic extrapolations is the meteorite strike. Meteorite strikes of sufficient size to cause repository disruption are reasonably randomly distributed and leave identifiable craters. The craters, of course, can be counted in order to provide estimates of probability. The earliest study which follows this procedure, by Claiborne and Gera (1974) determines the frequence of impacts producing craters larger than 1 kilometer (km) in diameter on the basis of ancient Canadian meteorite craters. It then calculates the probable number of craters of different diameters on the basis of the frequency distribution observed for the moon. This gives the probability of a meteorite capable of creating a crater 600 meters deep as 2×10^{-14} km^{-2} yr^{-1}. This result is repeated by several other authors (GEIS, 1980; Cohen, 1977; Logan and Berbano, 1978; ADL, 1980).

Logan and Berbano (1978) require the meteorite to be able to exhume material from a depth of 800 meters. For this reason, their probability estimate is one-half the value cited by Claiborne and Gera. KBS (1978) cites an estimate of 1×10^{-13} km^{-2} yr^{-1} for meteorites which can cause craters at least 100 meters deep. The highest probability of

1.4×10^{-12} km^{-2} yr^{-1} is given by Arthur D. Little, Inc., (ADL, 1982). This last analysis does not study the direct exhumation of the waste, but the increase in water transport through the repository due to the fracturing of the overlying rock by the impact. Hartmann (1979) has developed relationships between crater size, depth, impact energy, fracture depth, and seismic disturbance.

Direct release by volcanism has been considered by several studies (Claiborne and Gera, 1974; Smith and Kastenberg, 1976; Logan and Berbano, 1978; ADL, 1980). ADL estimtes that the national average probability of the formation of a volcanic vent in 1.25×10^{-11} km^{-2} yr^{-1}. It notes that repository site selection could reduce this probability. An important exception to the foregoing premise is a repository site location in basalt, e.g., the Hanford Site, where the probability estimate is at least six times higher than for other media or $\_7.5 \times 10^{-11}$ km^{-2} yr^{-1}, because the presence of basalt is indicative of significant volcanic activity in the past. These estimates are obtained from the average number of volcanic vents per square kilometer fromed within the coterminous United States within the last 10^7 years. According to Crow (1980), the probability of volcanic activites in nonvolcanic areas is undoubtedly quite low and could probably be bounded, but a realistic estimate is beyond the current state of the are in volcanology.

The likelihood of disruption by meteorites or volcanoes should be placed in perspective. If the universe is approximately 10 to 20 billion years old, these is one in 100 chances or less that the events discussed above would occur in the known history of the universe. Given their remote probability, it is surprising the various studies continue to consider these events. The Swedish studies reject them because of their improbability (KBS, 1978). The repeated reworking by other investigators suggests that these scenarios are the rare ones for which data can be obtained.

Whereas natural events such as those discussed above leave easily identifiable remains and have occurred over such long periods of time that estimates of probability can be obtained for even extremely rare events, such is not the case with human actions. Although the human genus has existed for several million years, the change from hunting-gathering to an agriculturally-based community has only occurred within the last 10,000 years. The possible range in human technology is dramatically apparent in the present society with some cultures capable of space exploration while others are still in the "stone age". This range, together with the short observation time and inherent difficulties of predicting conscious actions implies that human intrusion scenarios are the most difficult to predict.

The most commonly mentioned type of human intrusion is drilling and interception of a waste canister at some point in time. WIPP (1979) lists the sequence of events which must occur to bring some of the repository contents directly to the surface:

- Institutional control is lost,
- Knowledge of the repository is lost,
- There is an incentive to explore in the area of the site,
- The repository area is chosen for drilling,
- The contents of the repository go unrecognized as radioactive material before and during drilling,
- Drilling intercepts a high concentration of radionuclides,
- The material brought up is left untreated and exposed.

If any of these events does not occur, then the direct release of radionuclides will not occur. There is no individual study, based on the current CERT review of the available literature, which estimated the probability of drilling at the repository site, the probability of intercepting an emplaced waste canister, and the consequences of the foregoing interception of the canister. The generic environmental impact statement on commercially generated radioactive waste (GEIS, 1980) consider only the last two factors, while the ADL study considers only the first two. A study of the possible scenarios for WIPP (Bingham and Barr, 1979) suggests a relative probability of 0.1 for drilling at the site after 1,000 years. However, the authors state that they do not feel is appropriate to use this estimate in a risk assessment. Given that drilling at a site occurs, the probability of intercepting a waste canister in a underground repository has been estimated by taking the ratio of total canister area to the total area of the site.

Most studies do not assign numerical probabilities to groundwater transport scenarios. These scenarios often involve subsurface phenomena which are more subtle than the aforementioned meteorite strikes and volcanic eruptions, and the difficulties in using hydrology predictively become greater. Stottlemeyer et al (1980) have described a variety of hydrologic and geological phenomena that can affect groundwater release scenarios and have summarized the evidence concerning the likelihood of their occurrence. Additional work on predicting certain natural phenomena (climate, sea level fluctuations, denudation, floods, landslides, glaciation etc.) is summarized by Scott and coworkers at PNL (1979).

Some investigators have attempted to assign probabilities to scenarios involving accelerated groundwater transport. Of such scenarios, probabilities are given most often for those in which release is initiated by earth movements along a fault. It is always assumed that a more permeable pathway is created along the fault. In studies of salt repositories, for example, the faulting commonly is assumed to lead to dissolution of the salt.

The probability of fault movement was first estimated by Claiborne and Gera (1974) in their study of the Delaware Basin. Two major faults have been noted in this basin, whose age is 2×10^8 years. Claiborne and Gera assume that two additional faults of the same length would be come active at random times in the next 2×10^8 years. A geometric analysis giving the probability that a random line segment will intersect a circular area the size of the repository is used to estimate the probability that a new fault will intersect the repository.

There are a number of weaknesses in this analysis. Primarily, faulting is not a strictly random phenomenon. Where faults exist, they are weaker than the surrounding rock, and stresses tend to be relieved through movement along existing faults rather than through creation of new ones. Furthermore, as noted by Claiborne and Gera, a new fault would not necessarily lead to containment failure. The fault would also have to create a permeable, continuous flow path, a significant amount of water would have to flow through it, and the host rock constituents would have to enter into aqueous solution more rapidly than the fault zone would be closed by creep. However, no estimates of the probabilities of the foregoing occurrences have been discovered in the literature, and it is suggested that no such estimates are presently available.

ADL (1980) uses the same general approach, although perhaps in a manner even more likely to lead to overestimates, to calculate the probability of formation of new faults in a variety of rock types. KBS (1978) uses this approach to calculate an upper bound on the rate at which additional fractures will form in already fractured granite.

Fracturing due to remobolization of an existing fault is discussed by ADL (1980) and Logan and Berbano (1978). ADL uses the same approach as for the creation of new faults, with the time since the fault last moved substituted for the age of the formation. Logan and Berbano combine a series of empirical models and extrapolations to obtain the probability from the rate of occurrence of small earthquakes. This reasoning requires a number of poorly supported assumptions.

An alternative approach is employed by Bertozzi et al (1977) for estimating the probability that a fault affects a repository in bedded salt. It is assumed that the faulting frequency for a tectonically stable zone is 2×10^{-12} km^{-2} yr^{-1}. The average fault length is computed from the observed statistical distribution of fault lengths. The probability of faulting within a sensitive area surrounding the repository is then computed. The sensitive area around the repository is derived from estimated dissolution rates for bedded salt following faulting.

D'Alessandro et al (1980) use the same method to estimate the probability that a repository in northeastern Belgium would be intersected by a fault. The probability that new faults will form is taken to be 5×10^{-9} km^{-2} yr^{-1}. This value seems to represent an extreme upper bound. The probability that the movement along the fault would be sufficient to breach a repository is also discussed.

These methods, although some times useful in setting upper bounds for probabilities of fault movement, are dependent on so many counterfactual assumptions as to be without value in providing realistic estimates.

According to Stottlemyre et al (1980), who summarize recent work in this area, estimates of faulting should be based upon the state of effective stress, material properties, recorded seismicity, observed cumulative deformation, average strain rates, and anticipated changes in strain rates. A numer of different models exist for predicting faulting frequency from this data, but the degree of uncertainty in any estimate remains high.

Attempts to quantify the probability of other types of groundwater scenarios have been made by Bingham and Barr (1979) and ADL (1980). Again the degree of uncertainty in these estimates and the extent to which the estimates rely on expert opinion and recent theoretical work should be emphasized. Not only is it difficult to predict natural geologic events but scenarios also depend on the extrapolated performance of engineered features of a repository. For example, the probability of failure of boreholes and shaft seals must, to a large extent, be estimated by the use of engineering judgment. Another problem area involves the possibility that certain features, such as faults and brecchia pipes in the host rock formations, could go undetected during site exploration.

Further difficulties are presented by the possibility that human activities might affect groundwater release scenarios. Bingham and Barr, for example, have acknowledged these issues and emphasize that their results are "intended only to establish relative likelihood for the scenarios; they have little significance."

A list of estimates of generic scenario probabilities for high-level nuclear waste repositories, including both direct releases and groundwater transport, are summarized in table 2-30 (Koplik et al, 1982).

2.2.5.3 Preliminary Disruptive Release Scenario Identification for High-Level Nuclear Waste Repository at Hanford Site Identification of potential disruptions to a repository within the Hanford Site and classification of such disruptions according to probability of occurrence and adversity of consequence has been derived from a rather extensive list of studies published since 1979. For example, Myers et al, 1979; Gephart et al, 1979; BWIP, 1981; Caggiano et al, 1980; Price, 1981; Bergstrom et al, 1982. Nevertheless, since our knowledge of the causes of geologic phenomena is incomplete, as previously discussed in the preceding section of this report, and the uncertainties in extrapolating our limited knowledge of present conditions over reasonably long distances and time duration equal to exceeding 10,000 years, the magnitude of predictive uncertainties cannot be minimized.

Considerable work has been done to identify potential disruptions. Numerous experts have characterized such scenarios using BWIP site-specific and generic data. An additional reference list of prior studies that provide fundamental support for the current and future work related to the identification and characterization of potentially disruptive release scenarios for a high-level nuclear waste repository at the Hanford Site is presented in table 2-31.

Table 2-30. SUMMARY OF GENERIC DISRUPTIVE RELEASE SCENARIO PROBABILITIES FOR HIGH-LEVEL NUCLEAR WASTE REPOSITORIES

Author/ system	Scenario	Cumulative probability	Comments
Claiborne and Gera (1974)/ Los Medanos bedded salt	Meteorite impact	10^{-10} @ 10^3 yr 10^{-7} @ 10^6 yr	
	Faulting—water intrusion—transport to well	10^{-7} @ 10^6 yr 10^{-4} @ 10^6 yr	
Girardi et al. 1977 Generic bedded salt and domed salt	Water intrusion—transport to surface water body	10^{-9} @ 10^3 yr 10^{-7} @ 10^6 yr	Probabilities of causative mechanisms not reported separately
Logan and Berbano (1978)/ Los Medanos bedded salt	Meteorite impact	10^{-10} @ 10^3 yr 10^{-7} @ 10^6 yr	
	Volcanic explosion	10^{-9} @ 10^3 yr 10^{-6} @ 10^6 yr	
	Volcanic transport to surface	10^{-8} @ 10^3 yr 10^{-5} @ 10^6 yr	
	Faulting—water intrusion—transport to surface water	10^{-4} @ 10^3 yr 10^{1} @ 10^6 yr	
KBS (1978a)/ generic granite	Meteorite impact	10^{-10} @ 10^3 yr 10^{-7} @ 10^6 yr	
	Fracture Formation	10^{-6} @ 10^3 yr 10^{-3} @ 10^6 yr	
Bingham and Barr (1979)/ Los Medanos bedded salt	Exhumation		
	Drilling	10^{-3} @ 10^3 yr	
(selected scenarios)	Meteorite impact	10^{-9} @ 10^3 yr 10^{-6} @ 10^6 yr	Values intended only to establish relative likelihood for the scenarios
	Two aquifer connection—transport to surface water body		
	Faulting	10^{-7} @ 10^3 yr 10^{-4} @ 10^6 yr	
	Shaft seal failure	0 @ 10^3 yr 10^{-7} @ 10^4 yr 10^{-5} @ 10^6 yr	
	Igneous intrusion	10^{-9} @ 10^3 yr 10^{-6} @ 10^6 yr	
	Drilling	10^{-1} @ $\geq 10^3$ yr	
	One aquifer connection—transport to surface water body		
	Meteorite impact	10-9 @ 10^3 yr	

Table 2-30. SUMMARY OF GENERIC DISRUPTIVE RELEASE SCENARIO PROBABILITIES FOR HIGH-LEVEL NUCLEAR WASTE REPOSITORIES (CONTINUED)

Author/ system	Scenario	Cumulative probability	Comments
Bingham and Barr (continued)		10^{-6} @ 10^{6} yr	
	Drilling	10^{-1} @ $\geq 10^{3}$ yr	
	Mining	0 @ $\leq 10^{5}$ yr 10^{-4} @ $\leq 10^{6}$ yr	
	Fracturing	10^{-3} @ $\leq 10^{5}$ yr 10^{-2} 10^{6} yr	
	Shaft seal fracture	0 @ 10^{3} yr 10^{-5} @ 10^{4} yr $.10^{3}$ @ 10^{6} yr	
	Natural salt dissolutioning	0 @ 10^{3} yr 10^{-5} @ 10^{4} yr 10^{-2} @ 10^{5} yr 10^{-1} @ 10^{6} yr	
	Capitan reef potash mine flood	10^{-6} @ $\geq 10^{3}$ yr	
ADL (1980)/ Generic, bedded salt granite, basalt, shale, and domed salt	Undetected borehole	10^{-5} @ $\leq 10^{6}$ yr 10^{-4} @ $\leq 10^{6}$ yr 10^{-3} @ $\leq 10^{6}$ yr	Bedded salt Granite, basalt, shale domed salt
	Drilling	0 @ 10^{2} yr 10^{-2} @ 10^{2} yr	All rock types Bedded salt, shale, domed salt
		10^{-3} @ 10^{2} yr	Granite and basalt
	Faulting	10^{-5} @ 2×10^{3} yr 10^{-2} @ 10^{6} yr	Bedded salt, granite, and shale
		10^{-4} @ 10^{3} yr 10^{-1} @ 10^{6} yr	Domed salt
		10^{-3} @ 10^{3} yr 1 @ 10^{6} yr	Basalt
	Volcanism	10^{-7} @ 10^{3} yr 10^{-4} @ 10^{6} yr	Bedded salt, granite shale, and domed salt
		10^{-6} @ 10^{3} yr 10^{-3} @ 10^{6} yr	Basalt
	Igneous intrusion	10^{-7} @ 10^{3} yr 10^{-4} @ 10^{6} yr	Bedded salt, granite, shale
ADL (continued)		10^{-6} @ 10^{3} yr 10^{-3} @ 10^{6} yr	Domed salt
		10^{-5} @ 10^{3} yr 10^{-2} @ 10^{6} yr	Basalt
	Meteorite impact	10^{-7} @ 10^{3} yr 10^{-4} @ 10^{6} yr	
	Breccia pipe	0 @<500 yr 10^{-8} @ 500 yr 10^{-5} @ 10^{3} yr 10^{-2} @ 10^{6} yr	Bedded salt only Bedded salt only Bedded salt only Bedded salt only

Table 2-31. REFERENCE STUDY COMPILATIONS FOR IDENTIFICATION AND CHARACTERIZATION OF POTENTIALLY DISRUPTIVE RELEASE SCENARIOS FOR A WASTE REPOSITORY AT HANFORD

Subject	Reference
Generic list of disruptive events and processes	Wallace and others, 1980, 1982; Benson, 1981; Stottlemyre and others, 1980; Greenborg and others, 1978; Arnett and others, 1980; Lee and others, 1978
Scenarios parametrically characterized by consultants	Davis, 1980; Mara, 1980; Bull, 1980; Bull and others, 1981; Johnpeer and others, 1981; Benson, 1981; Lee and others, 1978; Scott and others, 1979; Logan and others, 1982; Crowe, 1980; Leaming, 1981; Murphy and
Johnpeer, 1981	
Duration of consideration	Wallace and others, 1980
Modeled release consequences—bounding or nonbounding	Wallace and others, 1980
Whether scenario could lead to breach of repository	Wallace and others, 1980
Whether phenomena are natural or human-induced	Arnett and others, 1980; Lee and others, 1978; Benson, 1981; Stottlemeyre and others, 1980; Greensborg and others, 1978
Probability of event or process occurrence	EPA, 1982; NRC, 1982
Seismicity and faulting	Rothe, 1978; Brown, 1937; Campbell and Bentley, 1981; Smith, 1976; Slemmons and O'Malley, 1980; Slemmons, 1977; Farooqui, 1980, 1979; Malone, 1979a, 1979b; Caggiano and others, 1980
Volcanism	Johnpeer and others, 1981; Crandell and Mullineaux, 1975; Waters, 1973; Shannon and Wilson, 1976; Crowe, 1980
Flooding	Wallace and others, 1980, 1982; Skaggs and Waters, 1981; U.S. Army Corps of Engineers, 1969, 1951
Effects or erosion, denudation, uplift, and subsidence	Woodward-Clyde Consultants, 1982
Meteorite impacts	Wallace and others, 1980, 1982

As inferred in the preceding section of this report, solicitation of expert opinion affords an alternate approach to the identification, selection and classification of disruptive-release scenarios according to relative occurrence probabilities. Such an appraoch has been employed by Rockwell Hanford Operations for a preliminary evaluation of potential release scenarios at the Hanford Site (Davis et al, 1983).

After considering several methods of scenarios selection, Rockwell chose the Delphi approach to expert opinion consensus-forming. The Delphi approach (Helmer, 1966; Dalkey, 1972; Linstone and Turoff, 1975) has been widely used to elicit expert opinion about future developments or conditions. Briefly, the procedure consists of the selection of a panel of experts with knowledge of the questions at hand and the interative adminstration of a questionnaire to each panel member. The questionnaire is usually administered by mail and/or in a personal meeting; strict anonymity of individual panelist responses is thus maintained. Each administration of the questionnaire is normally referred to as a round or stage. After each round, the responses are collated and summarized for the use of the panelists in reconsidering their earlier responses during the next round. Ideally, this process continues until consensus is reached or opinion stabilizes. Responses of the final round are then summarized to determine the degree of consensus. Thus, the Delphi approach, a systematic auditable means of soliciting informed expert opionion originally developed by the Rand Corporation, was adopted by Rockwell for the following reasons:

- This approach clearly is the only means of analysis considered that is compatible with the variable degree of data reliability and imperfect understanding of various hydrologic or geologic processes and their predictability.

- This method permits an unbiased consensus of recognized expert opinion to be obtained that allows for anonymous but auditable documentation of opinion variation and/or consensus. The approach is well suited for interdisciplinary analysis of complex problems.

- Expert opinion methods are a direct means of identifying, in a defensible manner, disruption scenarios for use in quantitative modeling of potential releases. Current NRC and EPA rules and proposed standards form a framework for approaching disruptive scenario analysis that is suitable for a Delphi methodology.

- Past uses of Delphi methods for long-range forecasting confirm its widespread acceptance by the public and by scientific communities for guidance on questions in the area of public policy.

- The approach and its implementation can be structured to minimize weaknesses arising from (1) disagreements of the scope of panelist expertise, (2) bias due to irrelevant factors in selection of panelists, such as personal preferences, and (3) differing premises used by members of the panel in reaching decisions.

Therefore, the Delphi Method was utilized to evaluate a set of 45 potentially disruptive processes and/or events for the four classes or families of possible site-specific disruptive release scenarios previously outlined in Sections 2.2.2 through 2.2.4 of this report.

A total of 15 separate potentially disruptive processes and/or events are presented in table 2-32 that were evaluated by the Delphi panel for the class of scenarios associated with site characterization omissions and uncertainties (Family 1). Similarly, 16 separate items were subjected to evaluation by the Delphi panel for that class of disruptive scenarios related to natural phenomena (Family 2) at the Hanford Site as shown in table 2-33. Likewise tables 2-34 and tables 2-35 list nine and five processes and/or events, respectively, for Family 3 disruptive scenarios resulting from repository construction and operation and Family 4 disruptive scenarios evolving from human activities other than repository construction and operation.

The potential undetected site conditions and disruptions or changes from nominal conditions that were identified during the Rockwell study are summarized in figures 2-29 through 2-32. Of the 45 potentially disruptive release scenarios considered in the study and itemized in tables 2-32 through 2-35, 32 were initially identified by Rockwell and an independent peer review panel based upon a survey of available information most of which has been either been discussed or referenced in the preceding text. The other 13 scenarios were identified by the Delphi panelists during three consensus-forming rounds of questionnaire circulation. For a total of 45 scenarios thus identified, the occurrence probabilities of 26 were agreed upon by 75 percent or more of the panelist expressing an opinion. Of these, a complete consensus was reached in categorizing the site-specific occurrence probabilties of 9 scenarios as summarized in figures 2-29 through 2-32.

Table 2-32. PROCESSES AND EVENTS CONSIDERED IN DISRUPTION FAMILY 1 SCENARIOS - SITE CHARACTERIZATION OMISSIONS AND UNCERTAINTIES-BASALT WASTE REPOSITORY AT HANFORD SITE-DELPHI METHOD

Item No.	Description
1	Undetected flow breccia of areal extent less than 1/2 mi^2 which adversely affects groundwater traveltimes.
2	Undetected fault with movement periodicity greater than 1/10,000 years.
3	Premature shaft seal failure resulting from No. 2.
4	Undetected flow breccia of areal extent greater than 1/2 mi^2 which adversely affects groundwater traveltimes.
5	Undetected fault with movement periodicity less than 1/10,000 years.
6	Premature shaft seal failure resulting from No. 5.
7	Undetected major fault within the site.
8	Estimation uncertainty of greater than one order of magnitude in hydraulic conductivity.
9	Estimation uncertainty of greater than one order of magnitude in radionuclide-rock partition coefficient.
10	Estimation uncertainty of greater than one order of magnitude in the state of fracturing of undisturbed rock.
11*	Glaciation.
12*	Volcanism.
13*	Latent seismic activity triggered by changes in hydraulic pressure and rock stress.
14*	Estimation uncertainty of greater than one order of magnitude in convective dispersion through fracture or interflow systems.
15*	Fault movement with periodicity greater than 1/10,000 years that could intersect the repository.

* Item added by panelist during Round 1. Family assignment made by panelist.

** Item added during Round 2.

Table 2-33. PROCESSES AND EVENTS CONSIDERED IN DISRUPTION FAMILY 2 SCENARIOS-NATURAL SYSTEM DYNAMICS-BASALT WASTE REPOSITORY AT HANFORD SITE-DELPHI METHOD

Item No.	Description
1	Seismicity of less than 6.7 magnitude, with faulting.
2	Seismicity of greater than 6.7 magnitude, with faulting.
3	Celestial impacts.
4	Seismically induced failure of shaft seals due to faulting.
5	Groundwater chemistry changes with adverse effects on radionuclide flux.
6	Breach or premature failure due to net effects of surficial geologic processes.
7	Intrusive igneous activity.
8	Microseismicity with host rock fracturing.
9	Collapse of repository waste into undetected voids, such as lava tubes.
10	Diapirism of rock underlying repository host rock (e.g., shale, serpentine, evaporites).
11	Change in transport properties causing a decrease of more than 50% in groundwater traveltimes.
12	Adverse effects on groundwater traveltimes due to adverse effects on recharge, because of severe changes in precipitations.
13	Adverse effects on groundwater traveltime due to adverse effects on recharge, because of accelerated erosion or sedimentation.
14	Change in the course of the Columbia River that adversely affects the site's hydrologic system.
15*	Climate change; development of glaciers and ice sheets in the region.
16*	Adverse effects on groundwater traveltime due to adverse effects on recharge, because of severe changes in the local water budget (i.e., evaporation/precipitation ratio).

* Item added by panelist during Round 1.

Table 2-35. PROCESSES AND EVENTS CONSIDERED IN DISRUPTION FAMILY 4 SCENARIOS-HUMAN ACTIVITIES OTHER THAN REPOSITORY CONSTRUCTION AND OPERATION-BASALT WASTE REPOSITORY AT HANFORD SITE-DELPHI METHOD

Item No.	Description
1	Nuclear fuel recovery by deep mining methods.
2	Irrigation or other human-induced perturbation of the hydrologic system resulting in adverse groundwater system changes.
3*	Inadvertent entry by deep drilling.
4*	A combination of human error during construction or commissioning of the repository, together with natural disasters, such as an earthquake, that may possibly result in a permanent state of disrepair and radioactive release that would be excessively hazardous to remedy.
5*	Breaching by nuclear weapons.

* Item added by panelist during Round 1.

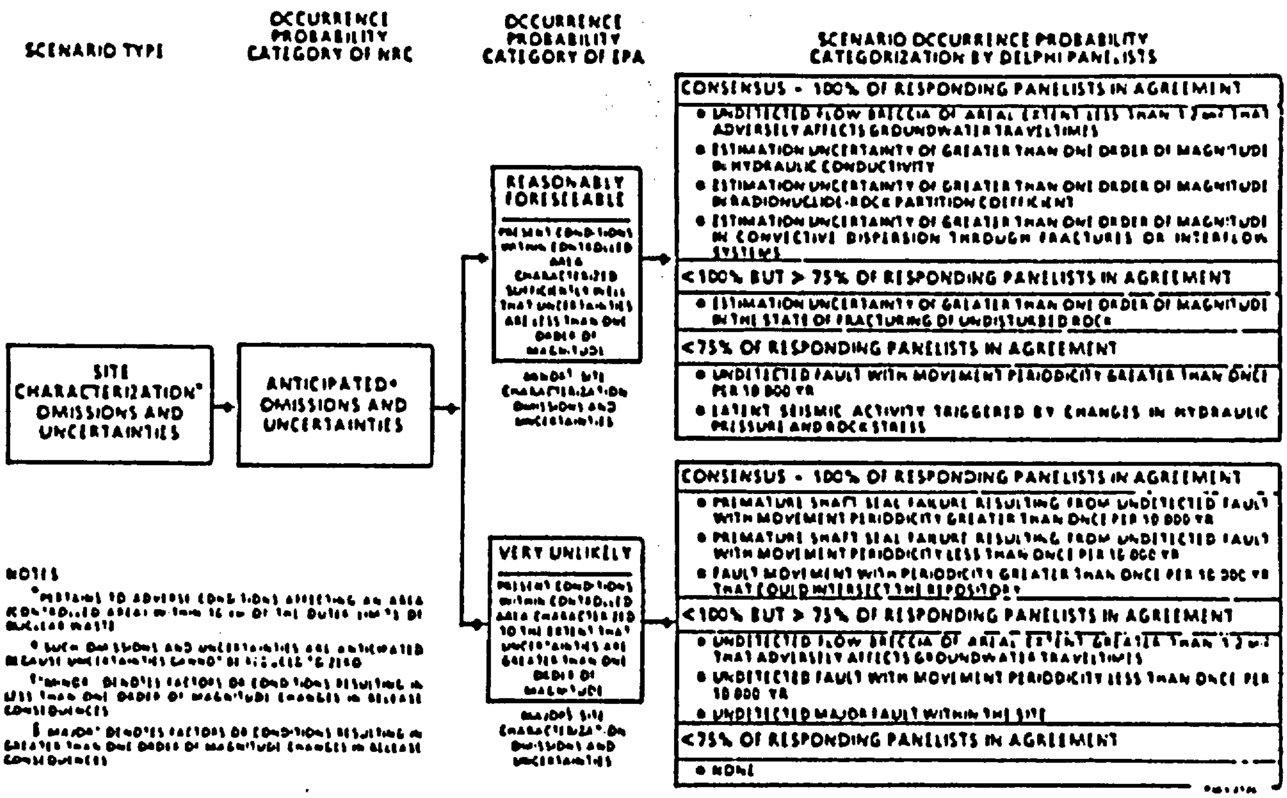


Figure 2-29. SUMMARY OF DELPHI PANEL DISTRIBUTION FOR RELATIVE OCCURRENCE PROBABILITIES OF POTENTIAL SITE CHARACTERIZATION OMISSIONS AND UNCERTAINTIES–HANFORD SITE BASALT WASTE REPOSITORY

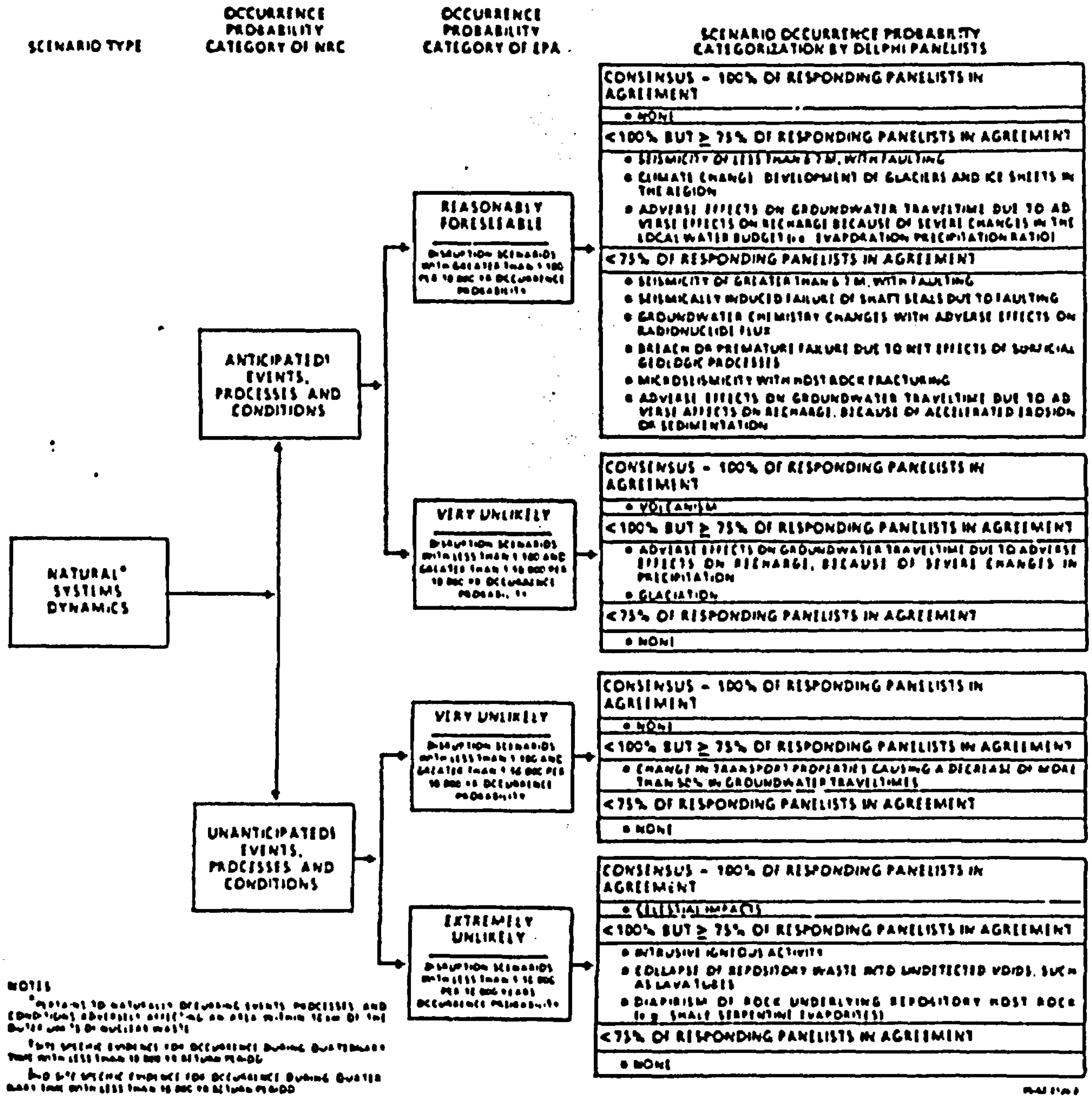


Figure 2-30. SUMMARY OF DELPHI PANEL DISTRIBUTION FOR RELATIVE OCCURRENCE PROBABILITIES AT HANFORD BASALT WASTE REPOSITORY SITE FOR SITE SPECIFIC DISRUPTION SCENARIOS DUE TO NATURAL SYSTEM DYNAMICS

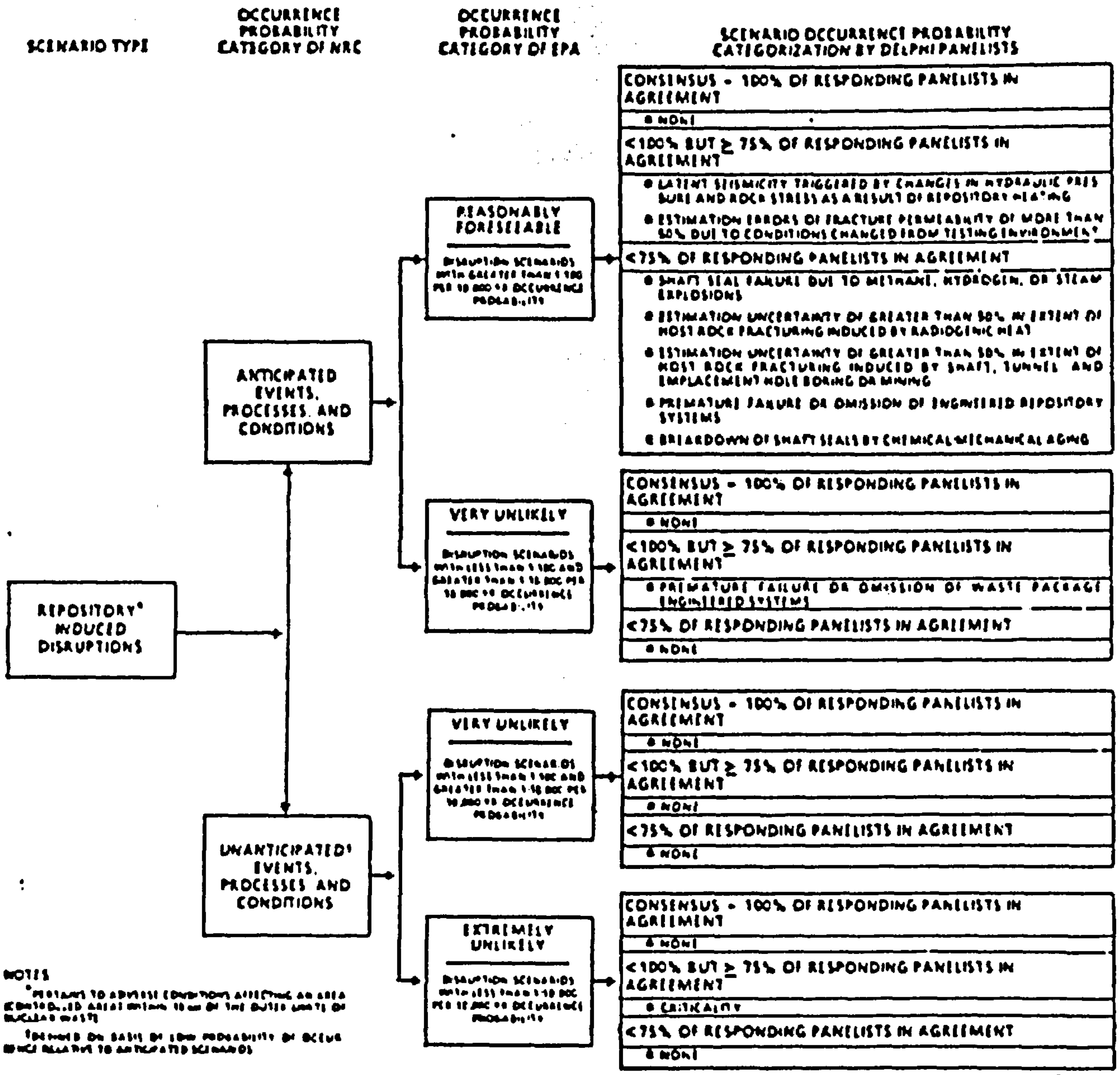


Figure 2-31. SUMMARY OF DELPHI PANEL DISTRIBUTION FOR PROBABILITIES OF SITE-SPECIFIC DISRUPTION SCENARIO OCCURRENCE-REPOSITORY CONSTRUCTION AND OPERATION-HANFORD SITE BASALT WASTE REPOSITORY

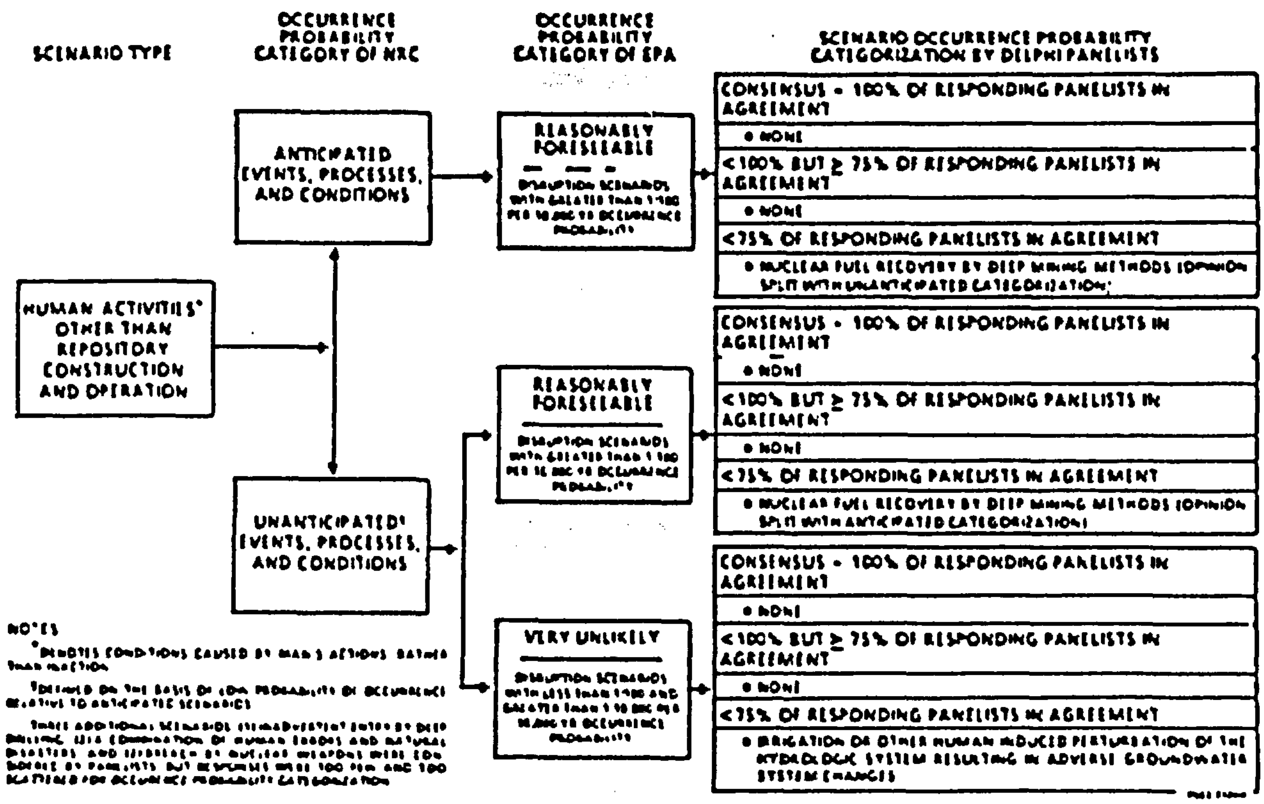


Figure 2-32. SUMMARY OF DELPHI PANEL DISTRIBUTION FOR PROBABILITES OF SITE-SPECIFC DISRUPTION SCENARIO OCCURRENCE-HUMAN ACTIVITIES OTHER THAN REPOSITORY CONSTRUCTION AND OPERATION-HANFORD SITE BASALT WASTE REPOSITORY

Considering each of the four types of scenarios in turn, the Delphi panelists reached majority agreement in selecting the following conditions, events, and processes as being potentially most adverse to repository isolation performance for each of the five occurrence probability categories (listed in order of decreasing consensus and decreasing probability):

- Reasonably Foreseeable - Anticipated

  1. Estimation uncertainties of greater than one order of magnitude in hydraulic conductivities (selected by 12 of 15 panelists, with 1 abstention) — Uncertainties and Potential Omissions Associated with Characterization of the Candidate Site.

  2. Estimation uncertainty of greater than 50% in host rock fracturing induced by shaft, tunnel, and emplacement hole boring or mining (selected by 8 of 15 panelists, with 2 abstentions) — Potential Disruptions Resulting from Repository Construction and Operation.

  3. Seismicity of less than 6.7 magnitude, with faulting (selected by 7 of 15 panelists, with 2 absentions) — Potential Disruptions Due to the Dynamics of Natural Site System.

  4. Nuclear fuel recovery by deep mining methods (selected br 5 of 15 panelists, with 5 absentions) — Potential Disruptions Induced by Human Activity Other than Repository Construction and Operation.

- Reasonably Foreseeable - Unanticipated

  1. Nuclear fuel recovery by deep mining methods (selected by 4 of 15 panelists, with 9 absentions) — Potential Disruptions Induced by Human Activity Other than Repository Construction and Operation (split of opinion with No. 4 above).

- Very Unlikely - Anticipated

 1. Undetected flow breccia of areal extent greater than 1/2 mi^2 (selected by 4 of 15 panelists, with 9 abstentions) — Uncertainties and Potential Omissions Associated with Characterization of the Candidate Site.

- Very Unlikely - Unanticipated

 1. Change in transport properties causing a decrease of more than 50% in groundwater traveltime (selected by 10 of 15 panelists, with 2 abstentions) — Potential Disruptions Due to the Dynamics of Natural Site System.

 2. Irrigation or other human-caused perturbation of the hydrologic system (selected by 4 of 15 panelists, with 7 abstentions) — Potential Disruptions Induced by Human Activity Other than Repository Construction and Operation.

- Extremely Unlikely - Unanticipated

 1. Criticality (assumes unreprocessed spent fuel in the repository) (selected by 5 of 15 panelists, with 9 abstentions) — Potential Disruptions Resulting from Repository Construction and Operation.

Considering all four types of scenarios collectively, panelists expressing an opinion reached majority agreement on the scenario judged likely to have the most adverse radionuclide release potential for three of the five occurrence probability categories. No clear majority was reached for the other two categories. The scenarios selected are as follows:

- Reasonably Foreseeable - Anticipated

 Estimation uncertainties of greater than one order of magnitude in hydraulic conductivities (selected by 10 of 15 panelists, with no abstentions).

- Very Unlikely - Anticipated

  Undetected flow breccia of areal extent greater than 1/2 mi^2 (selected by 7 of 15 panelists, with 2 abstentions).

- Very Unlikely - Unanticipated

  Change in transport properties causing a decrease of more than 50% in groundwater traveltime (selected by 9 of 15 panelists, with 2 abstentions).

