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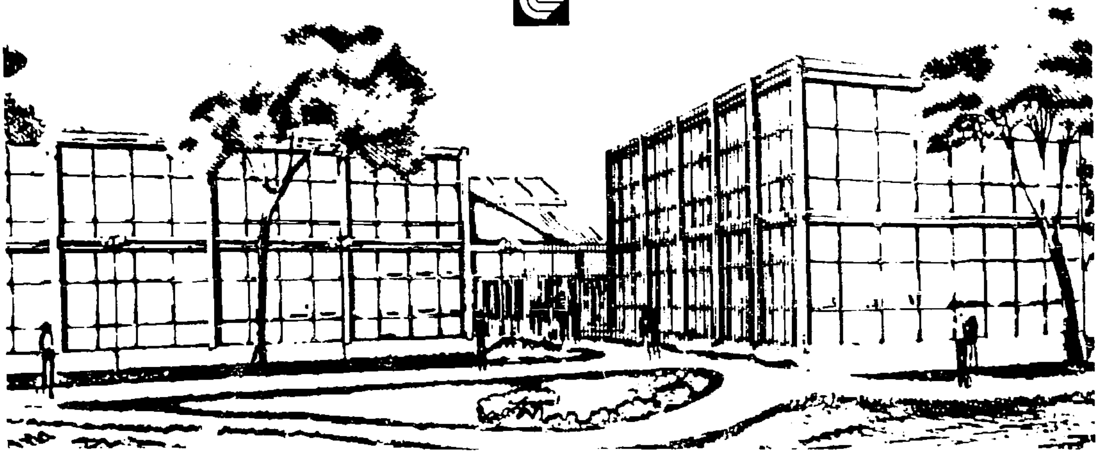

DETERMINATION OF PERFORMANCE CRITERIA FOR HIGH-LEVEL SOLIDIFIED NUCLEAR WASTE FROM THE COMMERCIAL NUCLEAR FUEL CYCLE: A PROBABILISTIC SAFETY ANALYSIS*

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ABSTRACT

To minimize the radiological risk from the operation of a waste management system for the safe disposal of high-level waste, performance characteristics of the solidified waste form must be specified. The minimum waste form characteristics that must be specified are the radionuclide volatilization fraction, airborne particulate dispersion fraction, and the aqueous dissolution characteristics. The results indicate that the pre-emplacement environs are more limiting in establishing the waste form performance criteria than the post-emplacement environs. The actual values of expected risk are sensitive to modeling assumptions and data base uncertainties. The transportation step appears to be the most limiting in determining the required performance characteristics.

INTRODUCTION

Prior to the Presidential decision to forego commercial spent fuel reprocessing, with its consequent recycling of plutonium and/or uranium, certain regulatory issues had to be addressed. These issues included the development of regulations for the performance criteria for high-level solidified nuclear waste from the commercial nuclear fuel cycle in order to close the "back end" of that cycle. The safe management of nuclear waste is clearly one of the pivotal issues facing the future role of power generated by the light water reactor nuclear system. Given the process flow sheet and the various operations associated with the solidification of high-level nuclear wastes, the regulator must decide what are the critical performance characteristics to be demanded of an allowable solid waste form. A decision mechanism to solve this problem is available using probabilistic analysis techniques.

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FUEL CYCLE

The operation of commercial nuclear reactors for the generation of electric power produces radioactive wastes. The "back end" of this fuel cycle includes all activities after removal of spent fuel rods from the reactor. The spent rods are taken from storage at the reactor site and transported to a fuel reprocessing plant. Here the unused uranium and the plutonium produced in the reactor are recovered. High-level radioactive waste is generated at the reprocessing facility. This waste must be solidified, stored to allow decay of radioactively generated heat, and finally transported to a Federal repository site for final disposition. Several alternative fuel cycles are under study. The results reported are concerned with the uranium recycle and plutonium storage reference fuel cycle. Closing the "back end" of this reference light water nuclear fuel cycle would provide for the operation of the spent fuel reprocessing plants; recycling of uranium by recovery and subsequent fabrication into nuclear reactor fuel rods; storage of the recovered plutonium, in some manner; solidifying and storing high-level wastes; transporting these wastes to the repository site; and finally, permanently disposing of the high-level wastes into deep geologic formations. This is shown graphically in Figure 1.

