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High-Level Waste Preclosure Systems Safety Analysis Phase 1, Final Report

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HIGH-LEVEL WASTE PRECLOSURE SYSTEMS SAFETY ANALYSIS

Phase 1, Final Report

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ABSTRACT

This report documents the results of the evaluations performed under the Program Planning Task - Phase I of the High-Level Waste Preclosure Systems Safety Analysis (HLW_PSSA) project for the U.S. Nuclear Regulatory Commission under contract to the Sandia National Laboratories.

The major effort for this phase of the project has been on the gathering, organizing, and assembling of information pertinent to the safety assessment of a nuclear waste repository during preclosure operations. Specific issues addressed in this report are:

1. Detailed analysis of a conceptual basalt repository design in order to identify potential initiating events/accident scenarios capable of causing radiological and/or nonradiological consequences;

2. Evaluation of radiological and nonradiological consequences relevant to a nuclear repository and recommendation of an approach for quantitative evaluation of these consequences;

3. Comparative evaluation of several importance ranking measures that had been used in the nuclear industry in order to select a measure to best meet the needs of the program (i.e., a measure which leads to an easily scrutable importance ranking to prioritize components/systems which are important contributors to safety);

4. Development of event and fault tree models for those initiating events which have passed the preliminary screening process (i.e., elimination of insignificant risk contributors);

5. Compilation of specific data such as initiating event frequencies, component/system failure rates and repair times, personnel injury, and basic information necessary for more detailed radiological consequence evaluations at a later time; and

6. Selection of a set of accident scenarios to be quantified in the next study phase to demonstrate the applicability of the proposed methodology that will identify and quantitatively prioritize structures, components, systems, and operations which are important to safety during the preclosure phase of a HLW repository.

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1.0 INTRODUCTION AND SUMMARY

This report documents the results of the evaluations performed under the Program Planning Task - Phase I of the High-Level Waste Preclosure Systems Safety Analysis (HLW-PSSA) project being performed by GA Technologies, Inc. for the U.S. Nuclear Regulatory Commission under contract to the Sandia National Laboratories.

The overall objectives of the HLW-PSSA project are (1) to develop and demonstrate the applicability of a systematic methodology that will identify and quantitatively prioritize structures, components, systems and operations which are important to safety during the preclosure phase of the HLW repository; and (2) to evaluate and rank the relative importance of repository components and systems in order to assist the NRC staff to make related regulatory planning and decisions. To meet these objectives, a methodology as illustrated in Fig. 1-1 has been proposed. The project tasks involve the integration of the various components of this methodology to provide the necessary information for the NRC decision-making process. The schedule for completion of these tasks is shown in Fig. 1-2. This schedule is intended to be responsive to the commitment dates proposed by the Department of Energy (DOE) for various reviews of the repository design concept.

The work performed in FY-84 consisted of the following tasks:

1. Task 1 - a literature review of the safety assessment methodology and supporting data as applied to the preclosure phase of nuclear waste repositories.

2. Task 2A - phase I of the program planning task which consists of evaluations required to develop the basic information necessary to demonstrate a comprehensive methodology for identifying components and systems important to safety and to rank them according to their degree of importance.

The results of Task 1 are documented in a separate report (Ligon, 1984). This task provided the background of available technical information needed to determine the direction taken in the performance of Task 2A. A more detailed description of the scope of work for Task 2A is given in Section 1.1.

Because only specific subtasks under Task 2 are addressed at this time, the results reported herein should not be considered as encompassing all the information required to perform a comprehensive safety evaluation of a nuclear waste repository. A number of equally important issues such as human reliability analysis, common-cause failure analysis, risk quantification, and component/system importance rankings will be addressed in the next study phase. Only when these studies will have been completed and integrated into other portions of the entire study can reasonable

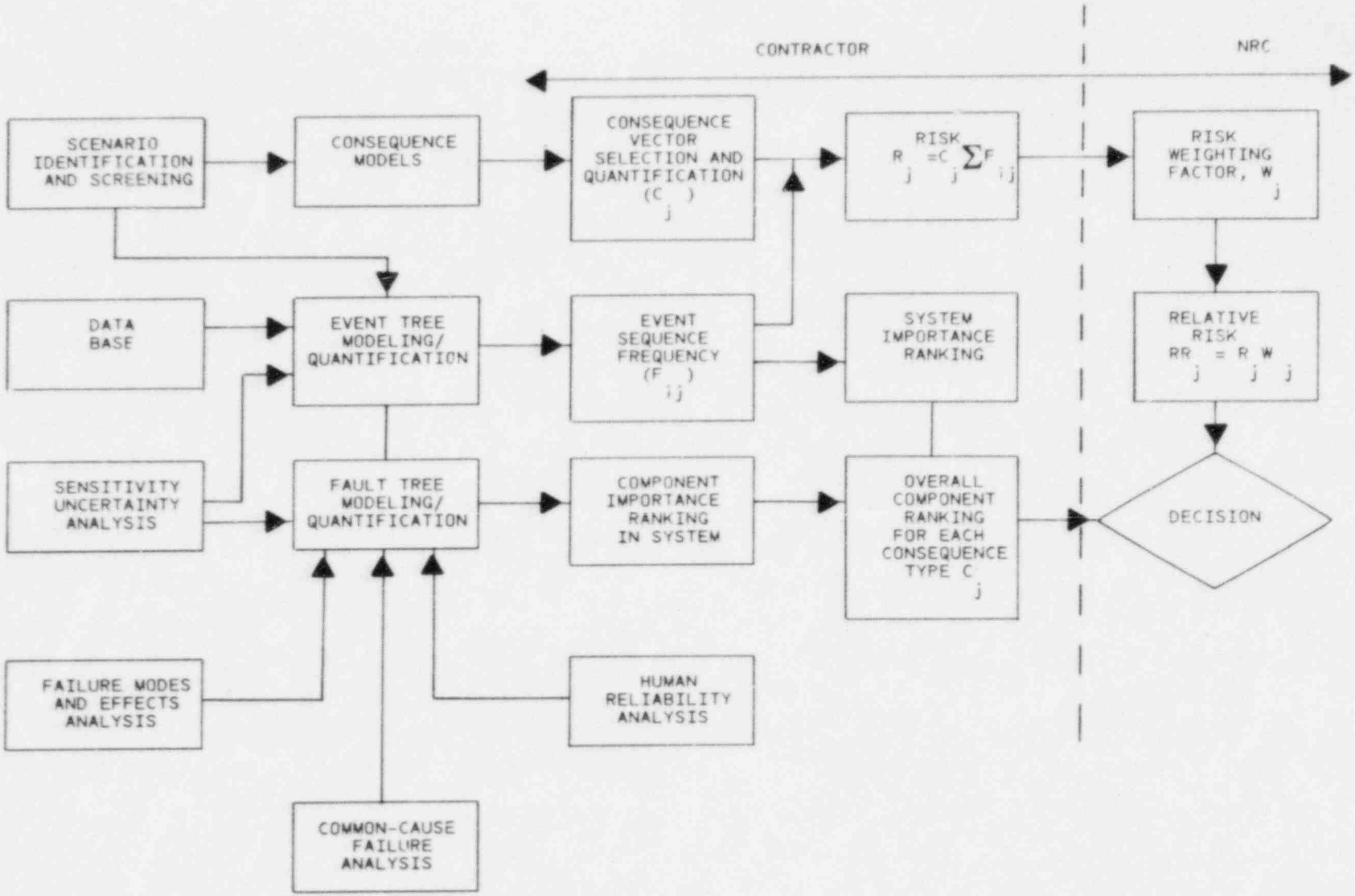

Fig. 1-1 Methodology for high-level waste preclosure safety systems analysis

	Task Description	Scheduled Completion FY-84	FY-85	FY-86	FY-87
TASK 1	Literature Review	____O			
TASK 2A	Program Planning, Phase I Includes accident scenario identification and screening, consequence type identification, importance measures evaluation, fault tree and event tree development, sample problem definition, and compilation of initial data base for event tree/fault tree quantification.	____O			
TASK 2B	Program Planning, Phase II Involves quantitative analysis of sample problem including human reliability analysis, common-cause failure analysis, sensitivity and uncertainty analysis, data base development, importance ranking calculations, consequence quantification, and demonstration of a comprehensive methodology.		____O		
TASK 3	Complete Analysis of the Basalt Repository			____O	
TASK 4	Scoping and Complete Analysis of Repositories in Other Geologic Media				____O

Fig. 1-2 HLW-PSSA Project Schedule

conclusions be made regarding the overall risk to the public of repository preclosure operations. Complete demonstration of the proposed methodology must await the results from the second phase of this project.

1.1 TASK 2A-PHASE I WORK SCOPE

The Task 2 work scope for FY-84 included the following subtasks:

Subtask 2.1 - Scenario Enumeration and Selection. This subtask involved a detailed analysis of a conceptual basalt repository design in order to identify potential initiating events/accident scenarios capable of causing radiological and/or nonradiological consequences. Relevant initiating events, i.e. those that may be high-risk contributors were selected by initially applying a qualitative screening process. In order to assure completeness, the entire process flow, including both surface, subsurface, and support activities, was considered.

Subtask 2.2 - Consequence Identification. The major objectives of this subtask were to: (1) evaluate the various radiological and nonradiological consequences relevant to a nuclear repository; (2) recommend an approach for quantitative evaluation of each consequence type; and (3) recommend the specific consequence types for quantitative evaluation.

Subtask 2.3 - Importance Measures Evaluation. This subtask involved a comparative evaluation of several importance ranking measures used in the nuclear industry for the purpose of selecting a measure to best meet the needs of the program; i.e., a measure which leads to an easily scrutable importance ranking to prioritize components/systems which are important contributors to safety.

Subtask 2.4A - Event Tree/Fault Tree Development. Initiating events that passed the preliminary screening process performed in Subtask 2.1 were developed into accident scenarios (corresponding to a particular consequence) using the event tree modeling approach. Each system represented in the event trees was analyzed further to the subsystem or component level (depending on data availability) by using fault trees. The quantification of these event trees and fault trees will be performed in the next study phase.

Subtask 2.7A - Data Base. This subtask involved the compilation of specific data such as initiating event frequencies, component/system failure rates and repair time, personnel injury, and basic information necessary for more detailed radiological consequence evaluations at a later time. The data base developed at this time focused mainly on events relevant to the accident scenarios selected for the reduced problem analysis (Subtask 2.11A). Human reliability and common-cause failures were not addressed because these topics will be the subject of a more detailed study in the next phase.

Subtask 2.11A - Sample Problem Definition. The main objective of this subtask was to select a set of accident scenarios to be quantified in the next study phase. These scenarios will be used to demonstrate the applicability of the proposed methodology (See Fig. 1-1) and to identify any refinements that may be required or problems that must be addressed before a complete analysis of the basalt repository is performed.

1.1.2 Report Organization

This report is organized into sections dealing with each of the major subtasks outlined previously plus a final section discussing the results of each subtask. Included in the appendix is the complete report addressing the selection of an importance measure for this project.

1.2 SUMMARY

The major effort for this phase of the project was placed in gathering, organizing, and assembling of all information pertinent to the safety assessment of a nuclear waste repository during preclosure operations. Some additional effort was expended to assemble the event and fault tree logic models required to quantitatively evaluate the safety of the repository concept.

The accident scenarios developed in this study were limited to the basalt repository concept. Material flow diagrams were developed for each waste handling process and used to examine each process step for identification of potential initiating events in the form of external events (e.g., earthquake, tornado, etc.), equipment failures, or human actions capable of process disruption, local personnel exposure/occupational injury, or challenge of plant barriers to radionuclide release. All potential initiating events were then subjected to a preliminary screening process to eliminate insignificant risk contributors (i.e., events having low frequency and low consequence). Screening was performed separately for each consequence type initially considered, allowing the subsequent development of accident scenario families for each consequence type. Initiating events surviving this screening process were developed into a group of event tree models. An accident scenario is a subset of a particular event tree, consisting of the intiating event and a unique path of assumed intermediate event successes and/or failures leading either to a consequence of interest or to accident mitigation.

Scenarios addressing emplacement and retrieval operations were kept separate. Retrieval is an option for repository operations, not an integral part of the waste processing activities. Although many events and corresponding accident scenarios considered in the retrieval operations were the same as those in the emplacement operations, they were treated separately in order to provide a measure of relative risk due solely to retrieval operations in addition to emplacement risk.

Although not all of the scenarios developed in this study will be used in the sample problem evaluation (Phase II), the logic model development is complete and suitable for future safety analysis of a basalt repository.

The consequence types initially evaluated in this study were: 1) radiological consequence to the public, 2) radiological consequence to the worker, 3) nonradiological consequence to the worker, 4) impact on repository availability, 5) compromise of a repository's ability for long-term geologic isolation of high-level waste, and 6) financial impact. The compromise of long-term repository viability is not a risk contributor in the preclosure phase and has been addressed in this study only to identify event sequences and/or operations that are potentially capable of generating this consequence in order to facilitate consideration of preventive procedures and operations at an early stage in the repository design/licensing process. The other consequence types were developed to the extent where a representative body of accident scenarios exists for each, including the models necessary for quantitative evaluation of the consequences.

Of the different consequence types considered, the radiological (public and worker) and nonradiological occupational consequences were recommended for further evaluation. However, in the interest of providing useful information to the NRC for licensing purposes in a time frame consistent with the DOE repository design schedule, emphasis should be placed on radiological consequences to the public and worker.

In the quantitative evaluation of radiological risk to the public and worker, the transport and behavior of radionuclides play a key role and sophisticated transport models have been developed in previous studies. Reliable estimates of release fractions, however, were difficult to obtain largely because of the accident - specific nature of the release and the lack of adequate experimental data to support the postulated releases. The large uncertainty in the release fraction should be recognized at this time and accounted for in future work.

Fault tree development requirements for this project were based on the systems identified as intermediate events in the event tree accident scenarios. Wherever possible, portions of fault trees developed in previous repository safety analyses were used to avoid duplication of effort. Systems not explicitly described in the "Conceptual System Design Description, Nuclear Waste Repository in Basalt" (SD-BWI-SD-005) were not modeled due to lack of information. Instead, these systems have been assigned failure probabilities from comparisons with similar systems that are currently in operation for other applications. Future phases of this project may include fault tree modeling of these systems as design information becomes available.

The systems modeled were primarily the various building and subterranean confinement exhaust ventilation/filtration systems. In order for a radiological incident to occur, the barriers to local and environmental radionuclide release must be defeated. For the repository concept, these barriers consist mostly of the air circulation/cleanup systems.

There appears to be several specific functions of these systems (based on the design description) that could be modified as system design matures to ensure the desired levels of redundancy and independence. They are discussed below.

The primary and secondary confinement exhaust ventilation/filtration systems are not completely independent and neither are their respective backup units. All primary and secondary fan units exhaust to a common plenum with a single tornado/bird screen and a single stack. These commonalities could cause all confinement exhaust systems to fail in the event of damage to the plenum or stack (e.g. earthquake). The subterranean confinement exhaust system has a similar commonality. The exhaust of five fan/filter assemblies is routed to a common plenum, tornado damper/bird screen assembly, and exhaust stack.

Both the waste handling building and the subterranean confinement ventilation/filtration systems require intake supply fans to provide sufficient circulation. Power to the intake supply fans appears to be supplied from normal power load centers. Standby power can only be supplied to these loads by closing a bus cross-tie breaker between standby and normal power buses. This is not only an additional equipment operation (probably manually initiated) that must occur but also adds other loads to the standby bus, possibly exceeding generator capacity. This condition is not certain from the design information available; however, to preserve the redundancy and independence of backup systems, both the exhaust and the supply intake fans should be directly powered by redundant bus arrangements.

Two fan/filtration units of the waste transport shaft exhaust system are used continuously (no installed backup) to draw air up the waste transport shaft and through the filter assemblies. The power distribution description indicates these fans are supplied from a normal power bus, requiring the same cross-tie operation described earlier to restore fan operation given a loss of off-site power. Final design should consider 1) a backup unit for this function and 2) the use of a standby power bus to supply fan loads.

The waste handling building confinement exhaust ventilation/filtration systems (both primary and secondary) can be cross-tied between the filter train outlets and the blower fan inlets. The schematic provided for these systems does not show a check valve on the inlet to the fan. In the case of the primary confinement system, a loss of the normal fan followed by the startup of the standby fan could allow bypass of the entire airflow path through the repository in favor of the lower resistance of pulling air back through the disabled fan and the cross-tie line. A check valve in the inlet to each fan (both primary and secondary systems) would eliminate this possibility.

A sample problem has been selected for quantification in the next study phase to demonstrate the overall methodology. The selected problem is a small subset of the complete family of repository accident scenarios

leading to public radiological exposure. These scenarios involve the following initiating events:

1. Train/truck collision or derailment.

2. Breached shipping cask undetected by radiation monitoring system in the yard area.

3. Windstorm damages arrival/storage yard.

4. Radwaste sampling line rupture or improper connection in receiving area.

5. Loss of seal between hot cell floor and shipping cask lip.

6. Liquid radwaste leak from process tank to secondary area.

7. Loss of seal between hot cell floor and transfer cask lip.

8. Transfer cask rupture during handling accident-subterranean transport.

9. Transporter collision during transport to placement.

10. Canister breach during borehole insertion.

11. Earthquake causes loss of seal between hot cell floor and shipping cask lip.

The sample problem was chosen to be small enough to permit verification by hand. It should be emphasized that the simplicity of this family of scenarios (due to omission of approximately 90 scenarios) requires that the final estimates of risk and importance rankings be used only for demonstration purposes. The results of this analysis do not represent a preliminary evaluation or order of magnitude estimate of repository safety. Final results will also depend on the human error analysis results, and the estimation of common cause contribution.

The Fussel-Vesely importance measure was selected to rank the relative importance of repository components and systems and its use will be initially demonstrated in the analysis of the sample problem discussed above.

In support of future tasks under this project, data concerning initiating event frequencies, intermediate event probabilities, component failures, radionuclide release fractions, etc. were compiled from various government and industry data sources. The data gathering task will continue in the next study phase with emphasis on human error and mine-related accidents.

2.0 ACCIDENT SCENARIO ENUMERATION AND SELECTION

Accurate estimation of the risks associated with nuclear waste repository operation depends on identification of all potentially significant contributors to all types of risk concerns. These contributors consist of an abnormal occurrence (the initiating event) coupled to system and operator actions (intermediate events) capable of influencing event consequences, together referred to as an accident scenario. The methodical identification and preliminary screening of all potential accident scenarios capable of generating risk to the public, repository personnel, repository investment, and long-term repository viability are addressed here.

The specific objectives of the scenario enumeration and evaluation task are to:

- o develop a detailed set of diagrams depicting all important steps in the process of receiving, preparation, emplacement, and retrieval of spent fuel and high-level waste from the reference basalt repository system design description;
- o use these process flow diagrams to identify potential initiating events/accident sequences capable of causing the consequences of interest;
- o compare these events/sequences with those developed for other risk assessment studies and utilize models wherever possible; and
- o develop additional models where required for complete representation of all contributors to the types of risk under consideration.

This analysis is currently limited to the basalt repository concept; other geologic media may be considered in later phases, subject to direction from the NRC. Subsequent sections discuss the basalt repository design and material flow process, initiating event identification and screening, including natural/external events, and scenario development and screening.

Individual operations/processes required for the placement of waste (and possibly its retrieval) have been obtained from the System Design Description (SDD) of the Nuclear Waste Repository in Basalt (NWRB) report (Davis, 1983). Material process flow diagrams have been developed from this SDD for use in identifying all potential initiating events (see Section 2.1). These diagrams depict (to the level of detail currently available) all significant activities with particular attention focused on human actions.

Potential accident sequence initiating events were identified from the activities associated with each processing step. Material flow diagrams from Section 2.1 were used to examine each process step for identification of equipment failures or human actions capable of process disruption, local personnel exposure/occupational injury, or challenge of plant barriers to radioisotope

release. All potential initiating events were then subjected to a preliminary screening criteria (Pepping, 1981) to eliminate obviously insignificant risk contributors.

The initiating events surviving this screening process were then compared with initiating events already developed into accident scenarios in logic models available from the literature review. Particular emphasis was placed on (1) the mitigative equipment/systems identified in existing logic models and their similarity with equipment systems described in the NWRB study; and (2) additional human errors not included in existing logic models but identified from the material flow diagram development. Accident scenarios were assembled from these comparisons.

The material flow diagrams were further reviewed for susceptibility to natural/external events (i.e., earthquake, fire, etc.). Process steps vulnerable to these events were treated by creating an additional set of accident scenarios. These were created by taking the existing scenarios developed for the specific process step, and modifying them to reflect the increased occurrence probability of the initiating event and increased failure probabilities of associated equipment/systems given the external event.