Therefore, the disruption scenarios that were judged by the Delphi panel to be the most likely to occur and to be potentially the most adverse to repository isolation performance must be considered the most representative set of potentially disruptive release scenarios developed to date for the proposed high-level nuclear waste repository at the Hanford Site. However, this selection of an initial group of disruption scenarios must be guided by the precept that the scope and depth requirements of completed consequence analyses will be developed by means of an iterative process that will be discussed in more detail in a later section of this report. Disruptions or changes from nominal conditions that have low probability of occurrence and high risk potential must be considered as should those disruptive scenarios having high probability of occurrence and low risks potential, since, ultimately, it is the product of these two factors that is of prime importance in total repository system risk assessments.

3.0 CHARACTERIZATION OF ENVIRONMENTAL CONCENTRATIONS

Once a set of release scenarios has been systematically developed and characterized for the two major classes of releases, i.e., (1) those associated with the high-level nuclear waste transportation system, and (2) those related to the construction, operation, permanent closure, and long-term waste storage of the proposed subsurface geologic repository, the transport of both radiological and non-radiological pollutants to the surrounding natural environment becomes a major consideration in the development of an overall methodology to assess potential risks.

Environmental surveillance at major nuclear facilities in the United States has constituted a significant part of the total environmental programs for more than thirty years. These programs have been used to provide data for scientific studies; to provide a supplementary check on the adequacy of containment and effluent controls; to determine compliance with applicable regulatory protection guides and standards; and, to assess actual environmental impacts, if any, on the specific site natural environment. Although inter-site sharing of concepts and methodology has occurred, the various facility management organizations over the years have exercised considerable latitude in tailoring the scope and methodology to meet, for example, the particular environmental monitoring needs at each individual site. However, the differing nature of various site operations and histories, the major differences in environmental conditions, exposure pathways and potentials, and the differences in the methodology of data collection at the various sites have made either area-wide or region-wide correlation of environmental results very difficult. In many instances, differences in methodology are a matter of professional judgment, and several adequate solutions may be available to resolve the same basic problem. Therefore, considering the relatively large land area of the Umatilla Reservation and its ceded lands as shown previously in in figure 2-2 of Section 2.1.1 of this report, it is essential that the Tribe give appropriate attention to the development of a environmental surveillance program to aid in the assessment of potential environmental impacts to their lands as a result of a high-level nuclear waste repository possibly being located at the proposed Hanford Site.

3.1 ENVIRONMENTAL SURVEILLANCE PROGRAM PLANNING OBJECTIVES AND RATIONALE

Any list of objectives for a tribal environmental program would include the following, in approximate order of importance for protection of the tribe.

- Evaluation of the adequacy and effectiveness of the containment and effluent control systems applied to facilities and operations at the proposed Hanford Repository Site.

- Detection of rapid changes and evaluation of long-term trends of concentrations in the environment, with the intent to detect failure or lack of proper control of releases and to initiate appropriate actions.

- Assessment of the actual or potential doses to man from radioactive materials or radiation released to the environment as a result of the DOE repository program or the estimation of the probable limits of such doses.

- Collection of data bearing on the history of contaminants released to the environment, especially with the intent of discovering previously unconsidered pathways and modes of exposure.

- Maintenance of a data base and capabilities for rapid evaluation and response to unusual releases of radioactivity.

- Detection and evaluation of both radioactive and non-radioactive contaminants from offsite sources in order to distinguish and compare the results of site operations.

- Demonstration of compliance with applicable regulations and legal requirements concerning contaminant releases to the environment.

Despite the second statement above, the time lag and generally lower concentrations in most environmental measurements make primary reliance on an environmental measurement as an action signal unwise other than for purposes of further investigation. With the exception of long-term accumulation of contaminants from source terms too dilute to be conventionally measured, all environmental measurements should be considered as an important supplement to effluent monitoring or other repository facility or process measurements. Even for record purposes, all environmental measurements may be considered as being theoretically redundant, in that complete and continuous control and measurement of all effluent releases, together with adequate knowledge of the subsequent history of radionuclides in the tribal environment, would make them unnecessary. In practice, some environmental measurements are vital, in part to demonstrate compliance with the stated objectives and in part because prior knowledge of the eventual fate of every potential contaminant released is incomplete. Measurements representing as much as possible the actual exposure vectors to people should therefore provide a more accurate though less precise, environmental dose estimate. In the extreme for radioactive contaminants, the latter would call for extensive dosimeter use by, and in-vivo monitoring of, the tribal population - a solution which would probably not be considered practical on a routine basis.

The natural environment is dynamic and heterogeneous, showing both spatial and temporal variations of nearly all constituents. It is impractical, for instance, to measure radioactivity routinely in all environmental media or even thoroughly in any one exposure pathway. For example, air sample networks around any point of release only rarely intercept as much as one millionth of the air which streams past; the fraction of a river taken for analysis may be similarly small. In consequence, radioactivity in the environment is generally characterized based on what is known or calculated with respect to contaminant release distributions in the environment.

A generic procedural flow diagram for an environmental surveillance program design process is presented in figure 3-1 as an aid to placing the required data inputs and the environmental pathway analysis procedures in the proper relationship to program planning. In figure 3-1, rectangles indicate data inputs, diamonds procedural steps. The many different kinds of data that must either be provided or estimated are apparent. Both the magnitude and complexity of the foregoing relationships are further illustrated in figure 3-2, which depicts the detailed radiation dose calculation procedure that must be incorporated into a predictive mathematical model for estimates of environmental concentrations in regions encompassed by a nuclear facility.

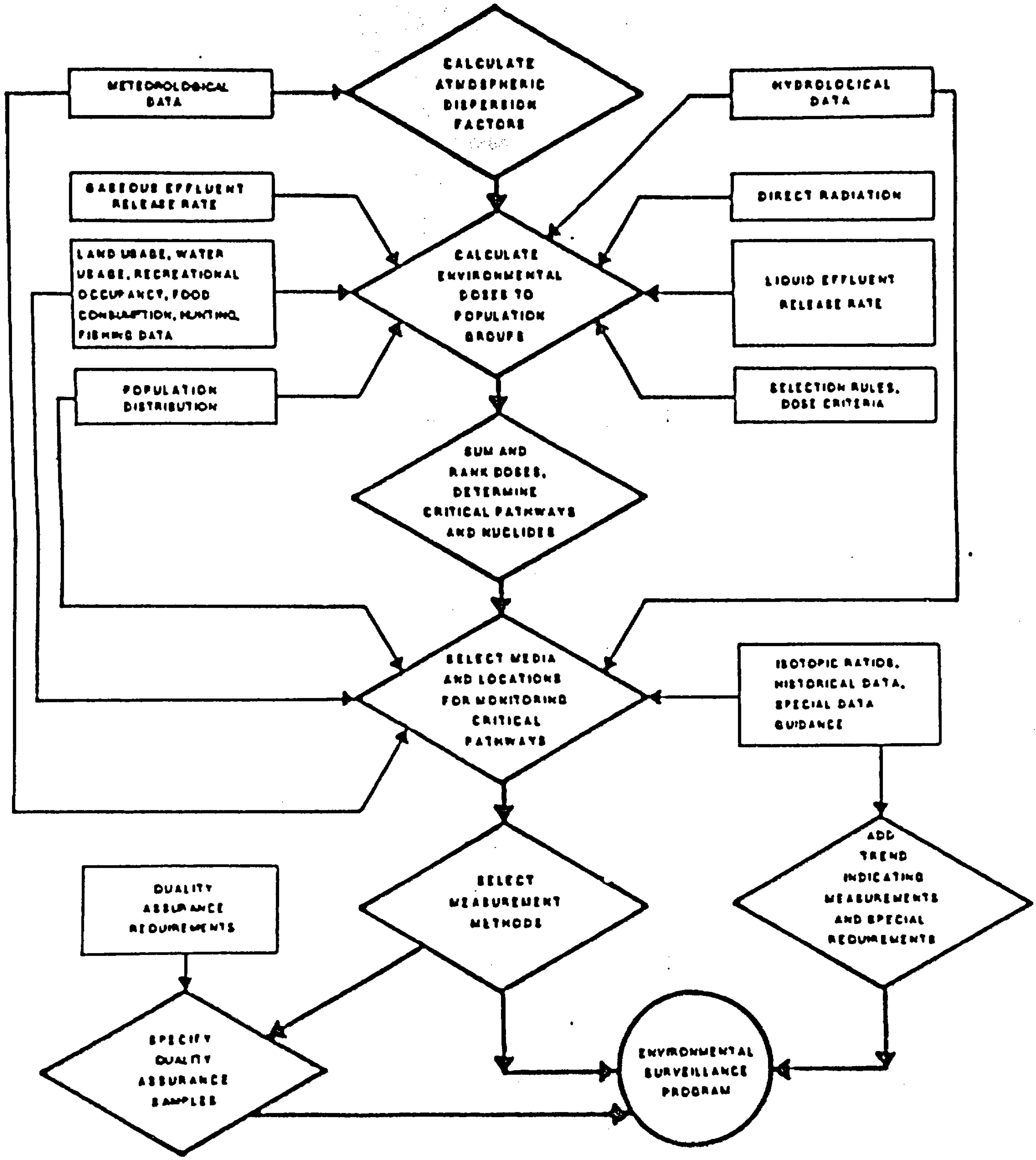


Figure 3-1. ENVIRONMENTAL SURVEILLANCE PROGRAM DESIGN PROCESS

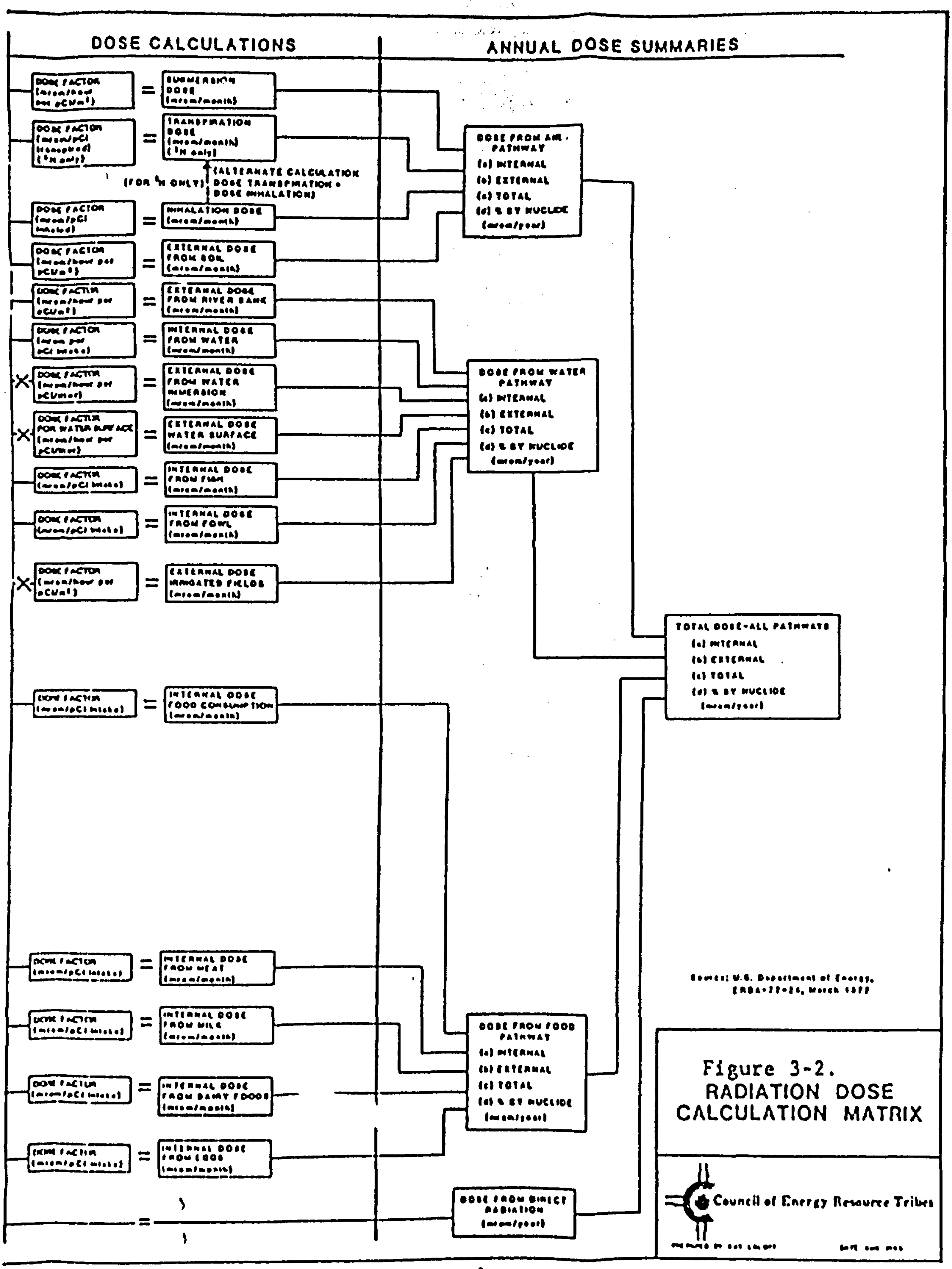
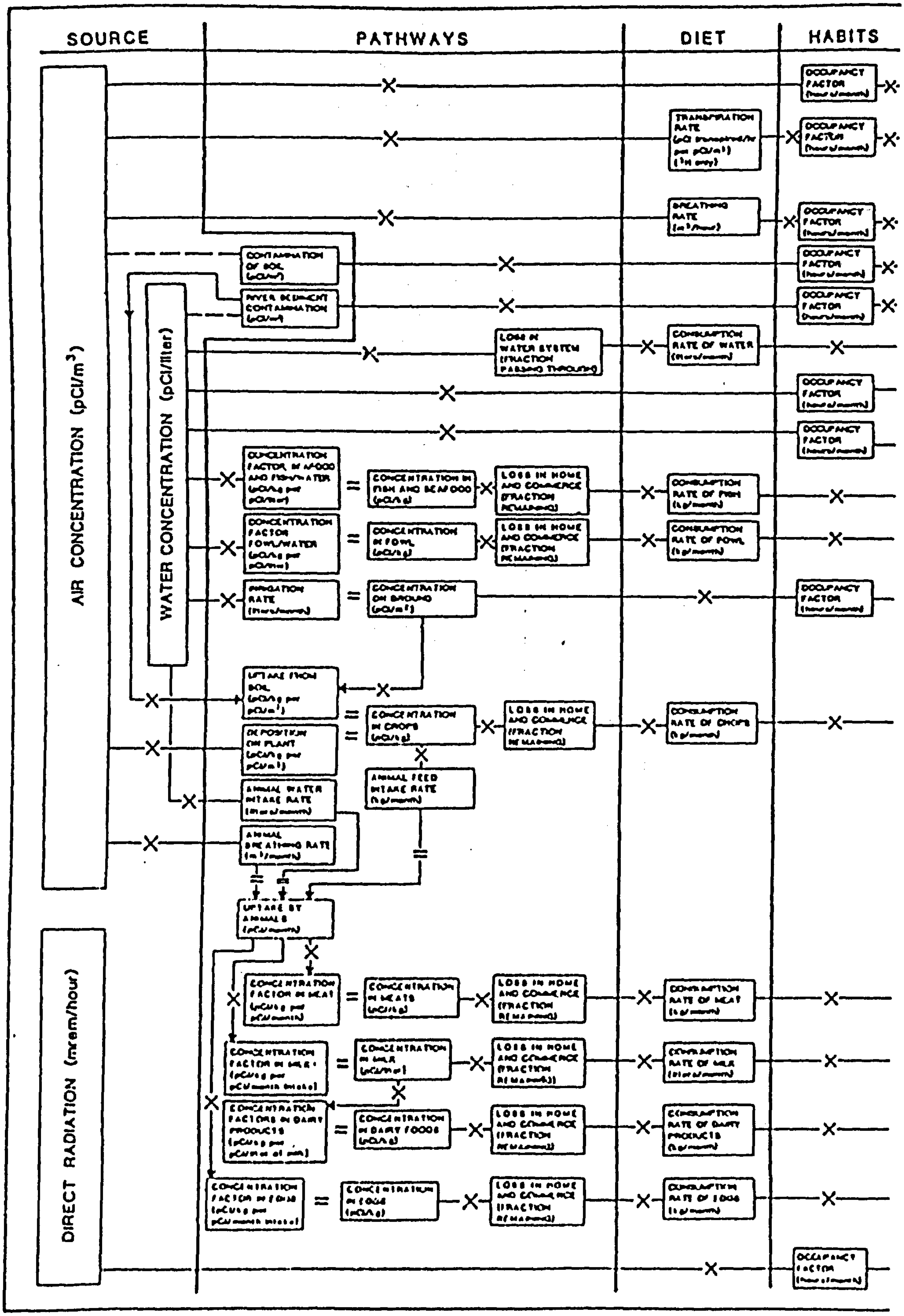


Figure 3-2.
RADIATION DOSE CALCULATION MATRIX



SOURCE
PATHWAYS
DIET
HABITS
AIR CONCENTRATION (pCi/m³)
WATER CONCENTRATION (pCi/liter)
DIRECT RADIATION (mrem/hour)
OCCUPANCY FACTOR
TRANSPIRATION RATE
BREATHING RATE
CONTAMINATION OF SOIL
RIVER SEDIMENT CONTAMINATION
LOSS IN WATER SYSTEM
CONSUMPTION RATE OF WATER
CONCENTRATION IN FISH AND SEAFOOD
CONSUMPTION RATE OF FISH
CONCENTRATION IN FOWL
CONSUMPTION RATE OF FOWL
IRRIGATION RATE
CONCENTRATION ON GROUND
UPTAKE FROM SOIL
DEPOSITION ON PLANT
CONCENTRATION IN CROPS
LOSS IN HOME AND COMMERCE
CONSUMPTION RATE OF CROPS
ANIMAL WATER INTAKE RATE
ANIMAL FEED INTAKE RATE
ANIMAL BREATHING RATE
UPTAKE BY ANIMALS
CONCENTRATION FACTOR IN MEAT
CONCENTRATION IN MEATS
CONSUMPTION RATE OF MEAT
CONCENTRATION FACTOR IN MILK
CONCENTRATION IN MILK
CONSUMPTION RATE OF MILK
CONCENTRATION FACTORS IN DAIRY PRODUCTS
CONCENTRATION IN DAIRY FOODS
CONSUMPTION RATE OF DAIRY PRODUCTS
CONCENTRATION FACTOR IN EGGS
CONCENTRATION IN EGGS
CONSUMPTION RATE OF EGGS

Since all the basic radiation regulatory standards currently promulgated are given in terms of a radiation dose to people, the environmental program planning process must be addressed to the sampling, direct measurement, and/or predictive mathematical modeling of critical environmental pathways which may contribute to the radiation exposure to the public. In this context, the "critical" path (nuclide, organ, population group) is defined according to standard usage as the pathway (nuclide, organ, population group) providing the largest percentage of the applicable dose criterion. Selection of locations, frequency, media, and nuclides to be measured, and measurement methods to be used for critical pathway surveillance provides the basic requirements for the environmental surveillance program. To these will be added any special monitoring requirements, including trend indicators and those additional samples, measurements, and/or mathematical analyses which should be recorded so that the purpose and any limitations or interpretations of results will be clear.

In general, two major transport media, air and water, are of primary concern in any characterization of environmental concentrations derived from contaminant releases for any environmental surveillance program as shown graphically in figures 3-3 and 3-4. Figure 3-3 depicts the major pathways of atmospheric routing from contaminant releases irrespective of the specific release scenario. Similarly, figure 3-4 illustrates the major pathways of hydrologic routing from most contaminant release scenarios. Since the initial or baseline environmental measurement and monitoring program is being developed for the CTUIR in considerable detail within a specific task for the FY1986 high-level nuclear waste program, emphasis in this report will be placed on development of atmospheric and hydrologic mathematical modeling techniques to quantify potential transport and eventual environmental concentrations of radioactive releases at significant receptor locations on the CTUIR and its ceded lands from credible disruptive release scenarios as a consequence of various physical processes and events that are currently planned for within the present nuclear waste repository program.

In this regard, atmospheric release and subsequent transport are presently envisioned as being the most likely major category of disruptive accident scenarios resulting from the operation of a high-level nuclear waste transportation system as discussed previously in Section 2.1 of this report. Similarly, it is generally agreed that the most likely pathway by which wastes could be released from a subsurface geologic repository is transport to the surface by groundwater. Therefore, various atmospheric and hydrologic dispersion models either presently being planned, under current research and development, or

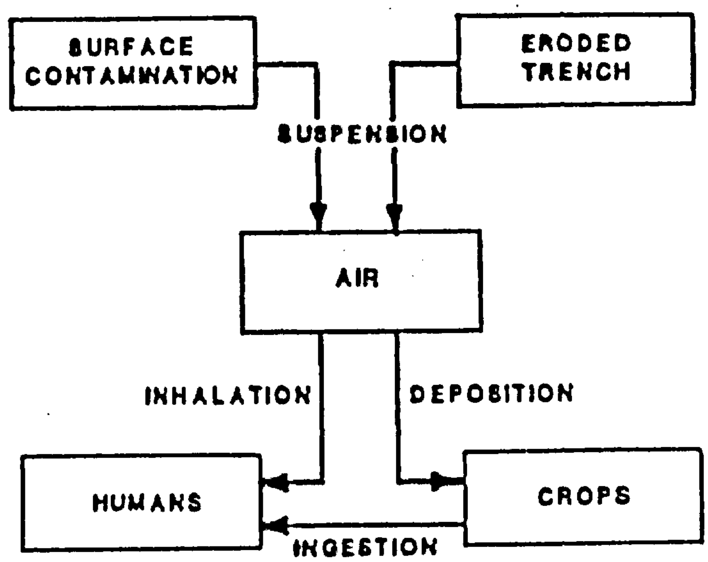


Figure 3-3. MAJOR PATHWAYS OF ATMOSPHERIC ROUTING

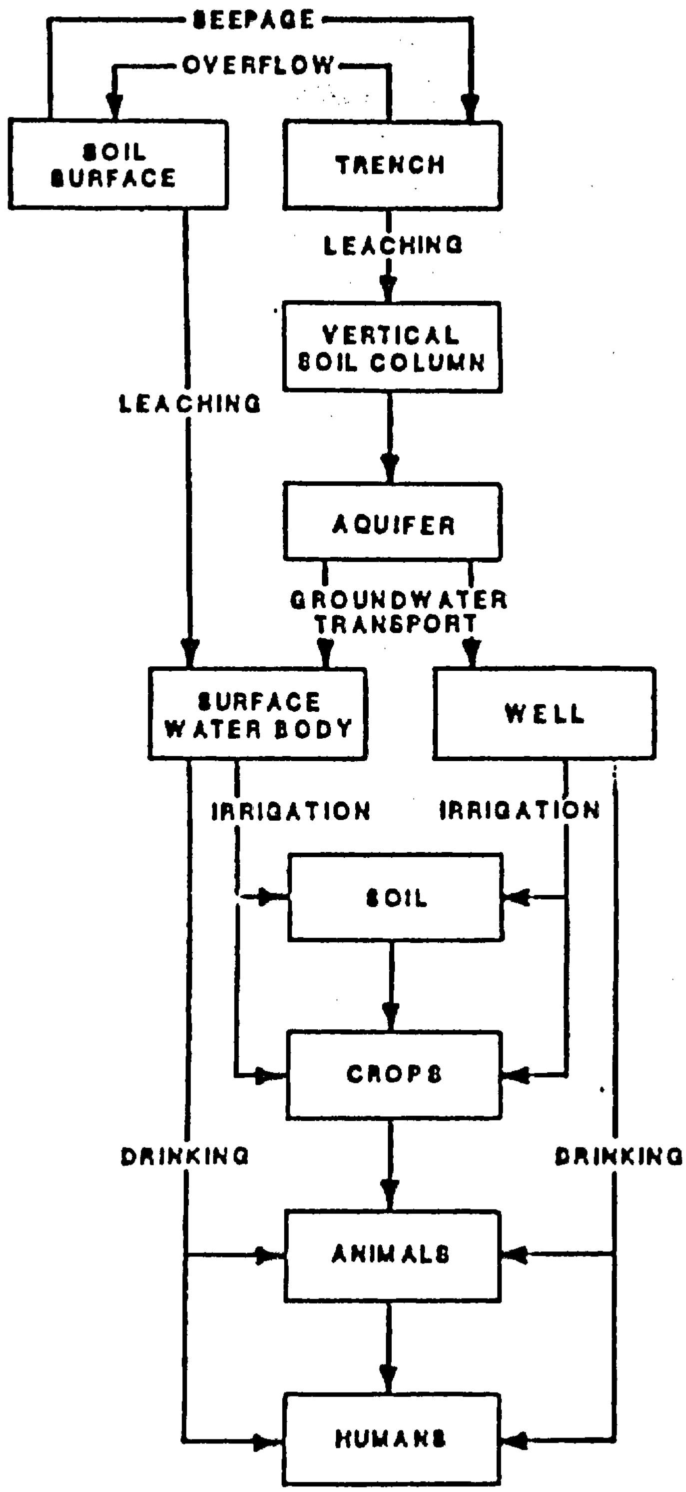


Figure 3-4. MAJOR PATHWAYS OF HYDROLOGIC ROUTING

being actively utilized to predict environmental concentrations, will be evaluated for possible incorporation into the overall risk assessment methodology under development for the CTUIR high-level nuclear waste repository program. It is contended that the above approach should be more effective in terms of both development time and cost for this element of the overall composite of mathematical modeling techniques that should ultimately comprise the risk assessment methodology for detailed analysis of the most likely disruptive release scenarios that could deleteriously impact the CTUIR and its ceded lands.

3.2 ATMOSPHERIC DISPERSION AND TRANSPORT MODELS

Atmospheric dispersion and transport models can be generically classified into two categories primarily according to source-to-receptor distances. Most computer-based near-field models utilize various mathematical forms of the Gaussian distribution to account for plume dispersion from a point, line, or area sou 'e.

Appropriately averaged meteorological parameters (wind speed, wind direction and wind stability) are input source terms in the mathematical model to simulate the magnitude, direction and geometric characteristics of the contaminant dispersion into the atmosphere as a function of time from the origin of release. Physical characteristics of the source pollutant, i.e., physical state (gas, solid or particulate or aerosol) and temperature (at the point of release) are also incorporated as source term inputs to the mathematical model.

Two of three-dimensional geometric coordinate systems are generally employed for both the source and receptor locations. Dispersed contaminant quantities or doses are generally calculated to either a point or sector-. 'eraged area receptor location. Two-dimensional coordinates geometries for the source and receptor locations assume that both the source and receptor are at the same elevation with respect to the ground surface and are generally termed "flat-terrain" models. Conversely "complex terrain" models account for variation in ground elevation between source and receptor locations as the plume disperses. Thus, complex terrain models mathematically attempt to account for alterations in the absolute quantities of the airborne contaminants in the plume as well as changes in the velocity and direction as a consequence of significant changes in elevations (hillsides, mountains, ranges, etc.) along the predicted line-of-sight traverse between the source and receptor.

Although the vast majority of the Gaussian models presently are designed to simulate releases from elevated stacks of plant processing facilities, some of these models do provide the necessary mathematical descriptions in the source input parameters to accomodate near surface or ground surface contaminant releases. This will be an important consideration in the development of the overall tribal risk assessment methodology since, for example, potential releases from transportation accident scenarios on the reservation will occur at or near the ground surface.

In general, it has been established that well designed atmospheric dispersion models, employing a Gaussian distribution can accurately predict environmental concentrations of possible contaminants with source-to-receptor distances of 10-12 km (6.25 - 7.5 miles) in flat terrain. However, most Gaussian atmospheric dispersion models are commonly utilized by both government and industry analysts to predict airborne environmental concentrations at source-to-receptor distances of 80 km (50 miles) or more in flat terrain with the attendant uncertainty inherent in the mathematical model. Fortunately, the vast majority of "benchmark" model comparisons and experimental comparisons with modeling predictions have erred in the conservative direction. That is to say that predictive modeling environmental concentrations at specific receptor locations generally have been higher in most instances than experimentally-derived contaminant concentrations at identical source-to-receptor distances under similarly controlled conditions. However, the degree of uncertainty associated with Gaussian atmospheric dispersion model predictions at source-to-receptor distances from approximately 12 km to 80 km or greater can vary widely. Therefore, predicted vs actual concentrations can vary from 25 to 35 percent to several orders of magnitude depending on such factors as local meteorological conditions, terrain considerations, nature of contaminant release, etc.

At source-to-receptor distances appreciably greater than 80 km (50 miles) different mathematical atmospheric dispersion modeling techniques have been applied more successfully with reportedly more precise results. Most of the "regional" atmospheric dispersion employ a "puff" "particle-in-cell" technique. Generically puff particle-in-cell models solve the three-dimensional advection - diffusion equation in its flux conservative form (pseudo velocity technique) for a given mass-consistent advection field by finite difference approximations in Cartesian coordinates. The particle-in-cell technique represents the pollutants source concentrations statistically by imbedded Lagrangian marked particles in an Eulerian grid. The more sophisticated models of this type generally have two versions: (1) a moving, expanded grid version and a fixed grid version. The moving, expanding grid version is primarily designed to model single puff releases with the grid automatically expanding and traveling with the puff along its trajectory in the free atmosphere. This version is particularly suitable for the study of free-air bursts or the regional or long-range study of surface contaminant releases. The fixed grid version is suitable for the study of instantaneous of continuous releases near or on the surface over a range of scales, loosely defined from close-in or about 5 to 10 km (3 to 6 miles) to mid-range or 10 to 200 km (6 to 125 miles).

Finally, most of the more generally-accepted atmospheric dispersion models utilized presently to analyze airborne releases from nuclear facilities incorporate radiation dosimetry models into the overall dispersion and transport analysis model since the basic regulatory standards that govern the allowable environmental radionuclide concentrations at specified receptor locations are promulgated on the basis of currently acceptable levels of radioactivity in terms of human health effects. The radiation dosimetry included in the most comprehensive of these dispersion models takes into account all the major potential pathways to the human body (inhalation, ingestion) and the radiation effects of the major radionuclides that might be present in sufficient quantities to cause deleterious human health effects and/or significant damage to specific organs or areas of the body.

The general procedural flow for the numerical computations normally incorporated in a radiation dosimetry model is shown graphically within the radiation dose calculation matrix for predictive mathematical models in figure 3-2.

Although a totally comprehensive review and evaluation of all the available and currently acceptable atmospheric dispersion models was not possible within the time and monetary constraints of this project activity, a number of the more prominent atmospheric dispersion models that could be utilized to predict environmental concentrations of airborne radionuclide contaminants as a consequence of likely release scenarios either on or adjacent to the Umatilla Reservation and its ceded lands will be discussed briefly in the following section.

3.2.1 Gaussian Atmospheric Dispersion and Transport Models - General Overview

Selection of a single available, completely comprehensive atmospheric dispersion and transport model that would satisfy a detailed risk assessment methodology capable of assessing all the probable atmospheric release scenarios arising from a high-level nuclear waste repository program located at Hanford Works would not be possible at this time.

A primary reason for the prior statement stems from the fact that the most probable release scenarios have not been defined in sufficient detail as previously discussed in section 2.0. This is quite understandable in view of the overall scope and complexity of the repository program. However, several additional major factors, somewhat unique to the proposed repository location on the Hanford Reservation, also constrain the immediate selection of a "universal" atmospheric dispersion and transport model.

One of these factors is related to the general physiography of the Umatilla Reservation and its ceded lands. As previously discussed in the prior scoping study report (CERT, 1984), vast portions of tribal lands include wide variations in topography, thereby dictating the employment of a complex terrain model for prediction of radioactive contaminant concentration levels at receptor locations downwind from the location of a probable release of significant quantities of radionuclides. Assumptions of flat terrain considerations in the predictive model could possibly lead to overly optimistic results for receptor locations at CTUIR; i.e., measured concentrations conceivably could be a factor of 2 or more higher than results predicted by a flat terrain dispersion model; especially for annually averaged contaminant concentrations emanating from a continuous source of radioactive release and for release scenarios where the receptors are located in terrain at elevations of 1,000 ft or more higher than the elevation at the source of release.

Another possible constraint in choice of atmospheric dispersion models for utilization in the CTUIR risk assessment methodology is related to the format employed in the more sophisticated Gaussian plume dispersion models for the inclusion of joint frequency distributions of the major surface meteorlogical parameters; i.e., wind speed, wind direction, and wind stability. For the most part, hourly averaged measurements of these parameters are collected continuously over long-term yearly intervals at permanent monitoring station sites. Various averaging schemes can be developed for these data bases depending on the input data requirements which, in turn, are dependent on the specific method employed by a given computer-based model. The more commonly used meteorological data set compilations include the following: daily, monthly, seasonal, and annual averages of joint frequency distributions of wind, speed, wind direction, and wind stability. A possible problem area arises from the fact that the Environmental Protection Agency (EPA) over the course of the last 10 to 15 years has developed a procedural format for compilations of the foregoing key surface meteorological data base parameters that has been incorporated in the vast majority of the atmospheric dispersion computer-based models currently in use throughout both government and industry. The adoption of the EPA format has evolved primarily because EPA has the fundamental responsibilities within the federal government for promulgating environmental regulatory standards for the nuclear industry in the United States.

However, the best long-term data base compilations for surface meteorological observations in the region encompassing the proposed site for the first subsurface geologic repository at Hanford Works have been prepared by the various DOE subcontractors responsible for Hanford Operations since about 1945. Since the foregoing surface meteorological data base constitutes one of the foremost long-term compilations of its kind in the United States, it should be utilized to the maximum extent possible in the development of tribal capability for quantitive risk assessment and to provide a sound basis for correlation of all climatological and meteorological data that will be compiled as a result of the forthcoming proposed tribal baseline environmental monitoring program for the CTUIR and its ceded lands.

Specifically, the major incompatibility in the Hanford-developed surface meteorological data base with the EPA data format lies in categorization of surface wind stability regimes. Historically, the long-term Hanford compilations have utilized four wind stability regimes; i.e., (1) VS (very stable), MS (moderately stable), N (neutral) and U (unstable). As a consequence the atmospheric dispersion and transport computer-based models developed and currently utilized by the DOE Hanford subcontractors employ the foregoing classification of wind stability regimes. In contrast, the data derived primarily from observaitons taken at. National Weather Service (NWS) stations utilize six and, in some instances, seven classifications of the wind stability regime based on the foregoing EPA-approved format as follows: (1) A (extremely unstable), (2) B (moderately unstable), (3) C (slightly unstable), (4) D (neutral), (5) E (slightly stable), (6) F (moderately stable), and optionally, (7) G (extremely stable). Although both wind stability classification schemes are normally based on lapse rates determined from measured temperature differences taken at two different elevations on a meteorological tower, as illustrated in table 3-1 (CERT, 1984), conversion of an extensive long-term meteorological data base from one wind stability classification scheme to another, would require the rather laborious and expensive task of manipulating large quantities of monitored data for several climatological/meteorological parameters. Thus, consideration must be given to such differences in format at this stage of preliminary development of tribal risk assessment methodology. Consequently, several of the more sophisticated Gaussian atmospheric dispersion and transport computer-based models that have been developed by Hanford Works personnel and that also employ the aforementioned HMS wind stability format, have been critically reviewed and evaluated in this report prior to possible inclusion in the CTUIR risk assessment methodology.

Although certain incongruities exist in the classification of the surface wind speed parameters within the various Gaussian atmospheric dispersion and transport models potential manipulation of one parameter vs several represents a somewhat less formidable task in formatting joint frequency distributions for any existing or future long-term meteorological data base.

3.2.2 Computer-Based Gaussian Atmospheric Dispersion and Transport Models

A preliminary review and evaluation of currently available, computer-based Gaussian atmospheric dispersion and transport models that might possibly be utilized in the development of an overall CTUIR risk assessment methodology has been conducted. A summary of the major available computer-based models is presented in table 3-2. The summary table highlights the salient features of each Gaussian model with respect to source and receptor input characteristics, flat and/or complex terrain capabilities, meteorological format, and radiation dosimetry capabilities. Only those models are included in table 3-2 which are believed to merit further consideration on the basis of the preliminary review and evaluation conducted within the framework of this study. Table 3-2 clearly illustrates that none of the major existing, available, candidate models exhibits all of the most desirable features necessary to precisely quantify airborne environmental concentrations of radioactive contaminants over the entire area encompassing the CTUIR possessory and usage rights area for the entire spectrum of both normal and accident release scenarios that could possibly be deemed credible within the high level nuclear waste program elements and activities as currently envisioned for

Table 3-1. CLASSIFICATION OF ATMOSPHERIC STABILITIY

Hanford Meteorological Station

Stability Classification	HMS Category	Temperature Change with Height (°F/197/ ft)[a]
Very Stable	VS	$\leq$ 3.5
Moderately Stable	MS	3.4 to -0.4
Neutral	N	-0.5 to -1.4
Unstable	US	$\leq$ -1.5

Yakima, Washing NWS and Pendleton, Oregon NWS

Stability Classification	Pasquill-Gifford Category	Temperature Changes with Height (°F/197 ft)[b]
Extremely Unstable	A	$\leq$2.1
Moderately Unstable	B	-2.1 to -1.8
Slightly Unstable	C	-1.8 to -1.6
Neutral	D	-1.6 to -0.5
Slightly Stable	E	-0.5 to 1.6
Moderately Stable	F	1.6 to 4.0

a HMS temperature differences at Hanford Meteorological Tower at 200 feet and 3 feet above ground surface.

b Yakima, Washington NWS and Pendleton, Oregon NWS data on PG stability categories adjusted to reflect measurement height differential at Hanford Meteorological Tower.

Table 3-2. CURRENT GAUSSIAN ATMOSPHERIC DISPERSION AND TRANSPORT MODEL SUMMARY - PRELIMINARY CTUIR RISK ASSESSMENT METHODOLOGY EVALUATION

Model	Sponsoring Group	Source Characteristics	Receptor Characteristics	Climatological/ Meteorological Data Format	Radiation Dosimeter Capability	Terrain Features
AIRDOS (include DARTAB and RADRISK)	EPA	Point and/or area geometries, ground or elevated release, short term or continuous dispersion, finite cloud.	Point and/or sector averaged, ground or elevated location, hourly, daily, monthly, seasonal and annual concentrations, annual concentrations.	Hourly, seasonal or annual averages, 6 wind speeds, 16 cardinal compass directions, 6 PG wind stability categories in std. EPA format.	Both internal and external dosimetry for major radioisotopes of concern for both chronic and acute releases.	Flat terrain only
KRONIC	Hanford/ PNL	Point and/or area geometries, ground or elevated release, short term or continuous dispersion, finite cloud.	Point and/or sector averaged, ground or elevated location, 1 year average concentrations.	Hourly, seasonal or annual averages, 6 wind speeds, 16 cardinal compass directions, 5 Hanford wind stability categories in std. Hanford format.	Both internal and external dosimetry for major radioisotopes of concern for both chronic and acute releases. (One-year doses)	Flat terrain only
SUBDOSA	Hanford/ PNL	Point and/or area geometries, ground or elevated release, short term or continuous dispersion.	Point and/or sector averaged, ground or elevated location, 1 year average concentrations.	Hourly, seasonal or annual averages, 6 wind speeds, 16 cardinal compass directions, 5 Hanford wind stability categories in std. Hanford format.	Both internal and external dosimetry for major radioisotopes of concern from acute releases.	Flat terrain only
DACRIN	Hanford/ PNL	Point and/or area geometries, ground or elevated releases, short term or continuous dispersion.	Point and/or sector averaged, ground or elevated location, 1 year average concentrations.	Hourly, seasonal or annual averages, 6 wind speeds, 16 cardinal compass directions, 5 Hanford wind stability categories in std. Hanford format.	Internal (inhalation) dosimetry for major radioisotopes of concern from chronic or acute doses. One-year doses, dose commitments, and accumulated doses.	Flat terrain only

Table 3-2. CURRENT GAUSSIAN ATMOSPHERIC DISPERSION AND TRANSPORT MODEL SUMMARY - PRELIMINARY CTUIR RISK ASSESSMENT METHODOLOGY EVALUATION (CONTINUED)

Model	Sponsoring Group	Source Characteristics	Receptor Characteristics	Climatological/ Meteorological Data Format	Radiation Dosimeter Capability	Terrain Features
VALLEY	EPA (UNAMAP Series)	Point and/or area geometries, ground or elevated releases, short term or continuous dispersion, finite cloud.	Point and/or sector averaged, ground or elevated location, or elevated location, hourly, daily, and annual concentrations.	Hourly, seasonal or annual averages, 6 wind speeds, 16 cardinal compass directions, 6 PG wind stability categories in std. EPA format.	Provides for radio-active decay during transport on a single radionuclide basis only.	Flat and complex terrain
COMPLEX I and II	EPA (UNAMAP Series)	Point and/or area geometries, ground or elevated releases, short term or continuous dispersion, finite cloud.	Point and/or sector averaged, ground or elevated location, or elevated location, 1-hour, 3-hour, and 24-hour concentrations.	Hourly averages, 6 wind speeds, 16 cardinal compass directions, 6 PG wind stability categories in std. EPA format (Hourly data for 1 calender year.)	No radiation dosimetry capability	Flat and complex terrain
COMPLEX/ PFM	EPA	Point and/or area geometries, ground or elevated releases, short term or continuous dispersion, applies potential flow theory (PFM) to the COMPLEX I/II Gaussian models.	Point and/or sector averaged, ground or elevated location, or elevated location, 1-hour, 3-hour, and 24-hour concentrations.	Hourly averages, 6 wind speeds, 16 cardinal compass direction, 6 PG wind stability categories in std. EPA format. (Hourly data for 1 calendar year.)	No radiation dosimetry capability.	Flat and complex terrain

a geologic repository site at Hanford.' For example, the most promising of the sophisticated models evaluated, e.g., EPA's AIRDOS and the Hanford KRONIC, DACRIN, and PABLM series of codes, consider radionuclide contaminant atmospheric dispersion and transport under flat terrain conditions only. Conversely EPA's Gaussian models that accomodate atmospheric dispersion and transport in complex terrain do not contain provisions for such significant items as radioactive decay during dispersion and transport as well as necessary radiation dosimetry computational matrices for environmental pathways analysis of specific radionuclides that represent contributions to the environmental dose at typical receptor locations of interest and concern.

A brief discussion of several of the more prominent Gaussian models included in table 3-2 that possibly could be utilized effectively in a CTUIR risk assessment methodology is presented in the following text.

3.2.2.1 EPA-AIRDOS The basic purpose of EPA-AIRDOS is to estimate environmental concentrations due to atmospheric dispersion from a specific source released either at or near the ground surface or at any arbitrary level above the surface.

Although EPA-AIRDOS specifically represents only one of three separate steps involved in the EPA process to assess doses and risks resulting from chronic and acute radionuclide releases to the atmosphere it has been used in a general sense in table 3-2 to describe the entire process. The three separate steps within this process are: (1) calculation of the estimated dose and risk due to a unit intake or unit external exposure to a radionuclide in the biosphere, (2) estimation of the radionuclide concentrations at a location that can cause intakes or external exposures, and (3) scaling the unit doses or risks from (1) to the intakes and exposures resulting from (2). The computer codes which are used to implement each of these steps are shown diagramatically in figure 3-5.