REFERENCE WASTE MANAGEMENT SYSTEM

The high-level waste management system has been defined as processing the solidified waste form through the interim storage operation; transportation to the repository site; handling at the repository; and finally, the future history after sealing the repository. Figure 2 represents a hierarchical approach to this waste management system analysis problem. Note that time flows to the right and functional detail down. This methodology allows us to compare the expectation of the overall radiological risk to the waste form release parameters. Assembly of a data base of characteristics and release parameters of candiate waste forms allows a comparison of the radiological risk expectation over all reasonable performance criteria specification. Thus, the computer simulation model may be used to determine the radiological risk inherent in the operation of the waste management system from a particular solid waste form.

PROBABILISTIC ANALYSIS MODEL

A probabilistic analysis model was developed which considers radioactivity releases for potential accident environments to which high-level waste solid matrices could be exposed during interim storage at the fuel reprocessing plant, transportation, handling and emplacement in a deep geologic medium, and the future history after sealing of the repository. Functional event trees were constructed for each waste management operation to identify potential release mechanisms, i.e., failure modes. Figures 3 through 7 give the event trees used, in analyzing the pre-emplacement risks. Published data for failure probabilities were used whenever available. Release functions were developed to quantitatively evaluate the severity of various failure consequences expressed as an expected value of radiological risk. Releases are normalized to Ci/Mwe-yr. Individual and population exposure doses were calculated on an expected value basis, normalized to per Mwe-yr. These expected value calculations provide a means of identifying the potential "pinch points" for setting criteria. The solid matrix performance criteria must consider the reduction of both the integrated risks and single event risks to the public to "as low as reasonably achievable" levels. The probabilistic model, as developed to date, treats only the technological issues involved in nuclear waste management and stops after defining the intermediate physical consequences from the environmental model. Treatment of the consequences of the radiological risks within the societal, ecological, or economic regime is not addressed in this study.

MODELING ASSUMPTIONS

We have studied the high-level nuclear waste management system associated with a technologically mature LWR nuclear power industry. We have limited ourselves, in this initial work, to the uranium recycle/plutonium storage LWR fuel cycle. The spent fuel is reprocessed 150 days after discharge from the reactor. The spent fuel is discharged after irradiation of 33,000 MWD/MTNU at 30MW/MTU. High-level waste is defined as raffinate from the first stage extraction in the Purex process and does not include intermediate or low-level wastes. Uranium and plutonium recovery is 99.5%, i.e., 0.5% remain in the waste stream. Consistent with 10CFR50, Appendix F, high-level liquid wastes will be solidified within 5 years after separation of the fission products from the irradiated fuel, and transferred to a Federal Waste repository within 10 years. High-level waste canisters are 0.305m in diameter and 3.05m long. Fill capacity is 0.2m or approximately the high-level waste generated from 100MWe-yr. of electric power generation. Canisters are fabricated from 304L stainless steel alloy. Only chemically, thermally, and radiolytically stable dry solids are allowed by Appendix F. Five waste forms were considered. Salt cake does not meet Appendix F requirements for stability and was eliminated. To insure that a sufficiently wide range of waste form characteristics were considered, but that the results were still bounded by reality, we used characteristics for spray calcine, fluidized-bed calcine, borosilicate glass, and supercalcine pellets coated with aluminum oxide and encased in a lead matrix, i.e., multi-barrier metal matrix.