All accident scenarios capable of generating any of the consequences of concern are listed in the results section. Each scenario has been assigned to one or more consequence categories identified in the consequence analysis task. These categories are discussed further in Section 3 of this report.

2.1 BASALT REPOSITORY DESIGN AND MATERIAL FLOW PROCESS

Movement of spent fuel and commercial high-level waste through various repository locations and the processes that occur at each location are described in this section. Figure 2-1 is a conceptual view of the surface facilities currently planned.[1] Figures 2-2 and 2-3 are detailed design drawings of the various areas comprising the surface facilities. The overall flow process for the repository is separated into activities occurring in the areas given in Table 2-1. Figure 2-4 provides a functional flow path of these activities in sequential order. Each area and its associated processing activities are addressed in the following sections.

2.1.1 Arrival Area

Radioactive waste shipments from commercial nuclear power plants will arrive at the repository primarily by rail car but truck shipments are also possible. The repository is currently being designed to handle either type of shipment.

Rail cars containing radioactive waste shipping casks will be shipped to the Burlington Northern railroad yard in Pasco, Washington where a special train will be assembled for the basalt repository located on the Hanford site. The Burlington Northern railroad will bring a daily train to the site boundary

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[1]Table 2-1 and Figs. 2-1 to 2-21 are located at the end of Section 2.1.

where repository personnel will move the train over the three-mile access road (see Fig. 2-2) to the repository receiving area. Shipping cask cross-country transportation risk is beyond the scope of this project; however, local handling and transfer functions are included. It is assumed that the definition of "preclosure operations" encompasses the transfer of the train at the site boundary from Burlington Northern to site personnel and the subsequent movement to the repository enclosure. Transportation risk studies separate from this project will address movement of the train from shipment origin to the Hanford site boundary railroad intersection. Truck shipments are not subject to these transfers at various boundaries and are assumed to become part of repository preclosure operat7K{s when the truck enters the repository access road.

The arrival area is shown on the bottom of Fig. 2-2. The function of this area is to provide an initial screening of incoming shipments for possible external radiation. Arriving truck and rail cars are routed past radiation monitoring systems located on the road and track, respectively. Units indicating external radiation are moved to the suspect rail and truck area, a location (see Fig. 2-3, C/D -9/10) surrounded by an elevated earth berm for shielding purposes.

The other function of the arrival area is to break arriving trains into individual rail cars for subsequent unloading. Uncoupling is performed with the switchyard locomotive used to bring the train from the intersection. Again, no similar transfer operation is required for truck shipments.

Figure 2-5 provides a material flow diagram for all steps/operations associated with the arrival area.

2.1.2 Yard Storage Area

The yard storage area, shown in Fig. 2-2 (A/B - 5/10), provides a large area for rail car and truck trailer storage prior to unloading. During this wait, each rail car and truck trailer is positioned over the respective inspection pit for exterior examination. This positioning maneuver is only one of several required to continually feed waste into the repository process. As more cars are moved into the waste handling building, those remaining in the yard are moved forward to generate room at the entrance for further shipments. Tractor trailer combinations are assumed to be reconnected for each movement. Rail cars are assumed to be positioned with the switchyard locomotive. Both the rail and truck inspection pits are located in the process path of moving the waste towards the waste handling building from the yard. No material flow figure is required for this area as the only process involved involved is a succession of attaching and pushing rail cars a short distance forward followed by uncoupling, or alternatively, a tractor hookup to a cask trailer followed by a short distance movement and uncoupling. During the course of these repeated maneuvers, every unit is eventually positioned over an inspection pit and subjected to exterior examination.

2.1.3 Washdown and Receiving Area

Rail cars and trucks containing waste shipping casks are shuttled one at a time into the waste handling building (see Fig. 2-6) from the yard storage area. Truck trailers are moved to the covered washdown area adjacent to the building entrance using truck tractors; rail cars are moved using a special type of low-speed tractor called a trackmobile. A normal switchyard locomotive is not used in the proximity of the waste handling building to minimize the possibility of high energy/momentum impact/crush accidents.

The washdown area is not isolated from the environment. Only external vehicle cleaning is performed to remove accumulated road dirt. Then the car or trailer and undisturbed cask assembly are moved inside airlock doors to the receiving area, again using a trackmobile or truck tractor to move rail and truck casks, respectively.

The primary function of the receiving area is to provide a controlled, sealed environment for the initial opening of the rail car or truck trailer. The vehicle is opened, the shock absorbing impact limiter around the shipping cask is removed, the externals of the cask smear tested for contamination, and a flexible pipe is connected to the cask for sampling the internal gases. A high activity level would indicate a combination of fuel pin and canister failure for a spent fuel shipment and would forewarn the operations personnel to expect high radiation/airborne activity levels in the hot cell when emptying this cask.

Following interior and exterior cask sampling, vehicles are moved through a second set of airlock doors to the cask unloading area, again by truck tractor or the special trackmobile for truck and rail shipments, respectively. Figure 2-7 outlines the major processing steps that occur in the washdown and receiving area.

2.1.4 Unloading Area

The unloading area provides a double barrier to the environment (two sets of airlock doors) for opening of the shipping cask and removal of SURF/CHLW canisters. The hot cell for canister handling is located directly above the unloading area.

Following movement of the vehicle/cask into the unloading area, it is automatically positioned by hydraulic arms under the hatch to the hot cell and the airlock doors are closed. Then, the vehicle is automatically leveled and clamps attached to prevent movement or misalignment. A power supply is connected to the cask rotation hydraulic drive assembly, and the upper and lower vehicle hatches removed to facilitate cask rotation (see Fig. 2-8). The cask is rotated to an upright position, a shielding collar is lowered from the primary hot cell and the pneumatic seals are inflated to insure an airtight seal. The hot cell shielding cover and the cask shielding cover are removed by the hot cell crane, exposing the canisters to the hot cell environment (see Fig. 2-8, D-9). Each canister is then raised into the primary hot cell,

checked at the automated smear test station for damage and surface contamination, and stored temporarily in the lag storage pit.

The interior of the empty shipping cask is smear tested for contamination using the primary hot cell crane, followed by replacement of the cask and hot cell shielding covers. The shielding cover is raised into the recess in the primary hot cell floor, the cask is rotated back to the horizontal (traveling) position, the vehicle upper and lower hatches closed, and the vehicle moved through the exit set of airlock doors into the cask preparation and shipping area (see Fig. 2-6, B/G - 4).

Laboratory analyses of the cask interior and canister exterior smear tests are performed locally (see Fig. 2-6, C-7) to determine cask decontamination and canister processing procedures. The material flow diagram for all functions associated with the unloading area is given in Fig. 2-9.

2.1.5 Secondary Area

The secondary area is an extension of the primary hot cell although the two can be sealed from each other. Canisters are either passed directly from the primary hot cell to the secondary area or through the intermediate step of the lag storage area (see Fig. 2-10). The secondary area provides direct access to either the transfer cask loading station or the process tank area. The transfer cask loading station is used to insert a single canister into the transfer cask for movement to underground storage. The process tank area is used to test canisters, place faulty ones in overpack containers, and seal/weld and test the overpacks. A succession of overhead crane movements in the primary hot cell and secondary areas are used to transfer the canisters as required.

The laboratory results obtained from canister smear and inspection tests are used to determine which process will be used for a specific canister. From the lag storage pit a canister will be transferred to one of two process tanks for any of the following operations:

- Welding the cover on a carbon-steel overpack for a CHLW canister.
- Rewelding a faulty weld, whether CHLW overpack or spent fuel canister.
- Overpacking; in case of an unlikely rupture of a CHLW overpack's 55.9 mm (2.2 in.) thick wall, the CHLW canister [324-mm (12.75 in.) O.D.] would be removed from the defective overpack and enclosed in a new overpack; a defective spent fuel canister [417 mm (16.4 in.) O.D.] could be overpacked in a container whose O.D. would be no more than the CHLW's O.D. of 457 mm (18 in.).
- Decontaminating if necessary.
- Transferring waste canister/overpacks from the possibly contaminated primary hot cell, through a decontamination station, to the secondary (and cleaner) hot cell where the clean canisters are loaded into the

transfer cask. The secondary hot cell is maintained clean to assure that the transfer cask remains clean and that clean canisters are emplaced.

The material flow diagram for the activities in the secondary area is shown in Fig. 2-11.

2.1.6 Holding Area

The completed, clean canister or overpack is lowered from the secondary hot cell into a transfer cask, which reduces radiation levels to 1 mrem/hr at 1 m (3 ft) from the cask surface. This transfer cask (see Fig. 2-12) shields the waste package during its journey from the hot cell to the waste transport shaft cage, down the shaft, and on to final placement. The cask assembly includes a telescopic, double-acting hydraulic cylinder with an electromagnet attached to the moving end, which will horizontally eject (and retract) a canister, overpack, dolly, or plug. When closed, the shielding door automatically locks by a spring-loaded pin, and is automatically unlocked when the cask is satisfactorily mated with either the hot cell or the borehole shielding assembly.

The transfer cask crane (see Fig. 2-13) moves the transfer cask to the holding area (area in contact with the hot cell outlet port, between the cask loading port and the waste cage) where a smear test is taken. The transfer cask then proceeds to the waste cage area through the hot cell airlock only if the smear test measurement is less than the allowed count rate.

2.1.7 Waste Cage Area and Waste Transport Shaft

The transfer cask is moved into the waste cage, where the cask rotation trunnions engage a vertical slot in the waste cage structure, minimizing cask movement. The waste cage is the largest area in the waste transport shaft which transports the transfer cask to the underground repository (see Fig. 2-14).

The shaft headframe is an integral part of the waste handling building consisting of a tower-mounted hoist and a service elevator in a concrete tower. It is of sufficient height to allow for cage, rope attachments, and overtravel. The tower and the emergency air intake duct are sealed from the ventilation zones of the waste handling building by an airlock door to maintain the pressure differential between the building and the mine-area ventilation circuits.

The hoisting system is designed to handle an expected receipt rate of 1,395 waste canisters and 1,600 CHTRU drums per year. The hoisting system is capable of raising or lowering a waste cage and payload weight of 37,682 kg (83,075 lb). It is designed to operate in a balanced mode and is equipped with a counterweight. The hoisting cage is locked in at the surface and at the underground shaft station before loading and unloading to prevent cable stretch or rebound.

The hoisting system consists of a friction hoist mounted in the headframe 41 m (134.5 ft) above the ground level. The hoist operates at a maximum speed of 2.5 m/sec (500 f/min) and a maximum acceleration/retardation of 0.6 m/sec^2 (2 ft/sec^2) to provide a cycle time of about 16 min., not including loading and unloading time.

The hoist is equipped with safety and overtravel devices. Normal safety devices included with a friction hoist are:

- o A motion switch connected to the deflection sheave or rope riding pulley that is activated when the "idle" drum moves faster than creep speed.
- o A tread wear switch that will open in the event of tread wear due to rope slippage.
- o A balance rope switch that opens if any of the loops rise.

Safety devices are also used on the cage to prevent cage movement while loading or unloading, or without closing the cage gate.

The shaft station for transferring canisters is 1,122 m (3,680 ft) below the surface. Areas for transporter maintenance and decontamination are adjacent to the station. An emergency shock absorber in the sump is designed to inhibit breaching of the transfer cask should there be a free-fall of the cage and its contents.

The transporter (see Fig. 2-15) receives the waste transfer cask at the subsurface level, moves it to the placement hole, ejects the waste canister/overpack from the cask to the borehole, and returns the empty cask to the cask receiving area. During travel in the placement rooms and entries, the transporter will normally be manned by two persons.

The material flow diagram associated with waste transport from the hot cell to the subsurface level is shown in Fig. 2-16.

2.1.8 Placement Area

2.1.8.1 Emplacement. The underground placement rooms for HLW are low, wide rooms, 3 m high by 6.1 m wide, with arched roofs. The waste canisters/overpacks are placed in horizontal, 762 m (30.0 in.) diameter holes extending from the room sidewalls. Placement holes are 61 m (200 ft) long. Two placement rooms, flanked by two reaming rooms forming three rows of placement holes, constitute a one-year waste panel.

Prior to emplacing a canister/overpack in a particular borehole, a crew attaches a shielding door assembly to the sleeve of that borehole, and inserts a dolly into the borehole between the skid rails and onto the V-rails adjacent to the sleeve.

The operators locate the placement position and align the transporter within about 7.6 cm (3 in.) of the ideal position. The cask cradle is then unlocked from alignment with the transporter vehicle frame by removal of the two safety shear pins. From inside the transporter, the operators activate the borehole interface phase of the automatic alignment system to rotate the cask horizontally and precisely align the cask with the shield door assembly. They manually lock the cask to the borehole sleeve, and plug in the power/control cable (transporter to borehole). Both borehole shielding and cask doors are opened, and the waste canister/overpack is hydraulically ejected from the cask onto skid rails inside the borehole. From there, the transporter dolly lifts the waste canister/overpack and carries it further into the borehole to its designated placement location. The dolly lowers the waste canister/overpack onto a supporting structure and returns to the borehole entrance, ready for the next canister/overpack.

After the borehole is filled, the dolly is retrieved and moved to the next empty borehole. The shielding plug is inserted in the full borehole and the shielding door assembly moved to the next empty borehole. At this stage, the emplaced canisters/overpacks may be retrieved readily by merely reversing the emplacement operation and using the same equipment. It is assumed the canisters/overpacks are intact.

The dolly has independent drive and lift motors (one for each drive axle and two redundant drives for the scissors lift mechanism). These motors and their control and position sensors are connected to the transporter through multiple contacts between the V-rails. In case of a break in the system, the dolly has a complete backup system consisting of batteries for power and radio signals for control. If the dolly still malfunctions, another dolly equipped with an electromagnet could be inserted into the borehole.

Between the emplaced waste canisters/overpacks and their borehole is approximately 152 mm (6 in.) of radial clearance, available for future pneumatic backfilling through a 76 mm (3 in.) pipe.

Figure 2-17 shows the specific material flow steps in the placement area. Equipment illustrations for the placement steps are shown in Figs. 2-18 through 2-20.

2.1.8.2 Retrieval. The option for retrieval of emplaced waste is currently required by NRC regulation (10CFR60). Retrieval must remain a valid option until such time as the NRC is satisfied as to the likely success of the isolation process. This is estimated to be approximately 50 years with the reservation that the NRC retains the option of changing the interval at their discretion.

Immediate retrieval of the waste following emplacement is the simplest task, as mentioned previously. The same equipment as that used for emplacement can be used for removal and transport to surface facilities. The only complication is in the multiple canister storage in a single borehole. Several or

all canisters in an individual borehole may have to be removed to obtain the desired canister.

Long-term retrieval poses a much more difficult problem. Potential complications such as backfilled boreholes, bulkhead rooms, canister degradation from corrosion, and possible borehole collapse make the analysis of later retrieval considerably more complex than a reversal of the emplacement process.

The material flow diagram for retrieval (see Fig. 2-21) considers several options for (1) time following emplacement, (2) backfilling and sealing (bulkheading) of full rooms, and (3) possible degradation of the canister/storage environment. The most complicated series of steps associated with a degraded canister in a collapsed borehole is described, along with references to steps that can be del··ed for intact canisters, retrieval prior to backfilling, etc.

The most restrictive conditions for canister retrieval would occur if a significant period of time (e.g., 20-30 years) had elapsed since emplacement, and the desired canister(s) resided in boreholes located in a set of rooms that had been full for a significant period of time. For these conditions, backfilling of the boreholes with a mixture of crushed bentonite and basalt would probably have been performed, followed by sealing of the room with a bulkhead panel.

The first step in the retrieval process is to gain access to the borehole of interest. This involves removal of the bulkhead assembly and location of the specific borehole containing the desired canister(s). If mine heaves, roof cave-in, or other underground movement phenomena have occurred, some remining of the emplacement rooms may be required. In addition, when the room is reopened to the remainder of the repository subterranean environment, an active ventilation system must be reestablished to remove possible airborne contamination, dust, harmful vapors, etc., and possibly to cool the area to a temperature acceptable for transporter activity. This temperature limit is currently estimated as <134°F based on allowing a transporter operator to leave the protection of the air-conditioned transporter cab and walk out of the area if the transporter should fail.

Once area access by transporter becomes possible, the borehole plug retainer ring must be removed and the status of the borehole determined. Preliminary radiation/contamination measurements must be made to determine working background level for transporter crews. If the borehole has been backfilled, an overcore or hole following debris removal machine will be required to tunnel a passage to and around each canister in the borehole. The assumption is made here that, if a borehole is opened, all canisters will be removed to the surface for decontamination and inspection prior to reemplacement. The debris removal machine must be capable, therefore, of coring out to 61 m (200 ft) from the machine bed (drive source) to reach all emplaced canisters in a basalt borehole. Further, the coring operation has to cut through basalt and remove and store contaminated mined material. Finally, if the borehole has collapsed, the coring operation will involve densely packed material (vs. a relatively "loose" backfill mixture) and an increased probability of a punc-

tured canister. Such an overcore machine of the required capacity, remotely operated with shielded mined material storage does not currently exist and will require development for repository operation. Influence of this activity on the scenario selection is significant; failure to free a leaking canister with subsequent blockage of the borehole by broken coring parts creates a direct source of radioactivity that would require permanent sealing of the room with no possibility of retrieval of other canisters in the area.

Given that the coring operation is either successful or not required (no backfill or collapse of borehole), the shield collar is then installed on the mouth of the borehole, a transfer cask containing a grapple is mated with and sealed to the shield collar, and the grapple extended into the borehole to electromagnetically connect to the canister. The free canister is withdrawn into the transfer cask, the cask and borehole shield covers closed, and the cask rotated to the travel position and manually secured with the safety shear pins. At this point, the question of canister integrity is dominant in estimating occupational exposure risk. Manual activities such as shear pin installation performed by transporter crews outside the shielded cab areas may have to be curtailed in the event of a leaking canister due to high background radiation levels.

The transporter is driven from the borehole location to the waste cage shaft where the cask is rotated to the upright position, removed from the transporter by the underground crane, and placed in the waste cage/hoist assembly. Transport to the surface requires approximately 16 minutes.

Upon arrival at the surface of the shaft, the surface crane unloads the transfer cask from the waste cage and moves through the holding area to position the cask under the hot cell loading port. The hot cell and cask shielding covers are removed and the hot cell crane removes the canister for inspection, decontamination, overpack, etc. The empty transfer cask is returned to the underground level, loaded on the transporter and moved to the borehole site for the next canister. The process is repeated until the borehole is empty (see Fig. 2-21).