The EPA-AIRDOS code estimates environmental concentrations due to atmospheric dispersion and terrestrial transport of radionuclides for releases from up to six sources per computer run. The code has provisions for calculating in either a rectangular or circular coordinate grid. The customarily used circular grid has 16 directions which proceed counterclockwise from north to north-northeast. Choice of grid radii are left to the user. Generally successive distances are chose with increasing spacing. It is important to realize that the calculational grid distances and the set of distances

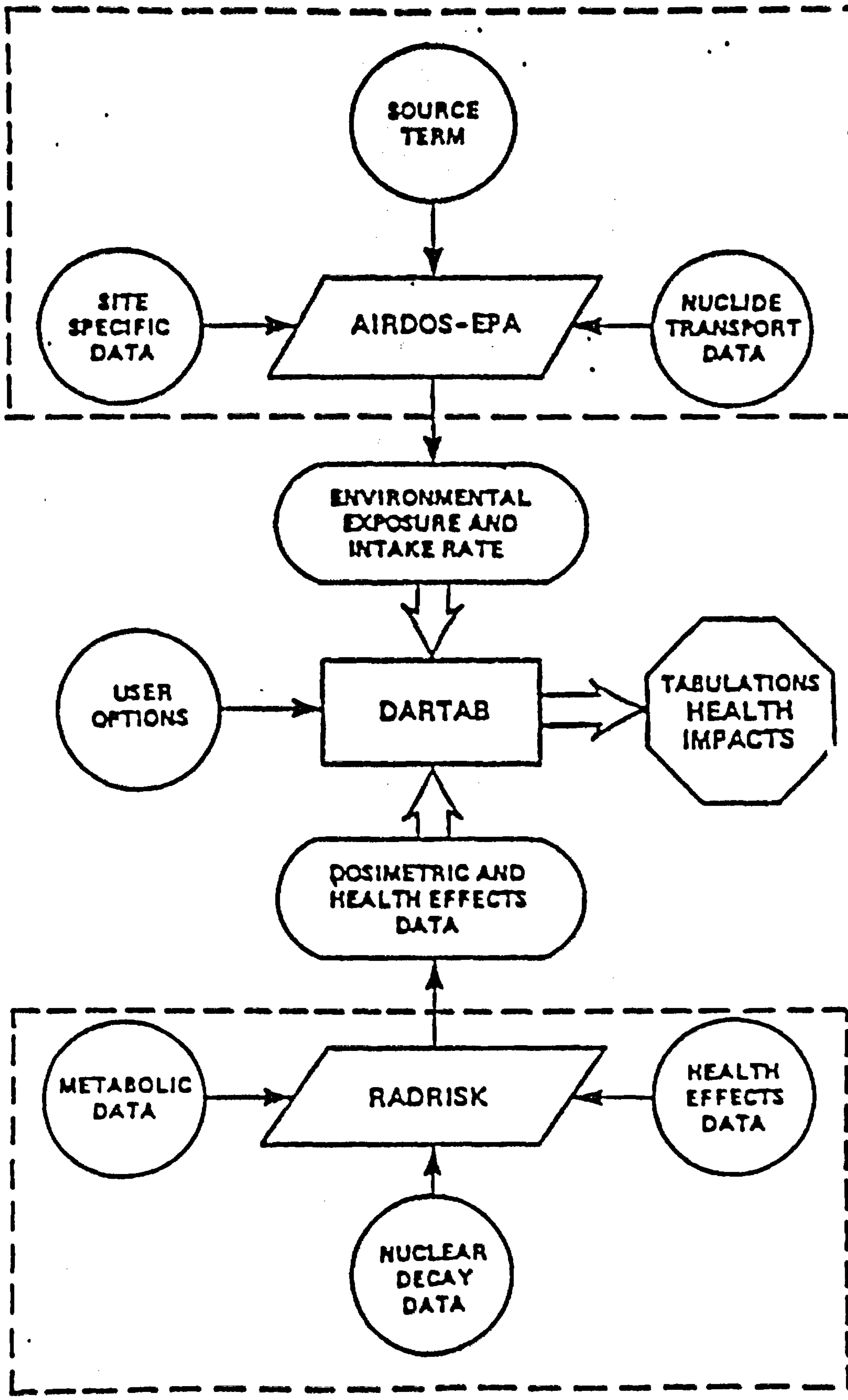


Figure 3-5. EPA SYSTEM FOR ASSESSING RADIOLOGICAL HEALTH IMPACTS

associated with population and food production data are one and the same. Hence, the concentration calculated for each grid distance must be the appropriate average value for the corresponding range of distances which are covered by the population and agricultural data.

An EPA-AIRDOS assessment for continuous release scenarios is based on what can be viewed as a snapshot of environmental concentrations after the assessed facility has been operating for some period of time. The choice of the environmental accumulation time affects only the pathways dependent on terrestrial concentrations, i.e., ground surface exposure and intakes. Usually the accumulation time for an individual assessment is chosen to be consistent with the expected life of a specific facility. For collective assessments, 100 years has been used customarily. Computing the population intake rate after 7 years of a constant unit release is equivalent to calculating the intake up to a time, T, following a unit release. For example, calculating the intake rate (Ci/yr) after 100 years of chronic release (Ci/yr) is equivalent to calculating the total intake (Ci) which would take place up to 100 years following a release (Ci) of the same value. This equivalence allows assessment values to be interpreted as the intake and exposures committed for an annual release.

Point sources are characterized by their physical height, and when desired, the parameters to calculate bouyant or momentum rise using Brigg's (Briggs, 1969) or Rupp's (Rupp et al, 1948) formulations, respectively. Alternatively, a fixed plume rise may be specified for each Pasquill Gifford atmospheric stability class A-G.

The area source model is similar to that of Culkowski and Patterson (1976) and transforms the original source into an annular segment with the same area. At large distances, the transformed source approaches a point source at the origin, while at distances close to the origin it approaches a circle with a receptor at its center.

Building wake effects and downwash are not included in the EPA-AIRDOS model. The same type of rise calculation (buoyant, momentum, or fixed) is used for all sources.

Releases for up to 36 radionuclides may be specified for EPA-AIRDOS sources. Each release is characterized by the radionuclide name, effective decay constant during dispersion, precipitation scavenging coefficient, deposition, velocity, and settling velocity as well as the annual release for each source. Decay products which are significant for the assessment of a radionuclide must be included in the list of releases. Their effective source strengths must be calculated separately since EPA-AIRDOS provides no explicit method for calculating radionuclide ingrowth during atmospheric dispersion. Parameters such as particle size, respiratory clearness class, and gastriointestinal absorption factor are entered in EPA-AIRDOS and passed on for use in the DARTAB human dose and risk assessment.

As briefly discussed in section 3.2.1 wind and stability class frequencies for each direction are the primary data for calculating atmospheric dispersion. The required data for EPA-AIRDOS can be calculated from a joint frequency distribution of windspeed and atmospheric stability class for each wind direction.

For assessment requiring long-term average dispersion values, the sector-averaged Gaussian plume option is generally used. The vertical dispersion parameter (σ_z) is calculated utilizing the aforementioned Brigg's formulas. Vertical dispersion is limited to the region between the ground and a mixing depth lid.

EPA-AIRDOS models both dry and wet deposition processes. It is assumed that once material is deposited, it is not resuspended. The dry deposition rate is the product of the deposition velocity and the near ground level air concentration while the wet deposition rate is the product of the precipitation scavenging coefficient and the vertically integrated air concentration. Therefore, wet deposition decreases monotonically with distance and is independent of the effective release height of the source, while the effect of the source height can be significant for dry deposition. For locations close to an elevated source, wet deposition can provide the principal source of radionuclide exposure. Concentrations are adjusted in the code for depletion due to deposition at each downwind distance.

EPA-AIRDOS then calculates ground surface concentration from the total (dry plus wet) deposition rate. The soil concentration is calculated by dividing this value by the effective ag.icultural soil surface density (kg/m^2). Both concentrations are calculated for the end of the environmental accumulation time, T, and can include the ingrowth from deposited parent radionuclides as removal due to radiological decay and environmental processes such as leaching.

Ingrowth from a parent radionuclide is calculated using a decay product ingrowth factor. The ingrowth factor is the equivalent deposition rate for a unit deposition rate of the parent radionuclide and is calculated prior to running EPA-AIRDOS.

Radionuclide concentrations in food are calculated using essentially the same models as in NRC Regulatory Guide 1.109. Changes from that model include consideration of environmental removal from the root zone, and separate values for food and pasture crops of the interception fraction, areal yield, and soil to plant transfer values. Concentration calculations for meat and milk use identical models to those found in NRC Regulatory Guide 1.109.

For a collective (population) assessment, both population and agricultural data for each grid location must be provided. EPA-AIRDOS calculates the collective assessment for agricultural products based on consumption by the assessment area population. The assessment can be based upon the amount produced by choosing utilization factors large enough to ensure that all items produced are consumed.

In addition to the consumption rate for different food categories (leafy vegetables, other produce, meat, and milk) the user may specify the fraction of vegetable meat, and milk which are (1) home grown, (2) from the assessment area, or (3) are imported from outside the assessment area. Those in the last category (3) are considered to contain no radionuclides. Those from (2) have the average concentration for that category produced within the assessment area while concentrations for (1) are those that would occur at each grid location.

Special consideration is given to the radionuclides H-3 (tritium), C-14 and Rn-222 (radon). Concentrations of H-3 and C-14 in produce and animal feed presume that the specific activity in these items is the same as that in air. The Rn-222 concentration in air is replaced by its short-lived decay product concentration in working level (WL) units using a fixed equilibrium fraction (typically 0.7).

3.2.2.2 RADRISK Internal exposures occur when radioactive material is inhaled or ingested. The RADRISK code implements contemporary dosimetric models such DARTAB in the EPA-AIRDOS computer modeling system (figure 3-5) to estimate the dose rate at various times to specified reference organs in the body from inhaled or ingested radionuclides. The dosimetric methods in RADRISK are based primarily on models recommended by the International Commission on Radiological Protection (ICRP). The principal qualitative difference is that RADRISK computes dose rates to specified organs separately for high and low linear energy transfer (LET) radiations. These time-dependent dose rates are used in the life table calculations of RADRISK.

In RADRISK, the direct intake of each nuclide is treated as a separate case. For chains, the ingrowth and dynamics of progeny in the body after intake of a parent radionuclide are considered explicitly in the calculation of dose rate. Consideration is also given to the different, metabolic properties of the various radionuclides in a decay chain.

3.2.2.3 DARTAB As previously illustrated in figure 3-5, the principal inputs to the dosimetric model, DARTAB, are the output file from EPA-AIRBOS and the set of unit dose and risk factors from RADRISK. The remaining input data serve primarily to select the quantities to be tabulated and the particular types of tables to be produced. There are three basis categories of tabulations produced by DARTAB: (1) summary tables, (2) detail tables, and (3) location tables. Factors which may be tabulated include organ specific doses and dose equivalents, genetic doses, somatic (cancer) risks, and genetic risks.

3.2.2.4 KRONIC The computer program KRONIC (Strenge and Watson, 1973) can be used to calculate air submersion doses primarily from routine or continuous releases of radionuclides by a Gaussian-based technique similar to that used on EPA-AIRDOS. A space integraiton over the plume volume is performed, but the plume width is determined by sector boundaries rather than by a Gaussian concentration gradient. KRONIC employs joint frequency distributions of windspeed and wind stability in a specified wind direction according to the Hanford meteorological format briefly described previously in Section 3.2.1. The gamma dose at a receptor is calculated as a tissue dose at body surface and at depths of 1 cm. and 5 cm. The 5-cm. dose is generally reported as a total or whole body dose. Normally, a one-year dose for both the maximum individual (MI) and the "regional" population are calculated.

3.2.2.4 SUBDOSA The computer program SUBDOSA (Strenge, Watson, and Houston, 1975) is used to calculate air submersion doses from accidental atmospheric releases of radionuclides. A space integration over the plume volume is performed. SUBDOSA also uses joint frequency distributions of surface windspeed, wind stability, and wind direction according to the Hanford meteorological format. Dose results are reported for skin, male gonads, and total body. Corresponding tissue depths are 0.007, 1.0 and 5.0 cm., respectively. Doses are calculated for releases within each of several release time intervals. Up to six time intervals can be accomodated, and separate radionuclide inventories and atmospheric dispersion conditions can be considered for each time interval. Normally, a one-year dose for both the MI and for the regional population are calculated.

3.2.2.5 DACRIN The DACRIN program (Houston, Strenge and Watson, 1975; Strenge, 1975) is used to calculate radiation doses from inhalation. The program uses the model of the ICRP Task Group on Lung Dynamics (ICRP, 1966) to predict radionuclide movements through the respiratory system. Once radionuclides reach the human blood stream, the doses to organs other than the lung are calculated using a single exponential retention function as given in ICRP Publication 2 (ICRP, 1959).

DACRIN can be used to calculate radionuclide concentrations in air using the Gaussian bivarate, normal distribution plume model with the accompanying joint frequency distributions of windspeed, wind stability and wind direction in the Hanford meteorological format. However, dispersion factors calculated externally to the program may also be entered by the user to generate the radionuclide concentrations.

Doses calculated in DACRIN are dependent upon the values of the release time and dose time interval as input. The doses that can be calculated for both an MI and the regional population include a one-year dose, dose commitments, and cumulative dose.

3.2.2.6 PABLM The PABLM program (Napier, Kennedy, and Soldat, 1980) is used to calculate potential doses from environmental contamination pathways, including direct radiation from contaminated water, sediment and soil surface; and ingestion doses from contaminated drinking water, aquatic food products, terrestrial farm products and animal products. PABLM combines and enhances the pathway modeling capabilities of PNL computer programs ARRRG and FOOD (Napier, Kennedy, and Soldat, 1980). It can also account for changing levels of environmental contamination with time from past or continuing deposition, and includes radioactive chain decay with daughter ingrowth. PABLM can be used to calculate dose commitments from one year of exposure and cumulative doses to either an MI or populations from multiple years of exposure. Some parameters included in the PABLM data libraries are specific to Hanford conditions.

3.2.2.7 EPA-VALLEY The VALLEY model (Burt, 1977) is a steady-state univariate Gaussian plume dispersion model designed for multiple point-and area-source applications. It calculates pollutant concentrations for each frequency disignated in an array defined by six stabilities (A-F), 16 wind directions, and six windspeeds for 112 program-designated receptor sites on a radial grid of variable scale. The output concentrations are appropriate for either a 24-hour or annual period as designated by the user. The model contains the concentration equation, the Pasquill-Gifford (PG) vertical dispersion coeffecients and the Pasquill stability classes as given by Turner and described previously in Section 3.2.1. Plume rise is calculated according to the Briggs formulations.

The most important aspect of the VALLEY program is its simulation of the effects of terrain on pollutant concentrations. For stable atmospheric conditions (PG stability categories E and F in this program) the model assumes that the plume height above the elevation of the release remains constant after final plume rise. Thus, as terrain rises the plume approaches the elevated surface; in effect the plume height decreases. Since the terrain elevations may vary from receptor to receptor, an effective plume height must be calculated for each receptor. All concentrations are then estimated as if the receptors were located at actual ground level at the respective geographical locations. However, it is further assumed that the plume centerline comes no closer than 10 m to the elevated terrain. If the terrain extends above the original plume height, the plume centerline is adjusted so that it remains 10 m above the ground. Any plume height which is initially within 10 m of the ground during stable conditions is assumed to remain at its initial height above ground, regardless of downwind terrain elevations. Any increase in concentration that would occur on the sides of the terrain obstable due to lateral deflection of the plume beyond the sector of immediate concern is ignored. Therefore, conservation of mass is not accomplished.

Deflection of the plume by terrain during stable conditions is simulated through the attenuation of concentration with height in the sector of immediate concern. This is accomplished by applying a factor based upon the relative elevations of the ground at the receptor and or the centerline of the undisturbed plume. The factor has value of unity at and below the elevation of the plume centerline in free air prior to encountering terrain effects, but decreases linearly with increasing height (from plume level) to zero at and above 400 m above the undisturbed plume centerline. The attenuation should not be inferred to represent pollutant decay or penetration into the terrain. This is an empirical scheme intended as a general representation of the blocking of air flow by significant terrain features. Therefore, in case of such plume impingement no attempt should be made to utilize the concentrations that will be calculated for the leeward side of a substantial hill or ridge, because the computer program has no memory regarding upwind terrain features.

The VALLEY program does have provisions for pollutant decay during plume transport. Through each successive period of travel defined by the half-life, I, the pollutant concentration in a given parcel of air is reduced by 50 percent. However, each specific radionuclide present in a release would constitute an individual computer run. The VALLEY program, in general, constitutes a relatively inexpensive technique for preliminary screening analysis in geographic regions where complex terrain conditions prevail.

3.2.2.8 COMPLEX I, II, PFM The COMPLEX models (COMPLEX I AND COMPLEX II) are multiple point source sequential terrain models formulated by the COMPLEX Terrain Team at the EPA Workshop on Air Quality Models held in Chicago in February, 1980. COMPLEX I is a univariate Gaussian horizontal sector averaging model (sector width = 22.5°), while COMPLEX II computes off-plume-centerline concentrations, according to a bivariate Gaussian distribution function. Joint frequency distributions of windspeed, wind stability and wind direction in the aforementioned standard EPA format as well as hourly source emission data are accepted by these programs. COMPLEX I is really multiple point source code with terrain adjustment representing a sequential modeling bridge between VALLEY and COMPLEX II.

Terrain treatment in COMPLEX varies with stability class. Neutral and unstable classes use a 0.5 terrain adjustment, while stable classes use no terrain adjustment when the recommended options are selected. With 22.5° sector averaging, COMPLEX I performs sequential VALLEY impingement calculations for stable cases. COMPLEX II plume impingement calculations are similar, with the exception that sector averaging is not used.

COMPLEX/PFM has the ability to utilize PFM calculations for neutral to moderately stable flows. The PFM option invokes either COMPLEX I, COMPLEX II or PFM computations depending upon the stability class and the Froude number. Unlike previous versions, however, all sources must be located at the same point (as in CRSTER).

COMPLEX II is invoked whenever the stability class is either 1, 2 or 3 (A, B, or C), regardless of the Froude number. In these cases plume growth is rapid and the details of terrain adjustment are not so important. A 0.5 terrain adjustment is an adequate representation of average plume behavior.

PFM is invoked for stability classes 4, 5, and 6 (D, E, and F) whenever the flow along the plume streamline has enough kinetic energy to rise against the stable density gradient and surmount the highest terrain elevation along the wind direction. Such a plume is said to be above the critical dividing streamline of the flow.

COMPLEX I is invoked whenever the plume is found to be beneath the critical dividing streamline of the flow. Plumes beneath the dividing streamline no longer pass over the terrain peak and therefore may impinge on the face of the hill somewhere. Thus, the PFM option defaults to VALLEY-like computations for impingement cases. This can potentially occur in conjunction with stabilitiy classes 4, 5, and 6; but, class 4 occurrences may be rare.

The PFM option enhances the ability of COMPLEX to perform complex terrain Gaussian plume dispersion computations in two important areas. Firstly, it incorporates plume deflections and distortions derived from potential flow theory. This enhancement approximates at least first-order terrain effects on plume geometry. And, because the streamline computations vary with obstacle shape, plume height and Froude number, plume distortions are coupled directly to meteorological variations and the approximate terrain in a way that no single terrain adjustment could be. Secondly, the use of the PFM option requires vertical temperature and velocity information to characterize the Froude number, the critical dividing streamline, and stable plume rise. Availability of the Froude number and the dividing streamline removes the assumption of coupling between the surface dispersion stability class and the dynamics of the flow aloft at plume elevation under stable conditions. It is not necessary to identify plume impingement with class E or F dispersion conditions.

3.2.3 Computer-Based Long-Range Atmospheric Transport Models

A number of pollutants emitted by energy-related activities tend to persist in the atmosphere for several days or longer in some form. Such substances have the potential to be transported in possibly significant quantities for several thousand kilometers or more. Although the current tribal list of probable release scenarios for the high-level nuclear waste repository program, as previously discussed in Section 2.0 of this report, does not indicate potential atmospheric releases of radioactive contaminants into the regional environment, there remains a general need to analyze their regional transport to determine whether the levels of atmospheric concentrations or deposition are sufficient to violate environmental standards or otherwise cause adverse effects. Depending on the pollutant form and the receptor, the mechanisms for effects may involve short-term peak values, longer term averages or cumulative measures of concentrations, deposition fluxes, or volume effects. Hence, the precise form of prediction or measure of impacts required of a computer-based model can only be identified after the nature of potential adverse effects has been clarified and/or standards have been established.

Methods for describing and predicting the transport of pollutants over long distances are still in a relatively early stage of development and validation. As evidenced in Section 3.2.2, models for predicting local behavior of plumes from point or other sources to distances of 10 to about 100 km have been reasonably well developed with certain restrictions over the last two decades. In addition, characterization of long-term global scale transport has been carried out for a number of long-lived pollutants. However, models for dealing with regional-scale atmospheric pollution in the range of 100 to several thousand km have been developed only recently (Meyers et al, 1979; Bass, 1980). Available models are limited in the range of pollutants treated, the temporal and spatial resolution, and the proven accuracy of prediction. No known regional-scale models have been tested and evaluated sufficiently to satisfy regulatory requirements.

Although regional models have not yet been approved for detailed regulatory decisions, a number of models are operational or being developed. Table 3-3 adopted from Bass (1980), summarizes many of the major models. Several of these models are operational and considered to be useful for assessment and general strategic analysis purposes. These can be generally classified according to their applicability to short-term air quality measures, usually with limited space and time coverage, or to broader geographic coverage and longer term averages.

Regional transport of any suspended matter that is nonreactive or subject to known decay and removal rates can be modeled reasonably well with sufficient data on atmospheric motions. Among important pollutants, many radioactive species, carbon monoxide, and emitted fine particulates should be predictable, although comprehensive observations needed to confirm fine particulate predictions are not yet available. Transport of sulfur oxides has received considerable attention and several operational models appear to give a reasonable representation of available observations of suspended concentrations.

A brief discussions of two of the more promising long-range or regional atmospheric transport models cited in table 3-3 will follow.

3.2.3.1 ADPIC-MATHEW In line with the widening need and effort in the field of pollutant transport and diffusion studies from local to extended regional scales, it has become apparent that fully three-dimensional computer programs must be developed.

Consequently, workers at Lawrence Livermore Laboratory (LLL) have developed such a code (Lange et al, 1980), ADPIC-MATTHEW. ADPIC-MATTHEW is a numerical three-dimensional, Cartesian coordinate particle-diffusion code capable of calculating the time-dependent distribution of air pollutants under many conditions including strongly distorted advection wind fields, calm conditions, space variable surface roughness, wet and dry deposition, radioactive decay, and space- and time-variable diffusion parameters.

Basically, the code solves the three-dimensional advection-diffusion equation in its flux convervative form (pseudo-velocity techniques) for a given mass-consistent advection field by finite difference approximations in Cartesian coordinates. The method is based on the particle-in-cell technique with the pollutant concentration represented statistically by imbedded Lagrangian marker particles in an Eulerian grid.

All instantaneous source problems are run on the expanded grid version of the ADPIC-MATTHEW program. In this version the grid automatically expands by a given ratio independently in any of the three coordinate directions as soon as a particle is entering a boundary cell. In this manner boundary velocities need not be known and are set equal to zero, therefore no particles ever leave the grid. No matter how small the source, it can be resolved by the ADPIC grid from the beginning, and remains well resolved provided the puff does not become too distorted.

The initial Gaussian distribution of the particles is generated by calling a random number generator. To generate a particle, the particle generator picks a random coordinate X,

Table 3-3. CURRENT LONG-RANGE TRANSPORT MODELS (USA)

Type	Name	Sponsoring Group
Short-Term Models		
Puff	Atmospheric Transport and Diffusion Model (ARL-ATAD)	NOAA Air Resources Lab
Puff	EURMAP-2	SRI
Puff	MESOPUFF	ERT
Plume Segment	Source-Transport Receptor Analysis Model (STRAM)	Battelle Pacific Northwest Lab
Plume Segment	MESOPLUME	ERT
Grid	Northern Great Plains	SAI
Grid	MESOGRID	ERT
Long-Term Models		
Statistical Trajectory	Advanced Statistical Trajectory Regional Air Pollution Control Model (ASTRAP)	Argonne National Lab
"Square" Puff	PNL Regional Pollutant Transport Model	Battelle Pacific Northwest Lab
Puff	ARL-ATAD	NOAA Air Resources Lab
Puff	EURMAP-1/ENAMAP	SRI
Puff/Vertical Finite Difference	SIRSOX	Brookhaven National Lab
Representative Research and Development Models		
Puff/Vertical Finite Difference	Mesoscale Trajectory and Diffusion Model	NOAA Air Resources Lab
Puff	PNL Regional Model	Battelle Pacific Northwest Lab
Plume Segment	Segmented Plume Model	Savannah River Lab
Particle-in-Cell	ADPIC-MATHEW	Lawrence Livermore Lab
Grid	SULFA3D	EPRI/ERT
Grid	SURAD	EPRI/ERT
Grid	Regional Transport Model	Teknekron
Grid	"Regional Supermodel"	EPA Meteorology Lab
Trajectory	Limited Area Mesoscale Prediction System (LAMPS)	Drexel-NCAR-BNL

and then computes the Gaussian probability $P_X = \exp\left(\frac{-x^2}{2\sigma^2}\right)$ that the particle would be found at that coordinate. It then picks another random number between zero and one. If this number is less than the probability P_X, the particle is "allowed." If the number is greater than P_X, the particle is "disallowed" and the new random coordinate X is picked, etc., until all particles are generated. To guard against too many disallowed particles at the edge of the distribution where $P_X \longrightarrow 0$, there is a cutoff on the coordinate X which is input into the code.

Continuous sources are generated in ADPIC-MATTHEW by releasing sequential puffs. The individual puffs are created each time cycle by a special source subroutine which is based on the number of particles generated for each puff. The fixed grid mode of the computer program is appropriate to treat continuous sources because the entire plume length of interest together with topography and meteorology must be covered by the grid mesh. It should be emphasized that the only purpose of this source particle treatment is to provide the model with a reasonably representative, extended source for the larger-scale three-dimensional phenomena such as varying meteorology (both surface and upper winds), diffusion parameters and topography over distances of hundreds of kilometers, the effects of which are not very sensitive to the exact shape of source distribution in a mathematical sense.

ADPIC-MATTHEW has been verified for a number of selected advection-diffusion problems for which analytic solutions are available and has been found to give results to within $\pm 5\%$ of the analytic solutions. Due to the nature of the foregoing mathematical techniques employed in ADPIC-MATTHEW, the program does require considerable mainframe computer storage.

3.2.3.2 ARL-ATAD The Air Resources Laboratories (ARL) of the National Oceanic and Atmospheric Administration (NOAA) has been in the process of developing the computerized model ATAD that calculated transport, diffusion, and deposition of effluents on regional and continental. The basic ARL model, described in detail by Hefter et al (1975), has also provided much of the theoretical basis for the Brookhaven National Laboratory (BNL) development of their regional transport model AIRSOX which is also included in table 3-3.

The ATAD model is designed to calculate the large number of trajectories required required to evaluate pollution problems and can also be used to investigate transport and dispersion characteristics of individual plumes. The current version of the model incorporates vertical temperature profiles along a trajectory to determine a mixing layer in which average transport winds are calculated to provide additional detailed data at the four daily observation times for the National Weather Service (NWS) upper atmospheric meteorological monitoring network.

The ATAD model calculates transport from any origin. A trajectory, composed of 3-hour segments is computed assuming time centered persistence of winds. The winds are averaged in a transport layer determined from vertical temperature profiles.

The top of the transport layers is defined as the base of any non-surface-based temperature inversion. A maximum inversion height is chosen by the user. If no inversion occurs below this height, the seasonal average for the top of the afternoon mixing layer (Holzworth, 1972) is used and winds are averaged in this transport layer.

The average wind in the layer is computed from the reported winds linearly weighted with respect to the thickness between mid levels. If winds at an observing station are not available at a reporting time (e.g., 06Z), the average wind for that station is interpolated from the average winds 6 hours before and after (e.g., 00Z and 12Z).

A trajectory segment is computed from all the average winds within a chosen radius of the segment origin. Each averaged wind contributing to a segment displacement calculation is weighted according to the distance from the segment to the wind station and the alignment of the wind with respect to the segment origin.

After trajectories have been determined in the transport section of the model, diffusion and deposition calculations are made using these trajectories. The effluent plume is represented by a series of puff. It is assumed that there is one puff for each trajectory and that a puff diffuses as it is transported along the trajectory path. In order to better represent the plume, an option is provided to linearly interpolate additional trajectories between those started at 6-hour intervals. Each puff diffuses according to:

$$C_m = (Q/2\pi\,\sigma_h\,Zm)$$

where = C_m = air concentration in the mixed layer.

Q = emission amount per puff.

σ_h = horizontal standard deviation.
Z_m = height of the mixed layer.
R = distance from puff center.

The concept of a deposition velocity is used to calculate dry deposition along a trajectory and an empirical scavenging ratio is used for wet deposition. The fraction of mass removed from the mixed layer by dry deposition is:

$$\dot{C}_m V_d \Delta t / C_m Z_m$$

where V_d is the dry deposition velocity and Δt is the time interval at which puff concentrations are calculated. The ari concentration in the mixed layer depleted by dry deposition, C'_m, is then given by:

$$(C_m \text{ EPA } \Delta t Zm)(C_m \text{ Zm})$$

The fraction of mass removed from the mixed layer by wet deposition is:

$$(C_m EPA \Delta t Z_m) C_m Z_m$$

where E is an empirically derived average scavenging ratio (Engelmann, 1970), P is the precipitation rate, and Z_p is the height of the precipitation layer. The air concentration in the mixed layer depleted by wet deposition, C^*_m, is then given by:

$$C^*_m = C_m \ (1-EPA\Delta t/Z_p).$$

When the effects of both wet and dry deposition are incorporated in the model, the expression used to calculate air concentrations depleted by deposition, C'^*_m, is:

$$C'^*_m = C_m(1-V_d \Delta t/Z_m) \ (1-EP\Delta t/Z_p).,$$

Various tracers have been investigated for use in long-range verification studies for th ATAD model. Kr-85 plumes released from the Savannah River production facilities in South Carolina have been measured at an ARL sampling station being operated at Murray Hill, New Jersey.

Reasonably good agreement between the foregoing experimental measurements and the ATAD model-calculated trajectories and air concentrations have been obtained.

3.3 HYDROLOGIC DISPERSION AND TRANSPORT MODELS

It is generally recognized that the most probable mode by which radioactive contaminants could be released from an underground geologic repository facility located at the Hanford Site is through the groundwater system. Thus, the principal objectives of long term (10,000 years or more) repository performance assessment are to quantify the degree of high level nuclear waste isolation achieved by the repository system, i.e., the engineered systems and the geologic medium. The basic set of system performance measures that will be used to quantify system performance will consist of the following: (1) groundwater flow paths and travel times from the repository to the accessible environment, (2) the rate of radionuclide release from the repository system, and (3), the total activity (of individual radionuclides) leaving the boundaries of a specified control or buffer zone around the repository as illustrated in figure 3-6.

One of the fundamental objectives of the long-term repository performance assessment is to determine the potential flow paths from the proposed repository and to estimate the travel times along these paths to the accessible environment. The accessible environment is defined in the EPA environmental standards (40 CFR Part 191) as "(1) the atmosphere, (2) land surfaces, (3) surface waters, (4) oceans, and (5) all of the lithosphere that is beyond the controlled area." The controlled area illustrated in figure 3-6 is also defined as (1) a surface location to be identified by passive institutional controls, that encompasses no more than 100 square kilometers and extends horizontally no more than five kilometers in any direction from the outer boundary of the original location of the radiactive wastes in a disposal system, and (2) the surface underlying such a surface location.

Hydrologic conditions generally considered favorable for waste isolation are long flow paths to the accessible environment, which are confined to the deep formations, and with travel times ranging from several thousands to hundreds of thousands of years. A minimum groundwater transit time of at least 1,000 years to the accessible environment

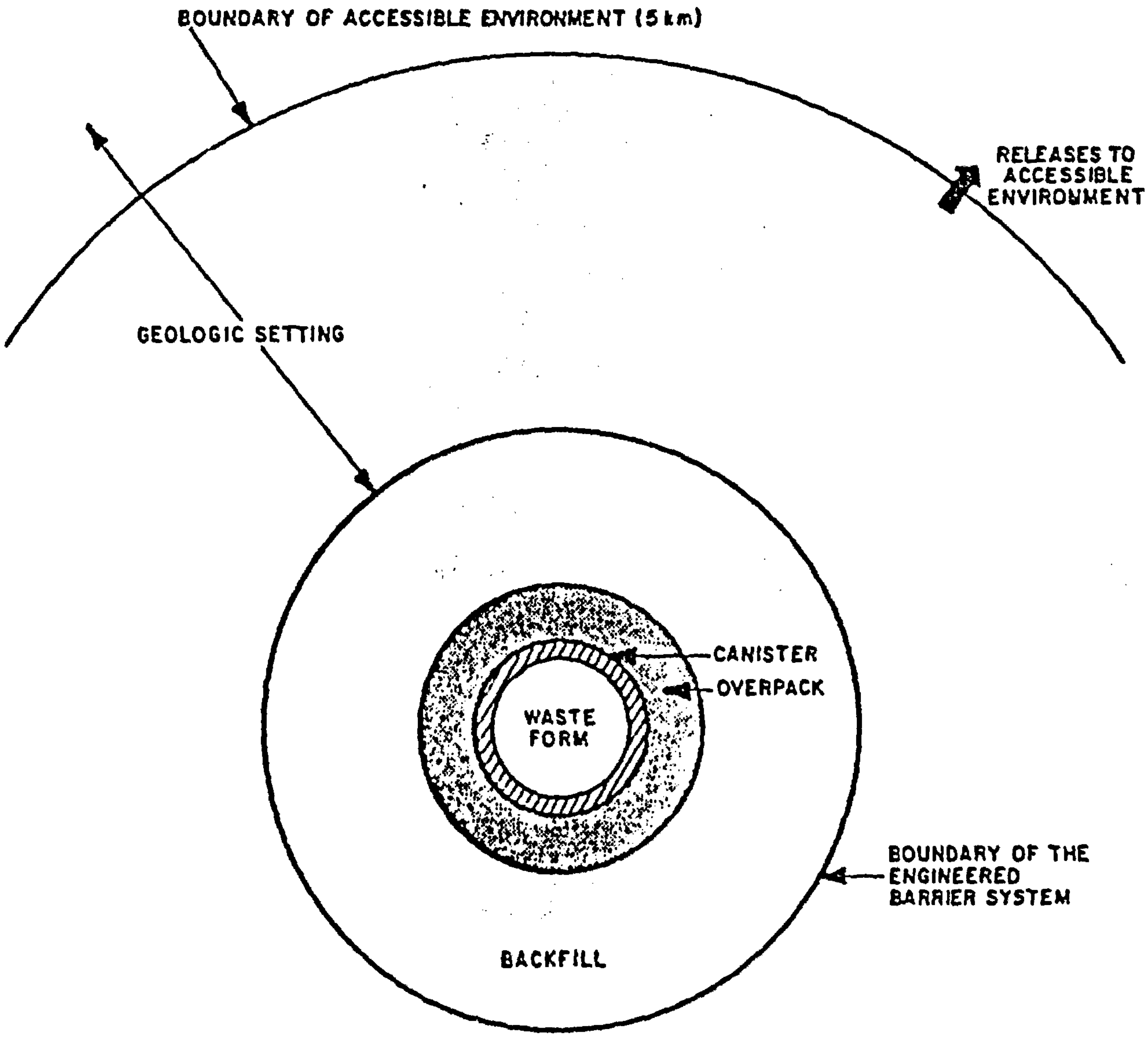


Figure 3-6. CONCEPTUAL DIAGRAM OF REPOSITORY SYSTEM AND REGULATORY CRITERIA

is a current technical criterion proposed by the USNRC. Thus, one of the foremost repository performance issues is to determine whether the pre-waste emplacement groundwater travel times near the repository are sufficient to assure compliance with both technical and regulatory criteria.

Although many factors determine the degree of long-term waste isolation achieved by the repository system, the basic factors are (1) the containment period provided by the engineered system, including the waste package and the underground facility and (2) the rate of radionuclide release from the emplacement horizon. The initial containment period (i.e., time period during which the nuclear wastes are confined to the engineered system) following repository closure is important because it mitigates any processes or events induced by the repository environment that adversely affect long-term waste isolation. After the containment period, it is assumed that any potential release will be controlled by the engineered barriers in the underground facility and the primary geologic barrier (i.e., emplacement horizon).

The long-term radionuclide release rate will be affected by the hydrologic and geochemical characteristics of the emplacement horizon; however, the period of containment depends on the engineered barriers and waste package designs. Thus, another significant repository performance issue relates to whether the very near-field interaction between the waste package and its components, the underground facility and the geologic setting, in basalt compromise waste package or engineered system performance.

To address this issue, predictive models for radionuclide hydrologic transport which take into account waste package degradation, waste from leaching, groundwater flow, and thermal conditions in the fractured, porous rock, must be applied to estimate the release rates and mass fluxes for a set of key radionuclides (Barney and Wood, 1980). Predictive estimates of these quantities of contaminant release and their variations over the entire waste isolation period as currently envisioned must be obtained for both the normal or controlled repository conditions and geologic setting as well as the off-normal or disruptive release scenarios which have been previously discussed in Section 2.2 of this report.

The aforementioned 5-kilometer control zone promulgation by EPA's 40 CFR 191 regulatory standard and previously shown in figure 3-6, sets the numerical limits on the allowable quantities of radionuclides released to the biosphere. Since the EPA regulatory standards also limit the quantity of radionuclides released over a 10,000 year period, a third repository performance issue becomes the assessment of the total amount of radioactive contaminants potentially releasable to the accessible environment in a 10,000 year period as a consequence of credible normal and/or accident repository release scenarios. Beyond the 10,000-year time period it is presumed that the radiological risk of the high-level nuclear waste is at an acceptable level because of the reduction of toxicity by decay and/or dilution. Longer time frames may be considered, however, for selected cases.

Since the amount of radionuclides leaving the designated buffer zone will depend on the repository release rates, groundwater flow paths, and travel times, the resolution of this important issue hinges on the degree to which the first two issues are resolved.

As indicated in the EPA 40 CFR 191 regulatory standard, a satisfactory resolution of this issue will require a comprehensive long-term risk assessment that: (1) identifies the plausible release modes, (2) estimates the probabilities of each release mode, and (3) conservatively bounds the consequences of releases. As part of the planned performance assessment, a large number of hydrologic simulations, considering a broad range of conditions, will be carried out to provide sufficient assurance that the model predictions compensate for uncertainties and thereby give a reasonable expectation of compliance with the EPA's regulatory standard.

The major stages of the long-term performance analysis approach consist of:

- identification and specification of plausible release modes for anticipated and postulated geologic conditions, and

- prediction of release consequences using numerical models for hydrologic, thermomechanical, and transport processes.

These stages should lead to a clear quantification of expected compliance with the numerical limits set in technical criteria and federal regulations.

In the first stage of long-term repository performance analysis, the objective is to identify the geotechnical factors and physical processes that have the most significant impact on containment and degree of isolation. Moreover, a quantitative understanding of the cause-and-effect relationships is developed between the potential release initiating events/processes and the rate of release. With this information, the consequences of radionuclide release and movement through the groundwater can be predicted to quantify long-term performance of the repository system. Because of the inherent uncertainties in such predictions, a conservative consequence analysis is required that is based on the use of both deterministic and probabilistic models. These consequence analyses should provide the information needed to quantify the likelihood of compliance with applicable criteria and regulations. Applying this approach to the set of release scenarios quantifies the radiologic risk of the repository system.

A re-examination of the potential repository scenarios presented previously in Section 2.2 of this report, indicates a wide diversity in the type and nature of the release-inducing phenomena. They vary from celestial events, such as meteorite impact, to undetected natural features and from weapons testing to resource develpment. In developing the initial scenario list, only a limited attempt was made to consider the specifics of either the host medium (basalt) or the particular setting of a potential repository system as they might influence any given scenario. Thus, it is a rather general list, which can be reduced by considering site-specific information.

Site-specific release scenarios are generally selected by means of an objective and consistent methodology that is documented in a step-by-step fashion. As a first step, it is advantageous to eliminate certain release-inducing phenomena by considering the credibility of plausibility of individual scenarios in a basalt setting. Selection of the most meaningful scenarios from those remaining is influenced by perceived as well as real risk. Also, certain cases can be justified on the basis of the bounding conditions they represent. The basic selection criteria used consider the following aspects:

- credibility of the event or process,
- probability of a significant release, and
- consistency with site-specific data and knowledge.

A long-term performance analysis of a repository site requires the evaluation of the consequences of potential releases. The problem centers on predicting repository system performance over large-space scales and very long time frames (i.e., tens to hundreds of kilometers and 10,000 years). The nature of the space and time scales virtually precludes the use of physical models or field experiments to predict the long-term performance of the performance of the repository system (underground facility and geologic medium). Mathematical models based on the principles of physics provide an efficient and versatile way to predict the long-term changes of the physical system. From such simulations, long-term performance measures are compared with the regulatory criteria and standards.

Because of the complexity of the processes, the overall long-term performance analysis problem must be broken down to one of analyzing the hydrologic processes in three subregions. These are very near field (canister to room scale), near field (repository scale), and far field (basinwide or regional scale). By this approach, mathematical models for each subregion can be developed that realistically portray the dominant physical processes, while accounting for less important processes in an approximate manner.

A typical consequence analysis approach than can be used to address the repository long-term performance analysis issues is presented in figure 3-7. The individual modeling approaches are designed to analyze the set of processes relevant to specific performance analysis issues. The very near-field and far-field models provide information needed for the near-field models, such as boundary conditions and source terms, whereas the near-field models provide flow paths starting locations for the far-field models. More specific descriptions of the various modeling approaches are presented in subsequent sections of this report.