Facilities housing solidification, interim HLW storage tanks, canister storage pool, and all solidification process equipment handling liquids are designed to Seismic category

1, i.e., 0.25g horizontal load. Interim HLW tank storage is provided prior to solidification as required by waste form thermal conductivity properties to meet canister centerline temperature limits. Internal canister fins will be provided as necessary to minimize interim liquid storage requirements. Canister storage at the reprocessing facility is an underwater storage pool until shipment time. Maximum allowed decay heat of canister is 3.5 Kw at shipment time, i.e., 10 years, to the Federal repository. Ages of entry into and exit from the interim storage pool at the reprocessing plant are as follows:

- Multibarrier metal matrix - 150 days and 10 years
- Borosilicate glass - one year and 10 years
- Spray and fluidized bed calcines - 5 years and 10 years

Four canisters are shipped per railroad cask. One canister is shipped per truck cask. No special transportation procedures are used. Cask shipments are by truck or train. Crush of massive casks and immersion as a damaging accident is considered an extraordinary event, i.e., one of low expected consequence; similary with puncture, after analyzing the expected release fraction to be zero. Canisters offer little protection, relative to the cask in impact accidents. Shipping distance from the reprocessing plant to the Federal repository is 1,500 miles. Transportation accidents take place with equal probability in urban and rural areas. All urban accidents occur at the city's center. Demography for urban airborne dose is the Dallas-Fort Worth area. Rural accident demography is assumed with a constant low population density. A puff meteorology model governs spread of airborne particulates released in impact accidents. A plume model is used for volatile released and airborne particulate for fire accidents. Radionuclide releases from transportation accidents are governed for the three release mechanisms by the appropriate release fractions.

1) Dissolution: Fraction of all radionuclide species present escape from glass and multibarrier waste forms. Only Sr and Cs escape from calcines by dissolution.

2) Volatilization: Fire accidents released Cs from all four waste forms and Ru is released from spray and fluidized bed.

3) Airborne particulate dispersion: Fractions of all species present escape by dispersion in impact and fire/accidents for the four waste forms, except no release by dispersion from glass due to fire accidents, i.e., glass melts.

All parallel accident sequences in event trees are assumed to be mutually exclusive events, thus allowing their individual expected doses to be added to compute overall risk. All dose calculations are consistent with procedures used in USNRC Reg. Guide 1.109. Biosphere transport and human dose has been modeled to study expected value of risk for both pre-emplacement accidents and post-sealing future events.

Dose to humans will depend on the movement of radionuclides through surface waters, concentration of various nuclides in the ecosystem, human living habits and diets, and the biochemistry of the radionuclides in the human body. Using our model we calculate doses due to external exposure, immersion or ingestion. Ingestion pathways include drinking water, irrigated farm crops, animal products, and aquatic foods. Aquatic foods include fresh water species, estuarine species, and pelagic fishes present in the "plume" of fresh water in the ocean near the river mouth. Note that spread of all surface waterborne radionuclides released by pre-emplacement accidents is calculated using the water transport model. Extrapolation of demographies associated with repository sites into the future is not possible. One can, however, develop potential population doses. Radionuclide pathways that dominate population doses include aquatic food, irrigated crops, and animal products. The doses therefore are dependent on harvest rates and irrigation rates, not on local demography. Therefore, population doses were calculated on usage rates of the water system rather than on population estimates.

SUMMARY

Tables 1 and 2 give values for expected whole-body dose from pre-emplacement accidents. Table 3 gives values for expected whole-body dose from post-emplacement releases. The condensed sensitivity results presented in Tables 1 and 2 represent uncertainties associated with both the modeling process and data inputs to those models for the pre-emplacement risks. The parameter sensitivities indicated in these tables reflect a methodology that involves computing values of expected dose by simultaneously perturbing all the non-linear factors to either their high-risk or low-risk response scenarios. These nonlinear factors were identified by our analysis to be accident severity probability distribution functions, radionuclide release functions, and cask/canister thermal failure loci for various transportation accident scenarios. These results are further decomposed in Table 2 to reflect the release pathways/ mechanisms that allow accidental introduction of radionuclides into the biosphere. Limitations on allowable performance characteristics of solidified high level waste forms provide the regulator with a way to minimize the radiological risk associated with the operation of a high-level nuclear waste management system. Specification of of limits for waste form radionuclide volatilization fractions, airborne particulate dispersions and aqueous dissolution characteristics will minimize risk. Expected values of risk from transportation accidents are given in Table 2 for four waste forms that cover the spectrum of waste form characteristics currently available.