TABLE 2-1
LIST OF KEY AREAS IN WASTE PROCESS

---

1) ARRIVAL AREA
   - MONITORING STATION
   - SUSPECT RAIL AND TRUCK AREA

2) YARD STORAGE AREA
   - INSPECTION PIT

3) WASHDOWN AND RECEIVING AREA

4) UNLOADING AREA
   - CASK PREPARATION AND SHIPPING AREA
   - SMEAR TEST STATION
   - LABORATORY AREA

5) SECONDARY AREA
   - LAG STORAGE AREA
   - PROCESS TANK AREA

6) HOLDING AREA

7) WASTE CAGE AREA AND SHAFT

8) PLACEMENT AREA

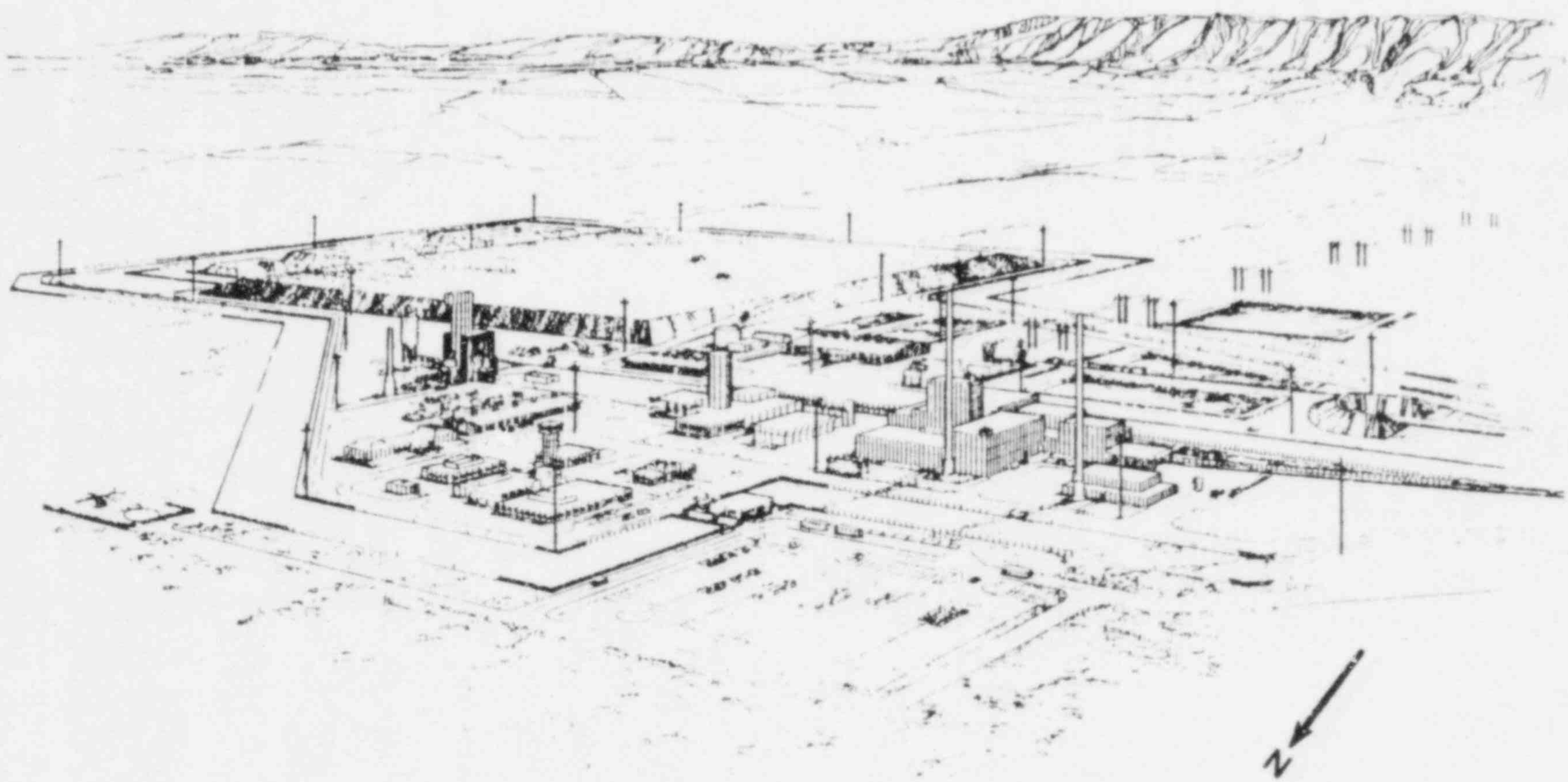


Fig. 2-1. Surface Facilities of the NWRB.

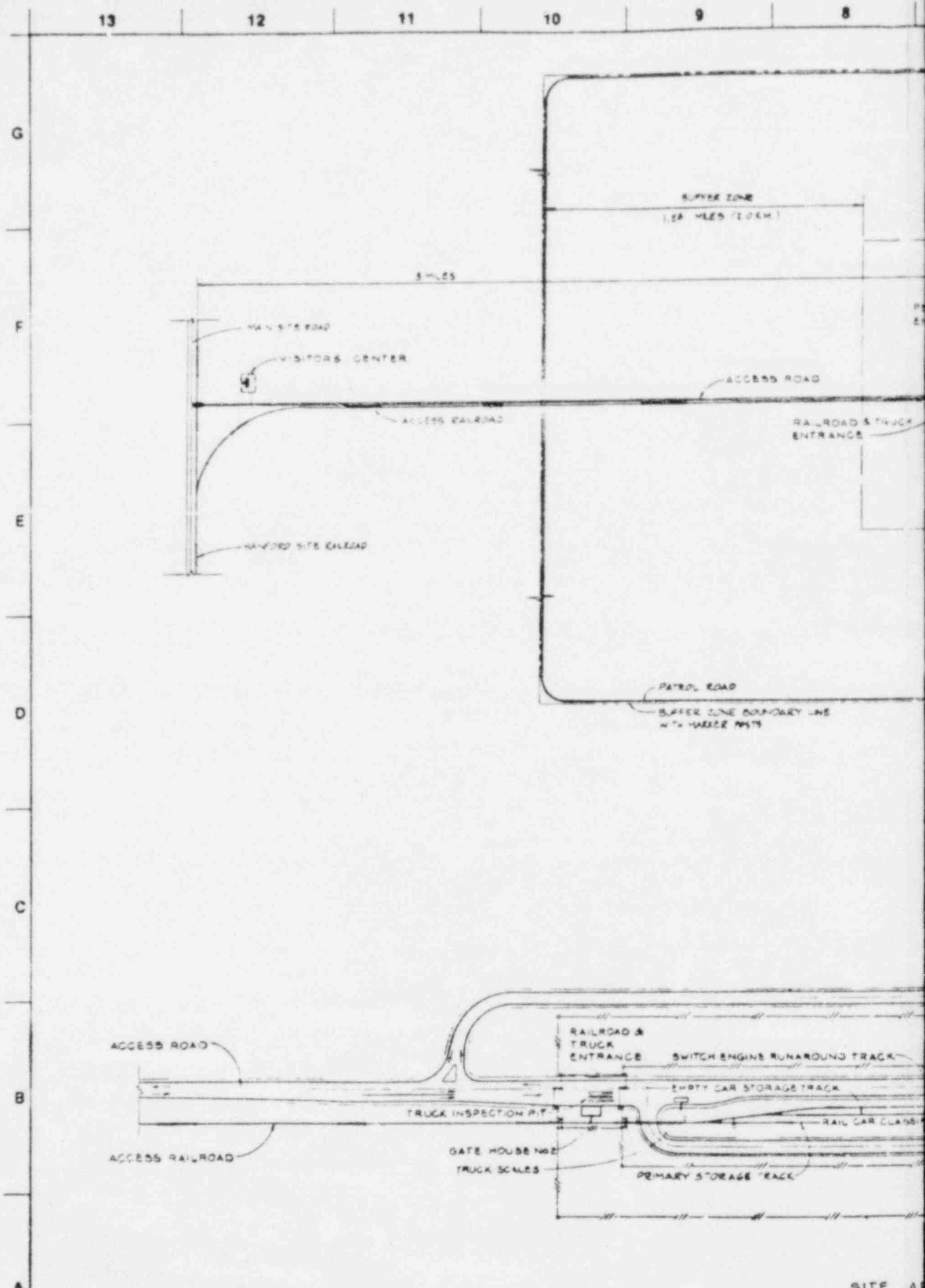
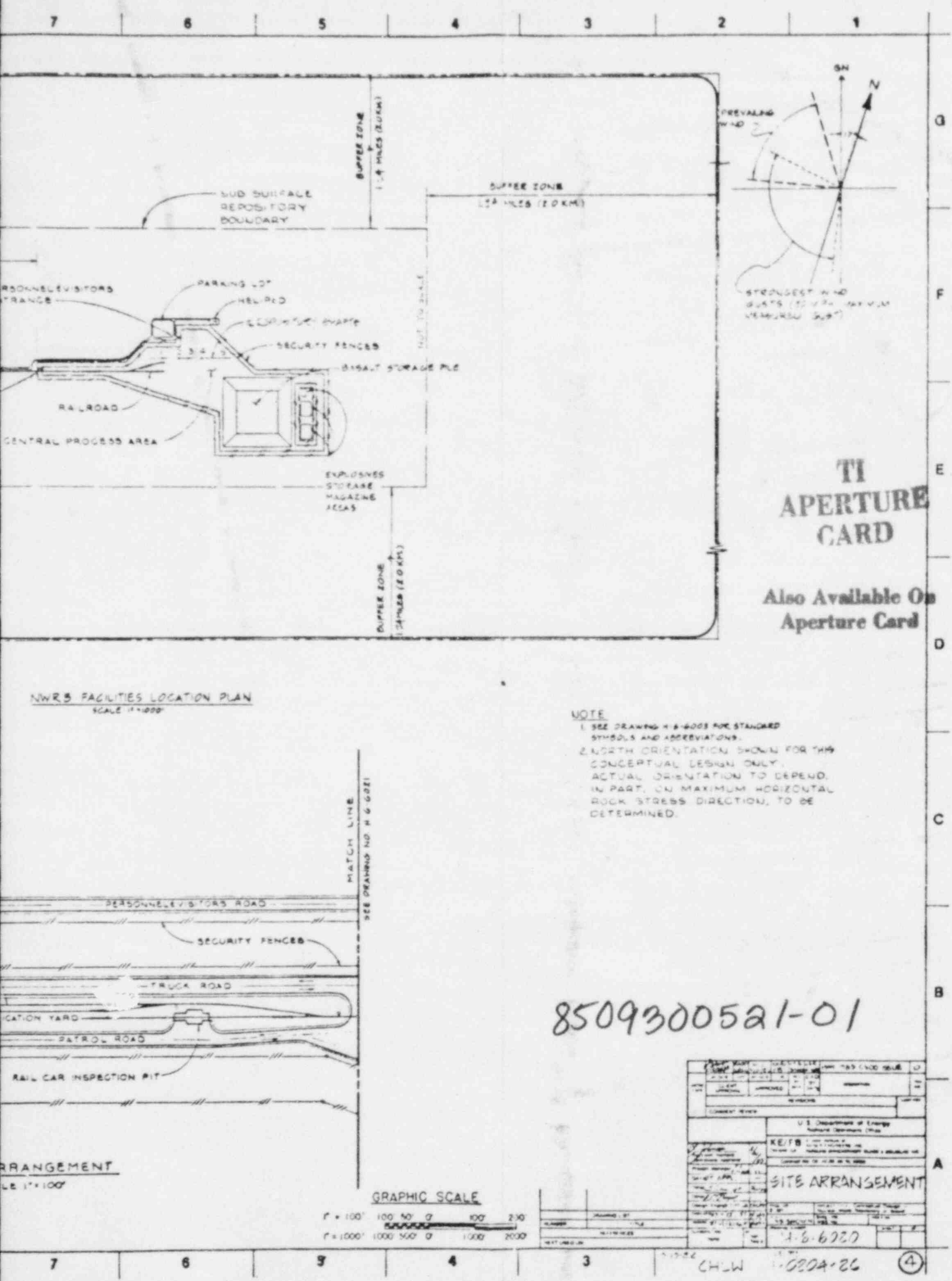


Fig. 2-2. Surface facilities site arrangement - View A.



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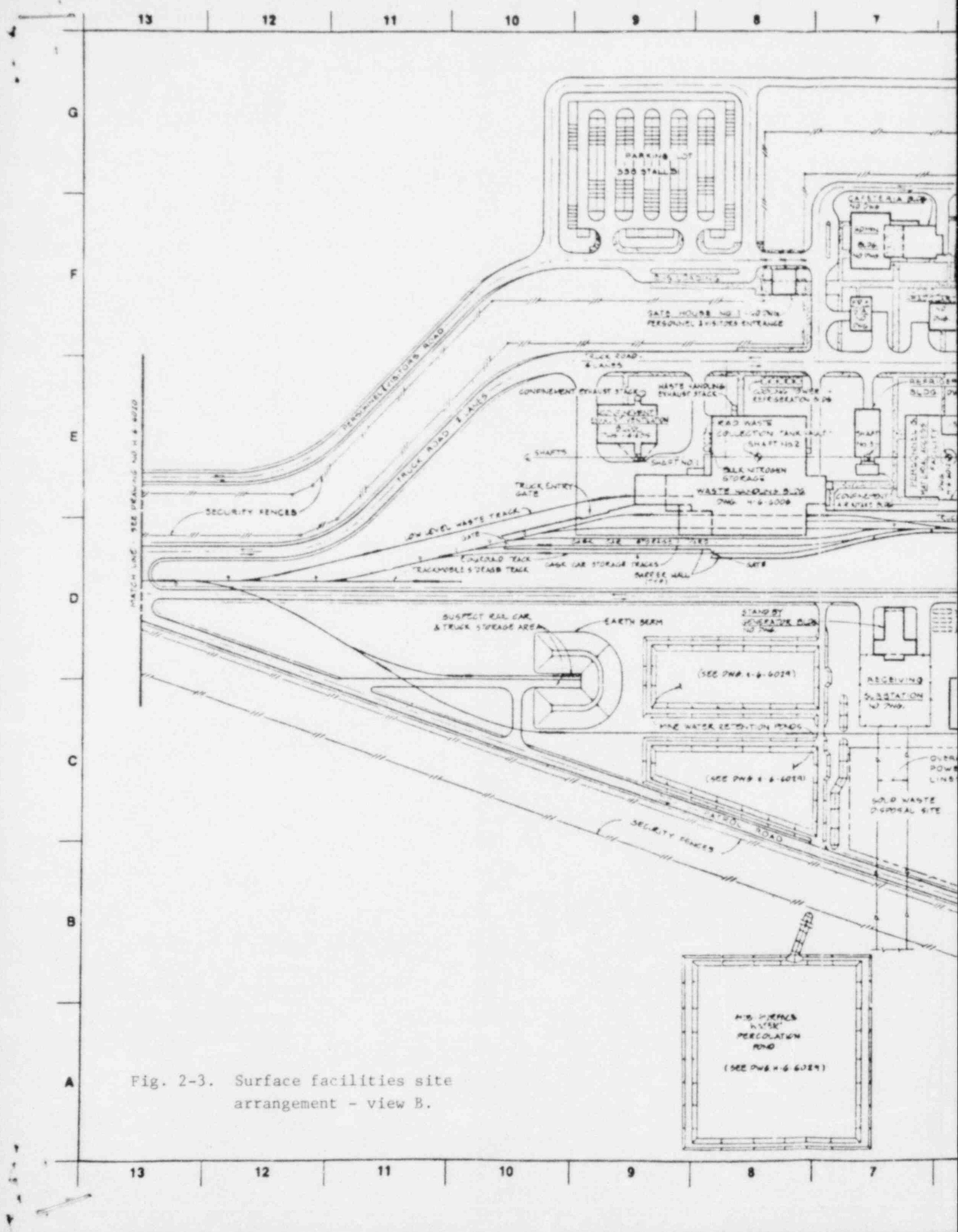
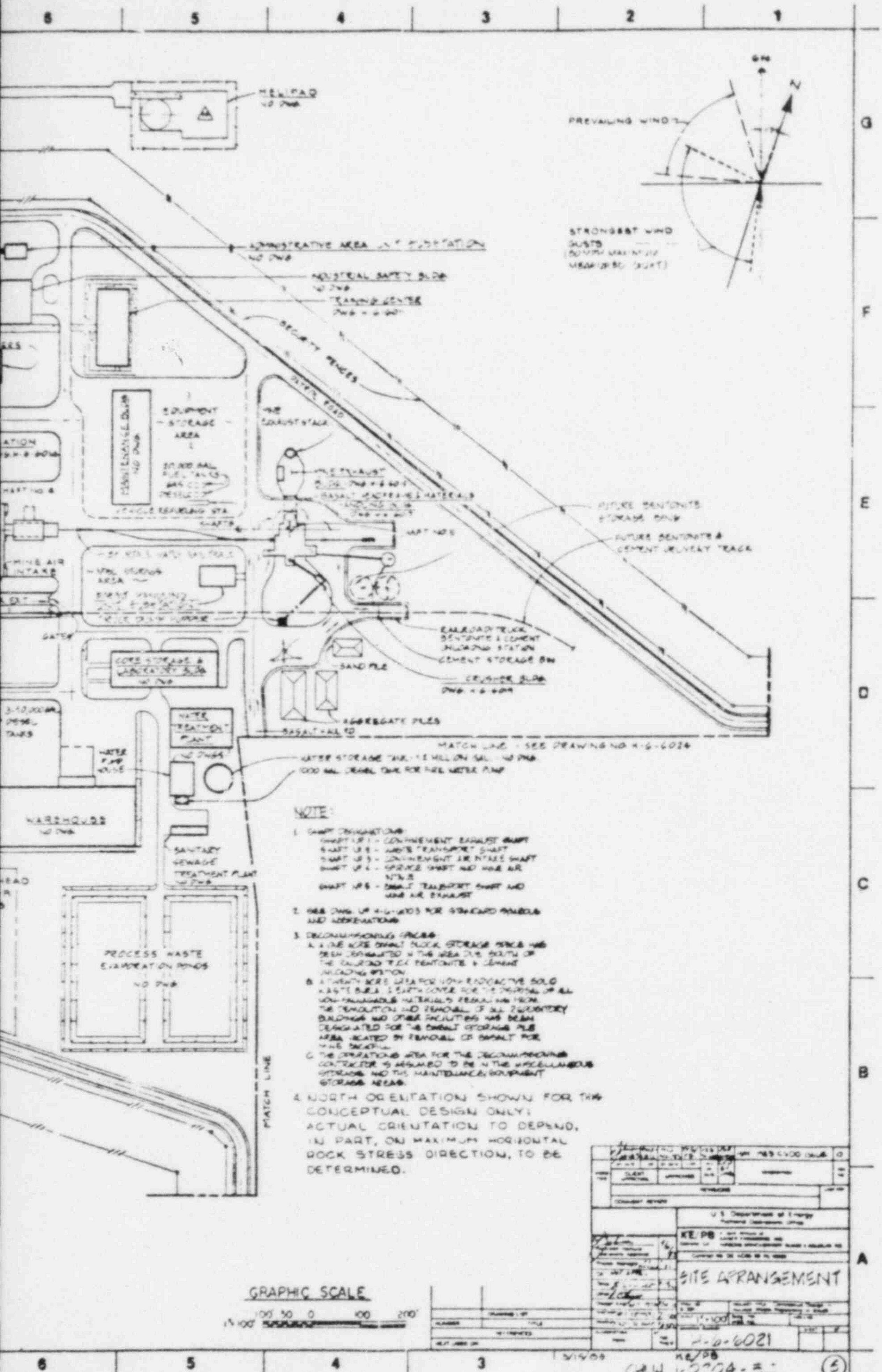


Fig. 2-3. Surface facilities site arrangement - view B.



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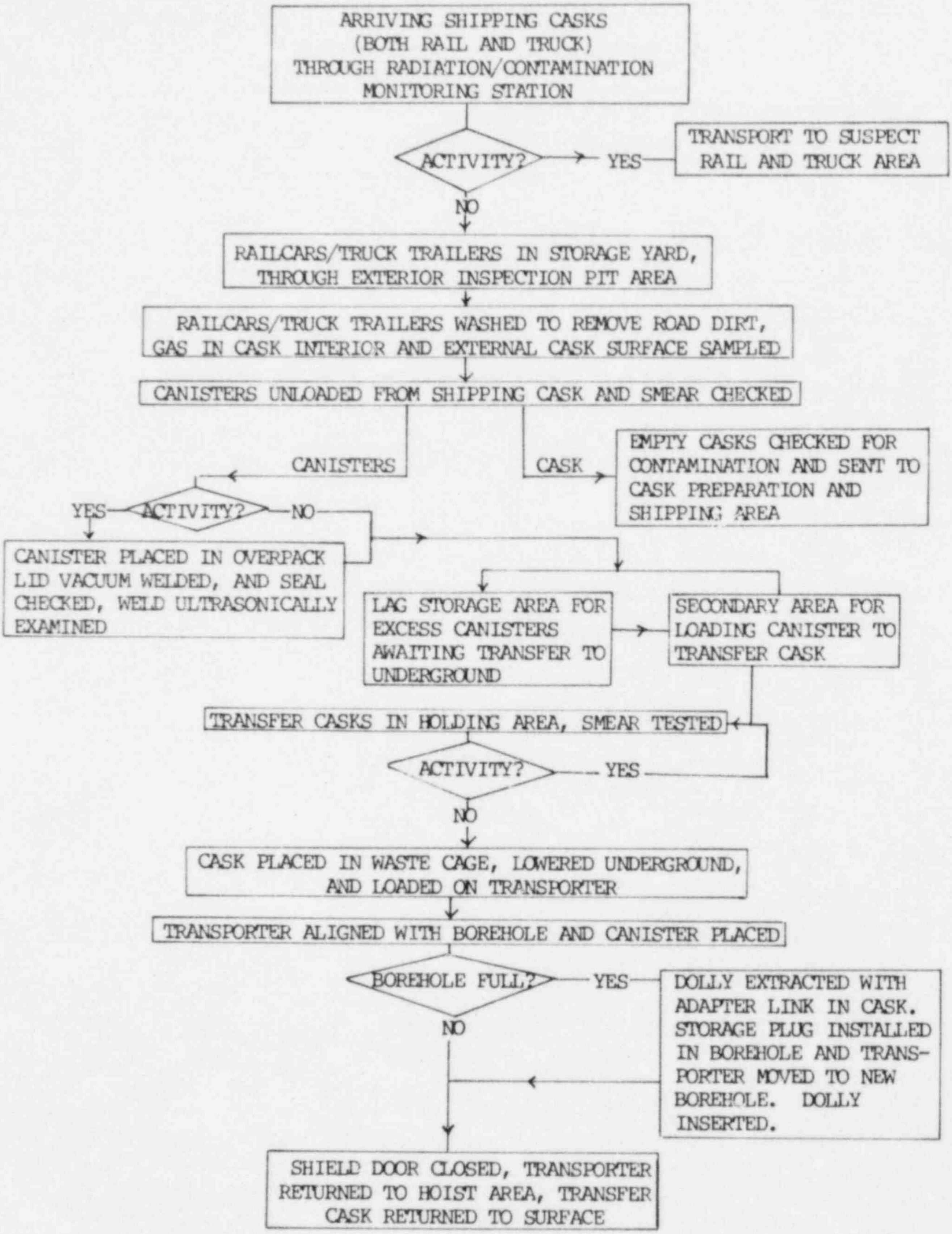


Fig. 2-4 Functional flow diagram of major processing steps.