Recognizing that the future decisions regarding the repository site will place much reliance on model predictions, consideration of uncertain elements in the consequence analysis is of fundamental and key importance. For the most part, the uncertainty in model predictions can be attributed to four sources:

- limitations in the mathematical models, including the computer codes that describe hydrologic and transport processes,

- random and systematic errors in field measurements of hydrologic properties,

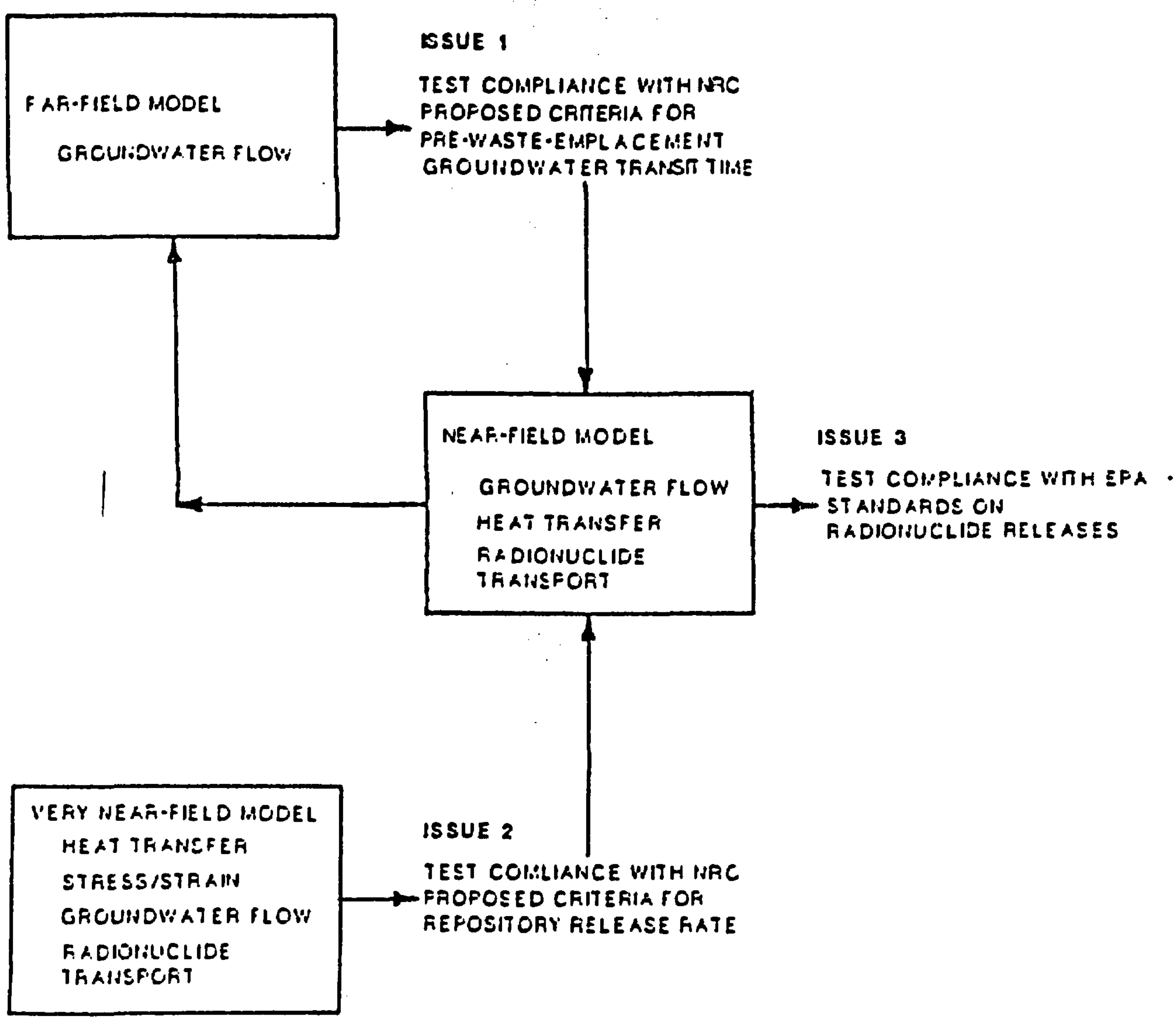


Figure 3-7. RELATIONSHIP BETWEEN PREDICTIVE MODELS AND PRIMARY LONG-TERM PERFORMANCE ASSESSMENT ISSUES.

- errors arising from subjective interpretations of the spatial variations of hydrologic parameters from discrete data points, and

- incompleteness of geohydrologic characterization.

The first source of uncertainty, which may be termed model uncertainty, can be addressed, at least to a limited degree, by code benchmarking and verification and by comparing computer-based model simulations with experimental data. These results, in turn, can be analyzed to determine the degree of correlation between measurement and calculation. The other three sources, which represent data uncertainty, can be evaluated using a variety of approaches. Various statistical techniques are available that estimate the impact of uncertain elements, given a probabilistic description of the model input (i.e., a probability density function for each hydrologic parameter). The last two elements can also be grouped into a descriptive uncertainty category, which is perhaps the most difficult to analyze in a rigorous fashion. Kriging techniques (Delhomme, 1976; Doctor, 1979), used in combination with a systematic scenario analysis, will provide a pragmatic approach to developing continuous representations of hydrologic data with uncertainty bounds and evaluating hydrologic significance of possible undetected geologic features.

The large quantity of measured data required for a rigorous uncertainty analysis appears to be a major obstacle in applying this technique to diverse geohydrologic systems. This indication is further reinforced by the simple fact that characterization of a candidate site may be limited to assure that natural barriers are not disturbed or compromised. An alternate approach to the problem of addressing predictive uncertainty is to adopt a systematic and conservative methodology that compensates for uncertain elements in the consequence analysis without assuming conditions that are not credible. Such a methodology should provide a framework for guiding the system simulations so that bounding estimates of radionuclide migration are obtained.

A methodology based on these concepts has been used historically for preliminary long-term performance analysis at candidate geologic repository sites in basalt, tuff and salt (SNL, 1983). The principal components of this analysis approach include:

- simulation models for coupled heat, groundwater flow, and radionuclide transport in fractured-porous media,

- parametric and sensitivity analysis of postulated release scenarios, and

- decision- or logic-tree strategy to guide parametric studies.

3.3.1 Predictive Hydrologic-Based Repository System Models

Four major processes and their interacting generally determine the degree of waste isolation achieved by any subsurface geologic repository system regardless of characteristics of the geologic medium. These processes are rock stress/strain, heat transfer, groundwater flow, and solute transport. The relationships between them are illustrated in figure 3-8. As indicated in the figure, the state of the repository system is characterized in terms of four basic state variables: (1) stress, σ_{ij}, and strain, ε_{ij}, (2) temperature, T; (3) hydraulic head, h (flow rate, q), and bouyancy, δ_b; and (4) radionuclide concentrations, C_i.

Several sets of mathematical models have been developed for use in detailed performance analysis of a repository in basalt at the three space scales. This section briefly outlines the theoretical framework of the mathematical models. Several of the most prominent numerical models that have been developed, and outlines of the salient capabilities and limitations of these models are then described in Section 3.3.2.

3.3.1.1 Generic Stress/Strain Models The mechanical behavior of the rock strata in the vicinity of the repository will change over the postclosure period, primarily in response to heat transfer from the waste form. These thermal stresses in the rock will generally have an effect on the hydraulic properties (e.g., fracture permeabilities and porosities) (Iwai, 1976), because of small-scale rock displacements.

The mathematical models of rock stress and strain are formulated on the basis of Newton's second law (Malvern, 1969) and Hooke's law (Timoshenko and Goodier, 1970). The actual governing equations may be found at Hardy et. al., (1978) and Baca et. al., (1980). The applicability of these mathematical models to basalt rock is based on the following assumptions:

- the jointed rock behaves mechanically as a continuous medium,

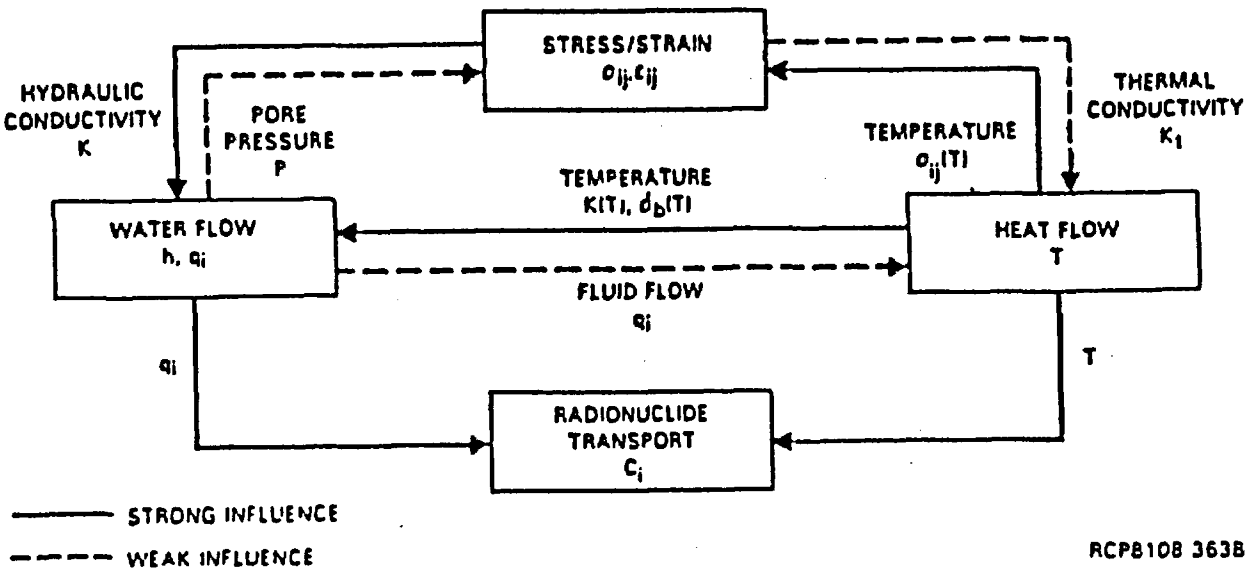


Figure 3-8. RELATIONSHIP OF IMPORTANT PHYSICAL PROCESSES OCCURRING IN THE VICINITY OF A GEOLOGIC REPOSITORY

- the thermal properties of the rock mass (e.g., thermal expansion coefficient, thermal conductivity, and specific heat) are homogeneous and isotropic, and

- the constitutive relations (stress/strain relations) are linear.

With regard to the last assumption, some recently developed stress/strain models can accomodate nonlinear behavior. However, the usefulness of such detailed models to performance analysis studies, in general, may be limited to the availability of required data.

3.3.1.2 Generic Groundwater Flow Models Groundwater flow in a nonisothermal regime is depedent on the temperature of the water-rock system. This coupling is particularly important in the near-field zone because variations in fluid density and hydraulic conductivity occur with temperature changes. Dependence of these properties on the thermal regime is made clear by considering the nonisothermal form of Darcy's law (Bear, 1972).

By combining Darcy's law and the equation for fluid continuity, one obtains the governing equations may be found in Baca and Arnett (1981) and Baca et. al., (1980). Over the temperature range expected in the vicinity of the repository, the fluid density changes by only a few percent; however, the fluid viscosity decreases by 20 to 30 times. Consequently, the hydraulic conductivity can change by a significant amount.

This particular formulation is based on the following set of basic assumptions:

(1) the fluid and rock together form a continuous system (i.e., an equivalent porous continuum), see Bear (1972),

(2) the groundwater flow regime is laminar and gradually varying (Freeze and Cherry, 1979; Bear, 1972),

(3) the fluid density variations are important only as they induce buoyancy effects (i.e., Boussinesq approximation, see Cheng, (1978), and

(4) the smallest spatial unit of analysis (i.e., representative elementary volume) is of such size that it possesses uniform hydraulic properties (Bear, 1972).

While the first three assumptions are well justified, the validity of the last assumption is limited to the near-field and far-field scales, where the representative elementary volume is relatively large. At the very near-field scale, however, the groundwater flow is determined by the "discrete" properties of the fractures and not by the "average" properties of the rock mass. For this more complex case, mathematical models have been developed and may be found in Baca et. al., (1981b).

The issue of deciding which type of groundwater flow model (continuous or discrete) applies to a particular space scale can be resolved on the basis of the so-called "scale effect" criterion (Snow, 1965; Maini, 1971; Roegiers et. al., 1979). In essence, this criterion provides a means of estimating the representative rock-mass size at which flow through discrete fractures can be represented by an equivalent continuum (i.e., Darcian flow) model. Typically, if the smallest characteristic rock size is 50 times (or more) greater than the fracture spacing, then the rock volume will generally have a high enough fractures density so as to behave like a porous continuum. A more rigorous criterion has been developed by Witherspoon et. al., (1981).

3.3.1.3 <u>Generic Heat Transport Models</u> Transfer of thermal energy from the repository to the surrounding geohydrological system will occur by advection and dispersion in the groundwater and by simple conduction through the rock mass. To describe these heat transport processes, the first law of thermodynamics (Malvern, 1969) is invoked and used with Fourier's law of conduction (Holman, 1980) to formulate the general mathematical model. The governing equation for heat transport in the water-rock system may be found in Baca et. al., (1981a, 1981b).

Basic assumptions made in this mathematical formulation are as follows:

- the temperature of the fluid and rock mass is the same at any point; (i.e., thermal equilibrium exists between fluid and solid phases (Cheng, 1978),
- the thermal properties of the rock mass are isotropic,
- the hydrologic regime is single phase, and
- the heat generation rate in the repository is a known function of time.

Although these assumptions are not necessarily well justified at the present time for the Hanford repository site location they can be used for preliminary analysis and have been applied to postclosure performance at all space scales.

In the case of a relatively low-porosity rock, such as basalt, the principal mode of heat transport is normally by conduction through the uniform rock mass-provided there are no major fractures or discontinuities in the rock mass. This fact may be confirmed by examining the Peclet number (Ozisik, 1977, Combarnous and Breis, 1975) characteristic of repository conditions.

3.3.1.4 <u>Generic Radionuclide Transport Models</u> Movement of radionuclides in groundwater is determined by the combined effects of various processes. To predict the rate and direction of potential radinuclide migration from the repository, the radionuclide transport model must describe the processes of advection, dispersion/diffusion, sorption, radioactive decay, and mass release. A mathematical model can be formulated from the principle of mass conservation (Malvern, 1969) and Fick's law (Bird et. al., 1960) for mass diffusion. The system of governing equations describing multicomponent radionuclide transport may be found in Gephart et. al., (1979) and King et. al., (1981).

Applicability of the mathematical model is constrained by certain basic assumptions:

- radionuclide (and groundwater) movement occurs slowly relative to the rate of chemical interaction (i.e., sorption),
- sorption of the dissolved radionuclide is described by a linear isotherm,
- transport properties are independent of fluid temperature,
- hydrodynamic dispersion is described by a Fickian-type law, and
- the smallest spatial unit of analysis is of such a size that it possesses average transport properties.

It is important to point out that the formulation of the transport model is generalized so as to apply to all types of radionuclides (activation and fission products) and to actinide elements that possess decay-chain couplings. In addition, the last assumption restricts the applicability of the formulation to near-field and far-field scales.

A more complex mathematical model has been developed for the very near-field scale. This model describes transport of any radionuclide of any radionuclide in fractured-porous media. The model considers transport through both continuum and discontinuum portions of the rock mass. The governing equations for this very near-field transport model may be found in Baca et. al, (1981a).

3.3.2 Specific Computer-Based Numerical Models

The principal computer codes either previously developed or currently under development for performance analysis of a repository in basalt are discussed in this section of the report. These computer-based models are classified into the previously described very near-field, near field and far-field categories.

The vast majority of this large suite of computer codes were developed and applied as part of the earlier BWIP studies at the Hanford site. The primary computer codes currently being adapted to long-term geologic repository performance analysis are presented in table 3-4.

Although only the far-field hydrologic models are of direct interest in terms of potential environmental doses to the CTUIR and its ceded lands from various repository release scenarios, the very-near field and near-field analysis codes must initially be employed to provide the necessary input source term characteristics for the subsequent far-field hydrologic modes.

Additionally, surface water hydrologic models are not included in the summary presented in table 3-4. However, surface water transport of radioactive contaminants could be an important transport medium for the far-field category of predictive modeling analysis at the Hanford repository site location in terms of potential impacts to the CTUIR and its ceded lands. Consequently, a brief discussion of potential surface hydrologic computer-based models will be included in this section of the report.

Table 3-4 SUMMARY OF CURRENTLY-AVAILABLE CODES FOR UTILIZATION IN BASALT REPOSITORY PERFORMANCE ANALYSIS

Computer Code	Approach		Stress/ strain		Ground-water flow		Heat		Radionuclide transport			Computational method			
	CO	DC	LI	NL	IS	NI	AD	DS	S	MC	DE	FE	FD	AL	DI
Very Near Field															
BETA	X		X		X	X		X				X			2
DAMSWEL	X		X	X	X	X		X				X			2
ANSYS	X	X	X	X	X	X		X				X			3
HEATING6	X					X		X					X		3
MAGNUM 2D	X	X			X	X	X	X				X			2
CHAINT	X	X							X	X	X	X			2
BARIER[a]	X					X			X				X		1
WAPPA[a]	X					X			X				X		1
Near Field															
. JRFLO	X				X	X	X	X	X				X		2
PATH[b]	X	X			X	X						X			2
MAGNUM	X	X			X	X	X	X				X			2
CHAINT	X	X							X	X	X	X			2
SWIFT	X				X	X	X	X	X	X	X		X		3
Far Field															
MAGNUM 3D	X				X							X			3
PATH 3D[b]	X				X							X			3
FECTRA	X								X			X			3
SWIFT	X				X	X	X	X	X	X	X		X		3
FE3DGW	X				X							X			3
WOOD/SALTER	X				X				X	X				X	1
NUTRAN	X								X	X				X	1

AD-Advectopm
AL-Analytical
CO-Continuum
DC-Discontinuum
DE-Decay Chains
DI-Dimensionality
DS-Diffusion
FD-Finite Difference
FE-Finite Element
IS-Isothermal
LI-Linear Properties
MC-Multicomponent
NI-Nonisothermal
NL-Nonlinear Properties
S-Single Component

[a]odes currently under development.
[b]omputes pathlines, streamlines, and travel times.

3.3.2.1 Very Near-field Computer Models

The probalistic methodology for estimating the radionuclide releases at the waste package subsystem boundary (the very near field) of a basalt repository consists of four submodels: (1) a container corrosion model, (2) a finite element model to obtain the probability distribution of releases from a single container failing at a specified time, (3) a model describing the random sequence of container failure in time, and (4) a model to integrate the releases from the set of containers in the repository having random failure times. This latter model generates the value of the radionuclide source term needed to model the mass transport in the near and far fields of the repository. Further details on this probabilities performance assessment method are found in Sagar et al. (1984).

In Sagar et al, (1984), the probabilities method is explained in some detail. The method is currently one of the most advanced methods reported for predicting waste package performance. The authors acknowledge that their mathematical models may have to be modified to account for other corrosion failure modes (e.g., uniform corrosion currently modeled) and that the preliminary performance results may change as the date base broadens.

A few other limitations to the overall method, in its current state of development, are noted:

- No credit is taken for the packing or waste form as engineered barriers,
- Disruptive events and manufacturing or handling defects are not included,
- Effects of a changing chemical state over time as affecting corrosion of the canister and overpack are not included,
- The overall failure mode is assumed to be an axisymmetric yield failure,
- Only selective radionuclides are analyzed, and
- Interaction of failure of one overpack on the chance of failure on another nearby overpack was not evaluated.

3.3.2.1.1 BETA Computer Model The BETA finite-element code is a modified version of a code developed by the University of Minnesota. The BETA code is designed to simulate the thermomechanical response of a continuous rock mass in two dimensions (i.e., cartesian- or cylinderical-coordinate systems). Stresses and strains in the rock mass surrounding the repository are computed as functions of stress boundary conditions, gravity loads, and transient thermal conditjons. Heat transport through the rock mass is assumed to occur by conduction only; advection of the groundwater and convective boundary conditions are not considered.

The specific governing equations, which may be found in most stress-analysis (Timoshenko and Goodier, 1970) and finite-element texts (Cook, 1981), deal with linear elastic behavior. Some of the basic features of the BETA code are:

- The continuous rock mass is represented by quadrilateral isoparameter elements.
- The model formulation accomodates plane stress and plane strain analysis.
- The computer code provides an option for isothermal stress, coupled stress, and temperature calculations.
- The transient heat transfer calculations accomodate arbitrary heat source loading.
- The computer code is easy and inexpensive to use.

3.3.2.1.2 DAMSWEL Computer Model The DAMSWEL computer model was developed by the Advanced Technology Group of Dames & Moore for thermomechanical analysis. Similar in application to the BETA code, DAMSWEL is a two-dimensional finite-element code. DAMSWEL, however, has the following major differences and advantages.

- The model formulation accomodates linear and nonlinear rock properties.
- The computer code calculates rock temperatures by solving the nonlinear heat equation.

- The computational algorithms used in the code are more advanced and sophisticated than those in the BETA codes.

- The code has been verified using problems with known analytical solutions.

3.3.2.1.3 ANSYS Computer Code The ANSYS computer code is a generalized stress-analysis code widely used in the nuclear industry. This proprietary computer code, developed by Swanson Analysis Systems, has a broad capability to analyze the thermomechanical response of the basalt rock. Some of the special capabilities of this computer code are:

- The model formulation is generalized to simulate coupled heat and stress transients.

- The model may be used to analyze stresses and strains in two or three dimensions.

- The model can consider linear and nonlinear rock properties.

- The code accomodates a continuous rock mass. Jointing may be modeled by means of gap elements.

The ANSYS computer code is very well documented with regard to theoretical basis, input instructions, and model use (DeSalvo, 1976; Kohnke, 1977; DeSalvo and Swanson, 1981).

3.3.2.1.4 HEATING6 Computer Model This code is designed to solve steady-state and/or transients heat conduction in one, two, or three dimensions using one of several finite-difference techniques. The principal application of HEATING6 in performance analysis studies has been to model the thermal environment of the waste package. The discussion of the governing equations, input and output description, and model use may be found in the report by Turner et al. (1977). Some of the general features and capabilities of this code are:

- The formulation is generalized to accomodate cartesian-, cylindrical-, or spherical-coordinate system.

- The code accomodates temperature-dependent thermal properties.

- The code can handle a wide variety of boundary conditions.

The HEATING6 code has been applied to a number of waste package studies and, consequently has, undergone considerable verification and benchmarking activities.

3.3.2.1.5 The MAGNUM 2D Computer Code The MAGNUM 2D code is a two-dimensional (2D) finite-element code designed to simulate transient groundwater flow and heat transport in fractured-porous rock systems. The theoretical frame-work of the model is based on concepts for a porous continuum and for discrete conduits. In particular, a dual-porosity approach is used to represent the continuous rock mass, where flow through planar conduits is described by Poiseuille's equation. The governing equations and finite-element solution techniques are presented by Baca et al. (1981b). The principal features of the MAGNUM 2D code are as follows:

- Continuous rock mass is represented with isoparametric finite elements; line elements are embedded along the sides of two-dimensional elements to represent discrete fractures.

- Model accomodates complex stratigraphic features with variable media properties.

- Computer code provides options for coupled or uncoupled solutions of heat and flow equations.

- Flow-field calculations are provided for input to pathline and transport models.

The MAGNUM 2D code has recently been modified to accomodate the use of Monte Carlo sampling techniques of various groundwater hydraulic input parameters (transmissivity, conductivity, head, etc.). Although the basic MAGNUM model is essentially deterministic - since for a given set of input parameters a specific set of output results are computed - a probabilities analysis can be performed by Monte Carlo sampling techniques; i.e., (random number generated distributions to determine how the uncertainties in the foregoing hydraulic input parameters to the MAGNUM model are manifested in the output results which are also computed and presented on the basis of a probability distribution.

Formal documentation for the MAGNUM 2D code currently contains a discussion of model theory, numerical techniques, verification, and validation test cases. This code is also interfaced with the PATH and CHAINT codes which will be subsequently discussed.

3.3.2.1.6 The CHAINT-MC Computer Model The CHAINT-MC code simulates multicomponent radionuclide transport in a fractured-porous medium. The processes modeled include advection, dispersion/diffusion, sorption, chain-decay coupling, and mass release. The computational method is based on a finite-element solution of the system of equations. Continuum portions of the medium are modeled as a single-porosity syste, using two-dimensional isoparametric elements. Discrete features are modeled using isoparametric line elements that are embedded along the sides of the two-dimensional elements. Principal input to this code is the groundwater flow calculations obtained with the MAGNUM 2D code (or a comparable nonisothermal flow model). The CHAINT code has the following major features:

- Model formulation is generalized to handle any combination of nuclides (actinides, fission, or activation products) with contrasting half-lives.

- Computational algorithm accomodates subzone calculations in which the region of active nodes, within the finite-element mesh, is varied with time as the problem progresses.

- Numerical algorithms are second-order accurate and fully implicit.

The CHAINT-MC code has been verified with boundary value problems and successfully compared with the PORFLO code. Additional work is proceeding to reduce computational times and to compare the model predictions with experimental data. The CHAINT-MC computer model has also been modified to allow probabilistic analysis by means of Monte Carlo sampling methods.

3.3.2.1.7 The BARIER Computer Model This code developed for the Office of Nuclear Waste Isolation (ONWI) (Lester et al, 1979) is specifically designed for use in performance analyses of the engineered barriers around a canister. The BARIER code takes into account material properties, geometry, corrosion, thermal expansion, internal pressure, creep strain, compaction, and temperature variations. This code, in conjunction with waste form codes, will provide a basis for estimating waste package release rates. Some of the advantages of the BARIER code are:

- The formulation considers various processes that determine waste packabe performance.

- The code is relativelv simple and inexpensive to use.

3.3.2.1.8 <u>The WAPPA Computer Code</u> The WAPPA computer-based model, also developed under the auspices of ONWI, is a generalized one-dimensional barrier degradation code for a waste package in a geologic repository. The code contains five complex degradation process models and generates waste package failure times and radionuclide release rates (source term) at the waste package/rock interface. WAPPA is essentially an extended version of the BARIER code with more extensive capabilities to describe the corrosion process, radiation effects, thermomechanical response of the waste canister, and leaching of the waste form.

Some assumptions of the model are:

- The repository temperature is constant.

- The repository is completely resaturated.

- The initial radionuclide concentrations in water outside of the waste package are zero. (This implies a large volume of water and/or high near-field water turnover rate.)

- The corrosion rates are dependent on temperature ranges and radiation level (i.e., linear corrosion).

- The stress field around the package is uniform.

- The backfill is intact at all times.

Although the WAPPA code will provide a general predictive capability, some modifications are required to aoapt the code to the basalt rock present at the proposed Hanford repository site. For example, necessary modifications to the WAPPA should include the following: (1) capability to handle temperature history at the waste package-basalt interface, (2) consideration of desaturation/resaturation phenomena, (3) laboratory bulk corrosion data, and (4) solubility limitation of waste form dissolution.

3.3.2.1.9 The ORIGEN 2 Computer Model The ORIGEN 2 computer model is a revised and updated version of the Oak Ridge Isotope Generation and Depletion Code. The model is a versatile point depletion and decay formulation for use in calculating radionuclide composition and porous decay as a function of time. The present version of the code provides file data input to the WAPPA code (Croft, 1980).

3.3.2.2 Near-field Computer Models

3.3.2.2.1 The PORFLO Computer Model PORFLO is a finite-difference code with options for modeling the coupled processes of groundwater flow, heat transfer, and radionuclide transport. The model is applicable to porous media or highly jointed rock systems that may be represented as an equivalent porous continuum. The finite-difference method is based on a nodal point integration technique used in conjunction with an alternating direction implicit method. Additional description of this model is contained in Baca et al. (1981a). Major features of the PORFLO computer code are as follows:

- The computer code is easy and inexpensive to use.
- The numerical method ensures energy and mass conversation at the grid-block level (Patanka, 1980).
- A donor-cell method is used to accomodate advection-dominated flow regimes (Runchal, 1972).
- The computer code computes the total activity crossing specified boundaries for the simulation period.

3.3.2.2.2 <u>The PATH Computer Model</u> Using the numerical results from a two-dimensional groundwater model, the PATH code computes the pathlines or streamlines for an arbitrary set of starting points in the study region. In addition to computing the particle trajectories, the model computes the cumulative time of travel along each trajectory (i.e., travel times). The program solves the pathline equations on a finite-element grid network, thereby tracing the particle trajectory from element to element. Major features of the PATH code are as follows:

- The pathline equations are solved using a predictor-corrector algorithm and finite-element shape functions.

- The computational algorithm accomodates two-dimensional isoparametric elements with one-dimensional line elements.

- The computer output is in graphic form with options provided for superimposing the finite-element mesh, rock-type boundaries, etc., as well as generating plots for subzone grids.

The PATH computer code is designed for interactive use on a standard graphics terminal. Versions of this program are currently interfaced with the MAGNUM 2D and PORFLO computer codes.

It should be noted that various near-field and far-field category computer models, such as MAGNUM 3D, CHAINT, and SWIFT, may also be applicable for near-field predicitve analysis.

3.3.2.2.3 <u>The EQ3/EQ6 Computer Model</u> EQ3 is a geochemical speciation code suitable for nuclear waste performance assessment. It computes chemical species and produces a model of the fluid by specifying concentration and thermodynamic activity of each species and calculates the saturation state of the fluid.

EQ6 is a geochemical reaction path code package tailored to nuclear waste performance assessment. It uses the ageous model routine from EQ3 as a starting point to predict changes in fluid composition, reactants, and mass transport (Wolery, 1979, INTERA, 1982b).

3.3.2.2.4 The TOSPAC Computer Model The TOSPAC computer model is a simple systems model of water flow, leaching, and nuclide migration in unsaturated and saturated media that is under development at Sandia National Laboratories (SNL). The code can be used to predict first order systems performance of the geologic and engineered barriers and the coupling between the various subsystems. TOSPAC uses statistical techniques for addressing uncertainties and will predict performance in the form of probability distributions (Tyler et al, 1980).

3.3.2.2.5 The NWFT Computer Model The Network Flow and Transport Code simulates groundwater flow and containment transport in a saturated porous medium and will accomodate radionuclide transport in one-dimension. The flow calculations utilize Darcy's law coupled with the conservation equations.

3.3.2.3 Far-Field Computer Models

3.3.2.3.1 The MAGNUM 3D Computer Model The MAGNUM 3D computer model has been developed to solve the three-dimensional form of the groundwater flow equation, using the same fundamental numerical procedures as the previously discussed MAGNUM 2D code. The present version of the MAGNUM 3D code is limited to isothermal conditions; future versions could probably be developed to accomodate three-dimensional, non-isothermal effects as required. The model is based on the continuum theory of porous media and is currently designed for analysis of flow patterns in large-scale groundwater basins such as the Pasco Basin which encompasses the proposed high level nuclear waste repository location at the Hanford Site. Some of the salient features of the MAGNUM 3D computer model are:

- The code accomodates complex three-dimensional geometry through the use of various three-dimensional isoparametric finite elements (e.g., tetrahedrons and parallelopipeds).
- The code can consider different types of boundary conditions (e.g., specified heads and/or fluxes).
- The code provides a three-dimensional flow field for input to pathline and transport codes.

3.3.2.3.2 The PATH 3D Computer Model The PATH 3D code is similar in numerical procedures to the previously described PATH computer model except that it calculates three-dimensional pathlines or streamlines. It can be interfaced with the MAGNUM 3D code.

3.3.2.3.3 The FECTRA Computer Model The FECTRA code analyzes radionuclide transport in porous media. This transport model is based on the dual-porosity approach that considers the interaction of radionuclides in the mobile and immobile phases. The mobile component is the dissolved radionuclide moving through the primary pores, whereas the immobile phase is contained in the secondary or dead-end pores (Coats and Smith, 1964). The theoretical framework considers the basic processes of advection, dispersion/diffusion, sorption, decay and mass release of a single species. The computer code was originally developed for application to the partially saturated flow regime. Basic features of FECTRA are:

- Two versions are available (two-dimensional and three-dimensional), using various isoparametric finite elements.
- Numerical techniques are second-order accurate and fully implicit.
- The code has been successful compared to other transport codes and verified with analytic solutions.

The FECTRA code is designed to be interfaced with a three-dimensional fluid-flow model such as MAGNUM 3D. It is being considered for simulation of the natural groundwater hydrochemical stratification as well as for far-field radionuclide transport.

3.3.2.3.4 The SWIFT Computer Model As evidenced in table 3-4, the SWIFT computer model developed by SNL is probably the most general simulation code for repository performance analysis that is currently available. The SWIFT code is capable of simulating the coupled processes of heat transport, groundwater flow, and radionuclide transport. The general governing equations (Dillon et al., 1979) are solved using finite-difference techniques. Several of the essential features of the SWIFT code are:

- The code can consider a variety of hydrologic regimes and boundary conditions.

- The code can simulate the transport of a variety of radionuclides (i.e., fission products, activation products, and actinide elements).

- The computer code has numerous options for solving the governing equations.

3.3.2.3.5 The FE3DGW Computer Model The FE3DGW computer model was developed by the Pacific Northwest Laboratory (PNL) as part of the Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program. FE3DGW is essentially a generalized code for groundwater flow analysis that solves the governing equations for three-dimensional flow in an isothermal-porous media. Salient features of this code are:

- The code can accomodate geohydrologic systems with complete geometries.

- The code uses a variety of types of finite elements.

- The computer code is highly modularized.

This code has been used in groundwater modeling studies of the Pasco Basin (Dove et al., 1981). The available documentation of FE3DGW provides a good discussion of model theory and application approach (Gupta et al., 1979).

3.3.2.3.6 The NUTRAN Computer Model The NUTRAN code is a generalized systems model for nuclear-waste-disposal consequence analysis (Berman et al., 1978). One of the components of this code is designed to model radionuclide transport from the repository to the accessible environment. The governing equations for radionuclide transport are solved using a Green's function approach. General factors and capabilities of the NUTRAN code (Ross et al., 1979) are:

- The code is easy and inexpensive to use.

- The code can be used in a Monte Carlo mode (i.e., probabilistic description of input parameters).

- The code can be interfaced with various submodels having the capability to develop a complete pathway analysis for environmental dose calcations at a surface receptor location.

The NUTRAN code has been applied in performance analysis studies for a repository site at Hanford.

3.3.2.3.7 The WOOD/SALTER Computer Model The WOOD/SALTER computer model has been primarily used to evaluate waste package performance requirements (Wood, 1980). The model is based on the closed-form analytic solution to the one-dimensional radionuclide transport equation developed by Haderman (1980). Basic features of the WOOD/SALTER code are:

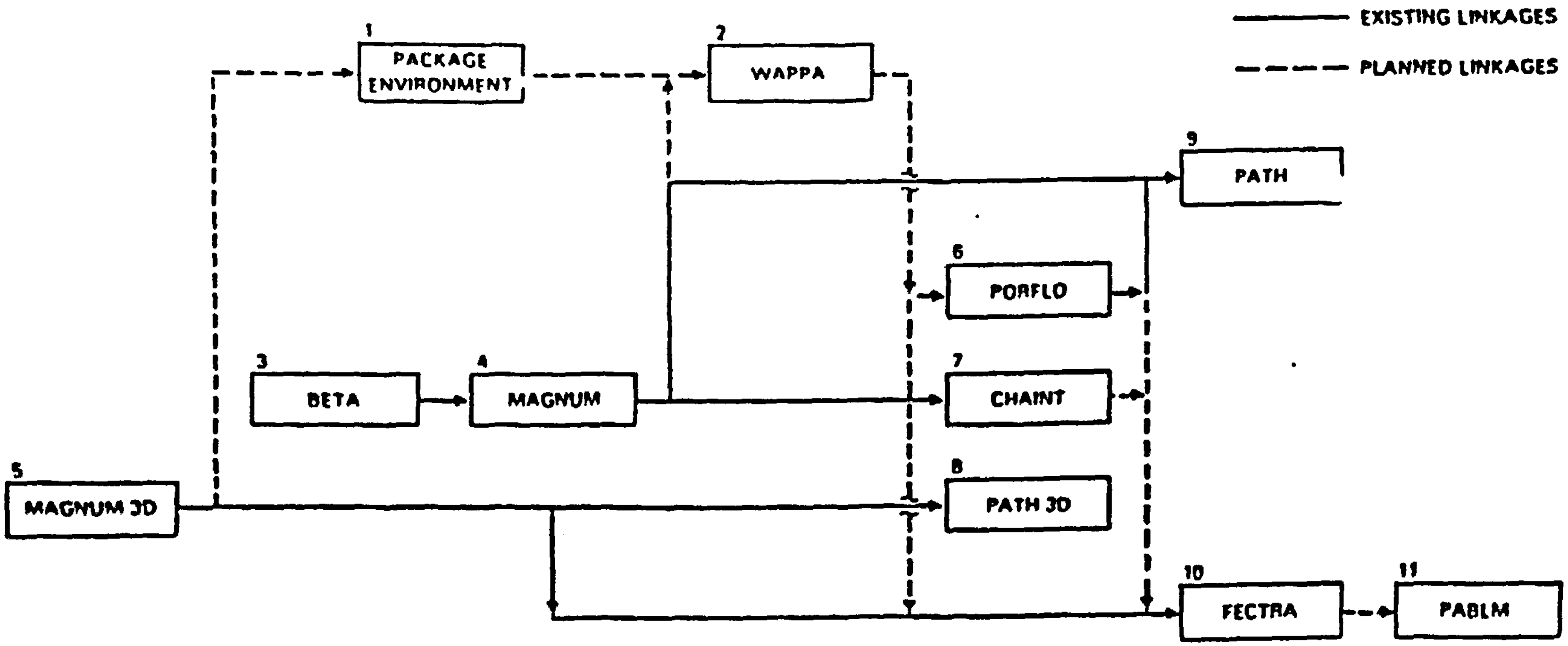
- The theoretical framework assumes a band radionuclide release rate.
- The code is applicable to a variety of sorbing and nonsorbing radionuclides.
- The computer code is simple and inexpensive to use.

The governing equations, basic assumptions, and analytic solution are documented in Haderman (1980).

3.3.3 Integrated Groundwater System Preliminary Computer Modeling Procedure-Proposed Hanford High-Level Nuclear Waste Repository Site

Selected computer models previously discussed in Section 3.3.2 can be integrated to produce a system model capable of effectively modeling the potential physical and chemical processes that can arise from the waste package environment through groundwater transport leading to environmental and human doses.

The interrelationships of the various subsystems computer models in a preliminary overall system model are conceptually presented in figure 3-9. The codes presented in figure 3-9 represent an integrated set of models specifically applicable to a repository system in basaltic rock. However, it must be emphasized that many unresolved uncertainties exist due to the current paucity of the experimental data base. Unless reliable boundary conditions and hydraulic parameters are determined during site characterization and until a defensible conceptual groundwater model is developed, little confidence can be attached from the results of any one, or a series of, computerized numerical codes.



1. WASTE PACKAGE ENVIRONMENT
2. RADIONUCLIDE RELEASE
3. FRACTURE ALTERATIONS
4. NEAR-FIELD AND VERY NEAR-FIELD GROUNDWATER FLOW FIELD
5. FAR-FIELD GROUNDWATER FLOW FIELD AND RESATURATION
6. NEAR-FIELD RADIONUCLIDE TRANSPORT
7. VERY NEAR-FIELD RADIONUCLIDE TRANSPORT
8. FAR FIELD GROUNDWATER PATHLINES AND TRAVEL TIMES
9. NEAR FIELD AND VERY NEAR FIELD GROUNDWATER PATHLINES AND TRAVEL TIMES
10. FAR FIELD RADIONUCLIDE TRANSPORT
11. DOSE TO MAN

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Figure 3-9. PRELIMINARY GROUNDWATER SYSTEM MODELING FORMAT-HANFORD HIGH-LEVEL NUCLEAR WASTE REPOSITORY SITE

For example, the current conceptual model, which is based on the assumed stratified nature of groundwater in basalt, does not consider many potentially important observed features. Among the alternative conceptual models that incorporate important aspects of observed geologic features are the following:

(1) An areally continuous, layered system with relatively high vertical leakage. In this conceptual model, the intraflow structures, such as fanning columnar joints in the entablature, are considered to permit significant vertical leakage between layers and reduce but not eliminate the assumed confining nature of basalt flow interiors. In all other respects this is a porous-flow-equivalent, continuum model, like that of the current DOE conceptual model.

(2) An areally discontinuous, layered system with relatively high vertical leakage that performs hydraulically as a large-scale, homogeneous, anisotropic system. In this conceptual model, the layered basalt system is laterally discontinuous because of intraflow structures and variable flow distribution. The high vertical leakage associated with intraflow structures would impart an anisotropy to this model system. To assume this to be a porous-flow-equivalent, continuum model, it is necessary to assume that these small-scale discontinuities would result in a homogeneous system on a large scale because of their high frequency and random distribution.

(3) An areally discontinuum, layered system bounded by high permeability structures. In this conceptual model, the layered basalt system is divided into a series of discrete blocks. The blocks are bounded by vertically disruptive features of high permeability (fault zones or tectonic breccias) that provide a direct means of recharge and discharge to and from aquifers. On the scale of the zone between the RRL and the accessible environment, this is a noncontinuum model for which the porous-flow-equivalent numerical modeling used in SCR could yield erroneous and nonconservative flow paths, travel times, and radionuclide fluxes.

(4) An areally discontinuous, layered system bounded by low permeability structures. In this conceptual model, the layered basalt system is divided into a series of discrete blocks separated by low permeability zones that

impede lateral groundwater movement. The low permeability barriers might consist of gouge zones along major faults or might represent simple juxtaposition of low horizontal hydraulic conductivity units (a dense basalt flow interior) against horizontal hydraulic conductivity units (a brecciated flow top). As with case 3 above, this is a noncontinuum model.

Thus, it may be concluded that, as more definitive experimental data is developed during site characterization, the various computer-base subsystem models comprising the overall groundwater system analysis format could change appreciably from the preliminary format outlined in figure 3-9.

3.3.4 Surface Water Transport Modeling Considerations

The surface water transport of radioactive contaminants from possible release scenarios envisioned for a proposed high-level nuclear waste repository program being implemented at the Hanford Site could conceivably impact the natural environment of the CTUIR and its ceded lands due to potential release scenarios from both major system categories; i.e., the transportation system and the repository storage system.

For example, radionuclide release from the repository could be transported initially by means of a groundwater pathway into a unconfined-surface aquifer in the accessible environment, e.g. a perched aquifer adjacent to a river or a stream. Thus, the radioactive contaminants upon reaching the river course could be subsequently transported by surface waters to the CTUIR and its ceded lands via several hydrologic routes as previously illustrated in figure 3-4.

In contrast, a transportation system accident release scenarios conceivably could occur where the radionuclides could be dispersed either directly into or near surface waters within the boundaries of the CTUIR and its ceded lands as previously inferred in Section 2.0.

Nevertheless, similar computer-based surface water modeling techniques should be amenable to assimilation into the overall CTUIR risk assessment methodology development with relatively minor revisions to existing, available surface water transport models. Several of these computer codes will be briefly discussed in the subsequent text. It must be emphasized that these computer-based models that will be reviewed are only representative of the relatively extensive list of modeling techniques that are both currently available and applicable to surface water transport analysis.

3.3.4.1 AQUAMAN Computer Model AQUAMAN is an interactive computer code developed by Oak Ridge National Laboratory (ORNL, Sheeffer et al., 1979) for calculating values of dose (50 year commitment) to man from aqueous releases of radionuclides from nuclear facilities. The data base contains values of internal and external dose conversion factors, and bioaccumulation (freshwater and marine) factors 56 radionuclides. A maximum of 20 radionuclides may be selected for any one computer model run. Dose and cumulative exposure index (CUEX) values are calculated for total body, GI tract, bone, thyroid, lungs, liver, kidneys, testes, and ovaries for each of three exposure pathways: water ingestion, fish ingestion, and submersion.

3.3.4.2 STTUBE Computer Program The STTUBE model developed by NRC (Codell, 1981) can be used for dispersion computations in unidirectional rivers with varying cross sections. Computations are performed in "stream-tube" coordinates, in which complex river cross sections are mapped into a new river discharge-based coordinate system so that their mathematical representation can be simplified. The relationship between stream-tube coordinates and geometric coordinates is developed in the subprogram TUBE. The method has been used successfully to simulate the dispersion of both conservative and nonconservative substances in a number of rivers.