Inspection of this table will indicate the relative tradeoffs between alternative waste forms under current consideration. Table 3 shows interim results from studies of post-emplacement risk from deep geologic repositories. The risk values have been calculated out to times of approximately three million years. The population dose was integrated over this time period. Peak individual, based on a 50 year exposure period, were also calculated and are shown with their corresponding times. Uncertainty analyses for the post-sealing phase continues. Space limitations do not allow an adequate review of the methodology being developed to address this important question. The reader is directed to reference 5 for greater detail.

The transportation step is the "pinch point" in the operation of a high-level nuclear waste management system. Interim storage at the reprocessing plant may be the next limiting step. Comparison of the risks associated with pre-emplacement operations, Table 1, with the baseline repository post-sealing risks, Table 3, shows that pre-emplacement operations will be most limiting in determining the waste form performance characteristics that will be required.

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TABLE 1
EXPECTED VALUES OF WHOLE-BODY DOSE FROM PRE-EMPLACEMENT ACCIDENTS
(man-REM/MWe-yr)

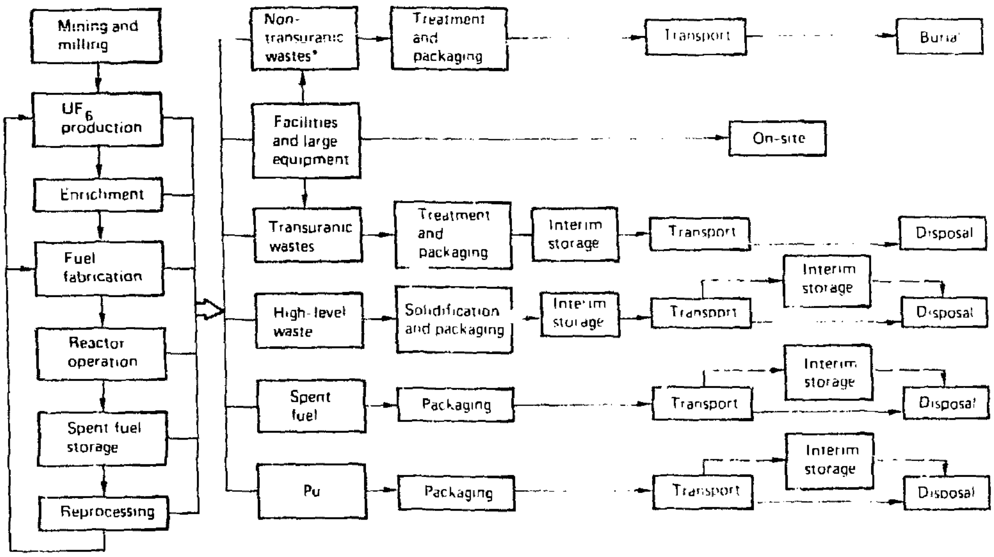
Waste Form	Parameter Sensitivity	Accident Type Transportation Truck	Transportation Train	Interim Storage Pool Drainage	Other	Total Shipment By Truck	Total Shipment By Train
Spray Calcine	upper	1.9E1	5.4E-1	0	---	1.9E1	5.4E-1
	baseline	1.7E0	4.1E-2	0	2.1E-13	1.7E0	4.1E-2
	lower	1.9E-3	9.4E-6	0	---	1.9E-3	9.9E-6
Fluidized Bed Calcine	upper	1.0E0	2.7E-1	0	---	1.0E0	2.7E-1
	baseline	3.6E-1	9.7E-3	0	2.1E-13	3.6E-1	9.7E-3
	lower	1.6E-1	5.5E-6	0	---	1.6E-3	6.0E-6
Borosilicate Glass	upper	3.2E-1	1.6E-1	2.5 E0	---	2.5E0	2.5E0
	baseline	8.5E-3	4.9E-3	0	1.1E-13	8.5E-3	4.9E-3
	lower	8.4E-6	7.0E-7	0	---	8.4E-6	7.0E-7
Multibarrier Metal Matrix	upper	1.2E-2	3.3E-3	1.5 E0	---	1.5E0	1.5E0
	baseline	1.6E-5	2.3E-7	0	1.8E-11	1.6E-5	2.3E-7
	lower	6.9E-8	1.6E-10	0	---	6.9E-8	1.6E-10