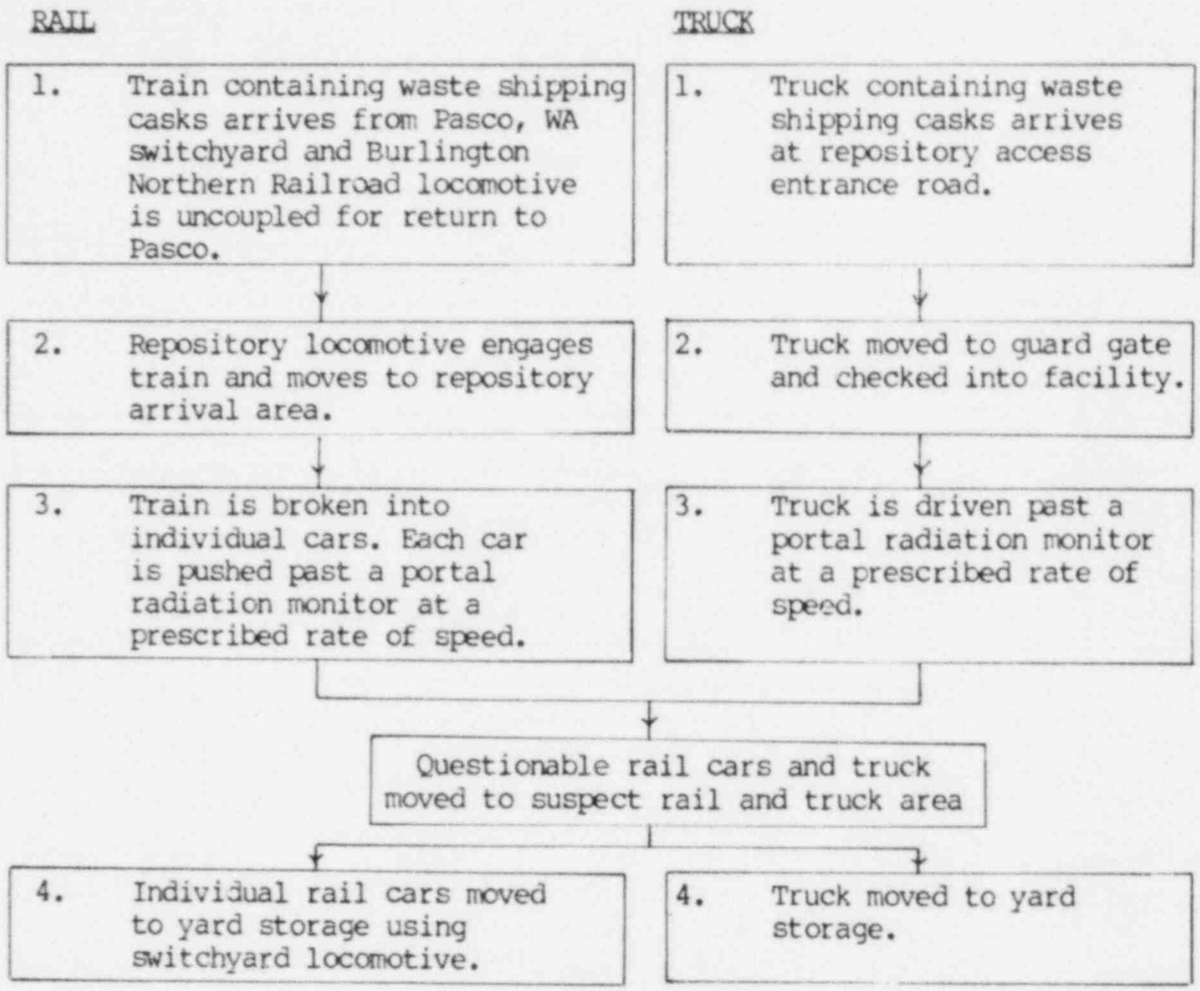


Fig. 2-5 Material flow diagram for arrival area.

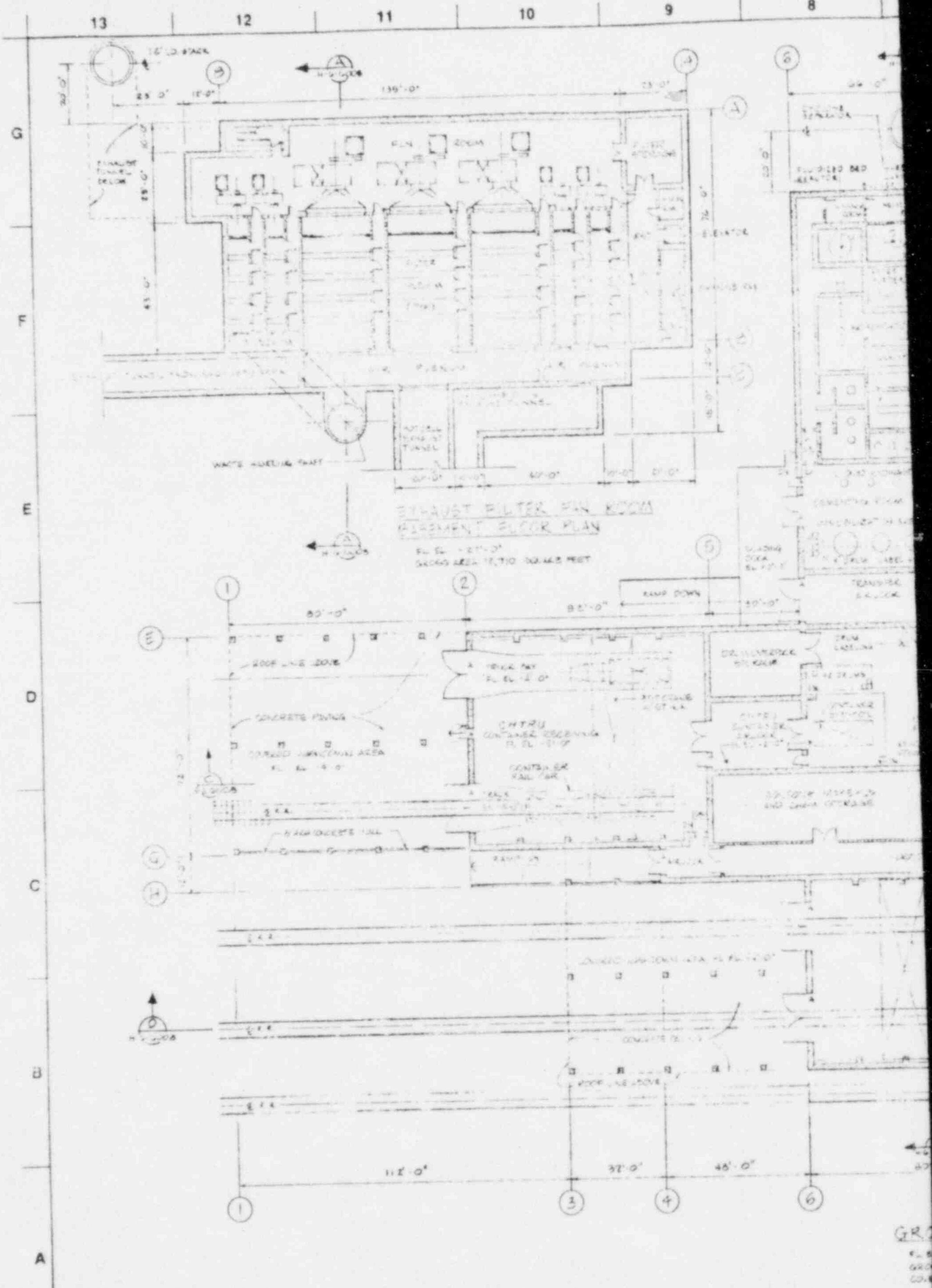
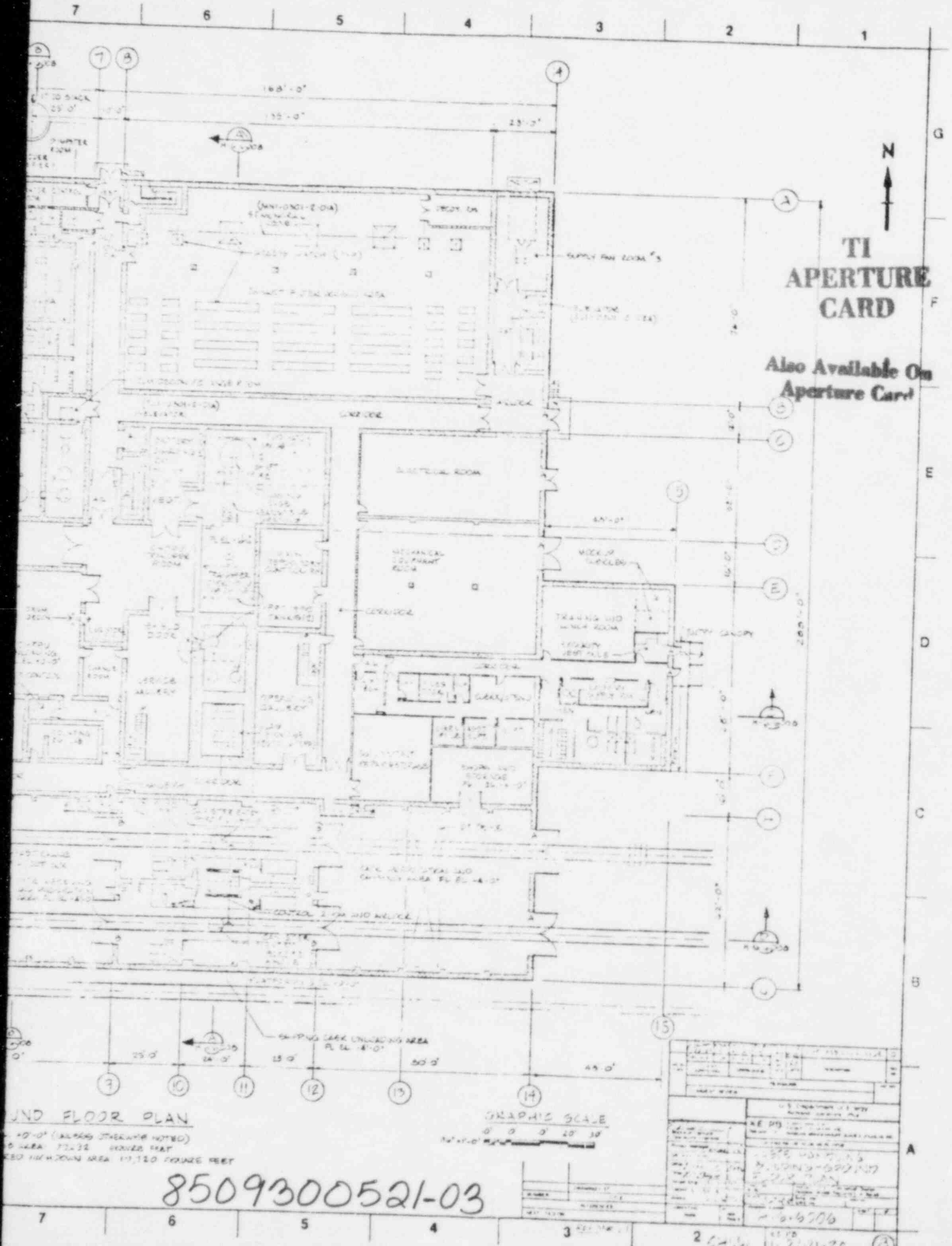


Fig. 2-6. Waste handling building, ground floor plans.



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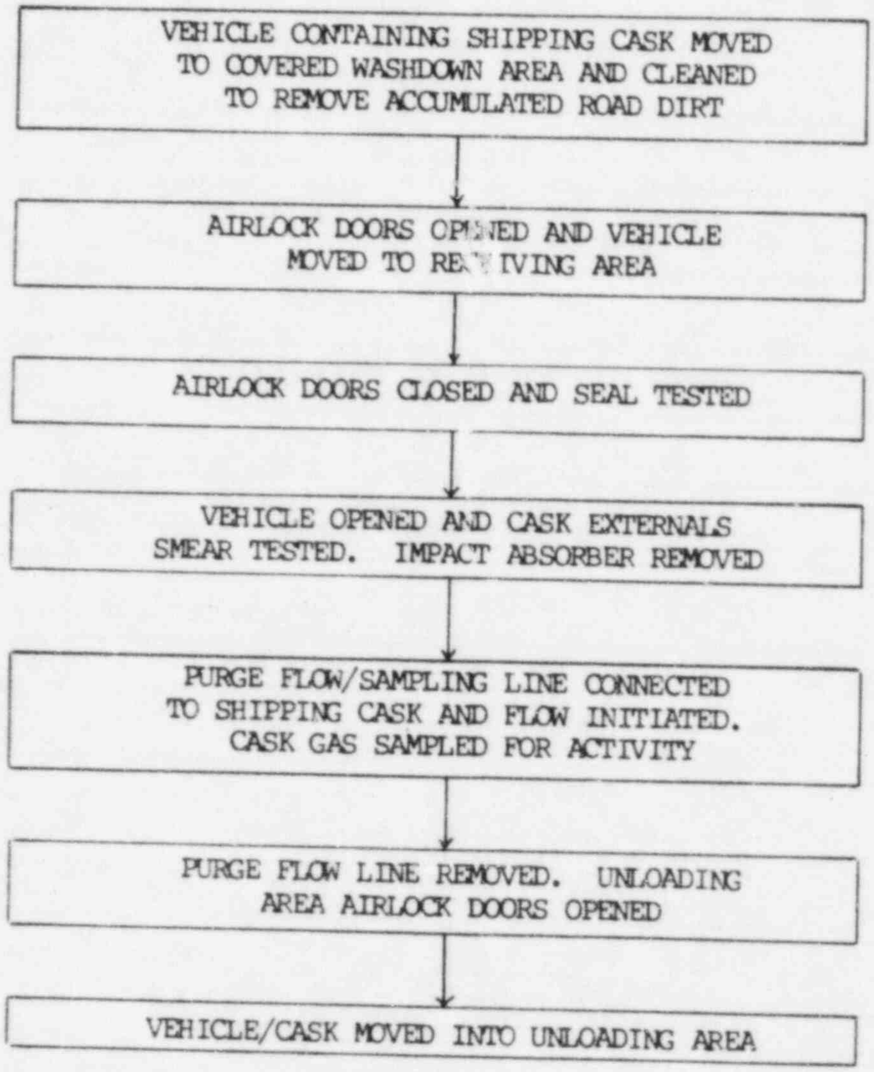


Fig. 2-7 Material flow diagram, washdown and receiving area.

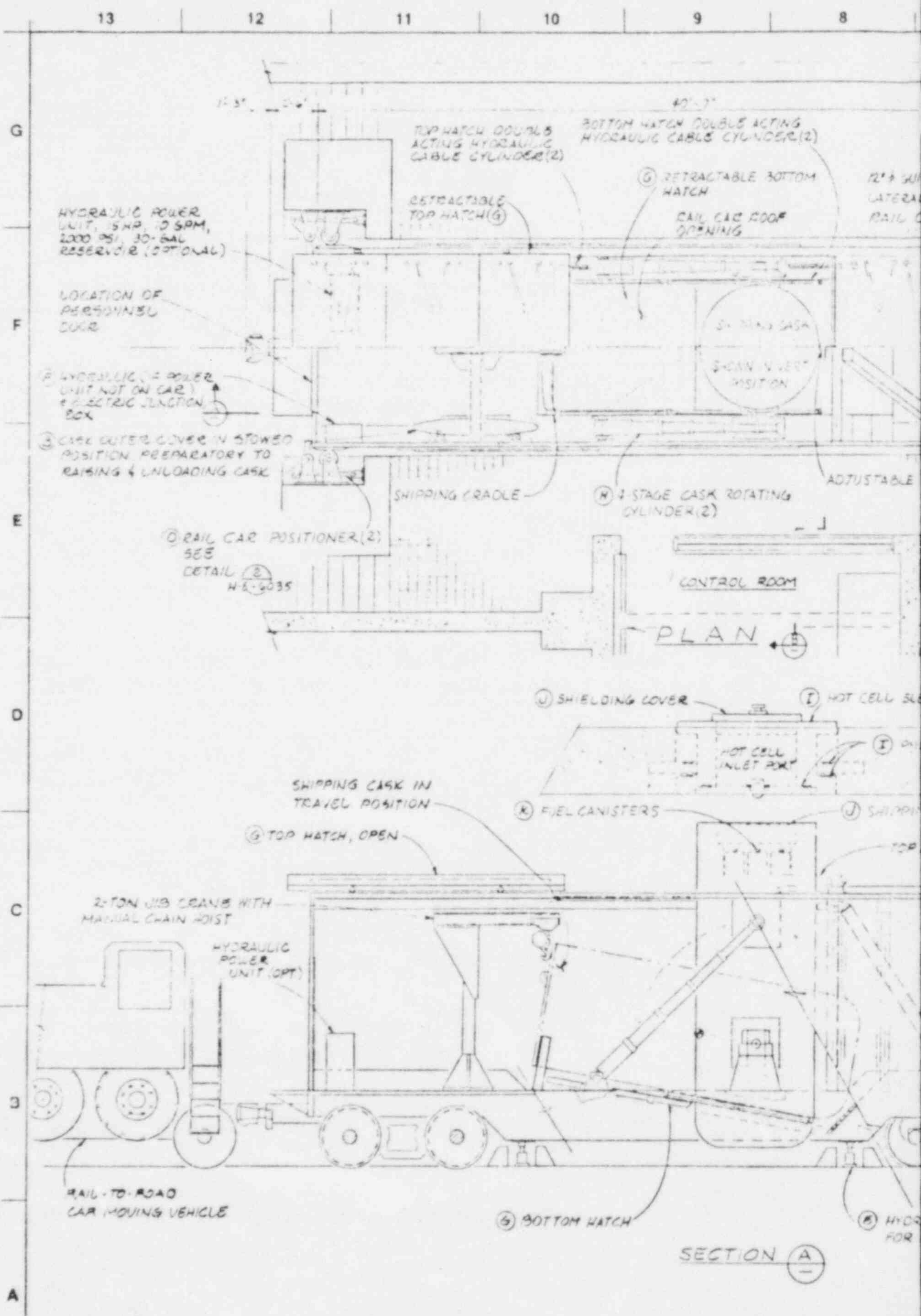
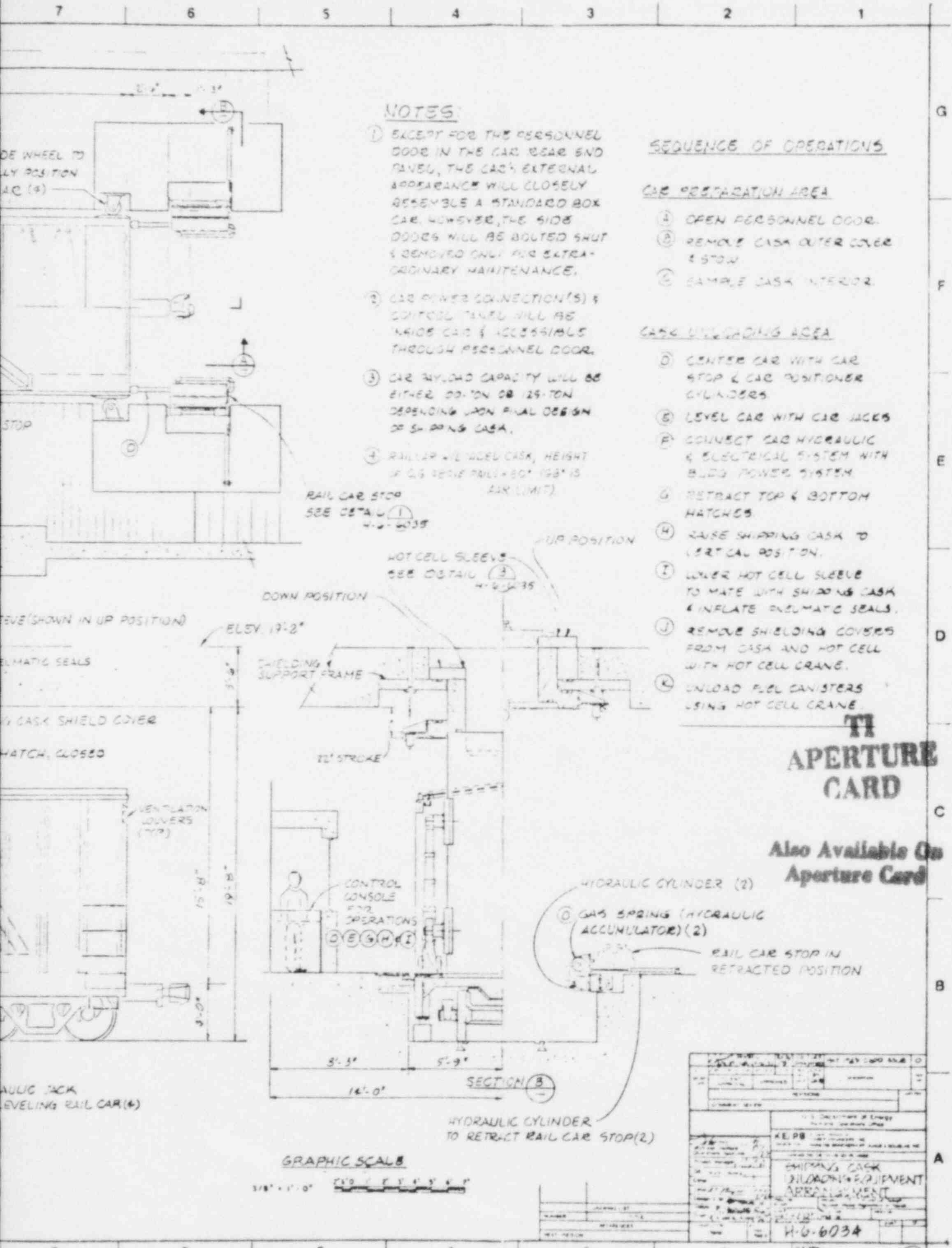


Fig. 2-8. Shipping cask unloading area equipment.