3.3.4.3 SCREENLP Computer Model The SCREENLP computer model developed by NRC (Codell, 1984) is a screening methodology that utilizes subsets of models from the Liquid Pathway Generic Study (LPGS). The SCREENLP program analyzes the potential contamination of surface water via the groundwater pathway resulting from a severe nuclear reactor core-melt accident (Class 9) and calculates a "surrogate" population dose resulting from three potential basic dose sources. These sources are drinking water, finfish and shellfish ingestion, and shoreline exposure. The calculated site-specific dose values are available to be compared with the generic site dose values that were calculated in a similar manner using LPGS parameters. This comparison provides the basis for determining if the site under study would pose an unusual liquid pathway hazard. The generic sites (land-based) are categorized in the four basic environments: river, Great Lakes, and estuary, and coastal. The dry site, that is, one located far away from a surface-water body, is not covered by the SCREENLP model. Although the

SCREENLP is specifically designed for analysis of reactor core-melt accidents, relatively minor adjustments to the source input can be made to make the SCREENLP computer model applicable to various release scenarios that might be encountered in the high-level nuclear waste repository program.

3.3.4.4 PRESTO-II Computer Model The PRESTO-II code is an extension of the PRESTO-EPA model which was developed under U.S. Environmental Protection Agency funds by ORNL to evaluate possible health effects from radionuclide releases from shallow radioactive waste disposal trenches and from associated areas contaminated by operational spillage. This model is designed to simulate transport of radionuclides form relatively low-level nuclear waste disposal sites and to predict radionuclide exposures and cancer risks for a 1,000 year period following the end of burial operations. PRESTO is versatile methodology for calculating risks to local and intermediate-range populations from both waterborne and airborne releases. The DARTAB code, discussed earlier in Section 3.2.2.1 in conjunction with the EPA-AIRDOS atmospheric and dispersion model, is also used by PRESTO-II as a subroutine to combine simulated radionuclide exposure values with dose and health risk factors to produce tabulations of dose and health risk.

The computer code format utilized in the simulations is modular and organized according to transport pathways. Near surface transport mechanisms currently considered in the model are trench cap failure, cap erosion, farming or reclamation practices, human intrusion, chemical exchange within an active soil layer, contamination from trench overflow, and dilute by surface streams. Subsurface processes include infiltration and drainage into the trenche, the ensuing dissolution of radionuclides, chemical interaction between trench water and buried solids. Mechanisms leading to contaminated water outflow include trench overflow and downward vertical percolation. The PRESTO-II model considers radiological exposures resulting from drinking contaminated aquifer and stream water as well as from irrigation and subsequent ingestion of crops.

The exceptional flexibility coupled with the structured modularity designed into this code make it worthy of additional, more detailed investigation for potential utilization, with appropriate modifications, in the future development of the CTUIR risk assessment methodology for the high-level nuclear waste repository program.

4.0 CHARACTERIZATION AND CLASSIFICATION OF HUMAN DOSE, HUMAN HEALTH EFFECTS AND HEALTH RISK

Previous sections of this report have pointed out that the primary federal regulatory standards governing the release of radioactive contaminants to the natural environment are directed toward the allowable radiation doses received by humans and the subsequent biological effects on both individuals and the general population that have been exposed to the radioactive release. Therefore, human dosimetry becomes a major consideration in the development of any risk assessment methodology designed to evaluate the potential environmental impacts related to a high-level nuclear waste repository program.

Although the general aspects of radiation dosimetry were briefly discussed in Section 3.0 in terms of their logical, sequential interface with the computation of environmental dose as prescribed by a number of the atmospheric and hydrologic dispersion and transport models that are presently being evaluated, a more fundamental development of human dose characterization is presented in Section 4.1.

Further characterization of the currently acceptable health effects to humans as a consequence of radiation exposure is then outlined in Section 4.2 followed by introduction of a preliminary conceptual technique for comparatively classifying and ranking potential health risks from predicted environmental concentrations and/or doses in Section 4.3.

4.1 CHARACTERIZATION OF HUMAN DOSES

Although mankind has produced many sources of nuclear radiation, natural background remains the greatest contibutor to the radiation exposure of the population of the United States today. Background radiation has three components: terrestrial radiation (external), resulting from the presence of naturally occurring radionuclides in the soil and earth; cosmic radiation (external) arising from outer space; and naturallly occurring radionuclides (internal) deposited in the human body.

The rate at which a person receives radiation from natural background is a function of the person's geographic location and living habits. For example, the dose-equivalent (DE) from terrestrial sources varies with the type of soil in a given area and its content of naturally occurring radionuclides. The penetrating gamma radiation from these radionuclides produces whole-body exposure.

In general, the conterminous United States can be divided into three broad areas, from the standpoint of terrestrial whole body rates: the Atlantic and gulf coastal plain, where terrestrial DE rates range from 15 to 35 mrems/yr; the northeastern, central, and far western portions, with DE rates ranging from 35 to 75 mrems/yr, and the Colorado Plateau area, in which terrestrial DE rates range from 75 to 140 mrems/yr.

Cosmic radiation includes both the energetic particles of extraterrestrial origin that strike the atmosphere of the earth (primary particles) and the particles generated by these interactions (secondary particles). By virtue of these interactions, the atmosphere serves as a shield against cosmic radiation, and, the thinner this shield, the greater the DE rate. Thus, the cosmic radiation DE rate increases with altitude. For example, the dose rate at 1,800m (5,900 ft) is about double that at sea level. Because of the variations in the earth's magnetic field, with which cosmic radiation also interacts, the DE rate also varies with latitude. Finally, the cosmic radiation dose rate also varies owing to solar modulation. For the United States, variations in the cosmic radiation dose rate due to the latter two influences amount to less than 10 percent. Because the components of cosmic radiation that reach the population are highly penetrating and are an external source, they result in whole-body irradiation. It is estimated that the average DE rate to the U.S. population from cosmic radiation is about 31 mrems/yr (disregarding shielding).

The deposition of naturally occurring radionuclides in the human body results primarily from the inhalation and ingestion of these materials in air, food and water. Such nuclides include radioisotopes of lead, polonium, bismuth, radium, radon, potassium, carbon, hydrogen, uranium and thorium, as well as a dozen or more extraterrestrially produced radionuclides. The heavier radionuclides are of particular interest in that they are widespread in the biosphere and they, or many of the shorter-lived members of their decay series, are alpha emitters.

Thus, in turn, the characterization of human dose in terms of exposure from the various elements or activities comprising the high-level nuclear waste repository program also are logically subdivided into an assessment of the external or whole body dose and the internal dose to specific organs or systems of the human body.

4.1.1. Internal Dose Characterization Internal doses are usually characterized by route of exposure (ingestion or inhalation), target organ, and radionuclide. The most widely accepted methods for the characterization of internal human dose have been developed in ICRP30 Part 1 and its supplement (1979). These methods develop internal dosimetry factors that are tabulated for radioisotopes as a function of pathway or route of exposure (oral or ingestion), fraction of ingested radioactive compound absorbed into the blood, retention in the pulmonary region and target organs.

The most common units of radiation dose are the rad, the unit of absorbed dose (1 rad = 100 ergs/g = 0.01 joule/kg), and the rem, unit of equivalent dose for different types of radiation (1 rem = 1 rad x a correction factor to equalize biologic effects). However, the reader should be aware that new units are coming into general use, in particular, the gray (1 Gy = 100 rads = 1J/kg) and the sievert (1Sv = 100 rems).

The International Commission on Radiation Units and Measurements (ICRU, 1971) states that, the activity, A, of a quantity of a radioactive nuclide is the quotient of dN by dT where dN is the member of spontaneous nuclear transformations which occur in this quantity in the time interval dt; i.e., $A = \frac{dN}{dt}$.

The special name for the SI unit of activity is the becquerel (bq), where 1 Bq = 1 dps (disintegration per second). Therefore, 1 Bq = 2.7×10^{-11} Ci.

The dose equivalent, H (ICRU, 1973) is the product of D, Q and N at the specific point of interest where D is the absorbed dose, Q is the quality factor and N is the product of any other modifying factors. Therefore, H = DQN, where the modifying factors Q and N are dimensionless.

For purpose of planning in radiological protection it is assumed that risk of a given biological effect is linearly related to dose equivalent. In these circumstances, risk of an effect is determined by the total dose equivalent averaged throughout the organ or tissue at risk, independent of the time over which that dose equivalent is delivered. For planning work with radioactive materials the Commission recommends that the appropriate period for integration of dose equivalent is a working life-time of 50 years. The total dose equivalent averaged throughout any tissue over the 50 years after intake of a radionuclide into the body is termed the committed dose equivalent. H_{50}, which is therefore given by

$$H_{50} = \sum_i \frac{\int^M D_{50,i} Q_i N_i \, dm}{\int^M dm} \qquad (1)$$

where M is the mass of the specified organ or tissue; and, for each type of radiation i.

$D_{50,i}$ is the total absorbed dose during a period of 50 years after intake of the radionuclide into the body in the element of mass dm of the specified organ or tissue;

Q_i is the quality factor; and

N_i is the product of all other modifying factors such as dose rate, fractionation, etc.

The quality factor, Q, is defined as a continuous function of collision stopping power in water (ICRP Publication 21). Therefore the value of Q_i will vary along the track of an ionizing particle and may be different for each element of mass dm in the irradiated tissue concerned. However, in view of the many uncertainties in estimating the dose to a tissue following the intake of a radioactive material, the Commission recommends (para. 20, ICRP Publication 26) that for internal exposure the value of Q for a given type of radiation may be considered constant and have one of three values as follows:

Q = 1 for beta particles, electrons and all electromagnetic radiation including gamma radiation, x rays and bremsstrahlung.

Q = 10 for fisson neutrons emitted in spontaneous fission and for protons.

Q = 20 for alpha particles from nuclear transformations, for heavy recoil particles and for fission fragments.

The Commission recommends that the product of all other factors, N, should be taken as 1 for values of dose equivalent less than or equal to the recommended primary limits.

The estimates of risks of radiation-induced cancer and hereditary disease on which the Commission's dose equivalent limits for stochastic effects are based were made using the hypothesis that risk of an effect in linearly related to dose equivalent. Therefore it is the total dose equivalent averaged throughout any organ or tissue, independently of the time over which that dose equivalent is delivered, which determines the degree of effect in that tissue. With regard to limits on the intake of a radioactive material into the body, the Commission has reconsidered the question of the time over which this total dose equivalent should be integrated and has concluded that the period of 50 years used heretofore is appropriate for an occupational lifetime. The total dose equivalent in any tisssue over the 50 years after intake of a radionuclide into the body is termed the Committed Dose Equivalent, H_{50}. It is emphasized that this is the dose equivalent which a Reference Man is assumed to receive if he lives for 50 years after his intake of the radioactive material and if no steps are taken to accelerate the removal of the radionuclide from his body.

Therefore, in order to meet the Commission's basic limits for the exposure of workers, the intakes or radioactive materials in any year must be limited to satisfy the following conditions

$$\sum_T W_T H_{50,T} \leq 0.05 \text{ Sv} \qquad (2)$$

and

$$H_{50,T} \leq 0.5 \text{ Sv} \qquad (3)$$

where W_T is the weighting factor shown for a specific organ or tissue in table 4-1 and H_{50} (in Sv) is the total committed doses equivalent in tissue (T) resulting from intakes of radioactive materials from all sources during the year in question.

Relationship (2) limits stochastic effects and relationship (3) non-stochastic effects arising from intakes of radioactive materials. With regard to (3), the limit for non-stochastic effects in any tissue is taken as 0.5 Sv (50 rem), since no case is known where eye lens opacity would be the factor limiting intake of radioactive material. It could possibly be the limiting factor when the body is irradiated from the exterior by submersion in a radioactive noble gas, e.g., Kr, Ar, Xe.

Therefore using the values of Q shown previously, which are constant for any type of radiation i, the expression for H_{50} shown in equation (1) simplifies to:

$$H_{50} = \sum_i Q_i \overline{D_{50,i}} \qquad (4)$$

Table 4-1. RECOMMENDED ICRP WEIGHTING FACTORS FOR STOCHASTIC RISKS

Organ or tissue	W_T
Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone surfaces	0.03
Remainder	0.30

where: D_{50} is the total absorbed dose during the 50 years after the intake of the radionuclide into the body averaged throughout the specified organ or tissue for each radiation of type i.

For each type of radiation i, $H_{50,i}$ in target organ T resulting from radionuclide j in source ogran S is the product of two factors

(a) the total number of transformations of radionuclide j in S over a period of 50 years after intake,

(b) the energy absorbed per g in T, suitably for quality factor, from radiation of type i per transformation of radionuclide j in S,

i.e. for each radiation of type i from radionuclide j

$$H_{50}(T \leftarrow S)_i = Qi\ \overline{D_{50}}(T \leftarrow S)$$
$$= U_s \times 1.6 \times 10^{-13}\ SEE\ (T \leftarrow S)_i \times 10^3\ Sv$$

where U_s is the number of transformation of j in S over the 50 years following intake of the radionuclide:

1.6×10^{-13} is the number of joules in 1 MeV;

SEE $(T \leftarrow S)_i$ (in MeV g^{-1} per transformation) is the specific effective energy for radiation type i, suitably modified by quality factor, absorbed in T from each transformation in S and 10^3 is the conversion factor from g^{-1} to kg^{-1}.

$$\therefore H_{50}(T \leftarrow S)_i = 1.6 \times 10^{-10}\ U_s\ \ SEE\ (T \leftarrow S)_i\ Sv. \qquad (5)$$

and for all types of radiation emitted by radionuclide j:

$$H_{50}(T \leftarrow S)_j = 1.6 \times 10^{-10} \left[U_s \sum_i SEE(T \leftarrow S)_i\right]_j \qquad (6)$$

When the radionuclide has a radioactive daughter j'

$$H_{50}(T\leftarrow S)_{j+j'} = 1.6 \times 10^{-10}\left[\left\{U_s \sum_i SEE(T\leftarrow S)_i\right\}_j + \left\{U_s \sum_i SEE(T\leftarrow S)_i\right\}_{j'}\right] \text{Sv.} \quad (7)$$

In general, for the intake of any mixture of radionuclides, i.e. parent with daughters and/or other radionuclides, H_{50} in target T from activity in source S is given by

$$\sum_j H_{50}(T\leftarrow S)_j = 1.6 \times 10^{-10} \sum_j \left[U_s \sum_i SEE(T\leftarrow S)_i\right]_j \text{ Sv.} \quad (8)$$

where the summation in j is over all the radionuclides involved. Finally, target T may be irradiated by radiations arising in several different sources S. The total value of H_{50} in target T is then given by

$$H_{50,T} = 1.6 \times 10^{-10} \sum_S \sum_j \left[U_s \sum_i SEE(T\leftarrow S)_i\right]_j \text{ Sv.} \quad (9)$$

4.1.1.1 Cellular Distribution of Dose Values of H_{50} derived in this report refer to the average committed dose equivalent in a target issue. For parts of the gastrointestinal tract the target tissue is considered to be the mucosal layer, for the bone the cells lying within 10 m of bone surfaces and for the skin the basal layer of the epidermis, taken to be at a depth of 70 um. In most other cases the position of sensitive cells within the target tissue has not yet been specified. It is recognized that there may be circumstances where the effects produced may be different from those expected fr considerations of average dose, e.g. for radionuclides that emit radiations of very shor range and which concentrate near radiosensitive microvolumes. With regard to radioactive particles, the Commission has expressed the view that, for late stochastic effects, the absorption of a given quantity of radiation is ordinarily likely to be less effective when due to a series of "hot spots" than when uniformly distributed (para. 33, ICRP Publication 26). Consideration also needs to be given to those compounds labelled with radionuclides such as ^{3}H, ^{14}C and ^{125}I which are incorporated into the nuclei of cells synthesizing DNA. In such cases, biological effects may arise from transmutation. i.e., the chemical change of the nuclide together with its sudden change of electric charge and recoil, as well as from the ionization and excitation produced by the emitted radiations.

4.1.1.2 Specific Effective Energy (SEE) In the dosimetric data for individual radionuclides values are given for

$$SEE(T \leftarrow 2) = \sum_i SEE(T \leftarrow S)_i$$

for a number of target and source organs. It is emphasized that the values shown refer only to the radionuclide concerned and do not include any contribution from daughter radionuclides. Values for daughter radionuclides are given separately. For any radionuclide j, SEE $(T \leftarrow S)_j$ for target T and source S is given by

$$SEE(T \leftarrow S)_j = \sum_i \frac{Y_i E_i AF(T \leftarrow S)_i Q_i}{M_T} \text{ MEV g}^{-1} \text{per transformation}$$

where the summation is over all radiations produced per transformation of radionuclide j in source organ S;

Y_i is the yield of radiation of type i per transformation of radionuclide j;

E_i (in MeV) is the average or unique energy of radiation i as appropriate;

$AF(T \leftarrow S)_i$ is the fraction of energy absorbed in target organ T per emission of radiation i in S. For most organs it is assumed that the energies from alpha particles and electrons are completely absorbed within the source organ. Notable exceptions are mineral bone and the contents of the gastrointestinal tract. The absorbed fraction of energy from photons is estimated by the use of data on specific absorbed fraction (absorbed fraction per g of target) given in ICRP Publication 23. The absorbed fractions for fission neutrons have been obtained from data given by Dillman and Jones (1975), and Ford et al. (1977);

Q_i is the quality factor appropriate for radiation of type i as previously presented; and

M_T (in g) is the mass of the target organ.

The masses of target organs are taken from ICRP Publication 23 and are listed in table 4-2. Except for ovaries and uterus they apply to the 70 kg Reference Man.

4.1.1.3 Number of Transformation in a Source Organ Over 50 Years The number of transformations of a radionuclide in any organ or tissue of the body during any period of time is the time integral of activity of the radionuclide within that organ or tissue over the stated period of time. The function describing uptake and retention of a radionuclide

Table 4-2. ICRP RECOMMENDED MASSES OF ORGANS AND TISSUES - REFERENCE MAN

Source Organs	Mass(g)	Target Organs	Mass(g)
Ovaries	11	Ovaries	11
Testes	35	Testes	35
Muscle	28,000	Muscle	28,000
Red marrow	1,500	Red marrow	1,500
Lungs	1,000	Lungs	1,000
Thyroid	20	Thryoid	20
ST content	250	Bone surface	120
SI content	400	ST wall	150
ULI content	220	SI wall	640
LLI content	135	ULI wall	210
Kidneys	310	LLI wall	160
Liver	1,800	Kidneys	310
Pancreas	100	Liver	1,800
Cortical bone	4,000	Pancreas	100
Trabecular bone	1,000	Skin	2,600
Skin	2,600	Spleen	180
Spleen	180	Thymus	20
Adrenals	14	Uterus	80
Bladder content	200	Adrenals	14
Total body	70,000	Bladder wall	45

in a body tissue following its ingestion or inhalation may be very complex and therefore it is convenient to describe the transfer of radionuclides within the body by relatively simple models which facilitate calculation and yet yield estimates of dose sufficiently accurate for purposes of this report. With certain exceptions, e.g. for alkaline earth radionuclides in bone, the assumptions in the dosimetric models that the body consists of a number of separate compartments are valid. Thus retention of an element in any organ or tissue will usually be described by either a single exponential term of the sum of a number of exponential terms, details of which are given in the metabolic data for individual elements.

For example, after a radionuclide has been inhaled or ingested it will be translocated to the body fluids at a rate determined by the rate constants for the different compartments in the repository and gastrointestinal systems and by the radioactive decay constant of the radionuclide. Its translocation thereafter to the compartments representing the various organs and tissues of the body is shown in figure 4-1.

The finite time taken for translocation to the organs and tissues of deposition following entry of a radionuclide into the body fluids is represented in the model by transfer compartment a, which is assumed to be cleared by first order kinetics with a half life of 0.25 day, unless otherwise stated in the metabolic data for a particular element. Transformations occuring in the transfer compartment are assumed to be uniformly distributed throughout the whole body of mass 70,000 g. Each organ or tissue of deposition is assumed to consist of one or more compartments, and from each of these compartments the radionuclide is translocated at an appropriate rate to the excretion pathways. For simplicity, it is usually assumed that there is no feedback to the transfer compartment either from the routes of excretion or from the organ compartments, although it is recognized that transfer to body fluids happens in practice, and thus no estimate is usually made of dose along the routes of excretion. It should be noted that, because of the above assumption the amount of a radionuclide in transfer compartment a at any time after inhalation or ingestion cannot be used to estimate the amount of the radionuclide present in the body fluids at that time.

From the above model the activity q(t) in any compartment at time t is derived using the following equations.

In transfer compartment a,

$$\frac{d}{dt} q_a(t) = \dot{I}(t) - \lambda_a q_a(t) - \lambda_R q_a(t) \qquad (10)$$

In tissue compartment b,

$$\frac{d}{dt} q_a(t) = b\,\lambda_a q_a(t) - \lambda_b q_b(t) - \lambda_R q_b(t) \qquad (11)$$

and so on, for any number of compartments.

where $\dot{I}(t)$ is the rate of entry of activity of the radionuclide into body fluids at time t after its inhalation or ingestion, and is calculated as described in Chapters 5 and 6;

λ_a is the clearance rate of stable isotopes of the element transferred compartment a;

b, c etc. are the fractions of stable isotopes of the element transferred from the body fluids to compartments b, c, etc.;

λ_b, λ_c, are the clearance rates of stable isotopes of the element from the compartments b, c, etc., and

λ_R is the radioactive decay constant of the radionuclide.

Values of b, c, etc.,; λ_b, λ_c, etc. can be derived from the metabolic data for individual elements. Any exceptions to this general method of deriving the activity in a compartment of the body will be noted in the metabolic data for that element, e.g. iodine.

In this presentation, H_{50} per unit intake, annual limit on intake (ALI), and derived air concentration (DAC) refer to the intake of the specific radionuclide alone. However, if the radionuclide has radioactive daughters, an allowance is made for the committed dose equivalent contributed by the build-up of daughters produced in the body from their

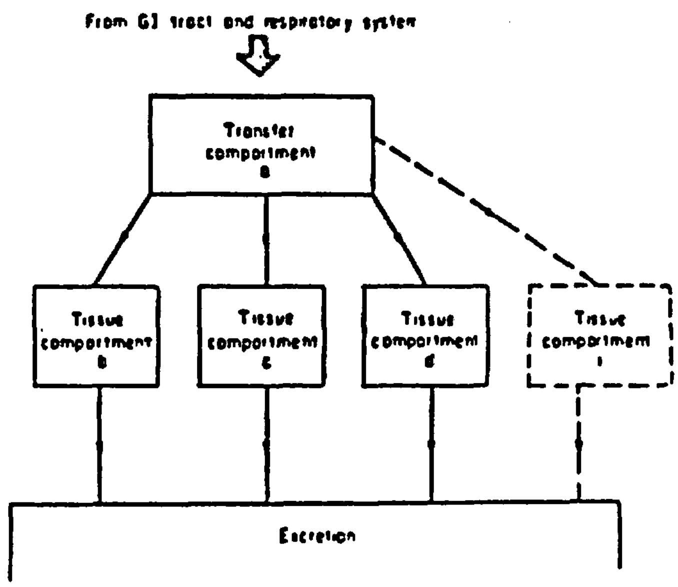

Figure 4-1. TYPICAL DOSIMETRIC MODEL TO DESCRIBE KINETICS OF RADIONUCLIDES IN THE GI TRACT AND RESPIRATORY SYSTEM

parent. In general there is little evidence to indicate whether these daughters will remain associated with, and behave as, their parent, or whether, upon being produced they will assume their own metabolic behaviour. When experimental evidence is available, e.g. concerning the behaviour of the nobel gases radon and thoron when produced from their parents ^{226}Ra and ^{224}Ra in the body, it is given in the metabolic data. In all other cases it is assumed that daughters and all subsequent progeny produced in the body (i.e. including the respiratory and gastrointestinal systems), stay with and behave metabolically like the inhaled or ingested parent radionuclide. However, if evidence to the contrary becomes available, it is recommended that this information should be used to calculate revised ALIs for the radionuclide or mixture of radionuclides in question.

Using this assumption the activity of a radioactive daughter q'(t) in the transfer compartment a, or in any organ or tissue compartment b at any time t after intake of the parent radionuclide can be obtained using the following equation:

$$\frac{d}{dt} q_a'(t) = I'(t) + \lambda_R' q_a(t) - \lambda_R q_a'(t) \qquad (12)$$

where I'(t) is the rate of entry of activity of the daughter radionuclide into the transfer compartment at time t, this activity having resulted from radionactive decay of the parent radionuclide within the respiratory system or the GI tract as described in Chapters 5 and 6 of ICRP Publication No. 30.

λ'_R is the radioactive decay constant of the daughter radionuclide;

q_a (t) is the activity of the parent radionuclide in the transfer compartment at time t;

λ_a is the biological rate of clearance from the transfer compartment of a daughter radionuclide produced in the body (assumed to have the same value as that for its parent); and

$q'_a(t)$ is the activity of the daughter radionuclide in the transfer compartment at time t.

$$\frac{d}{dt} q'_b(t) = b\lambda_a q'_a(t) + \lambda'_R q_b(t) - \lambda_b q'_b(t) - \lambda'_R q_b^!(t) \qquad (13)$$

where $q'_b(t)$ is the activity of the daughterr in tissue compartment b at time t;

b is the fraction of the daughter translocated from the transfer compartment to compartment b; it is assumed that this fraction is the same as that for the stable isotope of the parent radionuclide;

λ_a and λ_b are the biological rates of clearance of the daughter radionuclide from the transfer compartment and from compartment b respectively (assumed to be the same as those for the parent radionuclide;

λ'_R is the radioactive decay constant of the daughter radionuclide;

$q'_a(t)$ is the activitiy of the daughter in the transfer compartment at time t; and

$q_b^!(t)$ is the activity of the parent radionuclide in tissue compartment b at time t.

In a similar manner, a system of equations can be derived that describe the activities of a chain of parent and daughter radionuclides, the activity of each daughter being determined by the activity of its predecessor in the chain. The metabolic behavior of all the radioactive progeny is assumed to be the same as that of the ancestral radionuclide which was taken into the body. These equations, together with those given in ICRP Publication No. 30 for the repository system and GI tract, completely specify the models that can be used to calculate values of the number of transformations, U_s, in any source organ within the body. In the dosimetric data for any radionuclide, values of U_s are given for that radionuclide together with values of U'_s, U''_s, etc. for its daughter radionuclides which have built up in the body in the 50 years following ingestion or inhalation of unit activity of their parent.

It has been demonstrated that the dosimetry factors developed by the foregoing models are quite complete in that they represent the cumulative internal dose from a single intake over the subsequent 50 years. The cumulative dose includes radiation of target organs from all relevant organs in the body and from radioactive daughters.

Thus, dose is obtained from the ICRP -30 dosimetry models through the use of the Committed Dose Equivalent (H_{50}) which specifies the sieverts (Sv) accumulated over 50 years from exposure to a specific quantity of activity in bequerels (Bq). The H_{50} for a target organ is given directly for those organs which contribute greater than 10% of the H_{50} of any other tissues.

Since the ICRP model yields doses based on 50-year survival following exposure, it will under-or over-estimate doses for individuals surviving more or less than 50 years. It is based on the anatomy and physiology of an 70 Kg male or "Reference Man." As noted in ICRP-23 and ICRP-30, these specifications are not designed to be applied to the general population, which has many individuals who deviate from the "Reference Man." However, the reference man specifications (ICRP, 1975 and 1979) are the only source of dosimetry calculation parameters for many radioisotopes and are often used in general population dosimetry calculations. When applied the obtained doses must be considered to incorporate errors due to individual variations from the reference man.

The ICRP-30 model is restricted in its applicability, for a specific radioisotope, to certain types of particles and compounds. Because physiology and metabolic activity data are incorporated into the calculations, the use of the model is limited by the type of compounds and particles for which data are available. For example, strontium-90 (Sr-90) has retention values for Sr TiO_3 which differ from all other compounds due to its unique metabolic activity (ICRP, 1979).

Another aspect in which the ICRP-30 model is limited is in its emphasis on only organ systems which receive the largest doses. Organ systems are not considered if they receive less than 10% of the dose delivered to the tissue or organ receiving the maximum dose (ICRP, 1979). While this treatment may be adequate in cases where exposure is relatively small, when large exposures occur as in an accident release scenario, a dose of less than 10% of the maximum received may be a significant source of adverse health effects. Due to the potential for large exposures in some potential incident scenarios for the proposed high-level nuclear waste repository, the impact of this model deficit must be considered. A sample calculation for the radioisotope, Sr-90, based upon the ICRP-30 modeling approach is presented in Appendix B to introduce the reader to this analytical technique.

4.1.2 External Dose Characterization

In view of the greatly varying radiation sensitivity of different organs and tissues, the location and direction of a radiation field are significant. In particular, one distinquishes between whole-body exposure and localized exposure. In whole-body exposure, all of the body is assumed to be bathed in a uniform radiation field and to receive a specified average dose to all tissues. Such exposure can be assumed to have more serious consequences for the survival chances of the body than a comparable dose confined to a single organ. In fact, in cancer therapy the local dose often given to the malignant tissue would be fatal to the patient if all of his body were exposed. Similarly, in view of the lesser sensitivity of the extremities a high level of exposure confined to the hands or feet may be tolerated when it might cause serious damage to more sensitive areas. Hence, it is important in medical radiology and in the handling of radioactive materials to shield all portions of the body not directly involved in the radiation application. This applies particularly to such radiation-sensitive organs as the gonads, the lens of the eye and the bone marrow.

Radiation exposure may be chronic or acute. Chronic exposure implies continued exposure over long periods of time leading to a given total dose value. Acute exposure refers to sudden, perhaps massive, radiation exposure such as might arise in the case of a radiation accident or a nuclear explosion. At high radiation levels the rate of exposure is usually assumed to be unimportant and only the total absorbed dose is considered. Nevertheless, it is important, particularly in accident situations, to reconstruct the dose received and the duration of exposure.

Another important distinction arises from the limited range of some of the radiations, especially the heavier charged particles. If the source of radiation is external to the body, the dose to internal organs depends on the attenuation of the radiation by intervening tissue and the dose will be highest, in general, to the skin. In fact, for alpha particles or fission product nuclei and for low energy electrons, the range in tissue is so short that the skin is the only body part exposed to external sources. For gamma and X-rays up to approximately 150 KeV in energy, again the skin dose is highest and the dose to underlying tissue must be computed carefully, a central problem in radiological practice. As a consequence the risk of immersion in water or air containing only sources of strongly absorbed radiation is rather slight.

Whole body doses and skin doses are usually calculated using a cloud submersion model, making an additional assumption as to whether the cloud is infinite, semi-infinite, or finite relative to the receptor.

ICRP 30, Part 1 and Cember (Introduction to Health Physics, 1983) discuss a model which is useful for calculating whole body and skin doses for a receptor in an infinite hemisphere of uniformly distributed radioisotopes. This model also assumes that the density of emitted energy is equal to the density of absorbed energy and that the mass stopping power of tissue relative to air is 1.1. The whole body dose is calculated for gamma rays only, and the skin dose is calculated only for beta particle capable of penetrating the protective layer of the skin, i.e., a thickness of about 0.07 millimeters. Hence, only beta particles with energies greater than or equal to 0.07 Mev should be considered (Radiological Health Handbook, 1970). The following general equation can be used to calculate whole body gamma doses or beta skin doses:

$$H_E\,(\mathrm{Sv}) = \frac{1.6 \times 10^{-3}\, CK\,(E/1)\, g\, dT}{2\,(1.293)}$$

$$H_E = 6.18 \times 10^{-14}\, \frac{CE}{1}\, dt \quad (\text{for } k = 1.1 \text{ and } g = 1) \qquad (14)$$

where

k = mass stopping in tissue relative to air

H_E = whole body dose equivalent for gamma or skin dose equivalent for beta ≥ 0.07 MeV

C = activity in air in Bq/M^3

$E/1$ = average energy per transformation

dT = time of exposure

g = geometry factor for shielding by overlying tissues (assumed equal to 1)

When a person is submerged in a radioactive gas, the skin and other organs of the body may be irradiated both by external irradiation from gas absorbed into body tissues. The respiratory system and other organs may also be irradiated by gas contained in the lungs. These sources of irradiation, which are discussed below, limit the exposure of a worker to an inert radioactive gas and to elemental tritium (ICRP Publication 30).

4.1.2.1 <u>Relative Magnitudes of Dose-Equivalent Rates from External and Internal Radiation (ICRP Method)</u> Consider a person submerged in a radioactive cloud of infinite and extent and of volume concentration C Bq m^{-3}. Let the dose-equivalent rate to any tissue from external radiation be $\dot{H}_E$, from internal irradiation by absorbed gas be $\dot{H}_A$, and to the lung from contained gas be $\dot{H}_L$.

Then $\dot{H}_E$, in a small element of tissue in a person submerged in a radioactive cloud of infinte extent, is given by:

$$\dot{H}_E = C\ skg_E/\rho_A \qquad \text{Sv h}^{-1} \qquad (15)$$

where ρ_A, the density of air, is about 1300 g m^{-3};

s (in Sv h^{-1} is the dose-equivalent rate in a small element of any medium of infinite extent uniformly contaminated at a concentration of 1 Bq g^{-1};

k, which is usually close to unity, is the mass stopping power of radiations in tissue relative to their mass stopping power in air, and

g_E is a geometrical factor to allow for shielding by overlying tissues.

The value of g_E is always zero of β emissions from tritium and for all x-particle emissions, since these are unable to reach any of the sensitive tissues of the body, including the lenses of the eyes and the basal layer of the epidermis which, for the purposes of dosimetry, are taken to be at depths of 3 mm and 70 μm respectively (paras. 62 and 64, ICRP Publication 26). For most β emissions and for low energy photons, g_E is about 0.5 for tissue near the surface of the body and tends to zero for deep-lying tissues. For very penetrating photons, g_E approaches unity for all tissues of the body.

After prolonged exposure of the person to the cloud an equilibrium is reached between the concentrations of gas in air and tissue. Under these conditions it may be shown that the concentration of gas in tissue C_T is givey by:

$$C_T = \delta C/\rho_T \qquad \text{Bq g}^{-1} \qquad (16)$$

where ρ_T the density of tissue, is about 10^6 g m^{-3}, and

δ is the solubility of the gas in tissue expressed as the volume of gas in equilibrium with unit volume of tissue at normal atmospheric pressure.

This solubility coefficient increases with the atomic weight of the gas, e.g. in water at body temperature its value from about 0.02 for hydrogen to about 0.1 for xenon (Kaye and Laby, 1956). These values may be increased by a factor of 3-20 in adipose tissue (Lawrence et al., 1946). Thus the dose-equivalent rate in tissue from absorbed gas, $\dot{H}_A$, is given by:

$$\dot{H}_A = s\,\dot{\delta} C g_A / \rho_T \qquad \text{Sv h}^{-1} \qquad (17)$$

where g_A is a geometric factor determined by the dimensions of a person and the range of the radiations concerned. For χ and β emissions and also for the low energy photons g_A will be approximately unity for tissues at the centre of the body and 0.5 for surface tissues. For more energetic photons, g_A is much less than 1 for all tissues and decreases with increasing photon energy.

The dose-equivalent rate in the lung from contained gas, H_L, is given by:

$$\dot{H}_L = sCV_L g_L / M_L \qquad \text{Sv h}^{-1} \qquad (18)$$

where V_L, the average volume of air contained in the lungs, is about $3 \times 10^{-3}\ m^3$;

M_L, the mass of the lungs, is taken to be 1000g (ICRP Publication 23);

g_L is is a geometrical factor which, or χ and β emissions and for low energy photons, is approximately unity. The value of g_L decreases with increasing photon energy.

4.1.2.1.1 Tritium Case For this nuclide, $\dot{H}_E$ given by equation (15) is zero for all relevant tissues of the body because of the short range of the tritium β emissions in tissue. The ratio of dose-equivalent rate in any tissue from absorbed gas to that in the lung from gas is, from equations (17) and (18), given by:

$$\frac{\dot{H}_A}{\dot{H}_L} = \frac{\delta g_A M_L}{V_L g_L P_T} \qquad (19)$$

Since ρ_T is about 106 $g\ m^{-3}$ and $M_L/V_L = 10^6/3\ g\ m^{-3}$, while g_A and g_L are both approximately unity for tritium β emissions, the expression reduces to:

$$\frac{\dot{H}_A}{\dot{H}_L} = \frac{\delta_g \dot{M}_L}{V_L g_L \rho_T} \qquad (20)$$

For tritium δ is about 0.02 for aqueous tissues and 0.05 for adipose tissues (Lawrence et al., 1946). Thus the dose-equivalent rate in lung from the tritium gas contained within it will be 60 to 150 times that in any tissue from absorbed gas. Therefore, in this report, submersion in tritium gas is limited solely by consideration of dose-equivalent rate in the lung. However, it is emphasized that the limit on exposure to tritiated water is very much less than that for elemental tritium and in most cases in practice exposure to tritiated water will be the limiting factor (see dosimetric data for hydrogen).

4.1.2.1.2 Noble Gases All the radioisotopes of the noble gases argon, krypton and xenon considered in this report emit either photons or β particles of considerable energy. Thus, for tissues near the surface of the body, including the skin, g_E will be about 0.5. From equations (15) and (18)

$$\frac{\dot{H}_E}{\dot{H}_L} = \frac{M_L k g_E}{V_L g_L \rho_A} \geq \frac{130}{g_L} \qquad (21)$$

and since g_L cannot be greater than unity, $\dot{H}_E$ is more than 130 times $\dot{H}_L$. Similarly from equations (15) and (17)

$$\frac{\dot{H}_E}{\dot{H}} = \frac{\rho_T k g_E}{\rho_A \delta g_A} \qquad (22)$$

and since $\delta \leq 2$ (Lawrence et al., 1946), $\rho_T/\rho_A \simeq 800$, and $g_A \leq 1$, H_E is more than 200 times $\dot{H}_A$.

Therefore, when applying the system of dose limitation described in Chapter 2 of ICRP-30, it is clear that, for exposure by submersion in radioisotopes of the noble gases, external irradiation will be of such overriding importance that it alone need be considered. Thus, in this report dose equivalents from absorbed gas and gas contained in the lung have been disregarded. The methods used to calculate dose-equivalent rate from external irradiation are discussed in Section 4.1.2.2.

4.1.2.1.3 Daughter Radionuclides If a daughter radionuclide produced by decay of a radioactive inert gas is itself an inert gas exposure by submersion in the daughter radionuclide will be limited by the dose equivalents in tissues from external radiation. If the daughter radionuclide is not an inert radioactive gas, it can be shown that in practice dose equivalents from daughters produced from their parent absorbed in body tissues will usually be small compared with the external dose from parent and daughter outside the body.

4.1.2.2 Dose-Equivalent Rates in Body Tissues from Submersion The energy spectra from sources of mono-energetic photons in an infinite extent of air have been calculated by Dillman (1971). From these data the dose-equivalent rate $\dot{H}$ Sv h^{-1} to organs and tissues of the body may be calculated for a Reference Man situated within a semi-infinite cloud bounded by the floor on which he stands using methods described by Poston and Snyder (1974).

The dose-equivalent rate in the lens is assumed to be the same as that in the skin, for the purposes of this calculation taken to extend from 0-2 mm depth (ICRP Publication 23).

No energy spectra are available for photon sources in a cloud of finite dimensions such as that in a room, but an approximate estimate of the dose-equivalent rate in the skin or internal organs is givey by:

$$2\dot{H}\left[1 - \exp\left(-\mu_A \rho_A r\right)\right] \qquad (23)$$

where $\dot{H}$ is the dose-equivalent rate estimated for a semi-infinite cloud (Poston and Snyder, 1974) and the factor 2 is used because for most room sizes the floor no longer limits irradiation by the cloud to a 2π geometry, at least for the head of a standing worker and most room dimensions;

μ_A is the mass energy absorption coefficient in air;

ρ_A is the density of air, and

r is the effective radius of the room.

When r is the radius of a sphere equal in volume to that of the dose-equivalent rate is overestimated by a small factor which depends on the shape of the room.

$\left[1 - \exp (-\mu_A \rho_A r) \right]$ is the factor by which the first interaction dose from an infinite cloud is reduced, to allow for the fact that there are no sources of photons outside the room.

In this report correction factors are used for room volumes of 100 m^3 (r = 2.9 m), 500 m^3 (r = 4.9 m) and 1000 m^3 (r = 6.2 m), where such factors are necessary.

<u>Electrons and Emitters</u>

The dose-equivalent rate in the skim at a depth of 70 μm and in the lens at a depth of 3 mm from a semi-finite cloud of electron sources is computed by integrating the point kernel of Berger (1971) over the appropriate locus in air where the emitter could contribute to this dose-equivalent rate. The point kernels were corrected for the ratio of mass stopping powers tissue/air taken to be 1.14 for all electrons energies (Berger, 1974). In most cases the correction for room size is small and it is not estimated for electrons in this report.

The energies of the bremsstrahlung for electron sources have been calculated by Dillman et al., (1973) and the dose-equivalent rates from these photon sources are estimated as described above. Their contribution to the total dose-equivalent rate is very small.

4.1.2.3 <u>Derived Air Concentration for Submersion</u> As previously discussed, exposure to elemental tritium in air in any year is limited by consideration of stochastic effects in the lung as follows,

$$w_{LUNG} \dot{H}_{LUNG} \int C(t)\, dt \leq 0.05 \text{ Sv} \qquad (24)$$

where w_{LUNG} is the weighing factor for lung given in table 4-1.

$\dot{H}_{LUNG}$ (in Sv m^{-3} Bq^{-1} h^{-1}) is the dose-equivalent rate to lung from exposure to unit concentration of tritium in air (i.e. 1 Bqm^{-3}), and

$C(t)$ (in Bq m^{-3}) is the concentration of elemental tritium in air at any time t and the limits on integration are over a working year.

Exposure to an inert radioactive gas in any year is limited by consideration of external radiation of the body as follows:

$$\sum_T w_T \dot{H}_T \int C(t)\,dt \leq 0.05 \quad \text{Sv} \qquad (25a)$$

and

$$\dot{H}_T \int C(t)\,dt \leq 0.05 \quad \text{Sv} \qquad (25b)$$

and

$$\dot{H}_{Lens} \int C(t)\,dt \leq 0.3 \quad \text{Sv} \qquad (25c)$$

where w_T is the weighting factor for tissue T and has the values given in table 4-1,

$\dot{H}_T$ (in Sv m^{-3} Bq^{-1} h^{-1} is the dose-equivalent rate in any tissue T, and

$\dot{H}_{LENS}$ is the corresponding value for the lens of the eye, resulting from submersion of Reference Man in unit concentration of the inert gas in air (i.e. 1 Bq m^{-3}).

$C(t)$ (in Bq m^{-3}) is the concentration of the inert radioactive gas in air at any time t and the limits on integration are over a working year.

For convenience, the Commission recommends values of derived air concentration (DAC) which are 1/2000th of the greatest value of $\int C(t)dt$ which satisfies relationship in equation (24) for elemental tritium or which satisfies the relationship shown in equations (25a, b, and c) for an inert radioactive gas.

When the DAC for an inert radioactive gas is determined by consideration of non-stochastic effects, the organ or tissue concerned (usually the skin) is named below the value of DAC. In all such cases a greater value of DAC is given in parentheses. This value is determined by consideration of the Commission's recommendations for limiting stochastic effects, i.e. equation (25a) above. This value of DAC determined by consideraiton of stochastic effects is useful when considering the limitation of exposure from several sources and also when application of the Commission's system of dose limitation (Section E, ICRP Publication 26) requires that a worker, for example, receives only a fraction of the dose-equivalent limit for stochastic effects.