TABLE 2
EXPECTED VALUES OF WHOLE-BODY DOSE FROM TRANSPORTATION ACCIDENTS
man-REM/MWe-yr

Waste Form	Parameter Sensitivity	Transport Mode: Truck Volatilization	Truck Airborne Dispersion	Truck Dissolution	Truck Totals	Train Volatilization	Train Airborne Dispersion	Train Dissolution	Train Totals
Spray Calcine	upper	1.2E-2	1.6E1	4.0E-1	1.9E1	1.5E-3	4.6E-1	7.6E-2	5.4E-1
	baseline	3.0E-5	1.7E0	5.4E-2	1.7E0	1.1E-7	3.4E-2	7.4E-3	4.1E-2
	lower	1.2E-2	1.5E-3	3.6E-4	1.9E-3	0	3.2E-6	6.2E-6	9.4E-6
Fluidized Bed Calcine	upper	1.2E-2	6.3E-1	4.0E-1	1.0E0	1.5E-3	1.9E-1	7.6E-2	2.7E-1
	baseline	3.0E-5	3.4E-1	2.2E-2	3.6E-1	1.1E-7	6.7E-3	2.9E-3	9.7E-3
	lower	0	1.5E-3	1.5E-4	1.6E-3	0	3.0E-6	2.5E-6	5.5E-6
Borosilicate Glass	upper	2.4E-1	7.7E-2	1.7E-3	3.2E-1	1.5E-1	2.3E-3	4.8E-4	1.6E-1
	baseline	7.7E-3	8.5E-4	1.6E-5	8.5E-3	4.9E-3	1.7E-5	2.9E-6	4.9E-3
	lower	1.7E-14	8.2E-6	2.1E-7	8.4E-6	6.8E-7	1.6E-8	3.2E-9	7.0E-7
Multibarrier Metal Matrix	upper	6.6E-3	5.8E-3	1.7E-5	1.2E-2	1.9E-3	1.3E-3	4.8E-6	3.3E-3
	baseline	6.7E-6	8.8E-6	1.6E-7	1.6E-5	2.6E-8	1.7E-7	2.9E-8	2.3E-7
	lower	0	6.7E-8	2.1E-9	6.9E-8	0	1.3E-10	3.2E-11	1.6E-10

TABLE 3
EXPECTED VALUES OF WHOLE-BODY DOSE FROM POST-SEALING ENVIRONMENTS

Repository Geologic Medium	Parameter Sensitivity	Integrated Population Dose manRem/MWe-yr	Peak Individual Dose REM/MWe-yr Dose	Time (yrs)
Shale	baseline	1.3E-3	6E-14	1.4E4
Bedded Salt	baseline	1.6E-3	8E-14	1.5E5