NOTES:

1. EXCEPT FOR THE PERSONNEL DOOR IN THE CAR REAR END PANEL, THE CAR'S EXTERNAL APPEARANCE WILL CLOSELY RESEMBLE A STANDARD BOX CAR. HOWEVER, THE SIDE DOORS WILL BE BOLTED SHUT & REMOVED ONLY FOR EXTRAORDINARY MAINTENANCE.
2. CAR POWER CONNECTION(S) & CONTROL PANEL WILL BE INSIDE CAR & ACCESSIBLE THROUGH PERSONNEL DOOR.
3. CAR PAYLOAD CAPACITY WILL BE EITHER 00-TON OR 125-TON DEPENDING UPON FINAL DESIGN OF SHIPPING CASK.
4. RAILCAR WITH LOADED CASK, HEIGHT OF C.G. ABOVE RAIL = 80" (98" IS AAR LIMIT)

SEQUENCE OF OPERATIONS

CAR PREPARATION AREA

A. OPEN PERSONNEL DOOR.
B. REMOVE CASK OUTER COVER & STOW.
C. SAMPLE CASK INTERIOR.

CASK UNLOADING AREA

D. CENTER CAR WITH CAR STOP & CAR POSITIONER CYLINDERS.
E. LEVEL CAR WITH CAR JACKS.
F. CONNECT CAR HYDRAULIC & ELECTRICAL SYSTEM WITH BLDG. POWER SYSTEM.
G. RETRACT TOP & BOTTOM HATCHES.
H. RAISE SHIPPING CASK TO VERTICAL POSITION.
I. LOWER HOT CELL SLEEVE TO MATE WITH SHIPPING CASK & INFLATE PNEUMATIC SEALS.
J. REMOVE SHIELDING COVERS FROM CASK AND HOT CELL WITH HOT CELL CRANE.
K. UNLOAD FUEL CANISTERS USING HOT CELL CRANE.

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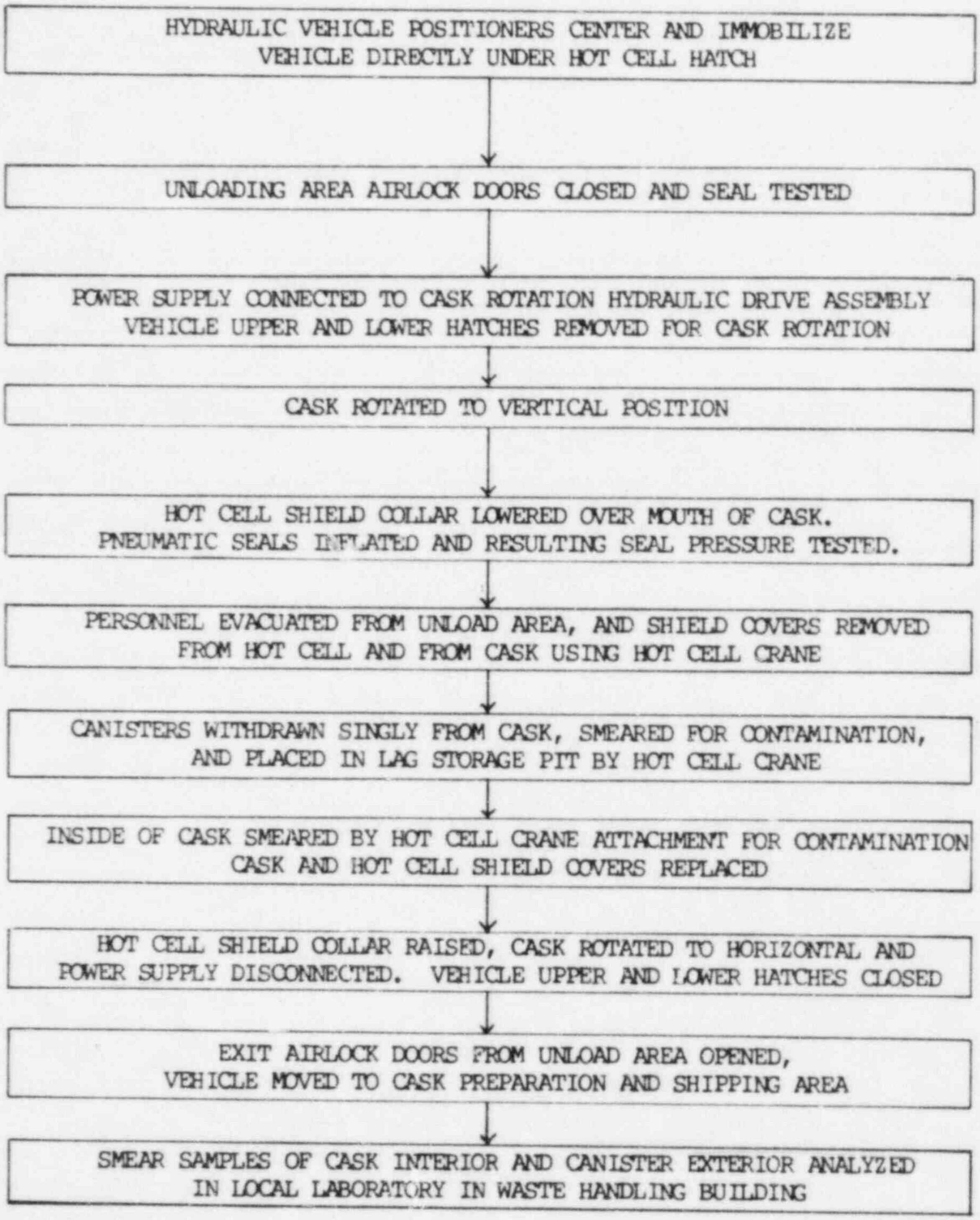


Fig. 2-9 Material flow diagram for unloading area.

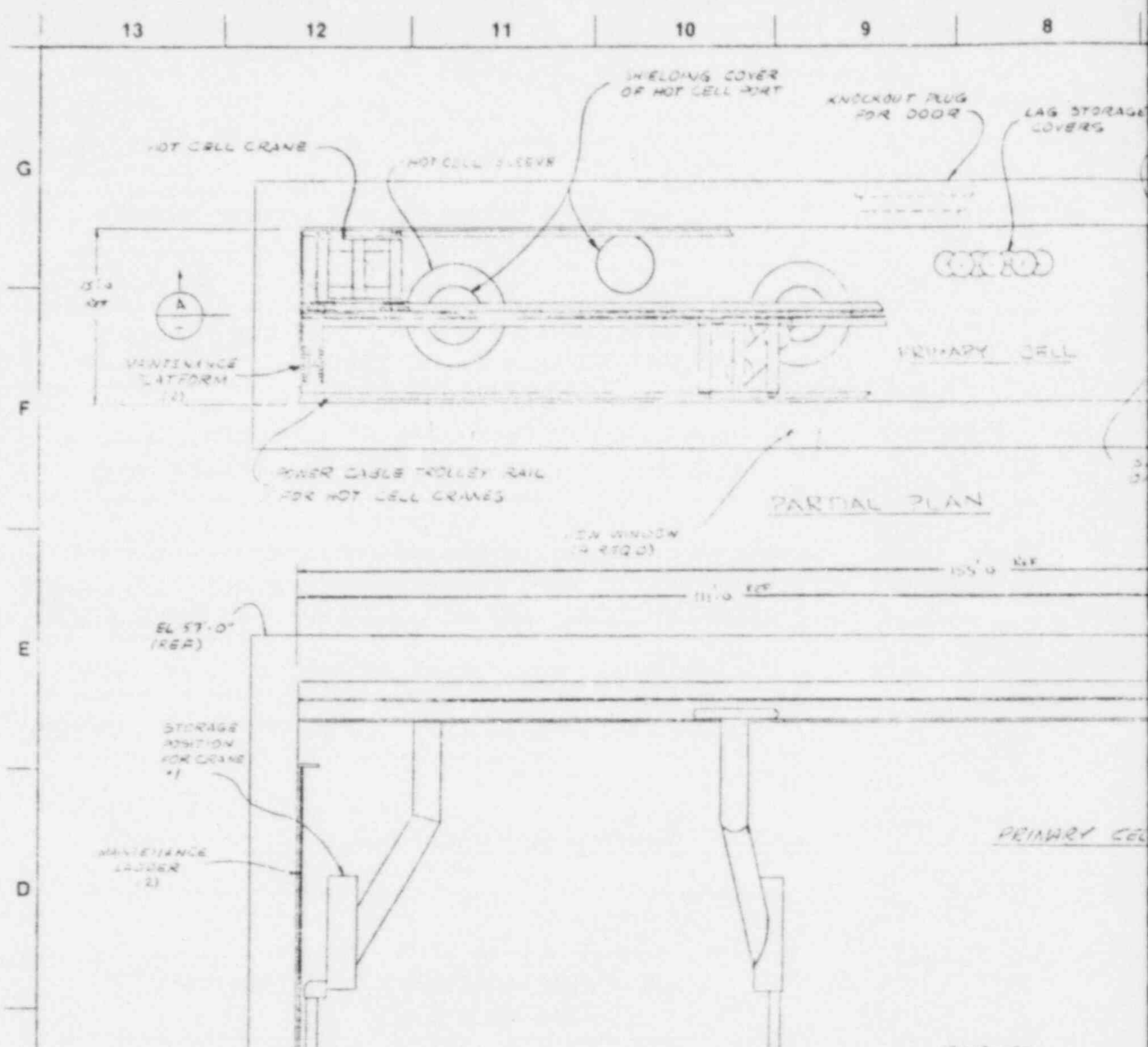
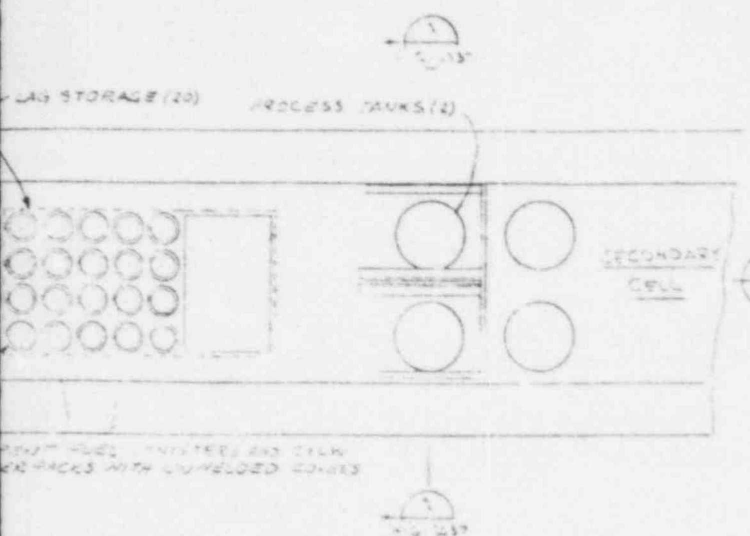


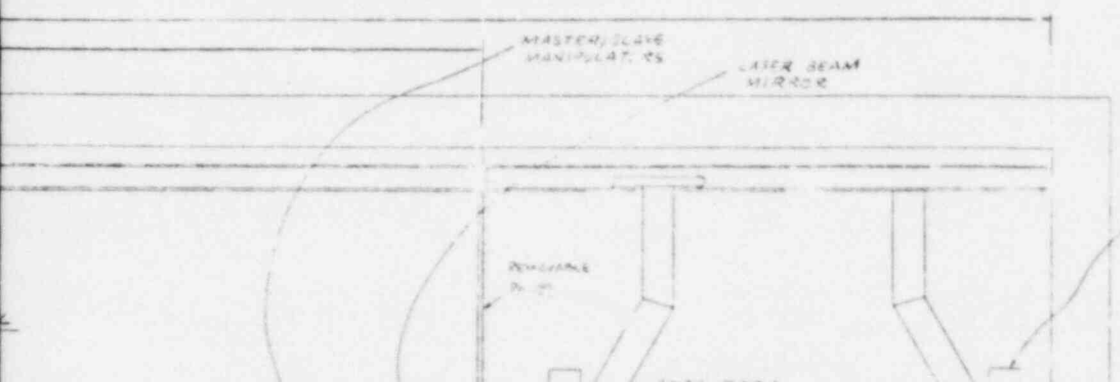
Fig. 2-10. Primary hot cells, secondary area, and lag storage area details.

7	6	5	4	3	2	1

SEQUENCE OF OPERATIONS

A. AFTER SEAL TEST REMOVE SHIELDING COVERS OF HOT CELL PORT & SHIPPING CASK USING HOT CELL CRANE
B. TRANSFER SPENT FUEL AND CHLW CANISTERS FROM THE SHIPPING CASK TO THE SMEAR TEST STATION FOR EVALUATION, AND THEN TO THE LAG STORAGE AREA. THE CHLW CANISTER WILL BE LOWERED INTO THE CHLW OVERPACK CYLINDER AND WILL THEN BE COVERED WITH THE CHLW OVERPACK COVER.
C. TRANSFER UNWELDED OVERPACK TO PROCESS TANK, USING CRANE.
D. POSITION DRIVE MECHANISM IN PROCESS TANK TO CENTER UNWELDED OVERPACK.
E. CLOSE COVER OF PROCESS TANK PNEUMATICALLY
F. EVACUATE PROCESS TANK AND LASER WELD.
G. HELIUM LEAK TEST WELDED ASSEMBLY IN PROCESS TANK (SEE NOTE 4).
H. ULTRASONICALLY TEST WELDED ASSEMBLY IN PROCESS TANK (SEE NOTE 3).
I. LEAK TEST IN SECONDARY CELL PRIOR LOADING TRANSFER CASK (SEE NOTE 5)





GRAPHIC SCALE

3/16" = 1'-0" 5' 0' 5' 10'

A

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HOT CELL EQUIPMENT ARRANGEMENT

H-6-6036

7	6	5	4	3	2	1

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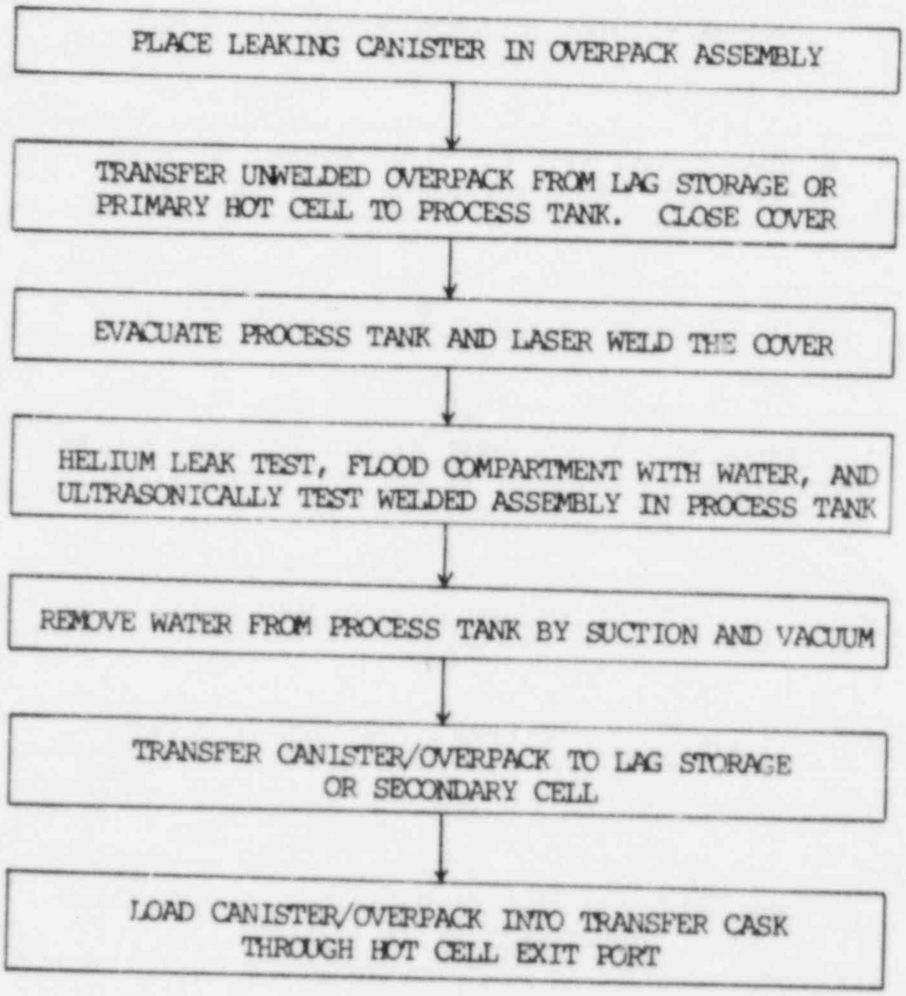


Fig. 2-11 Material flow diagram, secondary area.