It must be emphasized that DAC must always be used circumspectly. The overriding limit on exposure by submersion to elemental tritium is determined by the relationship shown by equation (23) and to an inert radioactive gas by the relationship presented in equation (24).

4.2 CHARACTERIZATION OF HUMAN HEALTH EFFECTS

The literature on the biologic affects of ionizing radiation is quite extensive and indicates that concern that has been manifest throughout the world about the potentially harmful effects of an expansion of nuclear technology and other applications of radiation. Indeed, it is reasonable to state that we have more scientific evidence on the hazards or ionizing radiation than on most, if not all, other environmental agents that affect the general public. Especially important is the evidence that has been obtained from studies of human populations that have been exposed to radiation for various reasons; however, the large body of experimental evidence on cell systems and animals is also important for our understanding of radiation effects on living systems.

Radiation effects have been classified traditionally as "somatic" if manifested in the exposed subject and "hereditary" or "genetic" if manifested in the descendants of the exposued subject. However, the term "genetic" is also applicable to effects that involve changes produced in the informational macromolecules of cells. Thus, some somatic effects of radiation may be mediated by genetics mechanisms that affect a wide range of body cells, whereas genetic effects involve only germ cells in the gonads.

The term "stochastic" is used to describe effects whose probability of occurrence in an exposed population (rather than their severity in an affected individual) is a direct function of dose. Stochastic effects are commonly regarded as having no threshold - that is, any dose, however small has some effect, provided that the population exposed is large enough. Heriditary effects and some somatic effects, such as cancer induction, are considered to be stochastic. The term "nonstochastic" is used to describe effects whose severity is a function of dose. For these effects, there may be a threshold - that is, there may be a dose below which there is no effect. Examples of nonstochastic somatic effects are cataracts, nonmalignant skin damage, hematologic deficiencies, and impairment of fertility.

4.2.1 Physical Aspects of the Biologic Effects of Ionizing Radiation

All ionizing radiation affects cells by the action of charged subatomic particles which dislodge electrons from atoms in the irradiated material, thus producing ions. By this mechanism, energy is transferred from the radiation to the material, and the amount of energy absorbed per unit mass of the material is the absorbed dose, D. Radiation is directly ionizing if it carries an electric charge that directly interacts with atoms in the tissue or medium by electrostatic attraction or repulsion. Indirectly ionizing radiation is not electrically charged, but results in production of charged particles by which its energy is absorbed. This kind of radiation produces high-velocity fragments of the atoms of the irradiated material; and these fragments become the source of energetic charged particles, which then act to ionize other atoms. It takes about 34 electron volts (eV) of energy to produce one ionization. Most human exposures to radiation are at energies of 0.05-5 million electron volts (MeV)-energies at which many ionizations occur as the radiation passes through cells.

A fundamental characteristic of charged particles produced directly or indirectly is their linear energy transfer (LET), which is the energy loss per unit of distance traveled, usually expressed in kiloelectron volts (KeV) per micrometer (μm). The LET, which depends on the velocity and the charge of the particle, can vary from about 0.2 to more than 1,000 keV/μm.

Some particles expend virtually all their energy at linear energy transfers of less than a few kiloelectron volts per micrometer. In human exposures, the most significant of these particles are μ-mesons (muons), which are the principal components of primary cosmic radiation, and electrons, especially those emitted by beta radiation. Such high-energy electrons, as well as the indirectly ionizing radiation that produces them (that is, x rays and gamma rays), are referred to as low-LET radiation. This radiation is responsible for most of the absorbed doses received by the general population and by radiation workers, but high-LET radiation also contributes. The most important directly ionizing high-LET radiation is alpha radiation emitted by internally deposited radionuclides. Neutron radiation is the principal kind of indirectly ionizing high-LET radiation; neutrons interact mainly by producing recoil protons. Low-energy electrons are produced by both direct and indirect ionizing radiation and are intermediate in LET.

Ionizing radiation interacts with matter along more or less straight charged-particle tracks, but the deposition of energy is not uniform, especially if small volumes and low absorbed doses are considered. In the latter case, the energy is delivered to this volume in only a small number of discrete interactions (i.e., only a few particle traversals). The nuclei of the cells in the human body, which are the loci believed to be primarily affected by ionizing radiation at low doses, have an average diameter of roughly 5 μm. At radiation levels that are of interest in human exposure, the energy absorbed in these structures can vary greatly and, thus, differ substantially from the mean. It is therefore necessary to consider the microdosimetric quantity specific energy, z, which, like the absorbed dose, D, is defined as energy divided by mass, but denotes values of this quotient in a localized region (in this case, the cell nucleus). The importance of this quantity becomes apparent if one determines the values of z in cell nuclei that have received about 1 yr of background radiation. This produces an absorbed dose of about 100 mrads of (mostly) low-LET radiation. In about two-thirds of the nuclei, z = O, that is, no ionizations have occurred; in the remainder, z varies over several orders of magnitude, with an average value of about 300 mrads. If the same dose, D, were delivered by fission neutrons, z would differ from zero in only about 0.2% of the nuclei; however, in these affected nuclei, it would average 50 rads, i.e., 500 times the average dose. It is evident that the heterogeneity of energy deposition depends greatly on radiation type.

4.2.1.1 Relative Biological Effectiveness Because $\bar{z}$, the average value of z, is always equal to D, microdosimetric considerations would be of little interest if the biologic effect* of radiation were simply proportional to z. In this case, the biologic effectiveness of radiation would be independent of LET, which is contrary to experience. The relative biologic effectiveness (RBE) of high-LET radiation relative to low-LET radiation is defined as D_L/D_H, where D_L and D_H are, respectively, the absorbed doses of low- and high-LET radiation required for equal biologic effect. The RBE is generally larger than 1, and values in excess of 50 have been reported for some types of cell effects at low absorbed doses. That is, high-LET radiation requires lower doses to produce equivalent effects. In general, increasing energy concentration in the cell results in a more than proportionally increased probability of effect. Probable exceptions to this are some effects on the genetic material that produce point mutations or cell transformations. However, for some genetic, as well as somatic, effects, the cell may respond to radiation energy in a nonlinear manner. Experimental evidence indicates that the response in these cases can be characterized as quadratic and is consistent with dependences on the square of the specific energy, z. The quadratic dependence on specifc energy might be due to a mechanism whereby biologic effects result from misjunction of pairs of broken DNA molecules. However, this interpretation must still be regarded as hypothetical, and we use here a conservative terminology that states that the basic action is one in which pairs of sublesions combined to form lesions.

If it is estimated that the average range of interaction of sublesions is roughly 1 μm and it is assumed that the yield of sublesions is proportional to the mean value of specific energy, i.e., to the absorbed dose, then E, the frequency of effects (numbers or probabilities of lesions that depend on the combination of two sublesions), is proportional to the square of the specific energy, Thus,

$$E = K\overline{z^2} \qquad (26)$$

It can be shown[13] that $\overline{z^2}$, the mean value of z^2, is given by

$$\overline{z^2} = \zeta D + D^2, \qquad (27)$$

where ζ is a microdosimetric quantity. Thus

$$E = K\ (\zeta D + D^2). \qquad (28)$$

In this model, if the critical specific energy is deposited in sites of 1 -μm diamter, the applicable values of ζ would range from 12.5 to 25 rads for low-LET radiation. Larger values would apply for small sizes. The value of for high-LET radiation on the basis of microdosimetry would typically be 100 times larger than the value for low-LET radiation. Therefore, the linear term would be much more important for high-LET radiation.

When D = ζ, the linear and quadratic, terms in equation (28) are equal. When D is less than 0.1 (i.e., the absorbed dose is low), the quadratic term becomes negligible, and the energy is deposited by single particles. Consequently, the fraction of the cells receiving energy is proportional to the absorbed dose and dose rates.

Equation (26) implies that the RBE should vary from approximately 1 at high absorbed doses to the ratio of the ζ values of high- and low-LET radiation at low absorbed doses. If this ratio were substantially larger than 1, there should be a considerable range of absorbed doses at which the RBE would be inversely proportional to the square root of the absorbed dose of high-LET radiation down to the doses where both the high- and the low-LET responses would be linear with dose. This behavior of the function relating RBE to the absorbed dose of the high-LET radiation, including RBE values up to 100, has often been observed experimentally for fission neutrons.

The above considerations and conclusions briefly summarize the theory of dual radiation action on autonomous cells. This simple form is, however, subject to qualifications and modifications. According to the simplified theory, at low absorbed doses any radiation effect on autonomous cells must be proportional to absorbed dose and independent of absorbed dose rate. This conclusion applies even if there is a variation in radiation sensitivity among the cells and even if repair processes are operative, and whether or not there is a quadratic response. On the average, an event in the nucleus carries a probability of producing a given effect, and the fraction of cells affedted is the product of this probability and the fraction of nuclei that could be affected. The latter fraction is proportional to th absorbed dose at low doses. However, when the absorbed dose is large enough for there to be an appreciable probability of multiple events, proportionality between absorbed dose and effect can no longer be expected, even for autonomous cells. According to equation (28) for a dose of $n(\zeta)$ rads, the effect will be

$\left[\frac{n(n+1)}{2}\right]$ times greater than the effect at ζ rads.

The relationship given by equation (28) is shown in a logarithmic presentation in figure 4-2, which indicates the magnitude of the error than can occur in linear extrapolation. The unit of absorbed dose is ζ i.e., the absorbed dose where the linear and quadratic components are equal, and the effect is plotted in units relative to the linear contribution at $D = \zeta$. It can be seen in figure 4-2 that there are about 2 decades of absorbed dose between the point where the slope of the curve is 1.1 and the point where it is 1.9. Precise radiobiologic data covering a hundredfold range of absorbed dose are rare, and thus it is not surprising that the entire transition from a linear to a quadratic dependence has rarely been observed, although this transition has been approached with low-LET radiation.

In general, data for yields, E, of cell effects can be satisfactorily fitted empirically to an expression of the form, similar to equation (28) as follows:

$$E = aD + bD^2 - C \qquad (29)$$

where C is the zero-dose incidence, and a and b are empirically determined coefficients. There is disagreement, however, over the meaning of the coefficients a and b, at least in the form in which they are determined by simple fitting of equation (29) to the experimental data points.

An alternative interpretation is that the end points in question—for example, mutations—may depend on the operation of more than one mechanism. That is, there may be more than one biologic mechanism involved in addition to the presumed "dual-action" mechanisms of physical absorption. There may be more than one class of events involved in point mutations, as discussed in BEIR I (1971). Furthermore, the end point, mutation, may result from the operation of both repair and damage mechanisms and may involve a variety of lesions. From this standpoint, it might be argued that the best estimate of damage at very low doses would be a linear extrapolation between the yield at the lowest dose for which there are reliable data and the yield at the zero dose. Such an estimate would not differ appreciably from that based on the quadratic relationship, provided that the value of bD^2 at the lowest measured dose is not appreciably different from zero.

A further complication at large absorbed doses is that radiation may produce a variety of effects. Because it has been assumed that each of these results from particular groupings of sublesions, it may be expected that, as the number of these increases, competition between effects may alter the dose dependence for one particular effect. An example of considerable practical importance concerns the interplay between malignant cell transformation and cell-killing within the same cell. Evidently, transformed cells cannot initiate tumors if they also have suffered reproductive death, which becomes increasingly probable at higher absorbed doses. Thus, dose-response data may show a decrease in effect at high doses—the so-called "cell-killing" efffect.

Several experiments on radiation-induced transformation of cells in cell culture have yielded dose-effect curves whose slopes decrease between the linear and quadratic regions shown in figure 4-2. This example illustrates the fact that the dose-effect curves for autonomous cells can have complex shapes and that extrapolation from high doses can

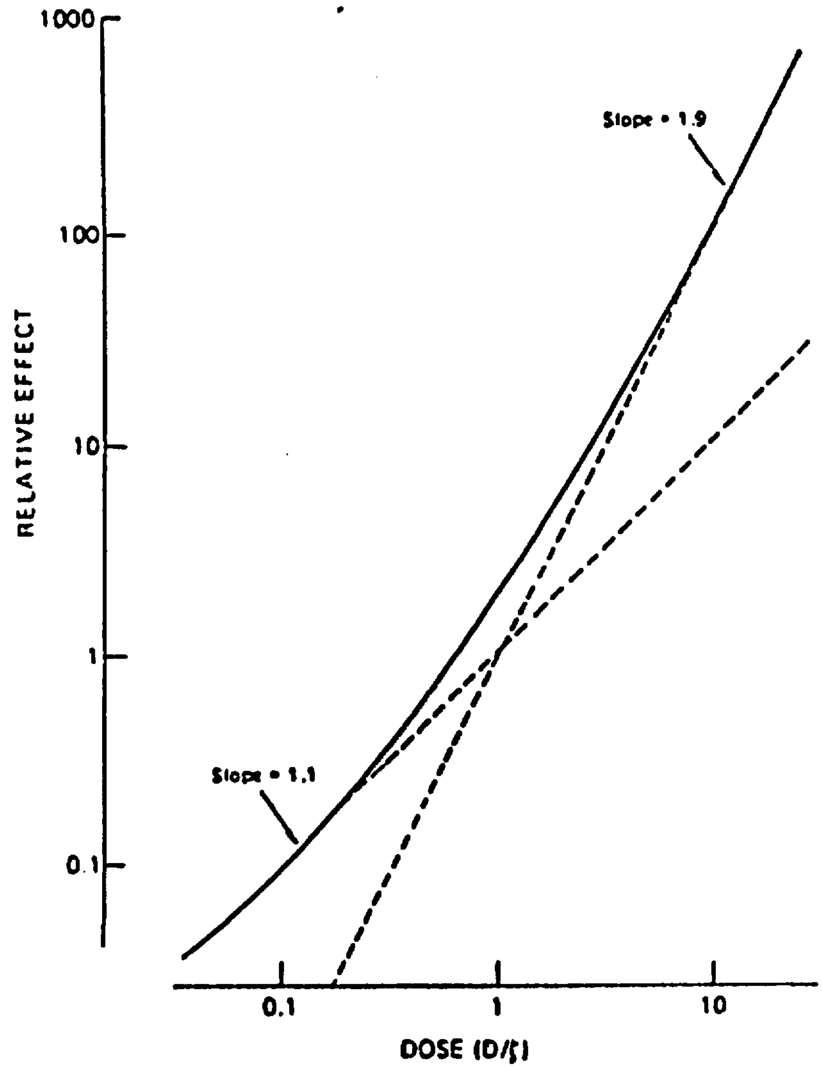

Figure 4-2. RELATIVE BIOLOGICAL EFFECT AND ABSORBED DOSE RELATIONSHIPS FOR AUTONOMOUS CELLS

lead to an underestimate of the effect of low doses. The effect can be explained in terms of competition for sublesions in which the alternative effect is not cell-killing, but one of a variety of possible nonlethal cell alterations. A related finding is that, if the total dose is given in several successive fractions, rather than all at once, the transformation rate is unchanged in the linear region at the lowest doses, reduced in the quadratic region at the highest doses, but increased in the intermediate region where the slope of the curve is less than 1. This is to be expected, if there is no interaction between the dose fractions. Finally, in such systems, the RBE could be less than would be deduced from the ratio of ζ values. If single high-LET particles produce increments of ζ that are comparable with the range of absorbed doses for which there is a relatively constant transformation rate, the RBE might be considerably less than expected on the basis of the considerations presented above.

4.2.1.2 <u>Relation Between Radiation Effects on Cell Systems and Mutagenesis or Carcinogenesis in Man</u> Some radiation effects are apparently due to damage to individual autonomous cells. In human radiation exposure, the most important example might be the mature gametes in the gonads. Other effects, such as cateractogenesis, are due to injury of several cells. Here, one would expect proportionality between dose and effect, whether or not the cells involved in the response were autonomous.

For the most important somatic radiation hazard, carcinogenesis, it is often assumed because the number of cells at risk is very large that transformation of an individual cell does not necessarily result in cancer. Among the various inhibitory mechanisms that have been considered is a requirement that several contiguous cells be transformed, or the action of immunologic or other host defenses be imparied. In the former case, a multicellular interaction would be involved; in the latter, the response of individual cells may not be autonomous—for example, if the effectiveness of the defense mechanisms is limited by the number of cells transformed.

In both situations, the dose-effect curve could have various forms at low absorbed doses. For example, a downward curvature of the dose-response relationship has been observed for radiation-induced mammary neoplasms in one strain of rat at absorbed doses of neutrons that are clearly much less than ζ, which indicates that this malignancy is not due to an autonomous-cell response. In this system, however, the RBE increases inversely with neutron dose in the same manner as observed for autonomous single cells. However, for both high-LET and low-LET radiation in dose ranges where the single-cell

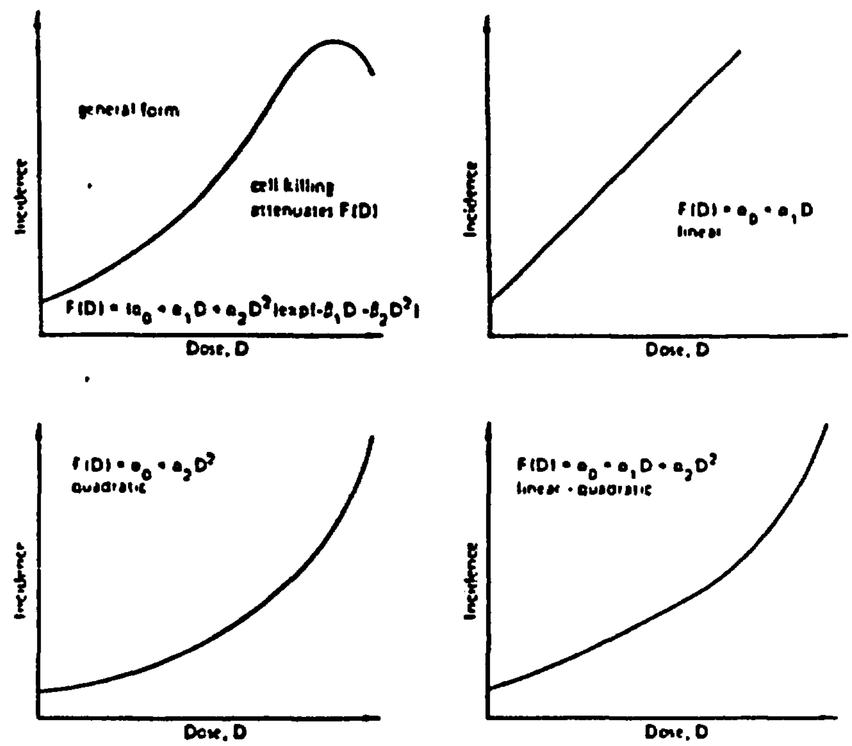

Figure 4-3. ALTERNATIVE DOSE-RESPONSE CURVES

the effects that appear much later is even more difficult. Furthermore, not only do individual cells vary in their response to radiation, but tissues contain many different types of cells and many biologic interactions occur with and among tissues, so we may expect the effects of cell damages to be very complex indeed. This section considers some of the biologic factors that may influence responses to radiation.

4.2.2.1 Cell Division An important effect of radiation, which accounts for the symptoms and causes of death from exposure to large doses of whole-body irradiation, is suppression of cell division. Nearly all lymphoid, bone-marrow, and intestinal epithelial cells responsible for rapid replacement of short-lived mature cells cease to be able to divide, and in these and many other tissues a substantial fraction of cells that would otherwise be capable of division die without further reproduction. If the organism is to survive these effects, the remaining stem cells must repopulate the tissues to overcome cell loss. An example of this process is the disappearance of granulocytes, as a result of suppression of cell division of precursor cells in the bone marrow, in the blood of persons irradiated at relatively high doses. Recovery may require days or weeks.

Cell killing and suppression of cell division are nonstochastic effects of radiation—the ultimate biologic effects depend markedly on the fraction of cells affected. At low radiation doses, only a small fraction of the dividing cells may be damaged, and in tissues this damage may lead to no detectable change in function. In tissues with rapid cell turnover, interference with normal function will occur only when the affected cells constitute a large fraction of those available for replenishment of cell stores. We anticipate that host factors play an important role in determining the fraction of cells required to produce serious physiologic or biochemical abnormalities in association with this disturbance in cell replacement, especially in the intestinal tract and in the population of white blood cells. Such host factors include general nutritional status (e.g., availability of nutrients important in cell growth), the presence or absence of preexisting infection, or exposure to chemicals to drugs that have effects on cell division similar to those of radiation.

Nevertheless, because these effects are observed at high doses of radiation, they are of limited interest in this report. An exception is the irradiation of the developing fetus.

4.2.2.2 Host Factors in Radiation Carcinogenesis Present evidence indicates that cancer induced by chemical or physical agents, such as ionizing radiation, involves a multistage process, with evolution of molecular and cellular changes leading to changes in the tissue as a whole. The earliest stage of this process is the so-called initiation phase, in which events leading to lesions in the DNA occur in a single cell or in a small group of cells. These cells have the capability of transforming into a neoplastic process—that is, normal growth constraints are altered in these cells. There are control mechanisms in tissues that act to prevent development of transformed cells into a malignant tumor. These regulatory processes involve the normal cells adjacent to the transformed cells, as well as hormonal, immunologic, and other influences in the tissue or the body. Inherited traits can influence all stages of cancer by modifying tissue responses to initiation, as with the DNA repair mechanism, or by variations in the regulatory mechanisms.

The process that affects the regulatory control exerted on the transformed cell or cells in a way that permits them to begin uncontrolled growth leading to a cancer is referred to as "promotion." Some physiologic disturbance of the tissue frees the potentially rapidly dividing cell or cells from constraints on cell division. Such disturbances may include repeated damage to normal tissue, stimuli to cell proliferation (such as hormonal effects), or disturbances in recognition of immunologically transformed cells by immune processes.

This is a brief statement of the two-stage theory of carcinogensis. The first stage is initiation, associated presumably with eventual alteration in the cell genome, which causes loss of normal control of cell division in transformed cells. The second stage is promotion, a process by which a transformed cell is able to grow into a detectable cell mass identifiable as a cancer. These two stages may be separated by many years, a factor accounting at least in part for the long latent periods often observed in man between exposure to a carcinogen and development of a cancer.

Both the initiating and the promotion steps can be modified by biologic factors, including those characteristic of the host, acting in concert with a carcinogen, such as radiation. The probability of an initiating event may be affected, for example, by whether the cell nucleus already contains viral nucleoproteins incorporated into the DNA. In this sense, viral infection may play a permissive role in the induction process—a necessary but not sufficient condition for carcinogenesis.

It is clear, however, that host factors are especially important in the promoting stage, where relatively nonspecific alterations of normal tissue function may be important.

Hormonal influences, which clearly exert great effects on all proliferation in normal tissues, are one factor of considerable significance, at least in some cancers. The importance of hormones is determined by the tissue type; for example, sex hormones regulate growth in the sex organs, and pituitary hormones influence cell proliferation in the gonads, as well as endocrine glands, such as the thyroid. The immunologically active lymphoid cells, which may suppress or destroy transformed cells if they are recognized as immunologically "foreign" to the host, may also be important.

Another factor in cancer promotion is the alteration of normal tissue integrity by a wide range of conditions, including irritant chemicals that reach epithelial structures, vitamin A deficiency, viral infection of the respiratory tract, and trauma. The precise role of any of these factors is not well understood in human carcinogenesis, but at least under experimental conditions their importance has been demonstrated for some neoplasms.

Finally, changes associated wtih the aging process have been postulated as predisposing to cancer through deterioration of tissue repair and loss of vitality of the normal cell complement.

This brief summary of mechanisms of carcinogenesis has been presented, because it is apparent that circumstances leading from cellular radiation effects to cancer involve many factors that may be highly variable in an exposed population. For this reason, we may expect sensitivity to cancer induction by radiation to be variable from individual to individual, as well as from time to time in the same individual. Thus, data on radiation dose versus cancer response obtained in cell systems or even in experimental animals must be applied to human population with considerable caution.

4.2.3 Epidemiologic Studies as the Basis of Risk Estimates for Effects of Ionizing Radiation

In assessing somatics affects of ionizing radiation, various research groups (BEIR, 1971) placed primary emphasis on studies of exposed human populations. In contrast, estimates of risks of heriditary effects on human populations have depended principally on evidence from animal experiments. However preferable it may be to have firm evidence of hereditary changes based on exposures of human populations to ionizing radiation, detection of increases in human mutations due to the action of any environmental agent is still difficult. For somatic abnormalties induced in utero by radiation, the position is somewhat intermediate—that is, some human data have been obtained, but we also depend on animal data.

The emphasis on human studies for determining the somatic effects of ionizing radiation remains valid, although theoretical and experimental studies continue to be important in extending our basic knowledge. For most types of health effects occurring in those exposed to radiation, we now have considerable human experience, as the balance of this report shows. Moreover, in terms of establishing human risk estimates, it is a well-recognized principle in the field of environmental toxicology that results obtained in animal experiments are not necessarily translatable directly to human populations. For example, the fact that the human population is genetically heterogeneous, with widely varying individual physiologic and biochemical characteristics, makes it likely that there are subpopulations at special risk from radiation exposure. It is difficult to simulate this kind of heterogencity in animal populations, other than by inferences drawn from species variation in responses or from differences is susceptibility between strains of a given species.

We lack adequate information on the effects of low radiation doses in human populations, and in this regard we still depend on concepts that have been developed on the basis of experimental studies.

Although epidemiologic studies constitute our principal source of information on somatic effects of ionizing radiation in human populations, one must recognize that there are problems in their use. The first problem arises from the fact that generally the group has been exposed to radiation because of some particular characteristics and thus may not be representative of the population at large. The reasons why those exposed to radiation are not typical of the general population may not affect radiation sensitivity, but an appropriate comparison group is nonetheless required.

The epidemiologic technique to deal with the scientific problem of an potentially biased sample is to obtain a control group matched as nearly as possible to the exposed persons. In radiation epidemiology, considerable effort has been made to deal with the question of the suitability of a control group. For example, in the Japanese atomic-bomb survivors, the zero-dose groups (those in the cities at the time of the bombing, but so far away from the bomb detonation that they were not exposed) are useful controls, although in the Nagasaki sample they are comparatively few. An alternative method has been to consider the regression of effects (such as cancer rates) on radiation dose. Systematic differences in rates of cancer not related to radiation exposure, for example, might be expected to be uniform throughout all dose categories; thus, any trend associated with radiation dose would indicate a radiation-induced effect.

A second problem in studies of radiation effects on human populations arises because most of them are retrospective-that is, exposure to radiation has occurred in the distant past, so the exact dose of radiation delivered to individuals or to a group is often not known. This problem is common to all retrospective studies of effects of environmental agents on human populations. In the case of radiation exposures, it has often been possible to estimate the radiation dose after the fact. For example, for the Japanese atomic-bomb survivors, great efforts have been made to determine the radiation dose-distance relationships of the Hiroshima and Nagasaki bombs, to locate the site of exposure of each person in the city at the time of bombing, and to determine the degree of shielding by buildings or terrain that may have reduced the radiation exposure.

Despite the problems with radiation dosimetry in retrospective studies, determination of excess cancer is generally of value, even in groups lacking dose estimates. Studies that produce inconsistent results suggest that radiation exposure is not a principal causative factor or that other factors have a role in carcinogenesis. A degree of consistency of results in a large number of studies constitute major support for defining somatic risk.

A third problem in the use of epidemiologic data arises from the very long latent periods that may separate exposure to radiation and the development of effects in man. This is a problem especially if the latent period is influenced by demographic variables. For example, for some solid tumors, the latent period for cancer development may be longer for persons exposed to radiation when they are younger. A minimal latent period as long as 30 years or more after exposure means that the true health risk of radiation exposure can be assessed only with extremely long followup of the populations under study. In general, followup of irradiated groups has not proceeded this long, so the extent to which risks of radiogenic cancer have been identified is not clear. This is one of the principal reasons why risks based on current followup studies may be underestimated, especially for persons irradiated at earlier ages. Therefore, to use the epidemiologic evidence in human studies available for any particular followup interval, it is necessary to make some assumptions about the way in which further cases are likely to appear in later years.

Accordingly, two models for projecting the effect of radiation expousre at a particular level were used by the original BEIR Committee. The first of these was the so-called absolute-risk model. According to this model, if a population was irradiated at a particular dose, either all at once or over some period, expression of the excess cancer risk in that population would begin at some time after exposure (the latent period) and continue at a rate in excess of the of the expected rate for an additional period, the "plateau" or expression period, which may exceed the period of followup. In this model, the absolute risk is defined as the number of excess cancer cases per unit of population per unit of time and per unit of radiation dose, and although it may depend on age at exposure, it does not otherwise depend on age at observation for risk.

In the second model adopted in BEIR I, the so-called relative-risk model, the excess cancer risk for the interval after the latent period was expressed as a multiple of the natural age-specific cancer risk for that population. The chief difference between the two models is that the relative-risk model took account of the differing susceptibility to cancer related to age at observation for risk. For the entire period of actual observation, the risk estimates derived from the absolute-risk and relative-risk models are arithmetically consistent, and the choice of one or the other is a matter of convenience. For the period beyond that from which the estimates were derived, both models make assumptions that may or may not be appropriate. This problem is especially significant for persons exposed either in utero or in childhood, at a time when at least some kinds of cancer appear to be more likely to be induced by radiation than in adults. The assumption of a risk that persists over the life span of a person becomes an important determinant of the total risk, especially if the number of excess cases is proportional to the number of spontaneous cases, which may, for example, increase markedly with increasing age. With the additional evidence now available, we are better able to evaluate the applicability of these two models to the information at hand. It should be noted that, if epidemiologic followup through the entire lifetime is complete, both models will give the same result for lifetime risk.

Support for interpretation of risks as an absolute number of cases of cancer arising from radiation exposure came initially from the analysis of leukemia risks in the Japanese atomic-bomb survivors. It was found by the late 1960's that the number of excess cases of leukemia had risen to a peak about 8 years after the radiation exposure in 1945 and was declining toward the expected leukemia rate in a nonirradiated population. By the early 1970's, the excess risk of leukemia had nearly disappeared in this population. Later analysis of the leukemia excess in the Japanese population has shown that the number of cases per unit of population is a function of the age of the people irradiated.

For most types of cancer arising from radiation exposure, it is apparent with longer followup times that the excess cancer risk remains well beyond 30 years. Indeed, some types of cancer may not even appear in excess 20 year or more after exposure. Therefore, the question in determining final risk estimates is: For how long a period after exposure does an excess risk continue to accumulate? It is clear that the total number of excess cases that will be considered to arise from radiation is influenced by this period of expression, called the "expression time" of the radiation insult. Although for development of leukemia, and bone cancer arising from radium-224 exposure, we may be able to give reasonable estimates of the expression time for cancer production; for virtually all other radiogenic cancers this is not yet possible.

The relative-risk concept assumes that the risk of radiation-induced cancer varies by age at observation and is proportional to the risk of spontaneous development of cancer in the population. An immediate problem, of course, is the question of what constitutes the natural cancer risk in a population. For example, in the case of bronchial cancer, do we except the spontaneous risk as the current risk of lung cancer in a population containing a substantial proportion of cigarette-smokers, or is it proper to use the nonsmoking population as the basis for calculating the risk estimates? Related to this question is the extent to which radiation will either add to or multiply the effects of other cancer-causing agents in the environment.

A second question is whether the relative hazard of radiation applies also to groups that may on other grounds be susceptible to cancer, since the most important factor influencing the risk of spontaneous cancer is age. If the relative-risk model applies, then the age of exposed groups, both at the time of exposure and as they move through life, becomes very important. There is now considerable evidence in nearly all the adult human populations studied that persons irradiated at higher ages have in general a greater excess risk of cancer than those irradiated at lower ages, or at least they develop cancer sooner. In other words, the relative-risk model with respect to cancer susceptibility, at least as a function of age, evidently applies to some kinds of cancer that have been observed to result from radiation exposures. It should be emphasized, however, that this last conclusion depends on how long the populations have been studied; whether the risk remains proportional to the risk of spontaneous cancer in the older stated, i.e., the duration of cancer expressions, whether the temporal expression of risk is relative to the normal age-specifc rate, etc. Finally, wherever possible, the total effect of radiation on a population should be calculated from the age-specific risk of cancer per unit of dose.

4.2.4 Implication of Regulatory Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes

Finalization of the environmental standard 40 CFR 191 by EPA in September 1985 for the high-level nuclear waste program provides the regulatory basis for developing methods of assessing human health risks.

The individual protection requirements in the final rule limit the annual exposure from the disposal system to a member of the public in the accessible environment for the first 1,000 years after disposal, to no more than 25 mrems to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ. These limitations apply to the predicted behavior of the disposal system including consideration of the uncertainties in the predicted behavior assuming that the disposal system is not disrupted by human intrusion or the occurrence of unlikely natural events.

Waste management (not including transportation) and storage activities conducted in accordance with Subpart A, 40 CFR, 191, would limit the maximum risk to a member of the public in the general environment to a 5×10^{-4} chance of incurring a premature fatal cancer over a lifetime. The foregoing probability assumes continuous exposure to the aforementioned wholebody and critical organ regulatory standard continuously over a normal lifetime. Because EPA believes that such continuous exposure is very unlikely, the Agency anticipates much lower actual risks to the individual. Linear, non-threshold dose effect relationships were used by EPA to promulgate these standards for individual protection.

With regard to exposure of populations, the EPA has estimated the long-term health risks to future generations from various types of mined geologic repositories also using very general models of environmental transport and linear nonthreshold dose effect relationships between radiations exposures and premature deaths from cancer.

These health risks models were used to assess the long-term health risks from several different model repositories containing the wastes from 100,000 MTHM-which could include all "existing wastes and the future wastes" from all currently operating nuclear reactors. The Agency estimates that this quantity of waste, when disposed of in accordance with the standard would cause no more than 1,000 premature deaths from cancer in the first 10,000 years after disposal or an average of no more than one premature death every ten years. EPA further maintains that any such increase in the number of cancer deaths would be very small compared to today's incidence of cancer which reportedly kills about 350,000 people per year in the United States. Similarly, any such increase would be much less than the approximately 6,000 premature cancer deaths per year that the same linear, nonthreshold dose effect relationship predicts for the nation due to the natural background radiation.

As previously inferred, regulatory standards governing the transportation system for the high-level nuclear waste repository program are not included in 40 CFR 191; however, potential transportation release scenarios are included in Subpart B-Environmental Standards for the Uranium Fuel Cycle in EPA's 40 CFR 190. Subpart B states that operations covered by this subpart shall be conducted in such a manner as to possible reasonable assurance that the annual dose equivalent does not exceed 25 mrems to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any member of the public as the result of exposures to planned discharges of radioactive material.

Therefore, the basic individual standards for the transportation system and the repository system are identical on the basis of the regulatory standard for annual dose to an individual.

The NRC as part of its licensing and compliance responsibilities for nuclear facilities has several regulatory requirements that are worthy of consideration in the development of the CTUIR risk assessment methodology.

Maximum Premissible Concentrations (MPC) have been specified by the NRC for exposures to the general population from NRC licensed nuclear facilities for most radioisotopes in both air and water media in Appendix B, 10 CFR 20. Selected MPCs for many of the radionuclides of possible concern at the Hanford Site are listed in Appendix C. of this document. MPCs are based on limiting the maximum exposure of the general population to a level which will produce a dose no greater than 500 mrem per year from all sources.

Most of the MPC's listed in Appendix C are based upon dosimetry estimates developed in the 1950 to 1960 and rely heavily on the information contained in ICRP Publications 2, 10, and 10A. The NRC is currently developing new MPC exposure limits based upon recommendations that may incorporate the more recent data and method of International Committee on Radiological Protection (ICRP), specifically ICRP Publication 30, which include the procedures and techniques associated with Common Dose Equivalent (H_{50}) Annual Limits on Intakes (ALI's) and Derived Air Concentrations (DAC's) discussed in some detail earlier Section 4.1 of this report.

It must be emphasized that an important limitation of reliance upon MCP's as a guide for environmental risk assessment method development is the basic assumption that 0.5 rem is adequate for all individuals comprising the total population. This limit may be unacceptable for certain hypersusceptible groups within the population including the elderly, children and those individuals with impaired immune systems as cited previously in Section 4.2.3. Theoretically, a more comprehensive method of determining health impacts of radiation exposure would be to develop risk factors including those which fall below the foregoing regulatory standards promulgated by EPA in 40 CFR 190 and 40 CFR 191 and possibly the MPC limits when updated by the NRC. In many cases these lower exposure levels could be expected to generate only minimal risks. However, it is impossible to determine what the risks would be without a comprehensive analysis utilizing the more recent epidemiological and animal test data.

An additional consideration in determining "safe" levels of exposure or the specific potential health risks due to the proposed Hanford repository program is the level of radiation of exposure already existing in the area. Radionuclides have been found in both surface and ground water in the area due to naturally occurring uranium deposits (CERT, 1984 p. 193). In addition, routine releases of low levels of radionuclides into the Columbia River and its tributaries and unconfined ground water from other nuclear facilities on the Hanford Site are potential additional sources of radiation exposure (CERT, 1984, p. 193). While emissions from the proposed repository may not exceed the 0.5 rem limit, the sum of all radiation sources should be determined and potential health risks evaluated.

4.2.5 Thresholds for Non-Stochastic Health Effects

As previously defined earlier in this section, non-stochastic effects are those for which the severity of effect, rather than the probabilitiy of occurrences is regarded as a function of dose beyond a certain threshold. Examples of these radiation effects include erythema, sterility, conjunctivitis, keratitis, cataracts, and the hematopoietic, gastrointestinal, and central nervous system syndromes. A summary of the thresholds for non-stochastic is prescribed in table 4-3. Table 4-3 illustrates tht non-stochastic human health effects generally require radiation doses of 30 to 2,000 rads to produce the adverse threshold effect.

4.2.6 Carcinogenic Related Stochastic Effects

As previously discussed, stochastic effects are those for which the probability of occurrence, rather than the severity of effect, is regarded as a function of dose. No threshold is assumed for stochastic radiation effects. Even if the most stringent exposure standards are met, any exposure to ionizing radiation conveys some risk of developing a stochastic response to such as cancer or genetic damage.

Organ-specific cancers attributed to stochastic effects are presented in table 4-4. A detailed summary of many of these effects is contained in the aforementioned BEIR Report (1980). In addition, there is a vast compendium of more recent epidemiological and experimental animal data which would require compilation, review, and evaluation to establish the full scope of potentially adverse health effects attributable to ionizing radiation.

4.2.7 Possible Methods for Extrapolation of Stochastic Health Effects

As previously discussed, a stochastic effect occurs with a certain frequency which is a function of dose. At higher doses the frequency (risk, probability, and frequency are synonomous terms) of occurrence is sufficiency high that the effect can be observed in a finite number of exposed subjects. Therefore, theoretically, any mathematical model can be used to fit dose-frequency data in the range of observable effect. The only

Table 4-3. THRESHOLDS FOR NON-STOCHASTIC EFFECTS*

Effect	Threshold	Comments
Hematopoietic syndrome	200 rads of gamma, whole body	Death may occur within 1 to 2 months
Gastrointestinal syndrome	1,000 rads of gamma, whole body	Death within 1 to 2 weeks is likely
Central nervous system syndrome	2,000 rads of gamma, whole body	Death within hours to days
Erythema	300 R	Higher doses may cause pigmentation, blistering, epilation, necrosis, and ulceration
Male sterility	30 rads to testes	Sterility is temporary
Female sterility	300 rads to ovaries	Sterility is temporary
Eyes: conjunctivitis, keratitis	Several hundred rads, local	
Eyes: cataracts	500 rads of beta or gamma to the lens 200 rads mixed gamma and neutrons to the lens 15-45 rads of neutrons 200 rads of X-rays	

* Cember, H. Introduction to Health Physics, Pergamon, 1983.

Table 4-4. CANCERS ATTRIBUTABLE TO EXPOSURE TO IONIZING RADIATION-ORGAN-SPECIFIC STOCHASTIC EFFECTS

Breast	Salivary Gland
Thyroid	Parathyroid
Lung	Ovary
Leukemia	Uterus
Esophagus	Cervix Uteri
Stomach	Bone
Intestine	Bone
Intestine	Brain
Rectum	Skin
Liver	Paranasal Sinus
Pancreas	Mastoid Air Cell
Pharynx	Kidney
Larynx	Bladder

requirement is that the model be a good predictor of frequency of effect over the range of experimental doses. However, at lower doses the frequency of occurrence is sufficiently low that the effect cannot be observed unless extremely large numbers of subjects are exposed. Hence, the true shape of the dose-frequency relationship at very low doses presently cannot be determined from empirical data.

At least nine mathematical models have been used to characterize the dose-frequency relationship at very low doses. All have been used by various research investigators and federal regulatory agencies over the last 20 years. These models have the following names: probit, logit, Weibull, multistage, multihit, linear, one-hit, quadratic and linera-quadratic. Three of these mathematical models, i.e., the linear, quadratic, and linear quadratic, have recently been adapted by the majority of the analysts and researchers were discussed in previous sections of this report. The linear and one-hit models yield about the same result in the very low dose region and are the most conservative in that they yield the highest frequency of effect at a given low dose. Each of the models is based to varying degrees on a hypothetical biological mechanism for the causation of stochastic effects. All of the models must be considered arbitrary in that the true shape of the dose-frequency function at very low dose can never be known for a given chemical or radioisotope, and may be only poorly predicted by high exposure levels. The linear model, which is the current basis for the EPA regulatory standards, should be used due to its conservatism and simplicity when low dose extrapolation calculations are performed.

4.2.7.1 Probabilities and Excess Cases When a release scenario produces a dose which is within the range of doses encountered in an animal or human study, the animal or human study can be used to derive an equation relating dose and frequency. This equation may have any form, including simple linear regression, and can be used to calculate the probability of effect for a given dose.

It is more likely that a particular exposure scenario will produce a dose below the range of doses seen in human or animal studies. In this case, linear extrapolation should be used incorporating organ-specific unit risk factors. A recently developed set of organ specific risk factors utilized by the EPA (Smith, et al, 1987) in the development of the aforementioned regulatory standards (40 CFR 190, 40 CFR 191) is presented in table 4-5. When non-radioactive chemicals are the subject of a risk assessment exposure information (both duration and concentration) is usually sufficient to permit of health effects. This is because most, if not all, unit risk factors for chemicals are in units of risk per unit of concentrations adjusted for duration (= cases/person/unit of concentration), e.g. risk/mg/m^3 or usually and more simply $(mg/m^3)^{-1}$. Unit risk factors for radiation are given in units of cases/person/unit of dose, e.g. risk/rem or risk/Sv or usually and more simply rem^{-1} and Sv^{-1}.

Ideally, unit risk factors should be specific for not only a given organ but also dose rate, race, age at first exposure and sex. Also, unit risk factors should be derived from the lowest available experimental dose since some unit risk factors have been found to be a function of dose with lower doses producing higher unit risk factors.