*Low-Level Wastes

Fig. 1 LWR FUEL CYCLE AND WASTE MANAGEMENT SYSTEM

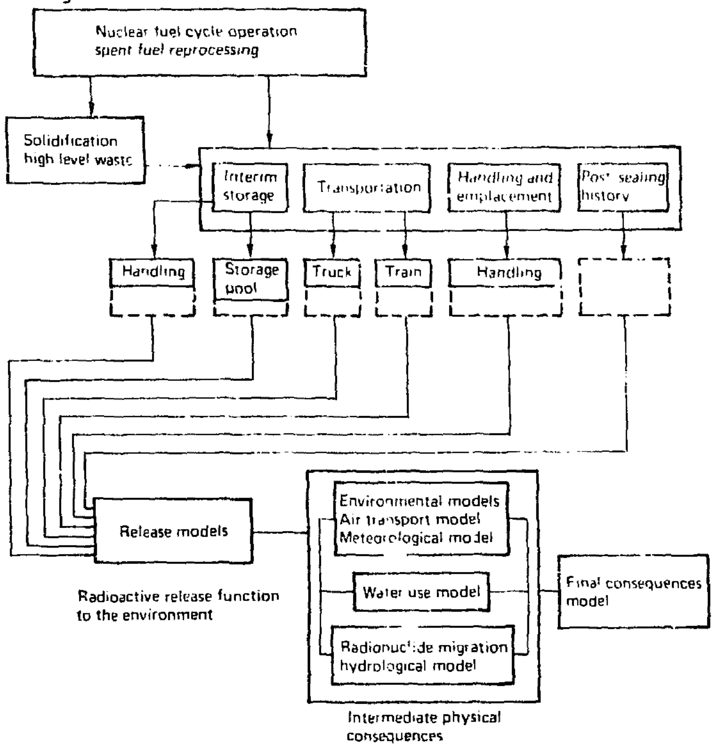


Fig. 2 WASTE MANAGEMENT SYSTEM ANALYSIS MODEL

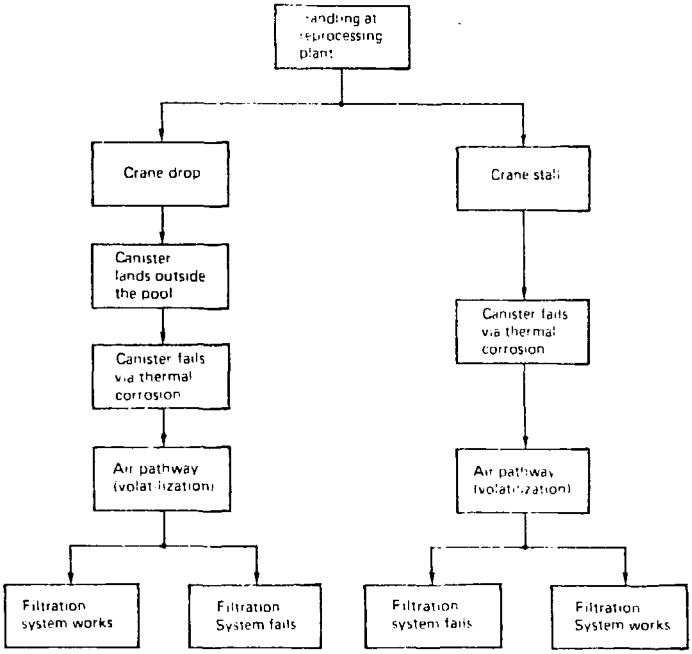


Fig. 3 EVENT TREE FOR HANDLING ACCIDENTS AT THE FUEL REPROCESSING PLANT