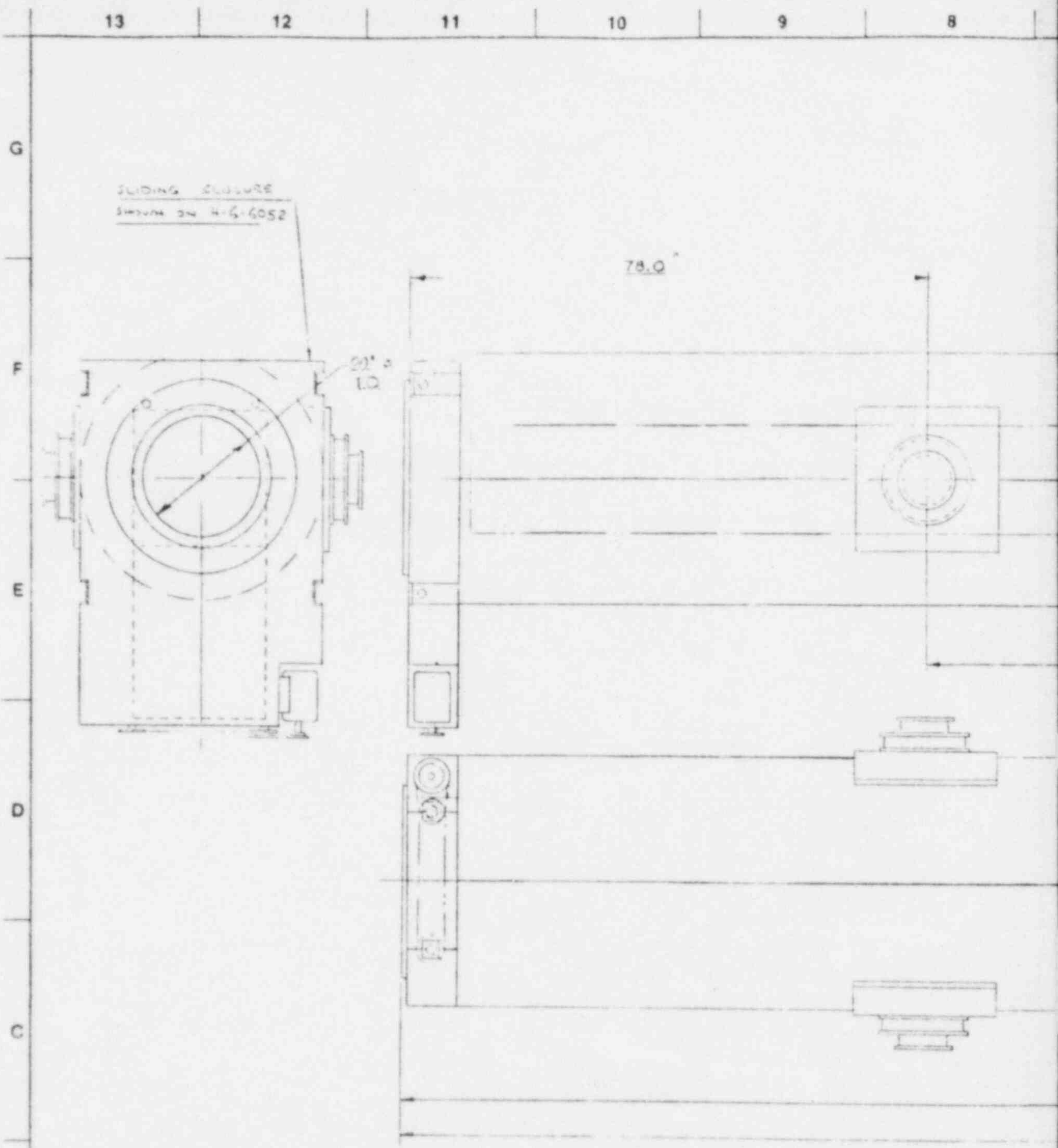


Fig. 2-12. Transfer cask.

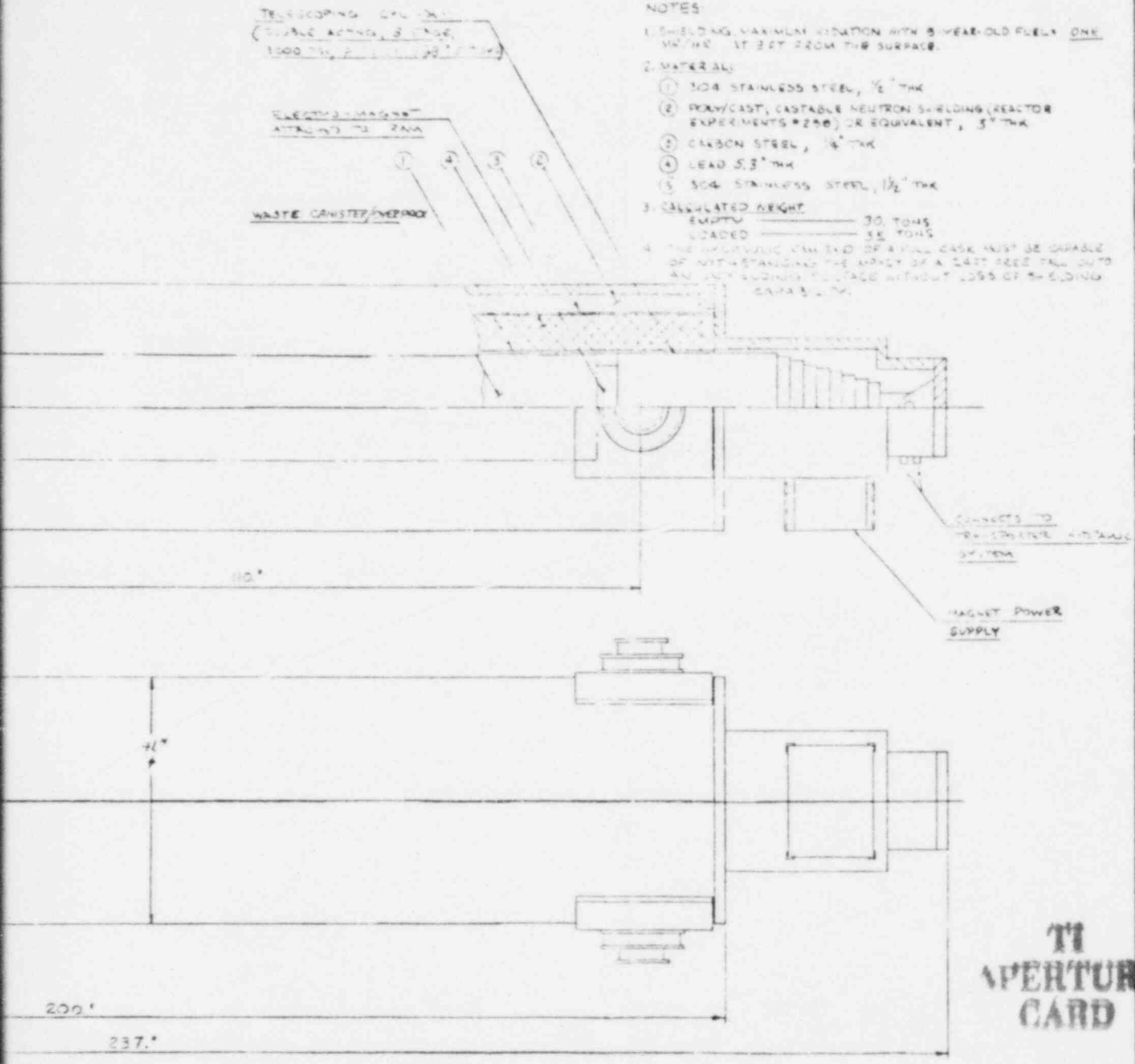
NOTES

1. SHIELDING MAXIMUM RADIATION WITH 5-YEAR-OLD FUEL: ONE MR/HR AT 3 FT FROM THE SURFACE.

2. MATERIAL:
   (1) 304 STAINLESS STEEL, 1/2" THK
   (2) POLY/CAST, CASTABLE NEUTRON SHIELDING (REACTOR EXPERIMENTS #250) OR EQUIVALENT, 3" THK
   (3) CARBON STEEL, 1/4" THK
   (4) LEAD 5.3" THK
   (5) 304 STAINLESS STEEL, 1 1/2" THK

3. CALCULATED WEIGHT
   EMPTY ———— 30. TONS
   LOADED ———— 35 TONS

4. THE HYDRAULIC RAM END OF A FULL CASK MUST BE CAPABLE OF WITHSTANDING THE IMPACT OF A 2-FT FREE FALL ONTO AN UNYIELDING SURFACE WITHOUT LOSS OF SHIELDING CAPABILITY.



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TRANSFER CASK

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2-23

CHLW

4-0105-3WH

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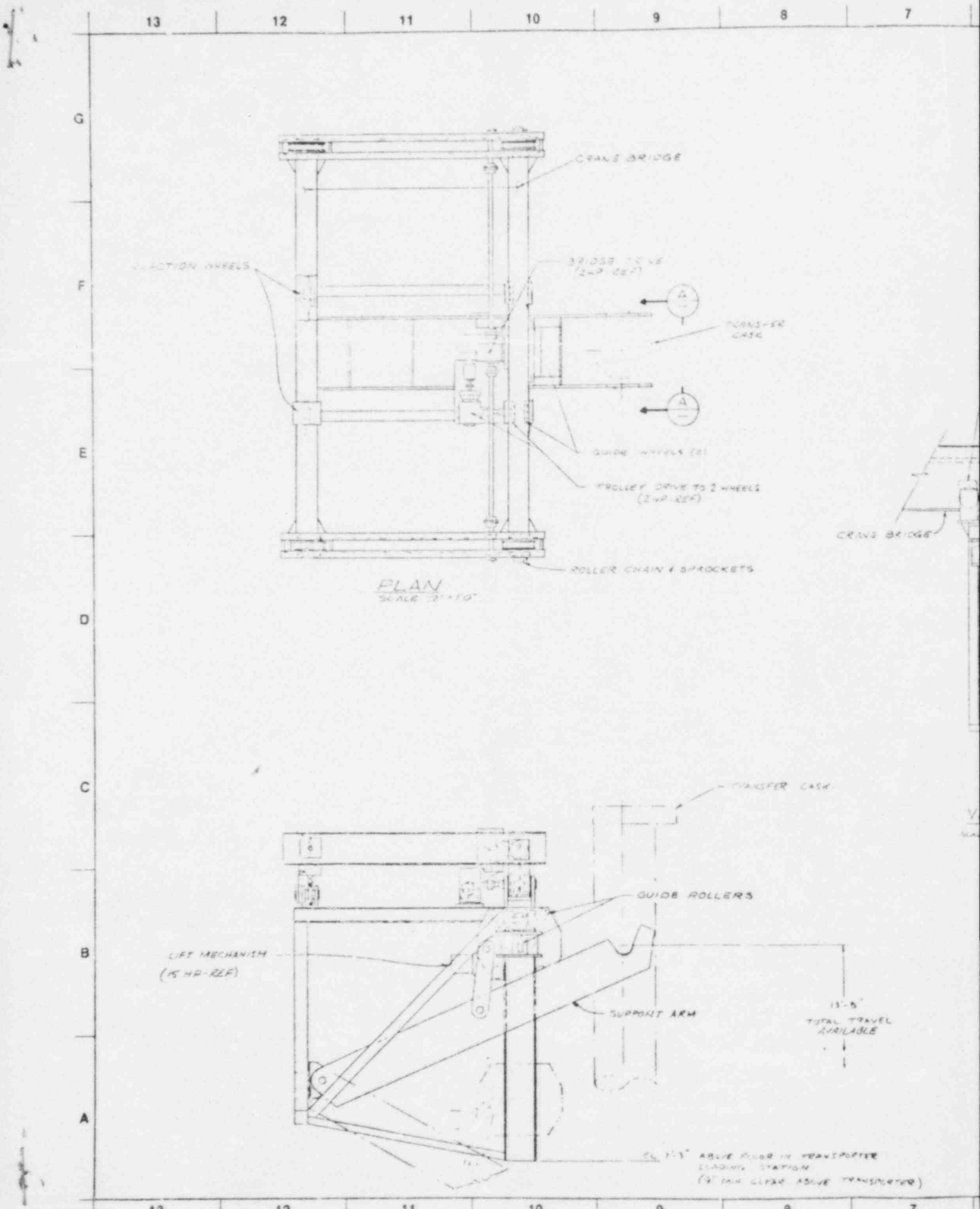


Fig. 2-13. Transfer cask crane.

NOTES:

1. DESCRIPTIONS OF TRUNNION LIFT POSITIONS
CAGE LOADING AREA, WASTE HANDLING BLDG.

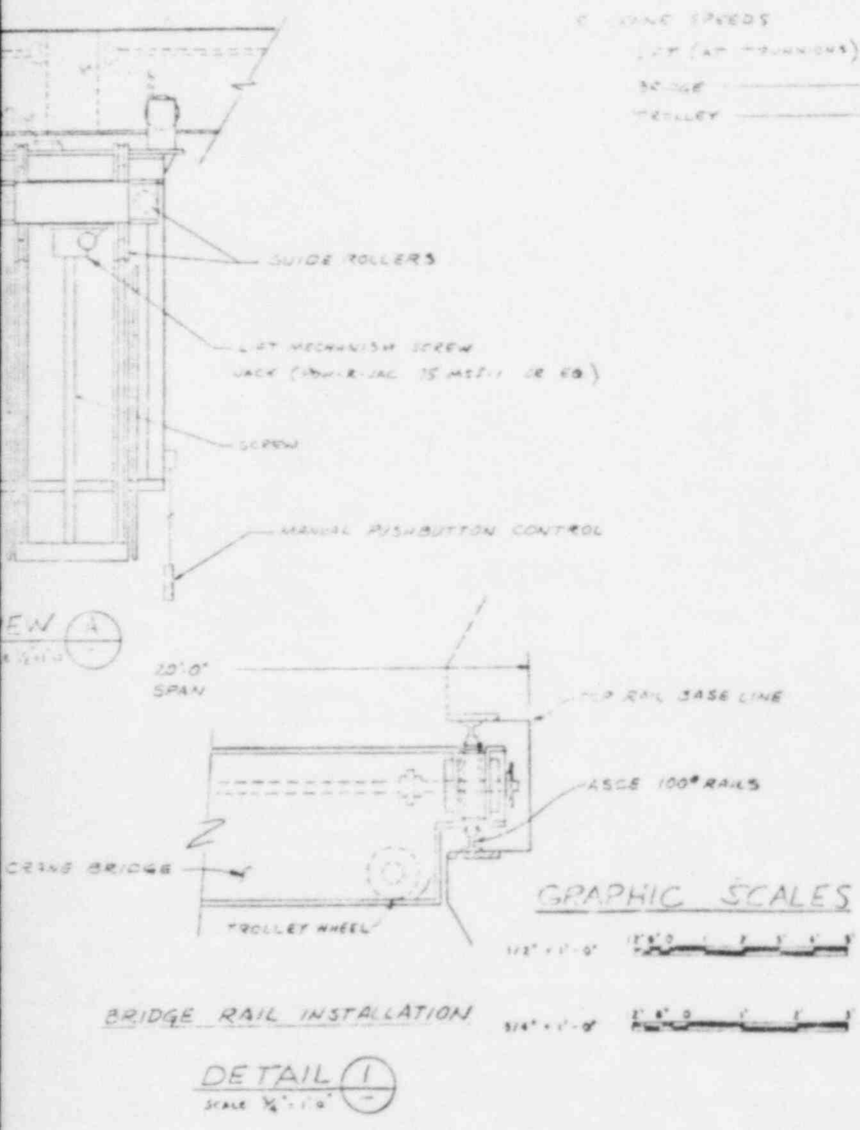
* 9'-8" CASK INTERFACE WITH HOT CELL.
* 7'-8" TRUNNION ENGAGEMENT W WASTE CAGE

TRANSPORTER LOADING STATION AT MINE LEVEL
ABOVE FLOOR

* TRUNNION ENGAGEMENT W WASTE CAGE
* 4'-9" HORIZONTAL CASK ON TRANSPORTER

2. NOMINAL CAPACITY: 35 TONS
3. TOTAL TRAVEL OF TROLLEY: APPROX 11 FEET.
4. THE TRANSFER CASK CRANES AT THE SURFACE AND AT THE MINE LEVEL ARE IDENTICAL.
5. CRANE SPEEDS

LIFT (AT TRUNNIONS)	2 FPM
BRIDGE	12 FPM
TROLLEY	15 FPM



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TRANSFER CASK CRANE

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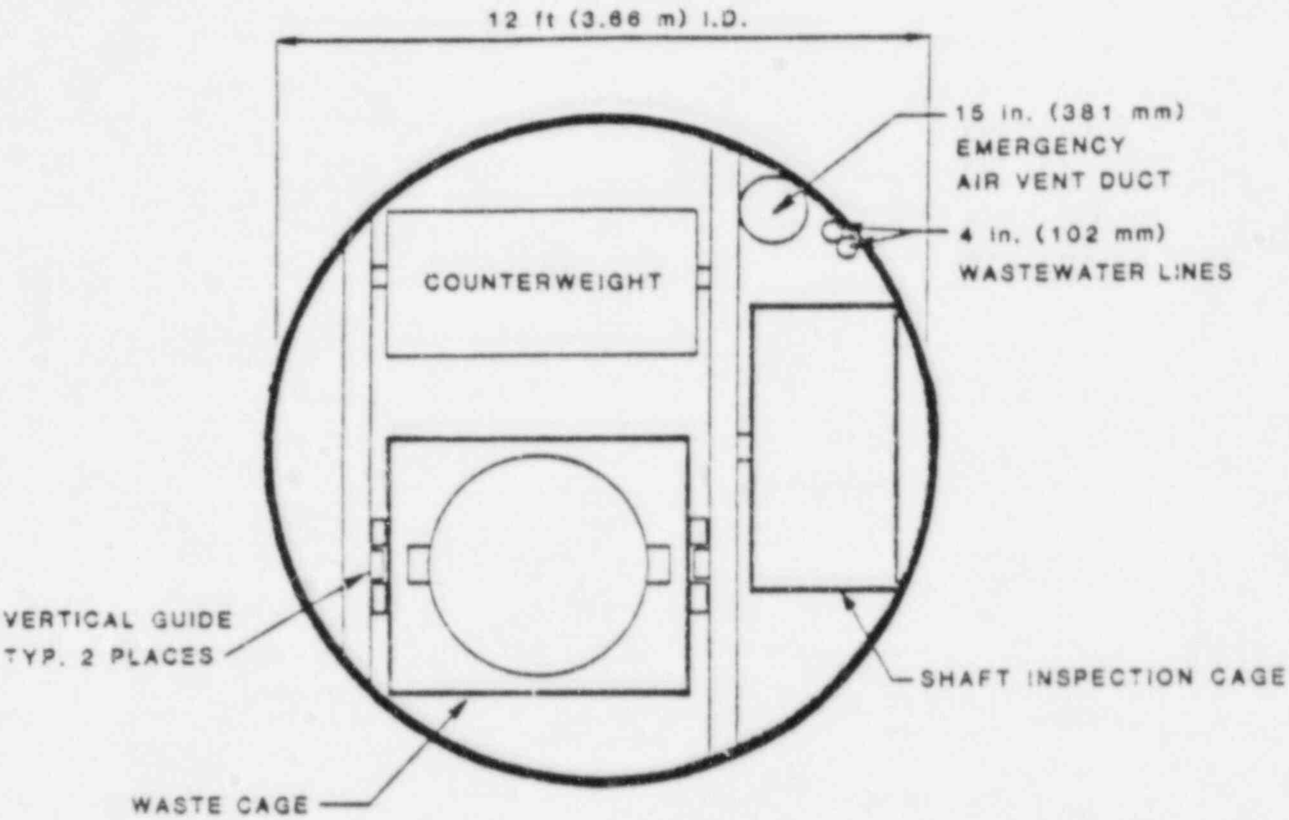


Fig. 2-14. Waste transport shaft.