Although ICRP 26 (1977) and BEIR (1980) contain several organ-specific unit risk factors, the other variables which define a unit risk factor (age, sex, race, dose-rate) are not completely discussed in those references. Epidemiological and experimental animal studies published since the 1980 BEIR report have substantially altered some specific unit risk factors, and provide data on other variables. Recent studies have indicated that higher risks may exist for exposure of some organs to certain types of radiation, e.g. alpha radiation and lung cancer (Radford and St. Clair Renard, 1984). Consequently, the most recent literature must be reviewed to determine these risk factors. These compilations should be revised and updated on a frequent basis, as more verifiable experimental data becomes available. Thus, if a properly defined unit risk factor is available, the probability of effect can be calculated for a given low dose. For example, if the lung cancer risk is desired, the following computation can be made:

$$R_{LC} = U_{LC} \times H_L \qquad (30)$$

where

R_{LC} = excess individual risk (= probability) of developing lung cancer at dose = H_L

U_{LC} = unit risk factor for lung cancer in excess cases/person/Sv or more simply in risk/Sv, risk/rem or Sv^{-1} or rem^{-1}

H_L = committed dose equivalent to the lungs in Sv or rem

Table 4-5. HEALTH EFFECTS CONVERSION FACTORS, HEALTH EFFECTS/MAN REM (FATAL CANCERS FOR ALL ORGANS EXCEPT OVARIES AND TESTES. GENETIC EFFECTS TO FIRST GENERATION FOR OVARIES AND TESTES)*

BONE	RED MARROW	LUNGS	LIVER	GI-LLI WALL	THYROID	KIDNEYS	OTHERORGAN	OVARIES	TESTES
1.000E-05	4.000E-05	4.000E-05	1.000E-05	2.000E-05	1.000E-06	1.000E-05	7.000E-05	2.000E-05	2.000E-05

NUCLIDE DEPENDENT INPUT DATA

NUCLIDE PATHWAY (INHALATION AND INGESTION-REM/CI INTAKE — DOSE COMMITMENT FACTORS AIR SUBMERSION-REM/Y PER CI/M**3 GROUND CONTAMINATION-REM/Y PER CI/M**2)

ORGAN

		BONE	RED MARROW	LUNGS	LIVER	GI-LLI WALL	THYROID	KIDNEYS	OTHERORGAN	OVARIES	TESTES
C-14	INHAL1	8.460E+00	2.420E+01	6.180E+00	8.880E+00	7.220E+00	6.480E+00	7.920E+00	1.410E+01	5.290E+00	5.420E+00
	INHAL2	8.460E+00	2.420E+01	6.180E+00	8.880E+00	7.220E+00	6.480E+00	7.920E+00	1.410E+01	5.290E+00	5.420E+00
	INGEST	1.170E+03	3.380E+03	8.490E+02	1.230E+03	1.460E+03	8.890E+02	1.060E+03	1.920E+03	7.360E+02	7.230E+02
	EXT AIR	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	EXT GND	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NI-59	INHAL1	1.290E+04	2.150E+03	8.470E+03	4.983E+03	7.120E+02	2.150E+03	2.150E+03	2.150E+03	2.150E+03	2.150E+03
	INHAL2	1.290E+04	2.150E+03	8.470E+03	4.980E+03	3.560E+02	2.150E+03	2.150E+03	2.150E+03	2.150E+03	2.150E+03
	INGEST	9.670E+03	1.610E+03	1.610E+03	3.320E+03	9.700E+02	1.610E+03	1.610E+03	1.610E+03	1.610E+03	1.610E+03
	EXT AIR	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	EXT GND	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SR-90	INHAL1	3.210E+05	1.210E+05	8.540E+06	1.930E+04	9.310E+05	3.740E+03	3.740E+03	1.510E+05	3.740E+03	3.730E+03
	INHAL2	3.000E+06	1.100E+06	4.920E+04	1.490E+04	5.500E+04	1.540E+04	1.540E+04	2.410E+05	1.540E+04	1.540E+04
	INGEST	1.200E+06	4.300E+05	1.570E-02	5.710E+03	1.980E+05	5.990E+03	5.990E+03	9.500E+04	5.990E+03	5.990E+03
	EXT AIR	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	EXT GND	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ZR-93	INHAL1	1.470E+03	1.750E+03	5.850E+04	2.930E+03	7.160E+03	1.600E+03	1.360E+03	2.500E+03	1.040E+03	1.470E+02
	INHAL2	4.120E+03	2.460E+03	3.080E+04	2.110E+03	6.980E+03	1.320E+03	1.320E+03	2.130E+03	1.460E+03	4.950E+02
	INGEST	1.970E+02	3.340E+02	3.900E+01	1.430E+02	1.750E+04	1.690E+01	1.990E+02	2.470E+02	1.360E+03	1.340E+02
	EXT AIR	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	EXT GND	1.780E+04	1.780E+04	1.780E+04	1.780E+04	1.780E+04	1.780E+04	1.780E+04	1.780E+04	1.780E+04	1.780E+04
TC-99	INHAL1	2.420E+02	2.150E+02	5.220E+04	4.210E+02	1.660E+03	9.460E+03	3.070E+02	8.870E+02	2.120E+02	2.120E+02
	INHAL2	2.420E+02	2.150E+02	5.220E+04	4.210E+02	1.660E+03	9.460E+03	3.070E+02	8.870E+02	2.120E+02	2.120E+02
	INGEST	3.610E+02	3.220E+02	0.0	6.280E+02	3.200E+03	1.410E+04	4.580E+02	2.140E+02	3.170E+02	3.170E+02
	EXT AIR	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	EXT GND	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

* Taken from Smith, et al., 1982

Table 4-5. HEALTH EFFECTS CONVERSION FACTORS, HEALTH EFFECTS/MAN REM (FATAL CANCERS FOR ALL ORGANS EXCEPT OVARIES AND TESTES. GENETIC EFFECTS TO FIRST GENERATION FOR OVARIES AND TESTES)* (CONTINUED)

		ORGAN									
		BONE	RED MARROW	LUNGS	LIVER	GI-LLI WALL	THYROID	KIDNEYS	OTHERORGAN	OVARIES	TESTES
SN-126	INHAL1	1.560E+05	1.580E+05	1.270E+06	4.190E+03	7.600E+04	1.230E+03	6.160E+03	6.160E+03	6.160E+03	6.160E+03
	INHAL2	1.560E+05	1.580E+05	1.270E+06	4.190E+03	7.600E+04	1.230E+03	6.160E+03	6.160E+03	6.160E+03	6.160E+03
	INGEST	8.570E+04	8.570E+04	3.110E+03	1.690E+03	1.150E+05	4.990E+02	2.830E+03	2.820E+03	2.820E+03	2.820E+03
	EXT AIR	1.150E+07	1.150E+07	1.150E+07	1.150E+07	1.150E+07	1.150E+07	1.150E+07	1.150E+07	1.150E+07	1.150E+07
	EXT GND	2.090E+05	2.090E+05	2.090E+05	2.090E+05	2.090E+05	2.090E+05	2.090E+05	2.090E+05	2.090E+05	2.090E+05
I-129	INHAL1	5.790E+02	6.050E+02	7.880E+02	4.660E+02	4.280E+01	5.000E+06	4.490E+02	2.050E+03	3.780E+02	3.570E+02
	INHAL2	5.790E+02	6.050E+02	7.880E+02	4.660E+02	4.260E+01	5.000E+06	4.490E+02	2.050E+03	3.760E+02	3.570E+02
	INGEST	9.020E+02	9.420E+02	1.790E+02	7.240E+02	6.700E+01	7.800E+05	7.020E+02	3.100E+03	5.920E+02	5.530E+02
	EXT AIR	1.450E+05	1.310E+05	4.850E+04	3.600E+04	1.150E+04	1.010E+05	5.380E+04	9.540E+04	3.400E+04	1.310E+05
	EXT GND	8.730E+03	7.870E+03	2.910E+03	2.160E+03	6.900E+02	6.040E+03	3.230E+03	5.730E+03	2.040E+03	7.880E+03
CS-135	INHAL1	7.470E+03	7.470E+03	6.400E+02	7.470E+03	8.510E+01	7.460E+03	7.470E+03	4.400E+03	7.470E+03	7.470E+03
	INHAL2	7.470E+03	7.470E+03	6.400E+02	7.470E+03	8.510E+01	7.400E+03	7.470E+03	4.400E+03	7.470E+03	7.470E+03
	INGEST	1.120E+04	1.120E+04	0.0	1.120E+04	5.550E+02	1.130E+04	1.120E+04	6.610E+03	1.120E+04	1.120E+04
	EXT AIR	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	EXT GND	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CS-137	INHAL1	4.540E+04	4.910E+04	1.620E+04	5.230E+04	1.600E+04	4.470E+04	5.130E+04	3.260E+04	5.000E+04	4.440E+04
	INHAL2	4.540E+04	4.910E+04	1.620E+04	5.230E+04	1.600E+04	4.470E+04	5.130E+04	3.260E+04	5.000E+04	4.440E+04
	INGEST	6.820E+04	7.380E-04	1.950E+04	7.870E+04	2.590E+04	6.720E+04	7.720E+04	4.510E+04	7.540E+04	6.680E+04
	EXT AIR	4.660E+06	4.450E+06	3.600E+06	3.180E+06	2.750E+06	4.020E+05	3.380E+06	3.810E+06	1.390E+06	4.240E+06
	EXT GND	8.250E+04	7.920E+04	6.400E+04	5.650E+04	4.900E+04	7.150E+04	6.030E+04	6.790E+04	2.490E+04	7.550E+04
SM-151	INHAL1	5.100E+02	2.090E-02	6.780E+04	1.900E+03	3.040E+03	1.920E+01	5.540E+02	1.090E+03	1.470E+01	1.070E+01
	INHAL2	4.910E+03	1.940E+03	1.590E+04	1.890E+04	2.810E+03	1.040E+02	5.380E+03	1.190E+03	1.050E-02	1.030E-02
	INGEST	4.910E+00	3.200E+00	1.050E-01	1.730E+01	5.850E+03	1.030E-01	5.520E+00	2.340E+01	5.660E+00	5.360E-01
	EXT AIR	2.440E+01	2.130E+01	4.240E+00	2.350E+00	2.920E+00	9.060E+00	7.020E+00	3.070E+01	3.920E+00	3.660E+01
	EXT GND	4.590E+00	4.000E+00	7.960E-01	4.410E-01	5.480E-01	1.700E+00	1.320E+00	5.780E+00	7.360E-01	7.300E+00
RA-226	INHAL1	1.100E+07	9.800E+05	2.810E+07	3.400E+05	1.000E+05	3.400E+05	3.490E+05	4.600E+06	3.400E+05	3.400E+05
	INHAL2	1.100E+07	9.800E+05	2.810E+07	3.400E+05	1.000E+05	3.400E+05	3.490E+05	4.600E+06	3.400E+05	3.400E+05
	INGEST	6.320E+07	2.140E+06	2.710E+02	1.870E+06	8.160E+05	8.010E+05	5.790E+06	7.790E+06	8.060E+05	8.010E+05
	EXT AIR	1.500E+07	1.390E+07	1.270E+07	1.120E+07	1.030E+07	1.280E+07	1.060E+07	1.180E+07	9.900E+06	1.130E+07
	EXT GND	2.520E+05	2.340E+05	2.070E+05	1.850E+05	1.690E+05	2.120E+05	1.750E+05	2.210E+05	1.630E+05	1.890E+05
U-234	INHAL1	2.000E+07	8.100E+05	2.730E+08	5.900E+05	5.480E+04	5.900E+05	8.700E+05	9.800E+06	5.900E+05	5.900E+05
	INHAL2	5.900E-07	2.400E+06	2.800E+07	1.700E+06	4.790E+04	1.700E+06	2.500E+06	5.500E+06	1.700E+06	1.700E+06
	INGEST	2.000E+07	8.000E+05	8.230E+02	5.800E+05	8.060E+04	5.800E+05	8.500E+05	1.700E+06	5.800E+05	5.800E+05
	EXT AIR	2.940E+03	2.640E+03	1.030E+03	7.640E+02	8.560E+02	1.280E+03	8.130E+02	2.490E+03	6.640E+02	2.050E+03
	EXT GND	5.630E+02	5.050E+02	1.970E+02	1.460E+02	1.640E+02	2.460E+02	1.560E+02	4.780E+02	1.270E+02	4.000E+02
NP-237	INHAL1	9.0[illegible]0E+08	3.010E+08	2.900E+08	4.020E+08	1.380E+05	3.000E+06	5.200E+07	8.500E+07	1.800E+06	5.800E+06
	INHAL2	2.240E+09	7.470E+08	3.000E+07	9.910E+08	1.260E+05	7.400E+06	1.280E+08	1.900E+08	4.600E+06	1.400E+07
	INGEST	1.900E+07	6.200E+06	8.870E+02	8.200E+06	1.460E+05	6.080E+04	1.100E+06	1.600E+06	3.900E+04	1.200E+05
	EXT AIR	3.270E+06	3.030E+06	1.790E+06	1.560E+06	1.130E+06	2.150E+06	1.500E+06	2.050E+06	1.020E+06	2.410E+06
	EXT GND	7.250E+04	6.720E+04	3.970E+04	3.460E+04	2.500E+04	4.470E+04	3.340E+04	4.570E+04	2.270E+04	5.350E+04

Table 4-5. HEALTH EFFECTS CONVERSION FACTORS, HEALTH EFFECTS/MAN REM (FATAL CANCERS FOR ALL ORGANS EXCEPT OVARIES AND TESTES. GENETIC EFFECTS TO FIRST GENERATION FOR OVARIES AND TESTES)* (CONTINUED)

		ORGAN									
		BONE	RED MARROW	LUNGS	LIVER	GI-LLI WALL	THYROID	KIDNEYS	OTHERORGAN	OVARIES	TESTES
PU-238	INHAL1	7.910E+08	2.640E+08	3.090E+08	3.550E+08	6.200E+04	2.600E+06	4.600E+07	7.600E+07	1.600E+06	5.000E+06
	INHAL2	2.030E+09	6.770E+08	3.200E+07	9.070E+08	5.510E+04	6.600E+06	1.170E+08	1.730E+08	4.100E+06	1.300E+07
	INGEST	5.000E-05	1.700E+05	7.890E-02	2.200E+05	1.100E+05	1.640E+03	2.910E+04	4.320E+04	1.030E+03	3.200E+03
	EXT AIR	1.260E+03	1.090E+03	3.020E+02	1.330E+02	4.450E+02	2.460E+02	1.770E+02	1.660E+03	1.860E-02	1.320E+03
	EXT GND	2.470E+02	2.140E+02	5.920E+01	2.600E+01	8.710E+01	4.810E+01	3.460E+01	3.250E+02	3.640E+01	2.560E+02
PU-239	INHAL1	9.120E+08	3.040E+08	2.940E+08	4.040E+08	5.780E+04	3.000E+06	5.200E+07	8.600E+07	1.800E+06	5.800E+06
	INHAL2	2.280E+09	7.610E+08	3.000E+07	1.000E+09	5.130E+04	7.400E+06	1.300E+08	1.920E+08	4.600E+06	1.500E+07
	INGEST	5.700E+05	1.900E+05	6.090E-02	2.500E+05	9.850E+04	1.850E+03	3.220E+04	4.820E+04	1.150E+03	3.600E+03
	EXT AIR	6.410E+02	5.610E+02	1.710E+02	9.380E+01	1.900E+02	1.890E+02	1.230E+02	7.220E+02	1.170E+02	6.110E+02
	EXT GND	1.220E+02	1.070E+02	3.240E+01	1.780E+01	3.600E+01	3.590E+01	2.330E+01	1.370E+02	2.210E+01	1.160E+02
PU-240	INHAL1	9.130E+08	3.040E+08	2.950E+08	4.050E+08	5.820E+04	3.000E+06	5.200E+07	8.600E+07	1.800E+05	5.800E+06
	INHAL2	2.280E+09	7.600E+08	3.100E-07	1.010E+09	5.170E+04	7.400E+06	1.300E+08	1.940E+08	4.600E-06	1.500E+07
	INGEST	5.700E+05	1.900E+05	8.320E-02	2.500E+05	9.930E+04	1.640E+03	3.220E+04	4.830E+04	1.150E+03	3.600E+03
	EXT AIR	1.160E+03	1.000E+03	2.890E+02	1.400E+02	3.850E+02	2.530E+02	1.780E+02	1.460E+03	1.600E-02	1.170E+03
	EXT GND	2.250E+02	1.960E+02	5.640E+01	2.720E+01	7.790E+01	4.930E+01	3.470E+01	2.850E+02	3.520E+01	2.260E+02
1-241	INHAL1	9.430E+08	3.140E+08	3.130E+08	4.190E+08	6.520E+04	3.100E+06	5.400E+07	8.900E+07	1.900E+06	6.000E+06
	INHAL2	2.350E+09	7.830E+08	3.200E+07	1.040E+09	6.110E+04	7.700E+06	1.340E+08	1.990E+08	4.800E-06	1.500E+07
	INGEST	1.900E+07	6.400E+06	1.270E+02	8.500E+06	1.100E+05	6.320E-04	1.100E+06	1.600E+06	3.940E-04	1.200E+05
	EXT AIR	2.720E+05	2.480E+05	1.010E-05	8.300E+04	5.680E+04	1.350E+05	8.800E+04	1.440E+05	8.510E+04	1.250E+05
	EXT GND	1.420E+04	1.300E+04	5.300E+03	4.330E+03	2.560E+03	7.210E+03	4.590E+03	7.500E-03	4.440E+03	5.570E+03
PU-242	INHAL1	8.650E-08	2.890E+08	2.800E+08	3.850E+08	5.510E+04	2.800E+06	5.000E+07	8.200E+07	1.600E+05	5.500E+06
	INHAL2	2.170E-09	7.220E+08	2.900E+07	9.560E+08	4.900E+04	7.100E+06	1.230E+08	1.640E+08	4.400E-06	1.400E+07
	INGEST	5.400E-05	1.800E+05	1.600E-01	2.400E+05	9.400E+04	1.760E+03	3.080E-04	4.600E+04	1.090E+03	3.420E-03
	EXT AIR	1.040E+03	8.930E+02	2.360E+02	9.370E+01	3.650E+02	1.770E+02	1.320E+02	1.390E+03	1.51[illegible]E+02	1.100E+03
	EXT GND	2.030E+02	1.750E+02	4.630E+01	1.840E+01	7.160E+01	3.470E+01	2.590E+01	2.720E+02	2.97[illegible]E+01	2.170E+02
AM-243	INHAL1	9.430E+08	1.560E+09	3.030E+08	4.210E+08	3.220E+05	3.100E+06	5.400E+07	8.500E+07	1.900E+06	6.000E+06
	INHAL2	2.340E+09	3.870E+09	3.100E+07	1.040E+09	1.500E+05	7.700E+06	1.2[illegible]E+08	1.990E+09	4.800E+06	1.500E+07
	INGEST	1.900E-07	3.200E+07	9.640E+02	8.500E+06	1.490E+05	6.340E+04	1.100E+05	1.600E+06	4.070E+04	1.200E-05
	EXT AIR	2.170E+06	2.010E+06	1.060E+06	9.140E+05	6.490E+05	1.330E+06	8.970E+05	1.290E+06	6.760E+05	1.410E+06
	EXT GND	5.290E+04	4.880E+04	2.630E+04	2.260E+04	1.610E+04	3.280E+04	2.210E+04	3.150E+04	1.650E+04	3.460E+04

The committed dose equivalent, H, can be calculated using ICRP 30-Part 1 and Supplement as discussed earlier in Section 4.0. For exposure to a given radioisotope, one need only predict the concentration of the isotope at a receptor coordinate (from an appropriate computer-based model identified in Section 3.0), the duration of exposure, rate of intake, chemical composition and size of the isotope, and target organs. Using the ICRP 30 data, the 50 year committed dose equivalent to the lung of, for example, a one-minute inhalation exposure to a radioisotope which causes lung cancer can be computed as follows:

$$H_{50,L} = (Sv/Bq)_L \times (Bq/m^3) \times (m^3/min) \times (1\ min)$$

If the total excess risk from <u>one</u> stochastic effect for an impacted region is to be computed, one simply sums over all years of source operation and all doses:

where

R_K (total excess risk due to <u>one</u> stochastic effect) =

$$\sum_{i=1}^{Y} \sum_{j=1}^{m} U_k \quad H_{ijk} \qquad (31)$$

Y = number of years of source operation

m = number of different doses in the region in a given year

H_{ijk} = j^{th} committed dose equivalent to the k^{th} target organ in the i^{th} year of source operation

U_k = unit risk factor for the k^{th} organ (k = a constant in this case)

If the total excess risk from <u>all</u> stochastic effects for an impacted region is to be computed, one sums over all years of source operation, all doses, and all unit risk factors:

R (total excess risk due to all stochastic effects) =

$$\sum_{i=1}^{Y} \sum_{j=1}^{m} \sum_{k=1}^{n} U_k \quad H_{ijk} \qquad (32)$$

where

Y, m, H_{ikj}, and U_k are the same as above

n = number of unit risks

If the excess cases are desired, the only additional data required for the number of exposed people. For example, if the number of excess cases of lung cancer is desired:

$$EC_{LC} = U_{LC} \times H_L \times N \qquad (33)$$

where

U_{LC} and H_L are the same as above

EC_{LC} = excess cases of lung cancer

N = number of exposed at H_L

If the total number of excess cases from one stochastic effect for an impacted region is to be computed, one sums over all years of operations, all doses, and all people exposed:

EC (total excess cases due to one stochastic effect) =

$$\sum_{i=1}^{Y} \sum_{j=1}^{m} \sum_{k=1}^{n} U_k \quad H_{ijk} \quad N_{ij} \qquad (34)$$

where

Y, m, U_k and Hijk are as above

N_{ij} = number of people of exposed to the jth committed dose equivalent during the i^{th} year of source operations. Assume N_{ij} is constant during the i^{th} year of source operations.

If the total number of excess cases from all stochastic effects for an impacted region is to be computed, one simply sums over all years of source operations, all doses, all people exposed, and all unit risk factors:

EC (total excess cases due to all stochastic effect) =

$$\sum_{i=1}^{Y} \sum_{j=1}^{m} \sum_{k=1}^{n} U_k \quad H_{ijk} \quad N_{ij} \qquad (35)$$

where

Y, m, n, U_k, and H_{ijk}, N_{ij} are as above.

4.3 PRELIMINARY CONCEPTUAL SYSTEM FOR CLASSIFYING AND RANKING POTENTIAL HEALTH RISKS FROM PREDICTED ENVIRONMENTAL CONCENTRATIONS

Conceptually a relatively large number of systems for classifying and ranking potential health risks from predicted and/or measured environmental concentrations can be developed for the Umatilla Tribe to assess potential adverse environmental impacts to their tribal lands as a consequence of the high-level nuclear waste repository program located at the Hanford Site. Nevertheless, a preliminary conceptual system is outlined in figure 4-4 which can be utilized as the basis for possibly more elaborate schemes in subsequent development of the CTUIR risk assessment methodology.

The fundamental unit for organizing health effects information is the release scenario as shown in figure 4-4. In order to make comparisons within and among scenarios of release, the health effects information must be collapsed into some kind of index, such as the cross-product of the probability of occurrence of a release scenario and the excess

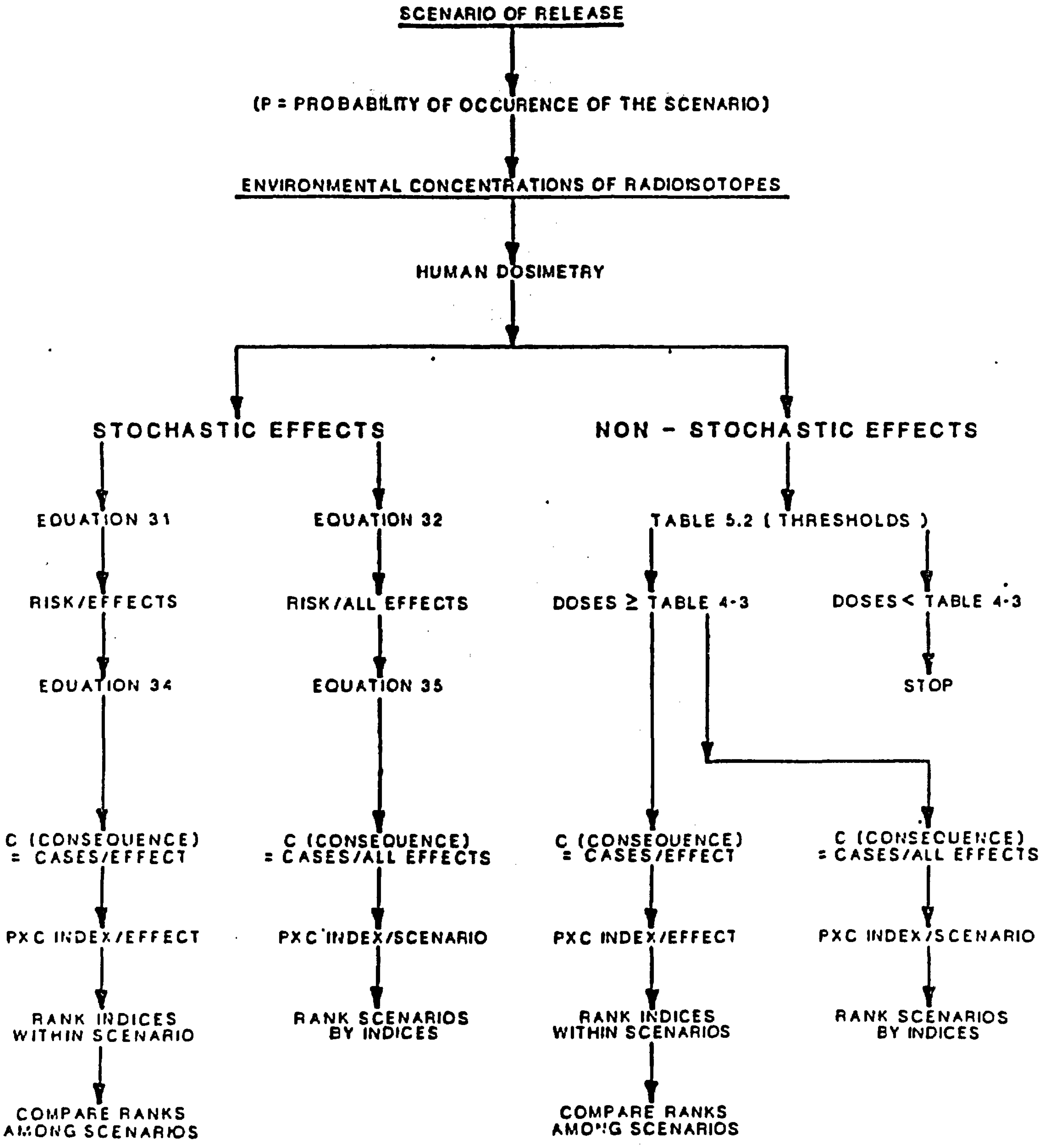

Figure 4-4. PRELIMINARY CONCEPTUAL SYSTEM FOR CLASSIFYING AND RANKING POTENTIAL HEALTH EFFECTS

cases generated by a scenario. This will be referred to as a PXC or probability consequence index for each effect and all effects. The PXC index allows an overall weighting of two factors such that a high probability/low consequence scenario would be ranked equally with a low probability/high consequence scenario. Figure 4-4 summarizes the preliminary methodology of obtaining PXC indices for both stochastic and non-stochastic human health effects using the techniques developed within prior sections of this report.

Therefore, PXC indices for <u>each</u> effect can be used to rank effects <u>within</u> a scenario and make comparisons <u>among</u> scenarios. For example, the index for lung cancer may be ranked lower in one scenario than in another. Ranking of the consequences <u>among</u> scenarios can be accomplished by using the PXC index for <u>all</u> effects generated by a certain scenario. In this case, each scenario would be characterized by one value of the PXC index.

5.0 CONCLUSIONS AND RECOMMENDATIONS

The skeletal framework for a systematic risk assessment methodology to evaluate the environmental and human health consequences of potentially significant releases of radioactive material to the CTUIR and its ceded lands as a result of a high-level nuclear waste repository program being located at the proposed Hanford Site has been structured for the two major categories of possible release scenarios; i.e. (1) transportation of high-level nuclear wastes via highway, rail, or barge through the CTUIR or its ceded lands to the mined geologic repository, and (2) site preparation, construction, operation, closure and permanent storage of the radioactive waste at that designated permanet repository location.

Major emphasis in this preliminary study effort has centered on: characterization of possible transportation system and repository system release scenarios; characterization of potential environmental concentrations on the CTUIR and its ceded lands as a result of the possible release of radioactive materials; characterization of the resultant radiation dose to humans residing on tribal lands and the concomitant health effects; and a system for classifying and ranking potential health risks from predicted environmental concentrations. More specific conclusions and recommendations in each of the foregoing primary areas of concern for the preliminary development of the CTUIR risk assessment methodology are presented in the following text.

5.1 CHARACTERIZATION OF RELEASE SCENARIOS

Preliminary compilation, review, and evaluation of potential transportation release scenarios has pointed out the need for more detailed definitions of accident release scenarios on the CTUIR for highway (truck) and rail transportation modes. Since both major highway and rail transportation routes to the proposed Hanford repository site pass through the CTUIR, the range of probabilities for radioactive release from high-level nuclear waste shipments must be defined initially on the basis of the following:

- most likely points of possible accident occurrence and subsequent radioactive waste releases to the environment for both truck and rail shipment modes,

- total shipping cask(s) radionuclide source inventories per truck or unit train on the basis of the type of high-level waste, i.e., nuclear reactor spent fuel, reprocessing wastes, commercial wastes, defense wastes, etc.,

- modal traffic volumes and estimated accident frequencies for each candidate shipment mode,

- sequential event tree analysis for each candidate transportation accident release scenario, and

- analysis of computational uncertainties entailed in the characterization and development of each candidate release scenario.

It is concluded that potential radioactive releases from truck or rail transportation of high-level nuclear waste to the proposed Hanford repository location would constitute the most immediate and most direct environmental impacts to the CTUIR since the point of release or occurrence of a transportation accident could transpire within CTUIR boundaries. Therefore, it is recommended that a more detailed computer-based environmental impact modeling analysis including a systematic health effects risk assessment be undertaken in future studies to develop a classification and comparative ranking of risks that could arise as a consequence of potentially significant high-level nuclear waste transportation release scenarios on the CTUIR and its ceded lands. Appropriate mitigation measures should be developed concomitantly to preclude the possibility of those candidate transportation release scenarios which exhibit significantly high risk on the basis of the foregoing environmental impact analysis and risk assessment.

It is concluded that potential release scenarios for the long-term, permanent high-level nuclear waste repository facilities at the Hanford Site have only been preliminarily characterized at the present time for the four general classes of possible release scenarios:

- uncertainties and potential omissions of significant consequence associated with characterization of the candidate repository site,

- potential disruptions due to natural system dynamics within the general area encompassing the candidate repository site,

- potential disruptive release scenarios resulting from repository construction and operations, and

- potential disruptive release scenarios induced by human activities other than repository construction and operation.

Presumably, most of the uncertainties and potential omissions of significant consequences associated with site characterization will be eliminated by the substantial DOE program planned for initiation at the Hanford Site in fiscal year 1987.

Only preliminary conceptual designs of the underground repository at the proposed Hanford Site have been released officially by DOE and its subcontractors. Therefore, more detailed characterization of repository construction and operation must await the release by DOE of more detailed design information in order to develop practical, sequential event trees for repository release scenarios within this class of possible release scenarios.

Therefore, it is recommended that near-term efforts in the characterization of repository release scenarios be centered on the development of candidate scenarios arising from disruptions associated with regional and area natural systems dynamics and/or potentially disruptive scenarios induced by human activities other than repository construction and operation. It is further suggested that the probabilities for combinatory scenarios of the two foregoing classes of disruptive releases also be investigated in greater detail in any subsequent task activities.

5.2 CHARACTERIZATION OF ENVIRONMENTAL CONCENTRATION

Although numerous computer-based mathematical models exist to characterize the environmental concentrations resulting from both radioactive and non-radioative contaminant dispersion and transport by both air and water media, important modifications must be made to both the atmospheric and hydrologic mathematical models in order to make them more amenable to a systematic method for CTUIR environmental impact analysis and health risk assessment. For example, one can envision transportation accident scenario releases occurring within the boundaries of the CTUIR. The RADTRAN code could be used in conjunction with a fault-or event tree model to derive mechanisms for release and the radioactive source terms for the postulated accident, but the code requires an adequate atmospheric dispersion and transport model, e.g. EPA-AIRDOS or KRONIC, to predict the appropriate downwind concentrations to the tribal environment and/or the tribal population at specific receptor locations or sector areas. However, neither of the above codes presently will accomodate atmospheric contaminant transport in complex terrain which is quite commonplace within the general area encompassing the CTUIR and its ceded lands. Thus, it may be concluded that a number of currently available computer-based atmospheric dispersion and transport modeling techniques can be utilized for subsequent CTUIR environmental assessments provided that appropriate modifications are developed for inclusion in existing models.

A similar dilemma arises when one considers hydrologic dispersion and transport from either transportation or repository releases scenarios. The characterization of environmental concentrations of potential significance to the CTUIR from possible repository release scenarios can be especially complex for cases involving groundwater contaminant dispersion and subsequent far field transport beyond the prescribed accessible environment for the proposed Hanford Site. The inherent inhomogensity of the basalt rock, the concomitant unpredictability of the possible groundwater pathways in the basalt rock media, and the contaminant sorption, desorption phenomena along the aqueous pathway all contribute to a very intricate, complex predictive mathematical modeling development. These factors, in turn, make quantification of the probabilistic uncertainties inherent in the subsurface modeling analysis very difficult at best since, in general, it may be concluded that all mathematical models that predict environmental contaminant concentrations share two basic problems: they become increasingly inaccurate with increasing distance from the source and increasingly inaccurate over long time frames.

Inasmuch as the near-term CTUIR environmental impacts assessments will concentrate on potential transportation release scenarios primarily involving atmospheric and, possibly, surface hydrologic dispersion and transport, it is concluded that the additional time required for necessary detailed experimental characterization of the basalt/water medium at the Hanford Site can be utilized to develop more precise mathematical modeling techniques that take into account more definitive site-specific physical hydrologic and hydrochemical data as it is developed.

5.3 CHARACTERIZATION OF HUMAN DOSE AND HEALTH EFFECTS

As previously described in the body of this report, there are two fundamental types of health effects: non-stochastic and stochastic. The threshold exposure and doses for non-stochastic effects summarized in this report must be further examined for accuracy, since they are central to making decisions concerning the occurrence of life-threatening early radiation effects.

The quality of the unit risk factors (cases/person/Sv) used in calculating probabilities of stochastic effects and excess cases over time are extremely important. For example, ICRP 26 (1977) and BEIR (1980) contain several organ-specific unit risk factors which are in current use. However, the bases for the numerical values of these unit risk factors are often not clearly described. Hence, it is of prime importance to define clearly the rationale for the numerical value of each organ-specific unit risk factor used in calculating probabilities of effects and excess cases due to stochastic effects. The ideal unit risk factor should be specific with regard to age, race, sex, and dose-rate. This degree of specificity is rarely attained.

This problem area is further aggravated by the fact that the current basic EPA regulatory standards governing the disposal of high-level nuclear wasrtes, namely, 40 CFR 190 and 40 CFR 191, do not specifically take into account non-stochastic health effects. Additionally, the above standards only utilize the criterion of those stochastic effects leading to early cancer fatilities.

Other potential difficulties in developing human health risk assessments that will be in compliance with NRC may arise, since, for example, NRC has not formally adopted the most current ICRP models for internal dosimetry available and utilized on an international basis - ICRP 30. The most significant results of the recent ICRP modeling revisions are:

- the hazard of some of the fission products (primarily Sr-90) is reduced,
- the hazard of several of the long-lived actinides is increased (especially Am-241, Am-243 and Np-237), and
- the hazard of Ra-226 is reduced and, as a result, the hazard of the original uranium ore is reduced.

It is recommended that a continuing effort be made within all CTUIR risk assessments to incorporate the results of the most recently validated epidemiological and experimental animal studies into the derivation of specific organ dose unit risk factors for utilization in the risk assessment methodology.

5.4 SYSTEM FOR CLASSIFYING AND RANKING POTENTIAL HEALTH RISKS FROM PREDICTED ENVIRONMENTAL CONCENTRATIONS

It is concluded that the presently prescribed system for classifying and ranking potential healths risks from predicted environmental concentrations that is described in this report provides a sound, fundamental basis for performing CTUIR risk assessments for potential release scenarios arising from a high-level nuclear waste repository program at the Hanford Site. It is believed that the potential radiation hazards imposed by the aforementioned program are best characterized by a PXC model. In addition, the system described in this report provides broad intrinsic flexibility, since the method presented relies on the classification of effects as either stochastic or non-stochastic, the calculation of the probability of occurrence of candidate scenarios, and the calculation of the number of excess cases (consequences) generated by a given scenario.

5.5 GENERAL CONCLUSIONS

The process of environmental risk assessment will most likely become a significant part of a much larger process. This process will employ all areas of human understanding in interpretating the significance of these numerical results and their subsequent impact on human action.

Some of the major uncertainties that exist in current risk assessment methodologies as they relate to the current high-level nuclear waste repository program have been discussed in this report. Perhaps the largest uncertainties, and often unquantifiable ones, arise from the use of presuppositions in the mathematical models. Hence, the total uncertainties in the final estimates of risk cannot be quantified with total confidence.

For example, estimates of the number of health effects may be precise within an order of magnitude for certain radionuclides, such as Carbon-14, but may vary by as much as 3 to 4 orders of magnitude for others such as plutonium and other transuranic elements. These uncertainties raise a number of questions, not the least of which is whether risk estimates are to be viewed as being not conservative enough or, alternatively, too low. These same uncertainties, however, apply to the current radiation protection standards which are set by the federal regulatory agencies.

For the time being, such questions are likely to remain controversial. All science contains areas of uncertainty and basic presuppositions which must be continuously challenged. The field of risk assessment must remain open to criticism, for the requirement to be correct in an absolute sense reveals a fundamental misunderstanding of the role of science. Society must develop a philosophy which deals with these inadequacies and which places the scientific basis for judgement in a proper perspective, perhaps within the context of practicality.

Risk assessment offers an excellent challenge for the development of such a philosophy. Society eventually will have to decide to what extent it views the results of this growing discipline as embodied in estimate of health effects, to make this scientific process a valid and necessary component in determining human and governmental actions.

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GLOSSARY OF RADIOLOGICAL TERMS

1. Absolute risk - Expression of excess risk due to exposure as the arithmetic difference between the risk among those exposed and that obtaining in the absence of exposure.

2. Absorption coefficient - Fractional decrease in intensity of a beam of x or gamma radiation per unit thickness (linear absorption coefficient), per unit mass (mass absorption coefficient), or per atom (atomic absorption coefficient), of absorber due to deposition of energy in the absorber; total absorption coefficient is the sum of individual energy absorption processes (Compton effect, photoelectric effect, and pair production).

3. Accelerator, particle - A device for imparting large kinetic energy to electrically charged particles, such as electrons, protons, deuterons, and helium ions; common types of particle accelerators are direct voltage accelerators, cyclotrons, betatrons, and linear accelerators.

4. Alpha particle - A charged particle emitted from atomic nucleus, with mass and charge equal to those of the helium nucleus: two protons and two neutrons.

5. Angstrom (symbol, Å) - Unit of length = 10^{-8}cm.

6. Anion - Negatively charged ion.

7. Atomic mass (symbol, μ) - The mass of a neutral atom of a nuclide, usually expressed in atomic mass units; atomic mass unit is one-twelfth the mass of one neutral atom of Carbon -12; equal to 16,604 x 10^{-24} .

8. Attenuation - Process by which a beam of radiation is reduced in intensity when passing through material-combination of absorption and scattering processes, leading to a decrease in flux density of beam when projected through matter.

9. Average life (mean life) - Average of lives of individual atoms of a radioactive half-life.

10. BEAR Committee - Advisory Committee on the Biological Effects of Atomic Radiation (precursor of BEIR Committee on the Biological Effects of Ionizing Radiations.

11. BEIR Committee - Advisory Committee on the Biological Effects of Ionizing Radiations.

12. Beta particle - Charged particles emitted from the nucleus of an atom, with mass and charge equal to those of an electron.

13. Bone-seeker - Any compound or ion that migrates in the body preferentially into bone.

14. Bremsstrahlung - Secondary photon radiation produced by deceleration of charged particles passing through matter.

15. Carrier - Nonradioactive or nonlabeled material of the same chemical composition as its corresponding radioactive or labeled counterpart; when mixed with the corresponding radioactive or labeled material, so as to form a chemically inseparable mixture, the carrier permits chemical (and some physical) manipulation of the mixture with less loss of label or radioactivity than would be possible in the use of undiluted label or radioactive material.

16. Cation - Positively charged ion.

17. Chamber, ionization - An instrument designed to measure quantity of ionizing radiation in terms of electric charge associated with ions produced within a defined volume.

18. Curie (abbr., Ci) - Unit of activity = 3.7×10^{10} nuclear transformations per second.

Common fractions are:

Megacurie	- One million curies (abbr., MCi).
Microcurie	- One-millionth of a curie (abbr., μ Ci)
Millicurie	- One-thousandth of a curie (abbr.,mCi)
Nanocurie	- One-billionth of a curie (abbr., nCi)
Picocurie	- One-millionth of a microcurie (abbr., pCi)

19. Decay, radioactive - Disintegration of the nucleus of an unstable nuclide by spontaneous emission of charged particles, photons, or both.

20. Dose - A General term denoting the quantity of radiation or energy absorbed for special purposes, must be qualified; if unqualified, refers to absorbed dose.

22. Absorbed dose - The energy imparted to matter by ionizing radiation per unit mass of irradiated material at the point of interest; unit of absorbed dose is the rad.

23. Cumulative dose - Total dose resulting from repeated exposure to radiation.

24. Dose equivalent (abbr., DF) - Quantity that expresses all kinds of radiation on a common scale for calculating the effective absorbed dose; defined as the product of the absorbed dose in rads and modifying factors; unit of DE is the rem.

25. Genetically significant dose (abbr., GSD) - The gonad dose from all sources of exposure that, if received by every member of the population, would be expected to produce the same total genetic effect on the population as the sum of the individual doses actually received; can be expressed algebraically as GSD $= \Sigma Di\ Ni\ Pi / Ni\ Pi$, where Di = average gonad dose to persons age i who receive x ray examinations, Ni = number of persons in population of age i who receive x ray examinations, Pi = expected future number of children of persons age i, and Ni = number of persons in population of age i; in 1964, the GSD was computed to be 55 mrads. per person per year for the United States; an estimated 55% of the population were receiving x rays at that time; thus, the average dose to those receiving medical radiation could be computeu to be approximately 80 mrads.

26. Maximum permissible dose equivelent (abbr., MPD) - The greatest dose equivalent that a person or specified part shall be allowed to receive in a given period.

27. Median lethal dose (abbr., MLD) Dose of radiaton required to kill, within a specified period, 50% of the individuals in a large group of animals or organisms; also called LD_{50}.

28. Permissible dose - The dose of radiation that may be received by an individual within a specified period with expectation of no substantially harmful result.

29. Threshold dose - The minimal absorbed dose that will produce a detectable degree of any given effect.

30. Doubling dose - The amount of radiation needed to double the incidence of a genetic or somatic anomaly.

31. Dose fractionation - A method of administering radiation in which relatively small doses are given daily or at longer intervals.

32. Dose protraction - A method of administering radiation in which it is delivered continously over a relatively long period at a low dose rate.