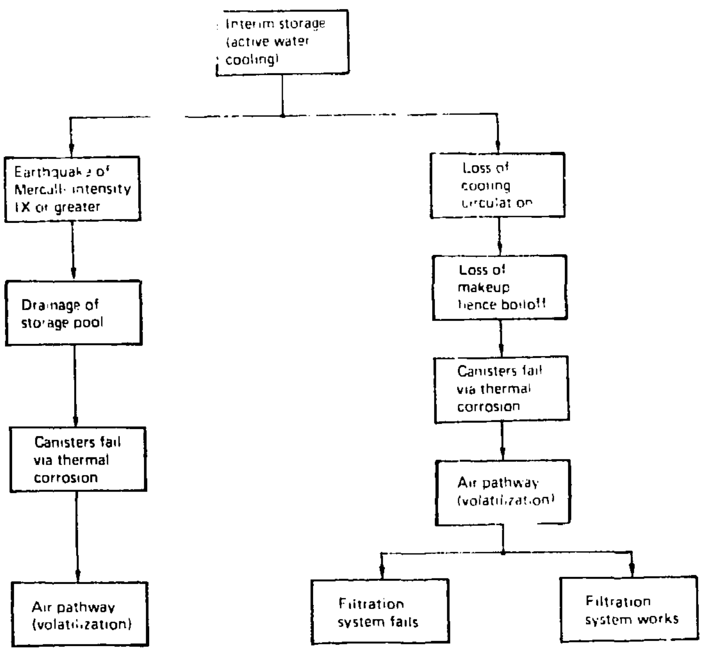


Fig. 4 EVENT TREE FOR INTERIM STORAGE AT REPROCESSING PLANT IN WATER POOL

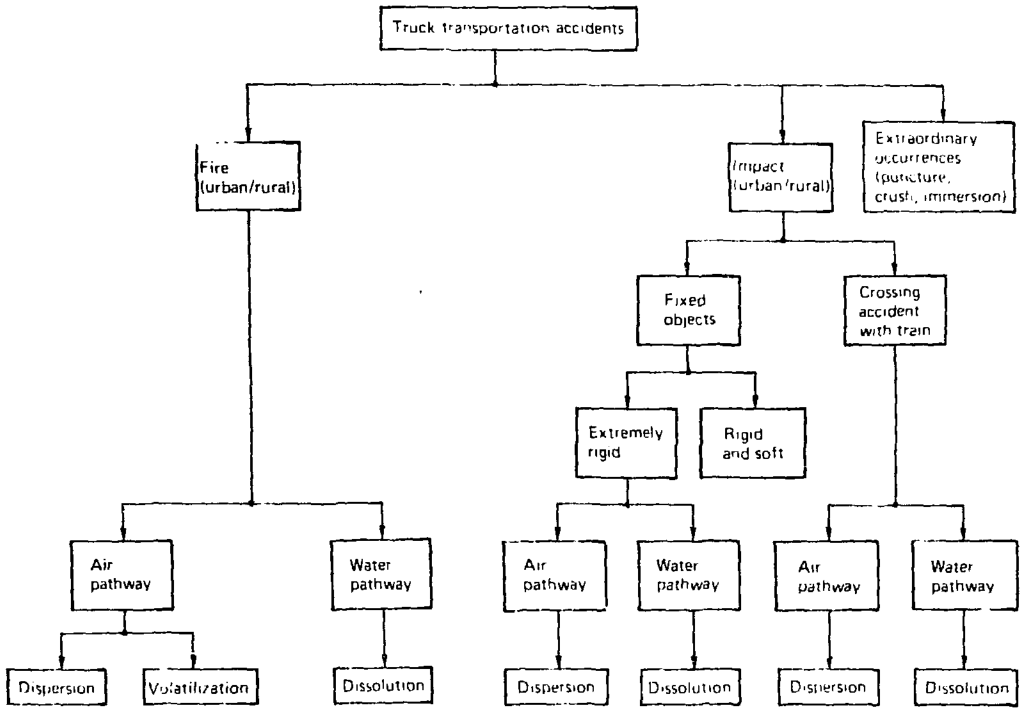


Fig. 5 EVENT TREE FOR TRUCK TRANSPORTATION ACCIDENTS

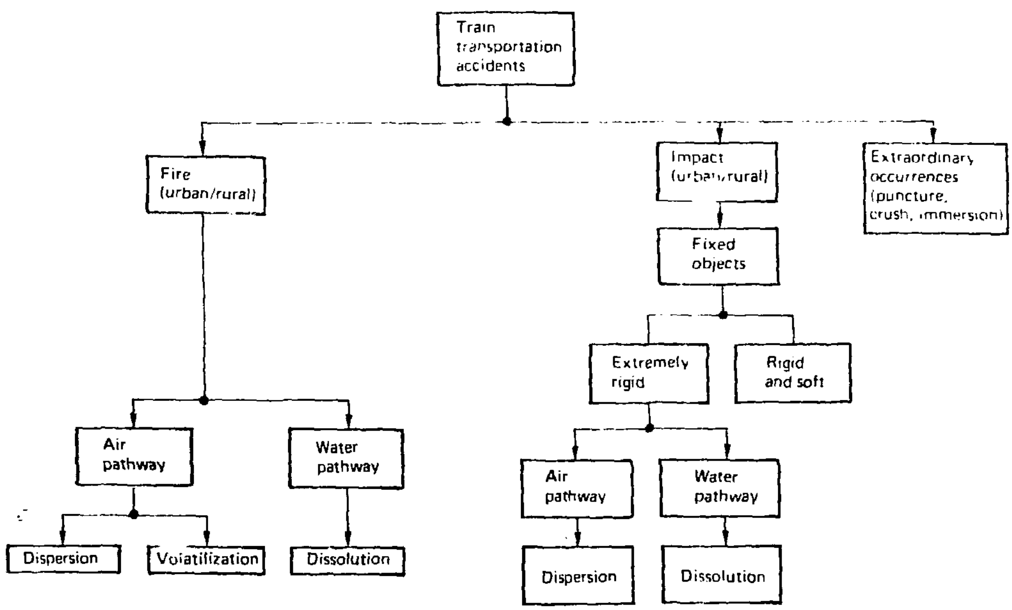