NOTES:

1. BOTH CABS SHIELDED TO REDUCE THE DOSE RATE TO A MAXIMUM OF 0.5 MR/HR.
2. GVW WITH 35 TONS LOADED TRANSFER CASK ≅ 60 TONS.
3. MAX OPERATIONAL SPEED = 10 MPH
4. POWER TRAIN
   - ENGINE: 250 HP, DIESEL
   - TRANSMISSION: 4 SPEED, POWER SHIFT FULL REVERSE W/ TORQUE CONVERTER.
   - EXHAUST: CATALYTIC CONVERTER AND/OR WET SCRUBBER.
5. THE TRANSFER CASK WILL BE AUTOMATICALLY ALIGNED WITH A BOREHOLE BY MEANS OF A LASER SENSING SYSTEM AND THE FOLLOWING CASK POSITIONING DRIVES
   - VERTICAL - LEVELING JACKS
   - LONGITUDINAL } AIR BEARINGS
   - TRANSVERSE } AIR BEARINGS
   - ROTATING } AIR BEARINGS
6. TRAVEL OF POSITIONING DRIVES FOR TRANSFER CASK ARE:

LONGITUDINAL	6" ±
TRANSVERSE	6" ±
ROTATING [illegible]	90°
ELEVATING	6"

7. CRAB AND COORDINATED 4-WHEEL STEERING PROVIDED.
8. TRANSPORTER TO HAVE SINGLE AXLE DRIVE.
9. TRANSPORTER TO OPERATE FROM EITHER CAB IN EITHER DIRECTION.
10. TURNING RADII:
    INSIDE 37'-0"
    OUTSIDE 48'-0"

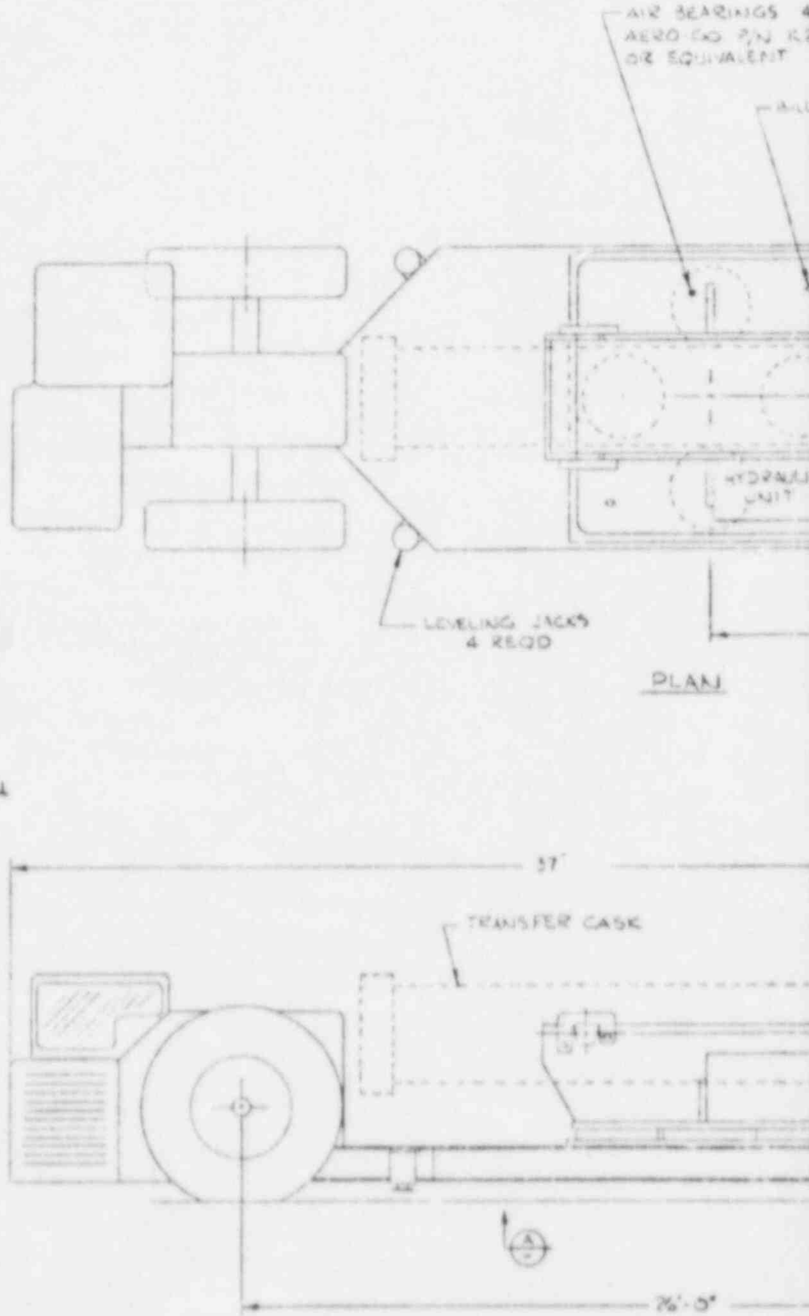
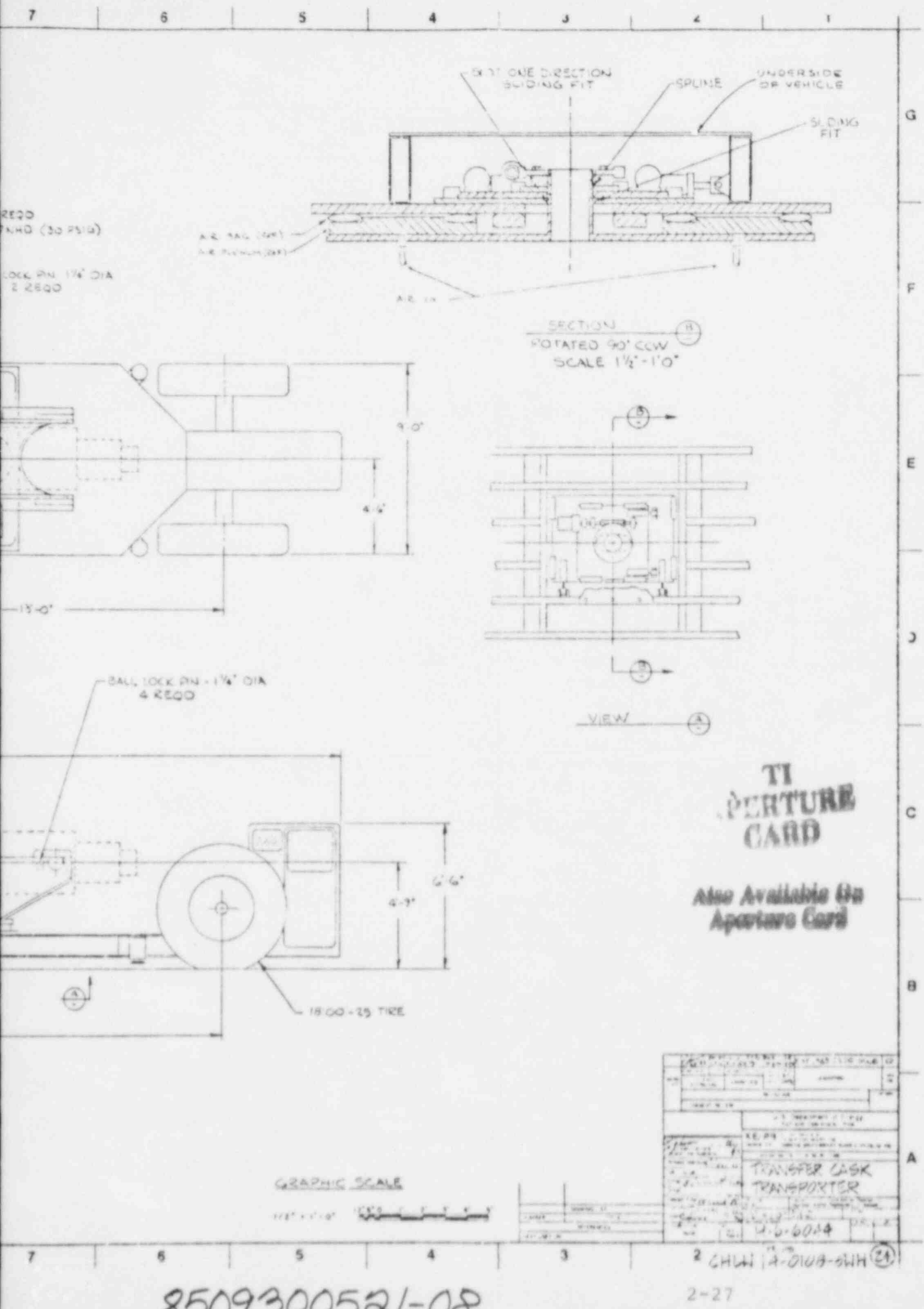


Fig. 2-15. Transfer cask transporter.



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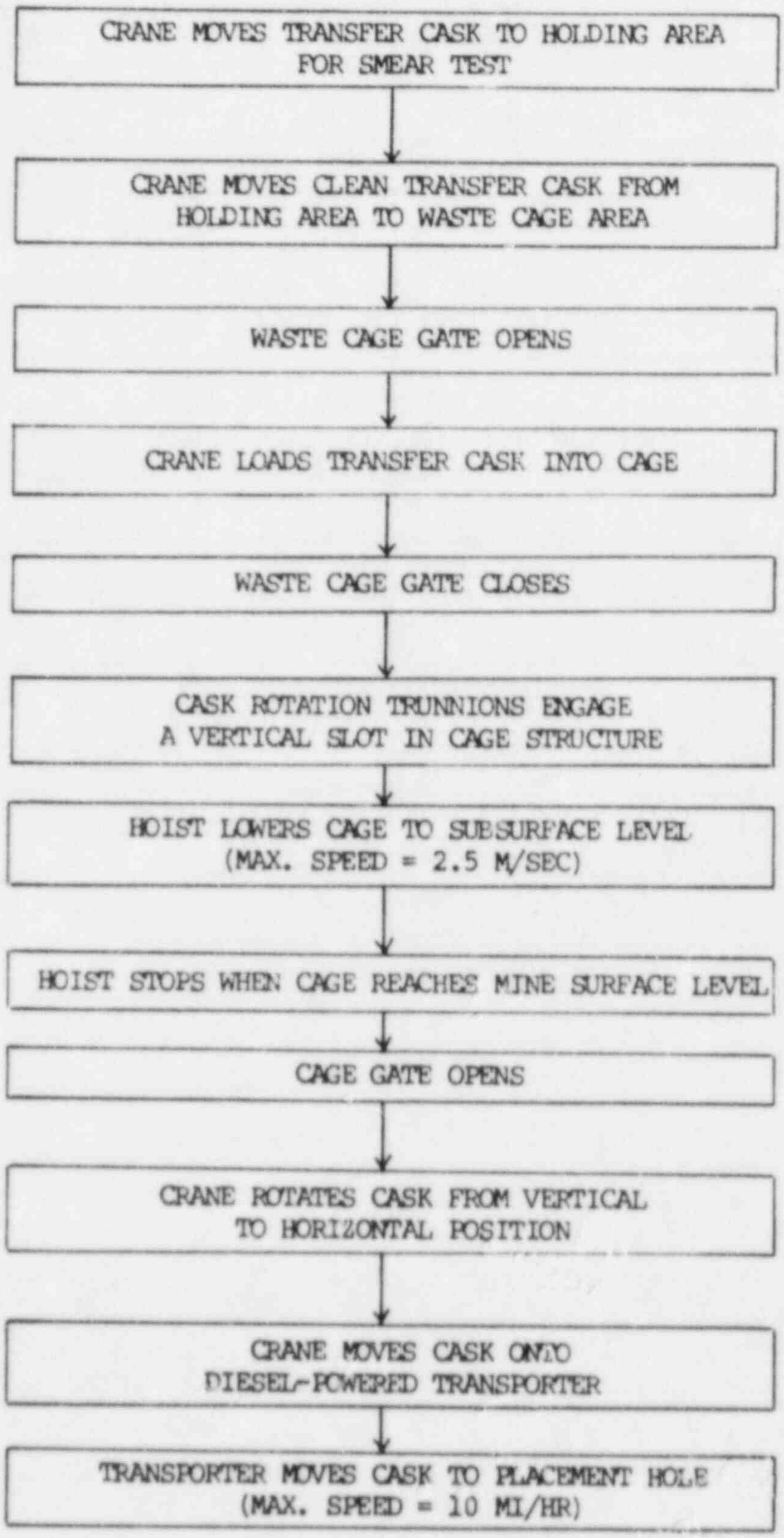


Fig. 2-16 Material flow diagram, hot cell area to placement area.

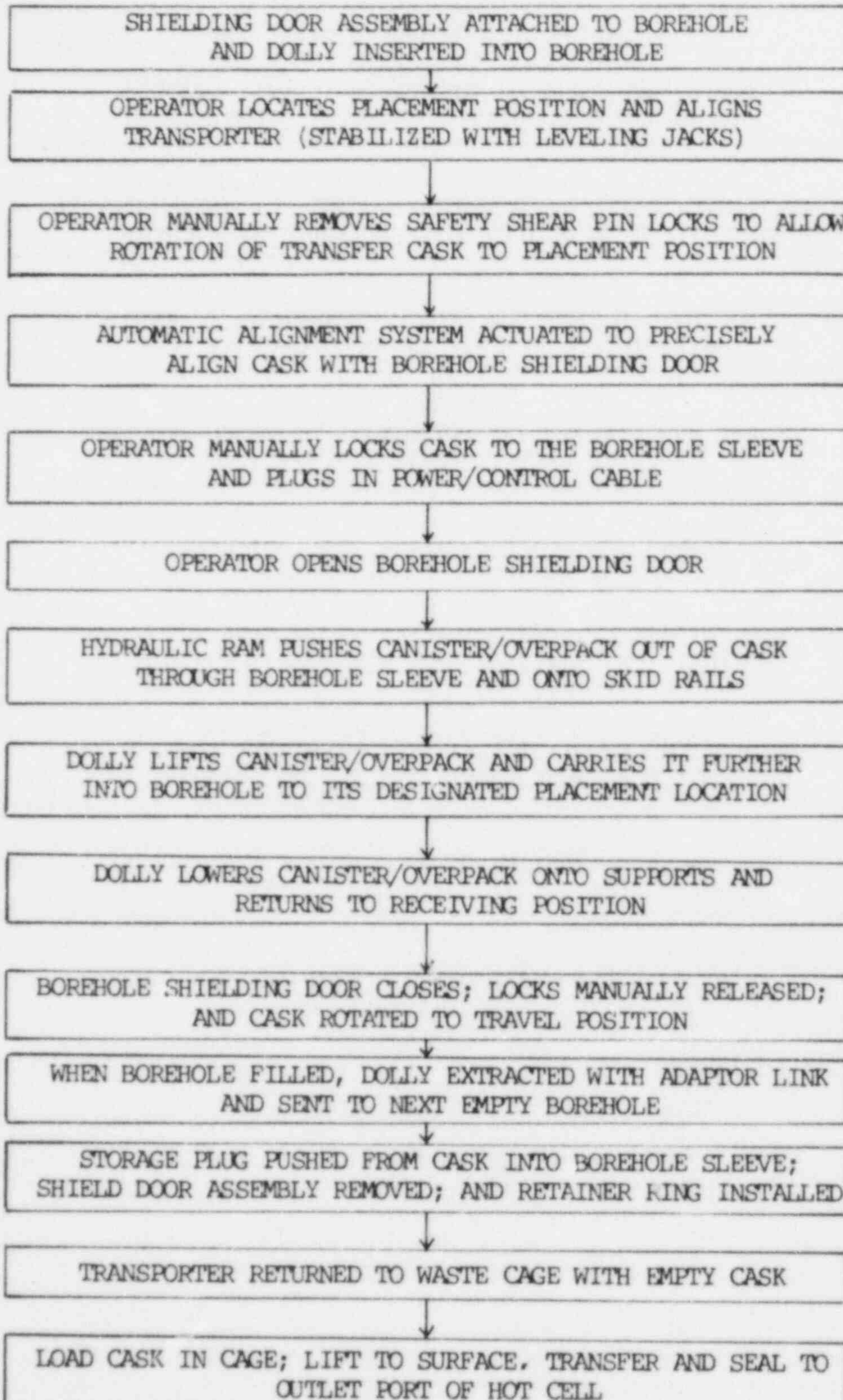


Fig. 2-17 Material flow diagram for the placement area.

1. PRIOR TO PLACEMENT, A DOLLY IS PLACED INTO THE BOREHOLE AND A BOREHOLE SHIELD IS ATTACHED TO THE SLEEVE. THE DOLLY REMAINS IN THAT HOLE UNTIL LAST CANISTER/OVERPACK HAS BEEN PLACED (SEE STEP 5)
2. THE SLIDING DOOR CAN NOT BE OPENED UNTIL IT IS INTERLOCKED WITH THE TRANSPORTER SHIELD.
3. THE BOREHOLE SHIELD & DOLLY REMAIN IN PLACE UNTIL THE LAST CANISTER/OVERPACK HAS BEEN PLACED IN THAT BOREHOLE AND THE HOLE PLUGGED (SHOWN IN STEP 6)

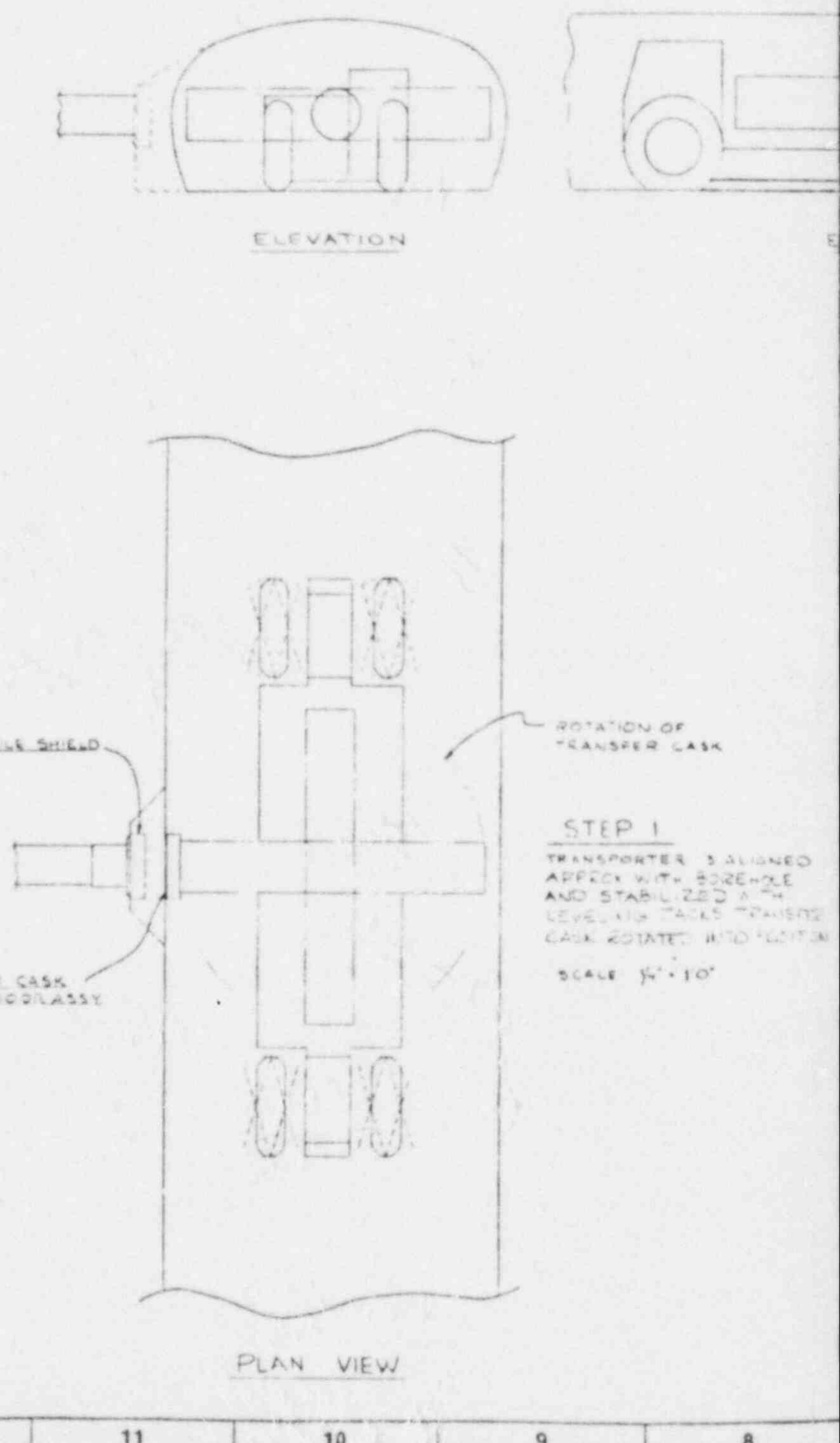
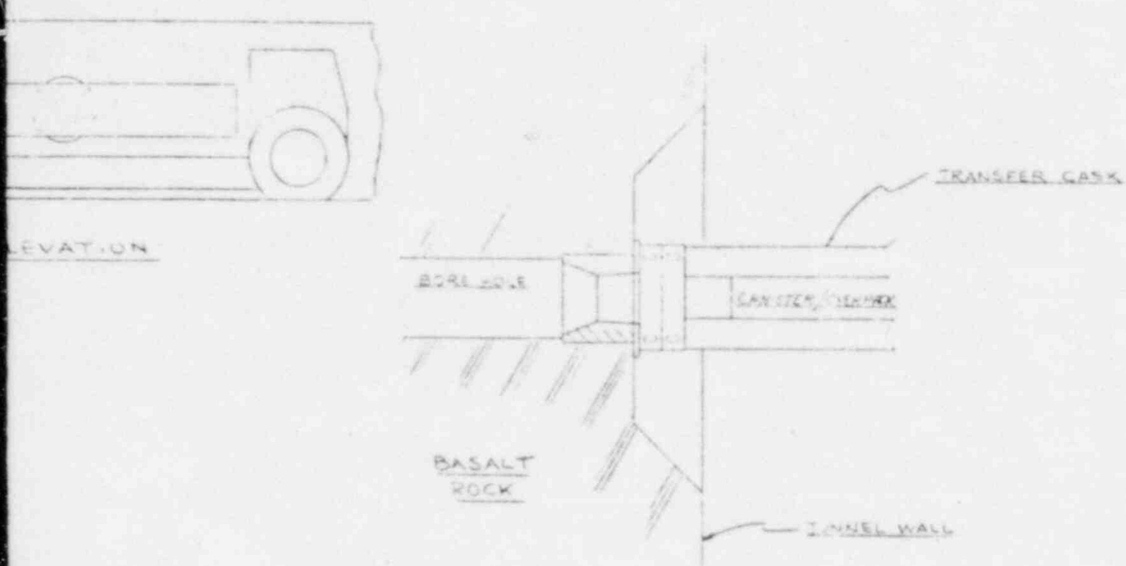


Fig. 2-18. Placement sequence 1.