33. Dose rate - Absorbed dose delivered per unit time.

34. Electron volt (abbr., eV). - A unit of energy = 1.6 x 10^{-12} ergs = 1.6 x 10^{-19}J; 1 eV is equivalent to the energy gained by an electron in passing through a potential difference 1 volt. 1 keV = 1,000 eV; 1 MeV=1,000,000 eV.

35. Exposure - A measure of the ionization produced in air by x or gamma radiation; the sum of electric charges on all ions of one sign produced in air when all electrons liberated by photons in a volume of air are completely stopped in air, divided by the mass of air in the volume; a unit of exposure in air is the roentgen (abbr., R).

36. Acute exposure - Radiation exposure of short duration.

37. Chronic exposure - Radiation exposure of long duration, because of fractionation or protraction.

38. Fission, nuclear - A nuclear transformation characterized by the splitting of a nucleus into at least two other nuclei and the release of a relatively large amount of energy.

39. Fission products - Elements or compounds resulting from fission.

40. Fission, nuclear - Act of coalescing of two or more nuclei.

41. Gamma ray - Short-wavelength electromagnetic radiation of nuclear origin (range of energy, 10KeV to 9 MeV).

42. Gram molecular weight (synonym. mole) - Mass, in grams, numerically equal to molecular weight of a substance.

44. Gram-rad - Unit of integrel dose = 100 ergs.

45. Gray (abbr. Gy) - Proposed unit of absorbed dose of radiation = 1J/kg = 100 rads.

46. Half-life, biologic - Time required for the body to eliminate half an administered dose of any substance by regular processes of elimination; approximately the same for both stable and radioactive isotopes of a particular element.

47. Half-life, effective - Time required for a radioactive element in an animal body to be diminished by 50% as a result of the combined action of radioactive decay and biologic elimination = (biologic half-life) + (radioactive half-life) .

48. Half-life, radioactive - Time required for a radioactive substance to lose 50% of its activity by decay.

49. Incidence - The rate of occurrence of a disease within a specified period; usually expressed in number of cases per million per year.

50. Ion - Atomic particle, atom or chemical radical bearing an electric charge, either negative or positive.

51. Ion exchange - A chemical process involving reversible interchange of ions between a solution and a particular solid material, such as an ion-exchange resin consisting of a matrix of insoluble material interspersed with fixed ions of charge opposite to that in solution.

52. Ionization - The process by which a neutral atom or molecule acquires a positive or negative charge.

53. Ionization density - Number of ion pairs per unit volume.

54. Ionization path (track) - The trail of ion pairs produced by ionizing radiation in its passage through matter.

55. Primary ionization - In collision theory, the ionization produced by primary particles, as contrasted with "total ionization", which includes the "secondary ionization" produced by delta rays.

56. Secondary ionization - Ionization produced by delta rays.

57. Isotopes - Nuclides having the same number; of protons in their nuclei, and hence the same atomic number, but differing in the number of neutrons, and therefore in the mass number; chemical properties of isotopes of a particular element are almost identical; term should not be used as a synonym for nuclide".

58. Kerma (Kinetic Energy Released in Mate-'al) - A unit of quantity that represents the kinetic energy transferred to charged particles by the uncharged particles per unit mass of the irradiated medium.

59. Labeled compound - A compound consisting, in part, of labeled molecules; by observation of radioactivity or isotopic composition, this compound or its fragments may be followed through physical, chemical, or biologic processes.

60. Latent period - Period of seeming inactivity between time of exposure of tissue to an injurious agent and response.

61. Linear energy transfer(abbr., LET) - Average amount of energy lost per unit of particle spur-track length.

62. Low LET - Radiation characteristics of electrons, x rays, and gamma rays.

63. High LET - Radiation characteristics of protons and fast neutrons. Average LET is specified to even out the effect of a particle that is slowing down near the end of its path and to allow for the fact that secondary particles from photon or fast-neutron beams are not all of the same energy.

64. Linear hypothesis - The hypothesis that excess risk is proportional to dose.

65. LSS - Life Span Study of the Japanese atomic-bomb survivors; sample consists of 109,000 persons, of whom 82,000 were exposed to the bombs, mostly at low doses.

66. Maximum credible accident - The worst accident in a reactor or nuclear energy installation that, by agreement, needs to be taken into account in deriving protective measures.

67. Medical exposure - Exposure to ionizing radiation in the course of diagnostic or therapeutic procedures; and in general, includes:

a). Diagnostic radiology (e.g. x rays).
b). Exposure to radioisotopes in nuclear medicine (e.g. Iodine -131 in thyroid treatment).
c). Theropeutic radiation (e.g., Cobalt treatment for cancer).
d). Dental exposure.

68. Micrometer (symbol, μm) - Unit of length = 10^{-6}m.

69. Morbidity
a) The condition of being deceased
b) The incidence, or prevalence, of illness in any sample.

70. Neoplasm - Any new and abnormal growth, such as a tumor; "neoplastic disease" refers to any disease that forms tumors, whether malignant or benign.

71. Nonstochastic - Describes effects whose severity is a function of dose; for these, a threshold may occur; some nonstochastic somatic effects are cataract inductions nonmalignant damage to skin, hematologic deficiencies, and impairment of fertility.

72. Nuclide - A species of atom characterized by the constitution of its nucleus, which is specified by the number of protons (Z), number of neutrons (N), and energy content or, alternatively, by the atomic number (Z), the mass number (A=N+Z), and atomic mass; to be regarded as a distinct nuclide an atom must be capable of existing for a measurable time; thus, nuclear isomers are separate nuclides, whereas promptly decaying excited nuclear states and unstable intermediates in nuclear reactions are not.

73. Person-rem(synonym, man-rem) - Unit of populatin exposure obtained by summing individual dose-equivalent values for all people in the population. Thus, the number of person-rems contributed by 1 person exposed to 100 rems is equal to that contributed by 100,000 people each exposed to 1mrem.

74. Plateau - A period of above-normal, relatively uniform incidence of morbidity or mortality in response to a given biologic insult.

75. Prevalence - The number of cases of a disease in existence at a given time per unit population.

76. Quality factor(abbr., QF) - The LET-dependent factor by which absorbed doses are multiplied to obtain (for radiation-protection purposes) a quantity that expresses the effectiveness of an absorbed dose on a common scale for all kinds of ionizing radiation.

77. Rad - Unit of absorbed dose of radiation = 0.01 J/kg = 100 ergs/g.

78. Radiation - a) The emission and propagation of energy through space or through matter in the form of waves, such as electromagnetic waves, sound waves, or elastic waves.
b) The energy propagated through space or through matter as waves;"radiation" or "radiant energy", when unqualified, usually refers to electromagnetic rediation; commonly classified by frequency - Hertzian, infrared, visible, ultraviolet, x, and gamma ray. c) Corpuscular emission, such as alpha and beta radiation, or rays of mixed or unknown type, such as cosmic radiation.

79. Background radiation - Radiation arising from radioactive material other than that under consideration; background radiation due to cosmic rays and natural radioactivity is always present; there may also be background radiation due to the presence of radioactive substances in building material, etc.

80. External radiation - Radiation from a source outside the body.

81. Internal radiation - Radiation from a source within the body (as a result of deposition of radionuclides in tissue).

82. Ionizing radiation - Any electromagnetic or particulate radiation capable of producing ions, directly or indirectly, in its passing through matter.

83. Secondary radiation - Radiation resulting from absorption or other radiation in matter; may be either electromagnetic or particulate.

84. Radiation sickness - A self-limited syndrome characterized by nausea, vomiting, diarrhea, and psychic depression; follows exposure to appreciable doses of ionizing radiation, particularly to the abdominal region; its mechanism is unknown, and there is no satisfactory remedy; usually appears a few hours after irradiation and may subside within a day; usually appears a few hours a day; may be sufficiently severe to necessitate interrupting a treatment series or to incapacitate the patient.

85. Radioactivity - The property of some nuclides of spontaneously emitting particles or gamma radiation or of emitting x radiation after orbital electron capture or of undergoing spontaneous fission.

86. Artificial Radioactivity - Man-made radioactivity produced by particle bombardment or electromagnetic irradiation.

87. Natural Radioactivity - The property of radioactivity exhibited by more than 50 naturally occurring radionuclides.

88. Radioisotopes - A radioactive atomic species of an element with which it shares almost identical chemical properties.

89. Radionuclide - A radioactive species of an atom characterized by the constitution of its nucleus; in nuclear medicine, an atomic species emitting ionizing radiation and capable of existing for a measurable time, so that it may be used to image organs and tissues.

90. Radiosensitivity - Relative susceptibility of cells, tissues, organs, and organisms to the injurious action of radiation; "radiosensitivity" and its antonym, "radioresistance", are used in a comparative sense; rather than in an absolute one.

91. Rav, Alpha: Beam of helium nuclei (two protons and two neutrons)

Beta:	Beam of electrons or positrons.
Delta:	Beam of electrons ejected by ionizing particles in passage through matter.
Gamma:	Beam of high-energy photons from radioactively decaying elements.
x :	Beam of mixed lower-energy photons.
Neuton:	Beam of neutrons.
Protons:	Beam of protons.

92. Reactor, breeder - A reactor that produces more fissile atoms from fertile atoms, but has a conversion ratio greater than unity.

93. Reactor, converter - A reactor that produces fissile atoms from fertile atoms, but has a conversion ratio less than unity.

94. Reactor, nuclear - An apparatus in which nuclear fission may be sustained in a self-supporting chain reaction.

95. Recoverv rate - The rate at which recovery take place after radiation injury; recovery may proceed at different rates for different tissues; among tissues recovering at different rates, those having lower rates will utimately suffer greater damage from a series of successive irradiations and this differential effect is considered in fractionated radiation therapy if neoplastic tissues have a lower recovery rate than surrounding normal structures.

96. Relative biologic effectiveness (abbr., RBE) - A factor used to compare the biologic effectiveness of absorbed radiation doses (i.e., rads) due to different types of ionizing radiation; more specifically, the experimentally determined ratio of an absorbed dose of a radiation in question to the absorbed dose of a reference radiation required to produce an identical biologic effect in a particular experimental organism or tissue; the ratio of rems to rads; if 1 rad of fast neutrons equated in lethality 3.2 rads of kilovolt-peak (KVp) x rays, the RBE of the fast neutrons would be 3.2.

97. Relative risk - Expression of risk due to exposure as the ratio of the risk among the exposed to that obtaining in the absence of exposure.

98. Rem - A unit of dose equivalent = absorbed dose (in rads) times quality factor QF times distribution factor times any other necessary modifying factors; represents quantity of radiation that is equivalent - in biologic damage of a specific part - to 1 rad of 250 KVp x rays.

99. Roentgen (abbr., R) - A unit of exposure = 2.58×10^{-4} coulomb/Kg of air.

100. Sievert (abbr. Sv) - Proposed unit of radiation dose equivalent

101. Softness - A relative specification of the quality or penetrating power of x rays; in general, the longer the wavelength, the softer the radiation.

102. Specific activity - Total activity of a given nuclide per gram of a compound, element, or radioactive nuclide.

103. Stochastic - Describes effects whose probability of occurrence in an exposed population (rather than severtiy in an affected individual) is a direct function of dose; these effects are commonly regarded as having no threshold; hereditary effects are regarded as being stochastic; some somatic effects, especially carginogenesis, are regarded as being stochastic.

104. Target theory (synonym, hit theory) - A theory explaining some biologic effects of radiation on the basis that ionization, occurring in a discrete volume (the target) within the cell, directly causes a lesion that later results in a physiologic response to the damage at that location; one, two, or more "hits" (ionizing events within the target) may be necessary to elicit the response.

104 Thermography - A noninvasive diagnostic radiologic imaging technique that uses infrared radiation to picture the heat emmitted by the surface, which characterizes the temperature distribution in the various underlying organs and tissues of the body.

106. Threshold hypothesis - The assumption that no radiation injury occurs below a specfied dose.

107. Ultrasonography - A noninvasive diagnostic radiologic imaging technique that uses acoustic radation and the acoustic properties of biologic structure to picture the structure and function of various organs and tissues of the body.

108. Working level (abbr., WL) - Any combination of radon daughters in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

109. Working-level month (abbr., WLM) - Exposure resulting from inhalation of air with a concentration of 1 WL of radon daughters for 170 working hours.

110. X ray - Penetrating electromagnetic radiation whose wavelength is shorter than that of visible light; usually produced by bombarding a metallic target with fast electrons in a high vacuum: in nuclear reactions, it is customary to refer to photons originating in the nucleus as gamma rays, and those originating in the extranuclear part of the atom as x rays; sometimes called roentgen rays, after their discoverer, W.C. Roentgen.

APPENDIX A

CHARACTERIZATION OF RADIOISOTOPES IN SPENT NUCLEAR FUEL AND DEFENSE WASTES

APPENDIX A

Characterization of Radioisotopes in Spent Nuclear Fuel and Defense Wastes

This appendix contains information on radionuclides expected to comprise waste from defense, commercial or nuclear power plant sources. They are listed by atomic number. Their half-lives, types of emissions, and energies of emissions are shown from all decays occurring at a frequency greater than .01%. Daughters are listed for most radioisotopes except those in the uranium decay series. An asterisk preceeds those radioisotopes with half-lives greater than 53 days. The list of radioisotopes and radiological data were taken from: UNSCEAR, 1977, 1982; Radiological Health Handbook, 1970; Wilson, 1966; CERT, 1984; Draft Environmental Assessment, 1984; Eicholz, 1976; IAEA, 1980; Nucleon, 1984; and Kocher, 1981.

KEY:	y	= years	M	= maximum energy
	m	= minutes	A	= average energy
	d	= days	IT	= isomeric transition
	s	= seconds	SF	= spontaneous fission
	$\beta-$	= beta	EC	= electron capture
	$\beta+$	= positron		* = T 1/2 $\geq$53 days
	α	= alpha		
	γ	= gamma		

Radioisctope	Daughter(s)	T1/2	Emission	Energies(Mev)
$\bullet H^3$		12.36y	β-	.0057(A)
$\bullet C^{14}$		5730y	β-	.045(A) .156(M)
Cr^{51}	Mn^{51}	27.8d	γ	.320-9%
Mn^{51}		45.2m	β+	2.17(M)
			γ	.511 (.04%)
Fe^{59}		45.6d	β-	1.57(M)(3%);.475(M)
			γ	.143 (.8%); .192(2.8%); 1.095(56.7%); 1.292 (43%)
$\bullet Co^{60}$		5.26y	β-	1.48(M)(12%);.314(M)
			γ	1.173(100%);1.332(100%)
$\bullet Se^{79}$		$\leq 6.5x10^4y$	β-	.16(M)
$\bullet Kr^{85}$		10.76y	β-	.67(M)
			γ	.514(43%)
Rb^{87}	Sr^{87}		β-	.274(M)
Sr^{89}		52.7d	β-	1.463(M)
Sr^{90}	Y^{90}	28.6y	β-	.546(M)
Y^{90}		64h	β-	2.27(M); .93(A)
$\bullet Y^{91}$		58.8d	β-	1.545(M)
			γ	1.21 (.3%)
$\bullet Zr^{93}$	Nb^{93m}	$1.5x10^6y$	β-	.06(M)
$\bullet Nb^{93m}$		13.6y	γ	Nb xrays
			e-	.011 .028
$\bullet Zr^{95}$	Nb^{95m}&Nb^{95}	65.5d	β-	.89(M)(2%); .396(M)
			γ	.724 (49%); .756 (49%)

Radioisotope	Daughter(s)	T1/2	Emission	Energies(Mev)
Nb^{95m}	Nb^{95}	90h	γ	Nb xrays .235
			e-	.216
Nb^{95}		35d	β-	.16(M)
			γ	.765 (100%)
*Tc^{99}	Ru^{99}	$2.12x10^5y$	β-	.292(M)
Ru^{103}	Rh^{103m}	39.5d	β-	.7(M)(3%); .21(M)
			γ	.497 (88%); .61(6%)
Rh^{103m}(IT)		56.12m	γ	Rh xrays .04 (4%)
			e-	.017; .037
*Ru^{106}	Rh^{106}	368d	β-	.039(M)
Rh^{106}		30.5s	β-	3.54(M)
			γ	.512(21%); .622 (11%); 1.05 (1.5%); 1.13 (.5%) 1.55 (.2%)
Pd^{107}		$7x10^6y$	β-	.04(M)
Ag^{110}		24.4s	β-	2.87 (M) neutron capture
				.658 (4.5%)
Ag^{110m}		255d	β-	1.5(M)(.6%); .53(M)(31%); .087(M)
			e-	.090; .113
			γ	.658(96%); .68(16%); .706(19%); .764(22% .818(8%); .885(71%); .937(32%); 1.384(21 1.505(11%)
Cd^{115}	In^{115m}	43d	β-	1.62(M)
				.485(.31%); .935(1.9%); 1.29(.9%

Radioisotope	Daughter(s)	T1/2	Emission	Energies(Mev)
*Sn^{121}		76y	β-	.42(M)
			e-	.007, .033
			γ	Sb xrays .037
*Sn^{123}		125d	β-	1.42(M)
Sn^{126}	Sb^{126}	10^5y	γ	.06; .067; .092
Sb^{126}		12.5d	β-	1.9(M)
			γ	.41; .069(3 γ rays)
*Sb^{124}		60.4d	β-	2.31(M)
			γ	.603(97%); .644(7%); .72(14%); .967(2%); 1.045(2.4%); 1.31(3%); 1.37(.5%); 1.45(.29 1.7(50%); 2.088(7%)
*Sb^{125}	Te^{125m}	2.7y	β-	.61(M)
			e-	.004; .03; .144; .395
			γ	Te xrays, .176(6%); .427(31%); .463(10%) .599(24%); .66(3%); .634(11%)
*Te^{125m}		58d	e-	.004; .03, .073; .105
			γ	Te xrays; .035(7%); .110(.3%)
Te^{127}		9.4h	β-	.7(M)
			γ	(xrays) .053(.01%), .21(.03%); .36(0.13%); .417 (1.0%)
*Te^{127m}		109d	γ	(Te xrays) .059(.19%), .089(.08%
			e-	.057, .084
			β-	.73(M)
In^{115m}		4.5h	β-	0.83(M)
			e-	.305; .331
			γ	(In xrays); .335(50%)

Radioisotope	Daughter(s)	T1/2	Emission	Energies(Mev)
Te129		68.7m	β-	1.45(M)
			e-	.022; .026
			γ	(I xrays) .027(19%); .275(1.75%); .455(15% .81(.5%)
Te129m(IT)	Te129	34.1d	β-	1.6(M)
			e-	.074, .102
			γ	(Te xrays) .69(6%)
*I129		1/7x10^{7}y	β-	.15(M)
			e-	.005; .034
			γ	(Xe xrays) .04(9%)
*Cs134		2.046y	β-	.662(M)
			γ	.57(23%); .605(98%); .796(99%), 1.038(1% 1.168(1.9%) 1.365(3.4%)
*Cs135		3/0x10^{6}y	β-	.21(M)
Cs136		13.7d	β-	.657(7%); .341(M)
			e-	.116; .126; .158; .302
			γ	(Ba xrays) .067(11%); .086(6%); .16(.36%) .273(18%) .340(53%); .818(100%); 1.05(82%); 1.25 (20%)
*Cs137		3.0y	β-	1.176(M)(7%); .514(M)
			e-	.624; .656
			γ	(Ba xrays) .662(85%)
Ba137m(IT)	Cs137	2.552m	γ	(Ba xrays) .662(89%)
			e	.624; .656
Ba140	La140	12.8d	β-	1.02(M)
			e	.024; .029
			γ	(La xrays) .03(11%); .163(6%); .305(6%); .433(5%) .537(34%)

Radioisotope	Daughter(s)	T1/2	Emission	Energies(Mev)
La^{140}		40.22h	β-	2.175(M)(6%); 1.69(M)(.5%); 1.36(M)
			γ	.329(20%); .487(40%), .815(19%); .923(10? 1.596(96%); 2.53(3%)
Ce^{141}		32.5d	β-	.581(M)
			e-	.104; .139
			γ	(Pr xrays); .145(48%)
Ce^{144}	Pr^{144}	284d	β -	.31(M)
			e-	.038; .092
			γ	(Pr xrays) .08(2%); .34(11%)
Pr^{144}		17.27m	β-	2.99(M)
			γ	.695(1.5%); 1.487(.29%); 2.186(.7%)
Pr^{143}		13.59d	β-	.933(M); .31(A)
*Nd^{144}		$2.4x10^{15}$y	α	1.83
Nd^{147}	Pm^{147}	11.06d	β-	.81(M)
			e-	.046; .034
			γ	.091(28%); .319(3%); .43(4%); .533(13%)
*Pm^{147}	Sm^{147}	2.62y	β-	.224(M); .07(M)
Pm^{148}		5.4d	β-	2.45(M)
			γ	.551(27%); .914(15%), 1.465(23%
Pm^{148m}	(IT-7%)	41.3d	β-	.69(M)
			e-	.031; .053; .091; .242; 503; .583
			γ	(Pm & Sm xrays); .289(13%); .413(17%); .551(95%); .630(87%); .727(36%); .916(21? 1.015(20%)
*Sm^{147}		$1.05x10^{11}$y		.223
*Sm^{151}		87y	β-	.076(M)
			e-	.014; .020
			γ	(Eu xrays); .022(.03%)

Radioisotope	Daughter(s)	T1/2	Emission	Energies(Mev)
*Eu^{152}		13.6y	β-	1.48(M)
EC 72%			e-	.075; .115; 1.20
B- 28%			β+	.71(M)
B+ .021%			γ	(Gd & Sm xrays); .122(37%) .245(8%); .344(.12%); .965(15%); 1.084(.2%); 1.113(14%); 1.408(21%)
*Eu^{154}		8.8y	β-	1.85(M)(10%); .84(M)
			e-	.073; .115; .122
			γ	(Gd xrays); .123(38%); .248(7%); .593(6%) .724(20%) .754(5%); .876(12%); 1.0(31%); 1.278(37%)
Eu^{155}		4.96y	β-	.25(M)
			e-	.011; .017; .036; .054; .078; .082
				(Gd xrays); .087(32%); .105(20%)
Eu^{156}		15.4d	β-	2.45(M)
			e-	.039; .081; .087
			γ	(Gd xrays) .089(8%), .646(7%); .733(6%); .812 (11%);1.15(14%); 1.24(16%); 1.97 (7%); 2.098(3%);2.1(4%)
*Tb^{160}		72.1d	β-	1.74(M)(4%); .86(M)
			e-	.033; .079; .085
			γ	(Dy xrays) .087(12%); .197(6%); .299(30%); .872(1%); 1.178(15%); 1.272(7%)
Tl^{206}		4.19m	β-	1.52(M)
Tl^{207}		4.79m	β-	1.44(M) .897(.16%)
Tl^{208}		3.1m	β-	1.8(M)
			e-	.187; .423; .495
			γ	.511(23%); .583(86%); .860(12%); 2.614(100%)

Radioisotope	Daughter(s)	T1/2	Emission	Energies(Mev)
Tl^{209}	Pb^{209}	2.2m	β-	1.99(M)
			e-	.03; .10
			γ	(Pb xrays) .12(50%); .45(100%); 1.56(100%
Pb^{209}		3.3h	β-	.635(M)
U^{232} (SF: $x10^3$y)	Th^{228}	72y	γ	5.32(68%); 5.27(32%) (Th xrays) .058(21%); .129(.082%)
			e-	.040; .054
*U^{233}	Th^{229}	$1.62x10^5$y	α	4.82(83%); 4.78(15%)
			γ	(Th x rays)
			e-	.023; .038
*U^{234} (SF: $2x10^{16}$y)	Th^{230}	$2.47x10^5$y	α	4.77(72%); 4.72(28%)
			γ	(Th L xrays) .053(.2%)
*U^{235} (SF:$1.9x10^{17}$y)	Th^{231}	$7.1x10^8$y	α	4.58(8%) 4.4(57%); 4.37(18%)
			γ	(Th x rays) .143(11%); .185(54%); .204(5%
*U^{236} (SF: $2x10^{16}$y)		$2.39x10^7$y	α	4.49(76%), 4.44(24%)
			γ	(Th L xrays)
			e-	.032; .045
*U^{238} (SF:$6.5x10^{15}$y)	Th^{234}	$4.5x10^9$y	α	4.2(75%); 4.15(25%) (Jh L xrays)
			e-	.03, .043
Np^{236}	Pu^{236}	22h	β-	.052(M)
			e-	.025; .040
			γ	U xrays; .642; .688
*Pu^{236}	U^{232}	2.85y	α	5.77(69%), 5.72(31%)
			γ	(U L xrays) .043(31%); .109(.01%
			e-	.028; .043
Pu^{237} Ec 99% α .0033%	U^{231}	45.6d	γ	(Np xrays) .06(5%)
			e-	.026, .032, .042, .056
				5.37, 6.66 relative intensities
*Pu^{238} (SF: $4.9x10^{10}$y)		86.4y	α	5.5(72%; 5.46(28%)
			γ	(U L xrays)
			e-	.024; .039

Radioisotope	Daughter(s)	T1/2	Emission	Energies(Mev)
*Pu^{239} SF:$5.5x10^{15}$y)	U^{235m}	24,390y	α γ	5.16(88%); 5.11(11.5%) U xrays .052(.02%)
			e-	.008; .019; .033; .047
U^{235m}	IT	26.1m	e-	≤ .0001
*Pu^{240} (SF: $1.34x10^4$y)		6580y	α γ	5.17(76%); 5.12(24%) U L xrays
			e-	.026; .040
*Pu^{241}	Am^{241};U^{237}	13.2y	β-	.021(M) U xrays
U^{237}	Np^{237}	6.75d	β-	.248(M)
			e-	.008; .011; .038; .089; .136
			γ	.026(2%); .06(36%); .165(2%); .208 (23%); .267(.76%) .332(1.4%); .37(.17%)
*Np^{237} (SF: 10^{18}y)	Pa^{233}	$2.14x10^6$y	α γ	4.78(75%), 4.65(12%) Pa L xrays .03(14%), .086(14%) .145(4%)
			e-	.009, .024; .036, .05; .067; .082
Pa^{233}		27d	β-	.568(M)(5%); .257; (M)
			e-	.013; .023; .036; .054; .065; .185 .197, .291
			γ	U rays .31(44%)
*Am^{241} (SF:$2x10^{14}$y)		458y	α γ e-	5.49 (85%); .5.44 (13%) (Np L xrays); .06 (36%); .101 (04 .022; .038; .054
Pu^{242} (SF: $7.1x10^{10}$y)		$3.79x10^5$y	α γ	4.9(76%); 4.86(24%) (U L xrays)
Am^{242}	Pu^{242},Cm^{242}	16.01h	β-	.67(M)(87%)
(E.C. 16%)			e-	.021; .037
			γ	(Pu & Cm xrays)

Radioisotope	Daughter(s)	T1/2	Emission	Energies(Mev)
*Cm^{242}	Pu^{238}	162.5d	α	6.12(74%); 6.07 (26%)
			γ	(Pu L xrays) .044(.041%)
*Am^{242m} (IT 99%)	Am^{242},Np^{238}	152y	β-	5.21 (.41%)
			e-	.028; .044
			γ	(Am & Np xrays) .049(.2%); .087(.036%), .11(.025%) .163(.025%)
Np^{238}	Pu^{238}	2.1d	β-	1.25(M)
			e-	.022; .039
			γ	1.01(42%)
*Am^{243}	Np^{239}	$7.95x10^3y$	α	5.28(87%); 5.23(11.5%)
			γ	(Np L xrays) .044(4.7%); .075(50
			e-	.011, .023, .052, .69
Np^{239}	Pu^{239}	2.346d	β-	.713(M)(4%); .437(M)
			e-	.02; .04; .048; .083; .086; .156
			γ	(Pu xrays), .106(23%); .209(4%); .228(12% .278(14%)
*Cm^{243}		32y	α	6.06(6%; 5.99(6%); 5.74(11.5%) 5.79(73%)
			γ	(Pu xrays); .209(4%); .228(12%); .275(14%
			e-	.02; .04; .045; .055; .106; .156
*Cm^{244} SF:$1.31x10^7y$	Pu^{240}	17.6y	α	5.81(77%); 5.77(43%)
				(Cm L xrays); .043(.02%)
			e-	.022; .038
*Cm^{245}	Pu^{241}	$9.3x10^3y$	α	5.36(80%); 5.31(7%
			γ	(Pu xrays); .13(5%); .173(14%)
*Cm^{246} (SF:$1.7x10^7y$)	Pu^{242}	$5.5x10^3y$	α	5.39 (81%), 5.34(19%)
			γ	Pu L xrays
*Cm^{247}		$1.6x10^7y$	α	4.8(76%), 4.9(3.6%); 5.2(20.7%)
*Cm^{248} (SF:$4.6x10^6y$)-11%		$4.7x10^5y$	α	5.08(82%); 5.04(15%)
			γ	(Pu L xrays)

APPENDIX B

ICRP - 30 MODEL - SAMPLE INTERNAL DOSE CALCULATIONS FOR SR-90

APPENDIX B

ICRP - 30 Model - Sample Internal Dose Calculations for Sr-90

The total intake of Sr-90 is assumed to be 50.87 curies. This value is based upon the concentration of Sr-90 specified for one gallon of defense waste sludge as referenced in the CTUIR scoping study report (CERT, 1984). The dose which results from ingestion or inhalation of a radioisotope is dependent upon many factors including metabolism and distribution in the body. The chemical nature and particle size of the compound containing the radioisotope can alter these factors. The ICRP=30 model, which is utliized in this example, incorporates current knowledge of these factors into the dose estimates (ICRP-30, 1979). This sample calculation is applicable to all compounds in inhalation Class D (a half-life of less than 10 days in the pulmonary region), and with an activity median aerodynamic diameter (AMAD) of 1 micrometer (um), excluding $SrTiO_3$ which has a different metabolic constant.

Dose is obtained from the ICRP-30 model through the use of Committed Dose Equivalent (H_{50}) which specifies the sieverts (Sv) accumulated over 50 years from exposure to a specific quantity of activity in bequerels (Bq).

Thus in Example A below, the H50 calculations are utilized. These include the dose due to exposure tó the daughter, Yttrium-90 (Y-90)emissions. H50 values are only available for the bone marrow and surface. The dose to the population due to inhalation is calculated in Example B. Because H50 values are not available for the lung, the calculations are based upon the number of transformations in source organs, and the Specific Effective Enegy (SEE) per transformation in a source organ which reaches the lung. The SEE in Mev/gram is then converted to rads or rems. The contribution from each source is calculated separately and summed to yield the total lung dose. Sr-90 and Y-90 both emit beta particles; consequently, a quality factor of 1 was used to convert rads to rems.

The calculated dose is most applicable to a "reference man" (see ICRP-23), and assumes adult male (70kg) anatomy and physiology in determining dose (ICRP, 1975). Modifications could be utilized for young and old individuals.

Sample B

Using the number of transformations in source organs over 50 years and SEE's, the general formula is:

$$\text{Dose (rem)} = \sum_{\substack{\text{all} \\ \text{radionuclides}}} \; \sum_{\substack{\text{all} \\ \text{source}}} \left[\begin{array}{l} \text{Transformations (T/Bq)} \\ \quad \times \text{ SEE (Mev/g/T)} \times \\ \text{Exposure (Bq)} \div \text{Mev/g/rad} \\ \quad \times \text{ 1 rem/rad} \end{array} \right]$$

The results of these calculations for the total lung dose from inhalation for Sr-90 is presented in table B-1.

Table B-1. Sr-90 TOTAL LUNG DOSE FROM INHALATION, EXAMPLE B

INHALATION, LUNG TARGET:

SOURCE	RADIOISOTOPE	T/Bq		SEE		Exposure		MeV/g/rad		DOSE (rems & rads)
Lungs	Sr-90	1.9×10^{4}	x	2×10^{-4}	x	1.882×10^{12}	÷	6.242×10^{7}	=	1.15×10^{5}
Lungs	Y -90	3.4×10^{3}		9.3×10^{-4}		same		same		9.5×10^{5}
ULI content*	Y -90**	8.6×10^{2}		4.3×10^{-18}		same		same		1.1×10^{-10}
LLI content*	Y -90**	3.1×10^{3}		1.5×10^{-19}		same		same		1.4×10^{-11}
cortical bone	Y -90**	1.9×10^{7}		7.2×10^{-13}		same		same		4.12×10^{-1}
trabecular bone	Y -90**	7.9×10^{6}		7.2×10^{-13}		same		same		1.7×10^{-1}

TOTAL LUNG DOSE FROM INHALATION: 1.1×10^{6} rems

* ULI = upper large intestine

LLI = lower large intestine

** no SEE values were provided for Sr-90

APPENDIX C

MAXIMUM PERMISSIBLE CONCENTRATIONS (MPC) OF SELECTED RADIOISOTOPES

APPENDIX C

Maximum Permissible Concentrations (MPC) of Selected Radioisotopes

The MPC's were obtained from two sources. The first two lines list the values specified in 10 CFR 20 for soluble and insoluble forms of each radioisotope. The subsequent organ specific data were obtained from an earlier guideline for occupational exposure (Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure, 1963).

RADIOISOTOPE		ORGAN**	40 HOUR WEEK WATER (UC/CC)	40 HOUR WEEK AIR (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) WATER (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) AIR (UC/CC)
Carbon 14	S*		2. E-2***	4. E-6	8. E-4	1. E-7
(Dioxide)	Submersion			5. E-5		1. E-6
	S	FAT	.02	4. E-6	8. E-3	1. E-6
	S	TOTAL BODY	.03	5. E-6	.01	2. E-6
	S	BONE	.04	6. E-6	.01	2. E-6
	Immersion	TOTAL BODY	---	5. E-5	---	1. E-5
^{90}Sr	S		1. E-5	1. E-9	3. E-7	3. E-11
	I*		1. E-3	5. E-9	4. E-5	2. E-10
	S	BONE	4. E-6	3. E-10	1. E-6	1. E-10
	S	TOTAL BODY	1. E-5	9. E-10	4. E-6	3. E-10
	S	G I (LLI)	1. E-3	3. E-7	5. E-4	1. E-7
	I	LUNG	---	5. E-9	---	2. E-9
	I	G I (LLI)	1. E-3	2. E-7	4. E-4	6. E-8

* S = Soluble
  I = Insoluble

** Abbreviations: LLI = Lower Large Intestine
SI = Small Intestine

*** 2. E-2 = 0.02

RADIOISOTOPE		ORGAN	40 HOUR WEEK WATER (UC/CC)	40 HOUR WEEK AIR (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) WATER (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) AIR (UC/CC)
^{90}Y	S		6. E-4	1. E-7	2. E-5	4. E-9
	I		6. E-4	1. E-7	2. E-5	3. E-9
	S	G I (LLI)	6. E-4	1. E-7	2. E-4	4. E-8
	S	BONE	10	5. E-7	4	2. E-7
	S	TOTAL BODY	80	3. E-6	30	1. E-6
	I	G I (LLI)	6. E-4	1. E-7	2. E-4	3. E-8
	I	LUNG	---	3. E-7	---	1. E-7
^{99}Tc	S		1. E-2	2. E-6	3. E-4	7. E-8
	I		5. E-3	6. E-8	2. E-4	2. E-9
	S	G I (LLI)	.01	2. E-6	3. E-3	7. E-7
	S	KIDNEY	.02	3. E-6	8. E-3	9. E-7
	S	LIVER	.3	4. E-5	.1	1. E-5
	S	TOTAL BODY	.4	4. E-5	.1	1. E-5
	S	SKIN	.7	7. E-5	.2	3. E-5
	S	BONE	.9	9. E-5	.3	3. E-5
	S	LUNG	4.	4. E-4	1.	1. E-4
	I	LUNG	---	6. E-8	---	2. E-8
	I	G I (LLI)	5. E-3	8. E-7	2. E-3	3. E-7
^{106}Ru	S		4. E-4	8. E-8	1. E-5	3. E-9
	I		3. E-4	6. E-9	1. E-5	2. E-10
	S	G I (LLI)	4. E-4	8. E-8	1. E-4	3. E-8
	S	KIDNEY	.01	1. E-7	4. E-3	5. E-8
	S	BONE	.04	5. E-7	.01	2. E-7
	S	TOTAL BODY	.06	7. E-7	.02	3. E-7
	I	LUNG	---	6. E-9	---	2. E-9
	I	G I (LLI)	3. E-4	6. E-8	1. E-4	2. E-8
^{129}I	S		1. E-5	2. E-9	6. E-8	2. E-11
	I		6. E-3	7. E-8	2. E-4	2. E-9
	S	THYROID	1. E-5	2. E-9	4. E-6	6. E-10
	S	TOTAL BODY	2. E-3	2. E-7	5. E-4	7. E-8
	S	G I (LLI)	.1	3. E-5	.04	9. E-6
	I	LUNG	---	7. E-8	---	2. E-8
	I	G I (LLI)	6. E-3	1. E-6	2. E-3	4. E-7

RADIOISOTOPE		ORGAN	40 HOUR WEEK WATER (UC/CC)	40 HOUR WEEK AIR (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) WATER (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) AIR (UC/CC)
^{134}Cs	S		3. E-4	4. E-8	9. E-6	1. E-9
	I		1. E-3	1. E-8	4. E-5	4. E-10
	S	TOTAL BODY	3. E-4	4. E-8	9. E-5	1. E-8
	S	LIVER	4. E-4	6. E-8	1. E-4	2. E-8
	S	MUSCLE	4. E-4	6. E-8	2. E-4	2. E-8
	S	SPLEEN	6. E-4	9. E-8	2. E-4	3. E-8
	S	KIDNEY	1. E-3	2. E-7	4. E-4	6. E-8
	S	BONE	2. E-3	3. E-7	7. E-4	1. E-7
	S	LUNG	4. E-3	5. E-7	1. E-3	2. E-7
	S	G I (SI)	.01	3. E-6	5. E-3	1. E-6
	I	LUNG	---	1. E-8	---	4. E-9
	I	G I (LLI)	1. E-3	2. E-7	4. E-4	7. E-8
^{137}Cs	S		4. E-4	6. E-8	2. E-5	2. E-9
	I		1. E-3	1. E-8	4. E-5	5. E-10
	S	TOTAL BODY	4. E-4	6. E-8	2. E-4	2. E-8
	S	LIVER	5. E-4	8. E-8	2. E-4	3. E-8
	S	SPLEEN	6. E-4	9. E-8	2. E-4	3. E-8
	S	MUSCLE	7. E-4	1. E-7	2. E-4	4. E-8
	S	BONE	1. E-3	2. E-7	5. E-4	7. E-8
	S	KIDNEY	1. E-3	2. E-7	5. E-4	8. E-8
	S	LUNG	5. E-3	6. E-7	2. E-3	2. E-7
	S	G I (SI)	.02	5. E-6	8. E-3	2. E-6
	I	LUNG	---	1. E-8	---	5. E-9
	I	G I (LLI)	1. E-3	2. E-7	4. E-4	8. E-8
^{151}Sm	S		.01	6. E-8	4. E-4	2. E-9
	I		.01	1. E-7	4. E-4	5. E-9
	S	G I (LLI)	.01	2. E-6	4. E-3	8. E-7
	S	BONE	2	6. E-8	.5	2. E-8
	S	KIDNEY	4	2. E-7	2	6. E-8
	S	LIVER	5	2. E-7	2	7. E-8
	S	TOTAL BODY	7	3. E-7	2	1. E-7
	I	LUNG	---	1. E-7	---	5. E-8
	I	G I (LLI)	.01	2. E-6	4. E-3	7. E-7

RADIOISOTOPE		ORGAN	40 HOUR WEEK WATER (UC/CC)	40 HOUR WEEK AIR (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) WATER (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) AIR (UC/CC)
^{241}Am	S		1. E-4	6. E-12	4. E-6	2. E-13
	I		8. E-4	1. E-10	3. E-5	4. E-12
	S	KIDNEY	1. E-4	6. E-12	4. E-5	2. E-12
	S	BONE	1. E-4	6. E-12	5. E-5	2. E-12
	S	LIVER	2. E-4	9. E-12	7. E-5	3. E-12
	S	TOTAL BODY	4. E-4	2. E-11	1. E-4	5. E-12
	S	G I (LLI)	8. E-4	2. E-7	3. E-4	6. E-8
	I	LUNG	---	1. E-10	---	4. E-11
	I	G I (LLI)	8. E-4	1. E-7	2. E-4	5. E-8
^{243}AM	S		1. E-4	6. E-12	4. E-6	2. E-13
	I		8. E-4	1. E-10	3. E-5	4. E-12
	S	BONE	1. E-4	6. E-12	4. E-5	2. E-12
	S	KIDNEY	1. E-4	6. E-12	5. E-5	2. E-12
	S	LIVER	2. E-4	9. E-12	7. E-5	3. E-12
	S	TOTAL BODY	4. E-4	2. E-11	1. E-4	5. E-12
	S	G I (LLI)	8. E-4	2. E-7	3. E-4	6. E-8
	I	LUNG	---	1. E-10	---	4. E-11
	I	G I (LLI)	8. E-4	1. E-7	3. E-4	5. E-8

RADIOISOTOPE		ORGAN	40 HOUR WEEK WATER (UC/CC)	40 HOUR WEEK AIR (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) WATER (UC/CC)	168 HOUR WEEK (CONTINUOUS EXPOSURE) AIR (UC/CC)
^{236}U	S		1. E-3	6. E-10	3. E-5	2. E-11
	I		1. E-3	1. E-10	3. E-5	4. E-12
	S	G I (LLI)	1. E-3	2. E-7	3. E-4	7. E-8
	S	BONE	.01	6. E-10	5. E-3	2. E-10
	S	KIDNEY	.03	1. E-9	.01	4. E-10
	S	TOTAL BODY	.04	2. E-9	.01	6. E-10
	I	LUNG	---	1. E-9	---	4. E-11
	I	G I (LLI)	1. E-3	2. E-7	3. E-4	6. E-8
$^{238\#}U$	S		1. E-3	7. E-11	4. E-5	3. E-12
	I		1. E-3	1. E-10	4. E-5	5. E-12
	S	G I (LLI)	1. E-3	2. E-7	4. E-4	8. E-8
	S	KIDNEY	2. E-3	7. E-11	6. E-4	3. E-11
	S	BONE	.01	6. E-10	5. E-3	2. E-10
	S	TOTAL BODY	.04	2. E-9	.01	6. E-10
	I	LUNG	---	1. E-10	---	5. E-11
	I	G I (LLI)	1. E-3	2. E-7	4. E-4	6. E-8
^{237}Np	S		9. E-5	4. E-12	3. E-6	1. E-13
	I		9. E-4	1. E-10	3. E-5	4. E-12
	S	BONE	9. E-5	4. E-12	3. E-5	1. E-12
	S	KIDNEY	2. E-4	7. E-12	6. E-5	2. E-12
	S	TOTAL BODY	4. E-4	2. E-11	1. E-4	6. E-12
	S	LIVER	6. E-4	2. E-11	2. E-4	8. E-12
	S	G I (LLI)	9. E-4	2. E-7	3. E-4	7. E-8
	I	LUNG	---	1. E-10	---	4. E-11
	I	G I (LLI)	9. E-4	2. E-7	3. E-4	5. E-8

\# Chemical toxicity may the the limiting factor for soluble mixtures of U-234, 235 and 238. See 10CFR20.