Fig. 6 EVENT TREE FOR TRAIN TRANSPORTATION ACCIDENTS

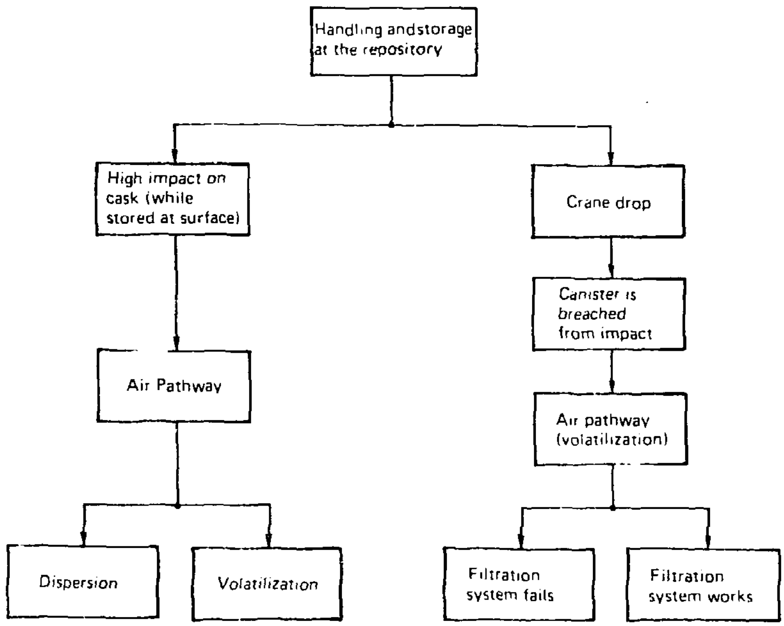


Fig. 7 EVENT TREE FOR HANDLING AND STORAGE ACCIDENTS AT THE REPOSITORY

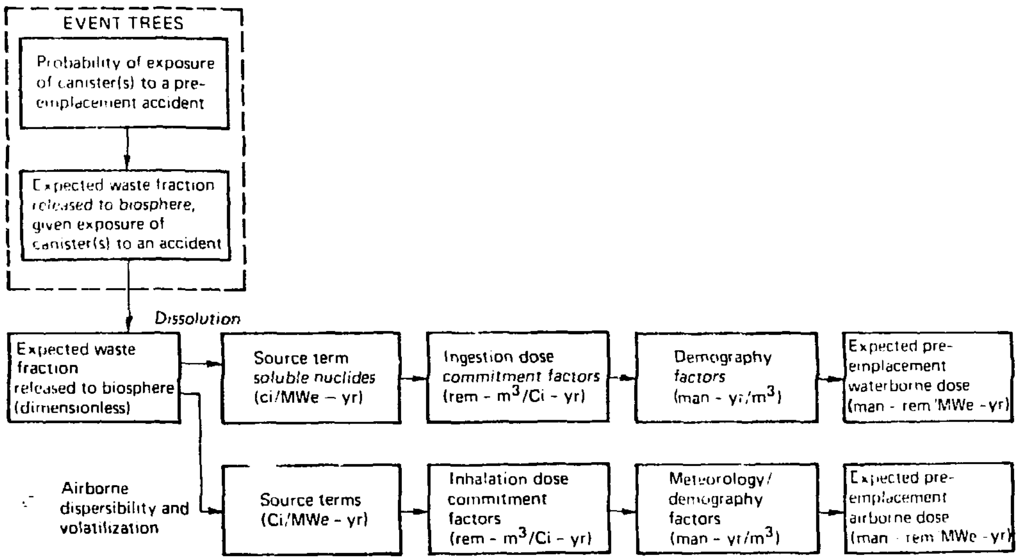


Fig. 8 RISK ASSESSMENT MODEL CALCULATIONAL SEQUENCE