STEP 2

TRANSFER CASK IS MOVED INTO PRECISE ALIGNMENT BY POSITIONERS ON THE TRANSPORTER. WHEN SHIELDS ARE ELECTRICALLY & MECHANICALLY INTERLOCKED DOORS MAY BE OPENED.

SCALE 1/2" = 1'0"

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GRAPHIC SCALES

1/4" = 1'-0"

1/2" = 1'-0"

PLACEMENT SEQUENCE

H-6-60A5

CHLW 4-0112-4WH

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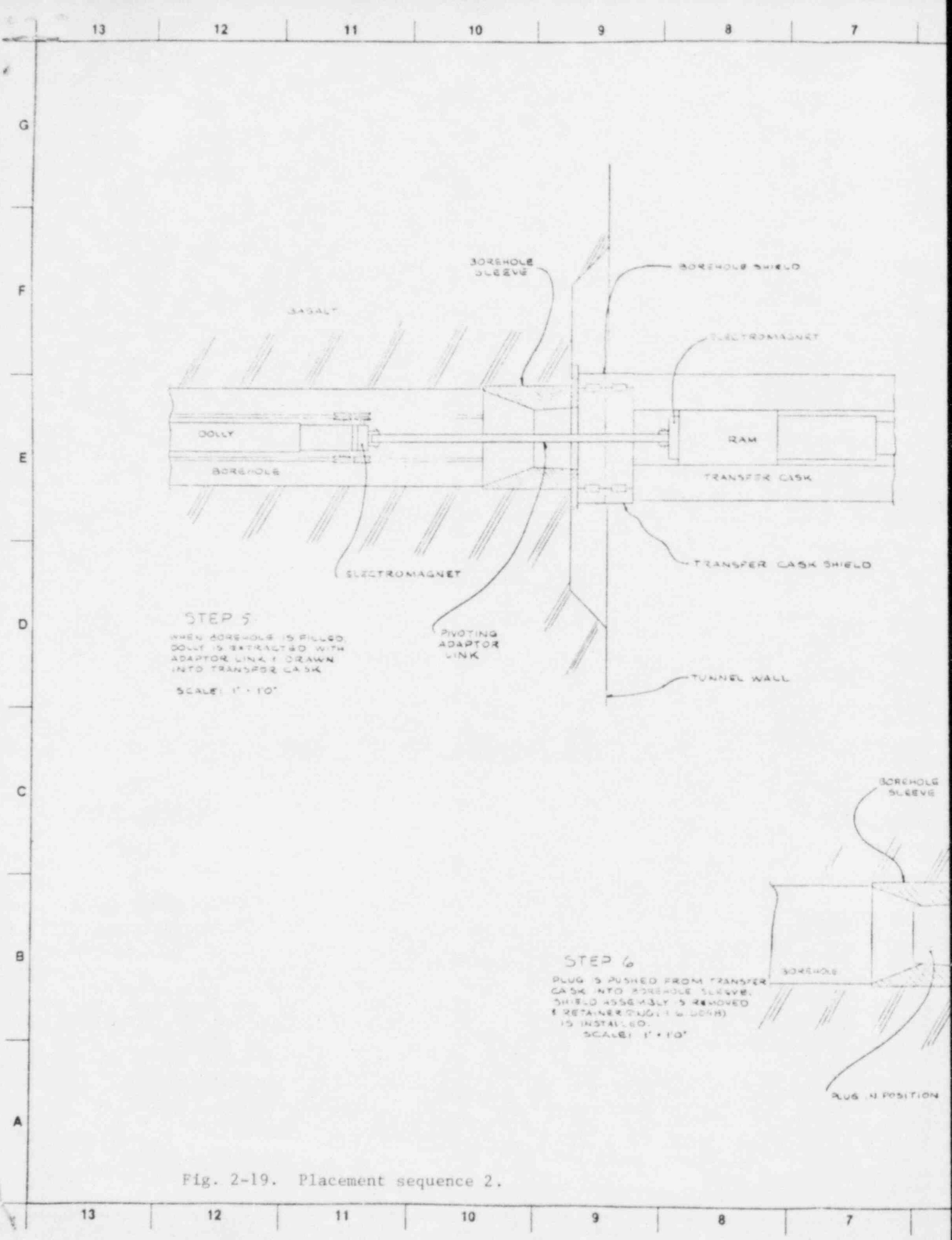
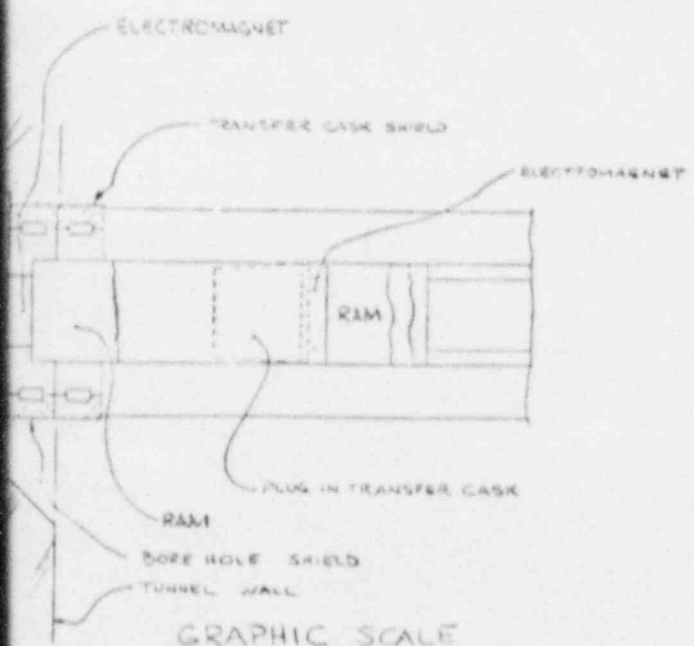


Fig. 2-19. Placement sequence 2.



PLACEMENT SEQUENCE

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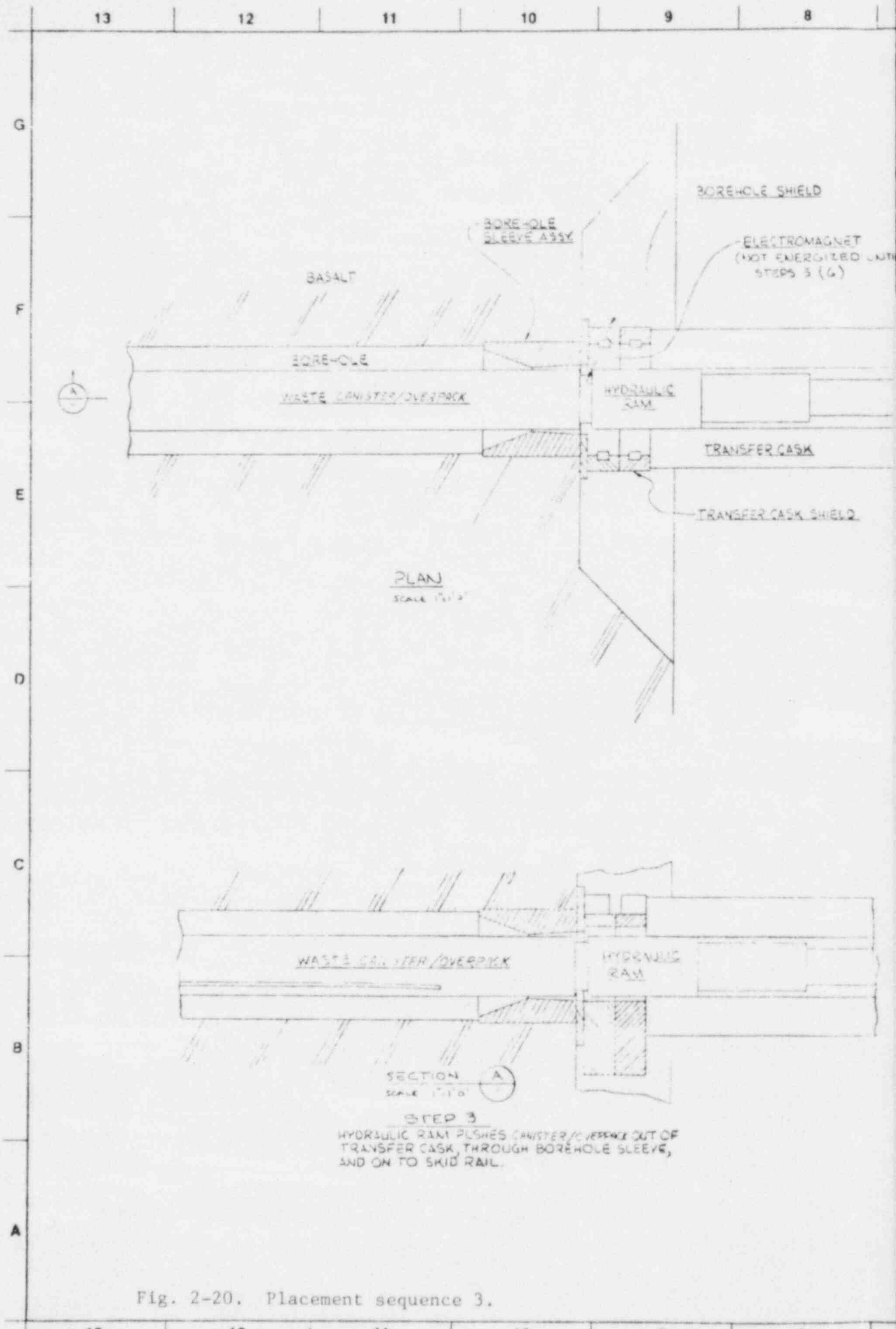
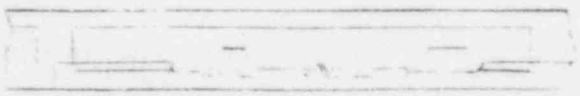


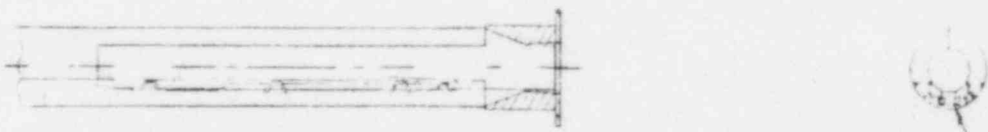
Fig. 2-20. Placement sequence 3.



SCALE 1/2" = 1'-0"

STEP 4B

AT DESIRED POSITION, DOLLY LOWERS CANISTER/OVERPACK ONTO SUPPORTS, THEN DOLLY RETURNS TO RECEIVING POSITION.



SCALE 1/2" = 1'-0"

STEP 4A

THE DOLLY, WHICH IS UNDER THE CANISTER/OVERPACK, THEN RAISES IT FROM THE SKID RAIL AND TRANSPORTS IT TO DESIRED POSITION.

GRAPHIC SCALES

U.S. Department of Energy

KE/PB

PLACEMENT SEQUENCE

H-6-6046

CHLW 4-0112-9WH 26

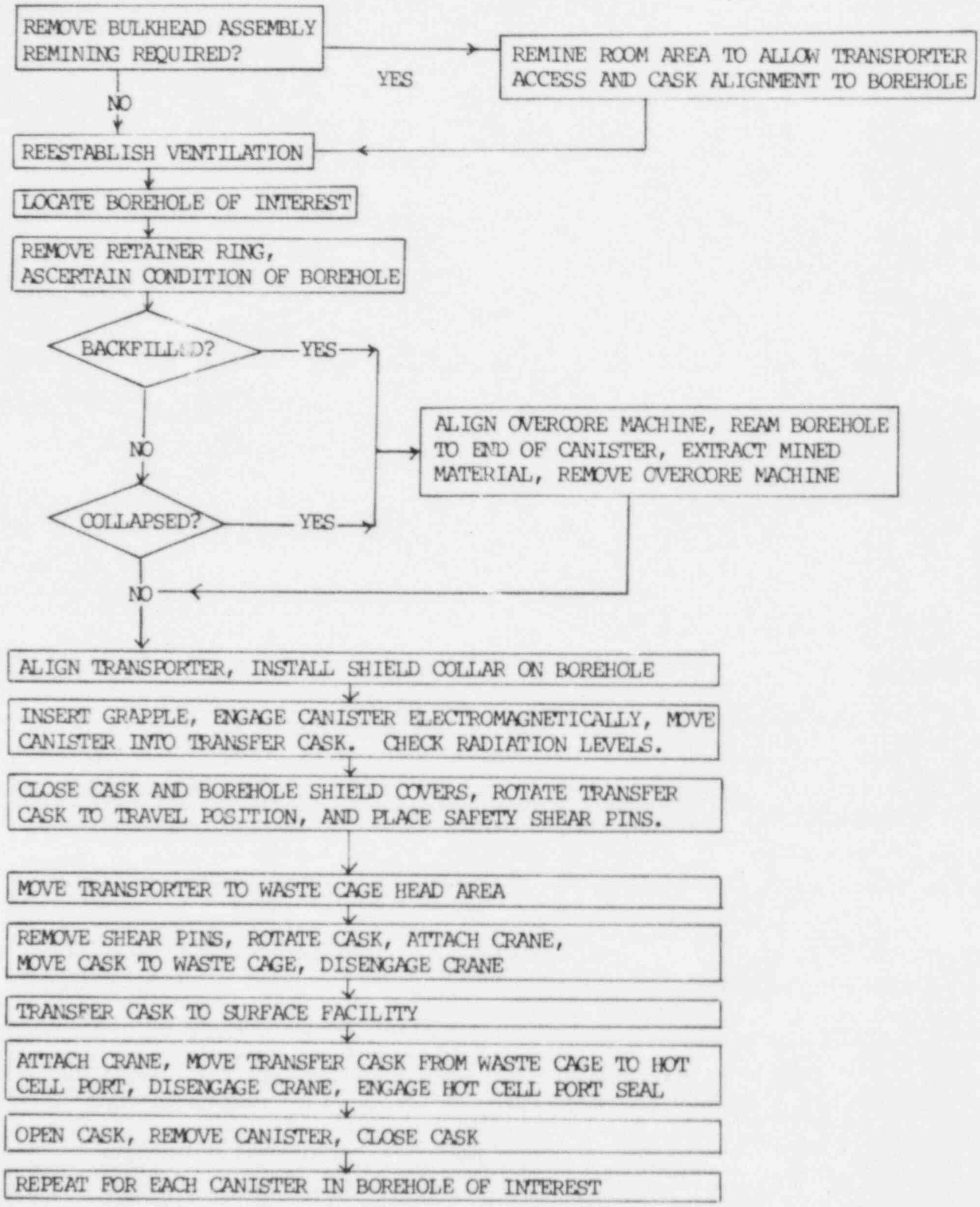


Fig. 2-21 Canister retrieval operations.

2.2 INITIATING EVENT IDENTIFICATION AND PRELIMINARY SCREENING

The material flow diagrams developed in the previous section for the process areas listed in Table 2-1 are each examined in detail to identify all plausible events, incidents, or occurrences capable of initiating an accident. The identification is performed by process area, first examining all internal (process generated) events followed by examining the vulnerability of the process and area to external and natural events.

The completed list of potential initiating events is then subjected to a preliminary screening process (Pepping, 1981). The objective of this screening is the early elimination of inconsequential risk contributors to keep the subsequent analysis tractable. Simple screening criteria are employed by assigning each event an occurrence probability ranking of low, medium, or high; determining what consequence types could result from the event; and applying a similar ranking for each consequence type. Consequences of interest are: (1) public radiological exposure, (2) occupational radiological exposure, (3) occupational nonradiological related injury, (4) repository availability, and (5) preservation of long-term repository viability.

Table 2-2[2] lists the external events identified from the literature search as potential contributors to overall risk. This list is divided into natural and human-induced events external to the processing of spent fuel and high-level waste. For the Hanford site, some can be dismissed immediately (i.e., tidal waves); the others are used to reexamine the processes in each area for vulnerability to that particular event.

2.2.1 Arrival Area and Storage Yard

Spent fuel and high-level waste shipments are brought to the repository by either rail or truck. Truck shipments are driven directly to the arrival area. Rail shipments are halted at the site boundary where rail crews are changed from commercial railroad personnel (Burlington Northern) to repository site personnel. It is assumed that locomotive switching will also occur, and commercial railroad crews will return to the Pasco, Washington switchyard with their locomotive(s) and possibly an empty train of shipping cask rail cars. Repository personnel will shuttle the empty/full rail trains, and move full incoming trains from the railroad intersection to the arrival area, a distance of about three miles, using switchyard locomotives. Each car passes through a portal radiation monitor in the process (see Section 2.1.1). This switching and shuttling operation has the potential for several accidents to occur.

Table 2-3 contains a listing of events, accidents, and occurrences that could occur and the possible consequences that could result from each.

Two types of collisions are considered: train/truck collisions with moving objects and train/truck collisions with stationary objects. Both reflect switchyard accidents in rail yards and truck depots that often occur.

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[2]Tables 2-2 to 2-10 are located at the end of Section 2.2.

Collisions involving both objects in motion are given a "medium" frequency while collisions with stationary objects are more frequent and, thus, assigned a "high" frequency.

Both accident types possess the capacity for energetic impact/crush of the cask due to the mass and momentum inherent in switchyard locomotives and diesel truck tractors and are, therefore, assigned a medium radiological consequence severity. Personnel injury potential is definitely high for these activities. Repository availability remains unaffected as the arrival area will contain sufficient backlog to insure processing for several days prior to requiring further shipments. Long-term repository viability is unaffected.

Shipping cask separation from the transport vehicle is considered a low probability event due to the qualification criteria and reinspection requirements a cask vehicle is subjected to following every shipment. Similarly, radiological consequences are considered low for shipping cask falls from the height of a rail car or a truck bed because all shipping casks must be designed to withstand a drop from 9.2 meters (30 ft) onto an unyielding surface landing in an orientation that does the most damage (10CFR71.36). Further stringent criteria are specified for resistance to puncture (1 m drop onto a 15.2 cm plunger). A cask falling off a rail/truck vehicle at rest is therefore extremely unlikely to be subject to a breach. Risk of personnel injury/death remains high, however.

Derailment of a rail car is a medium probability event for switchyard activities. Radiological consequences are considered minimal for derailments unless collision also occurs. Rail car collision events are addressed above. Occupational injury due to derailment does occur and is considered here. Effect on repository processing (both near- and long-term) is insignificant due to the backlog of other rail car/casks available to ensure continued throughput.

The possibility of a breached canister/shipping cask passing through the portal monitor undetected is considered to be low with present day redundant monitoring systems; however, if this does occur, subsequent local radiological release is considered highly probable and of medium probability at the site boundary. The potential for affecting repository availability is high, particularly if the transport car is opened in the receiving area, forcing evacuation and temporary facility shutdown.

Aircraft crash on to the arrival area is deemed a low probability event as it is assumed air space restrictions would be enforced for the airspace above the repository boundaries. However, the consequences of a crash would be high, particularly for energetic cask crush/impact breach and personnel injury. No long-term effects are anticipated and overall repository availability should be minimally affected.

Natural events such as earthquake and wind storms would hamper processing activities and pose a hazard to personnel but do not possess sufficient potential for shipping cask breach. Meteorite impact on the arrival area is an exceedingly low-probability event with severe radiological and personnel injury

consequences. Repository availability would also be affected but no long-term degradation of function is considered credible.

Fire and explosion are always potential hazards when motor fuels and/or mining explosives are present. Fire presents a more significant threat to a shipping cask than explosion. Boiling of coolant can produce unacceptably high pressures leading to rupture. Effects of explosions are reduced by cask design for impact resistance. Other natural events are of such a low-occurrence probability at the Hanford site that they are not considered justifiable hazards.

2.2.2 Washdown and Receiving Area

The function of the washdown and receiving area is addressed in Section 2.1.3 and the material flow diagram is given in Fig. 2-7. Transportation accidents involving vehicle movement are again potential contributors; however, movement in this area is restricted to (1) unidirectional flow of vehicles into the repository at one end and out the other end, and (2) low speed, restricted momentum by using the special track-mobiles rather than locomotives for rail car movement. Accidents in which both vehicles are in motion are therefore not reasonable for this area. Collision momentum is restricted to low-speed impact with stationary objects. Potential for personnel injury remains high as in any switching operation, but radiological consequences are at most of medium ranking (see Table 2-4). It should be noted that impact on repository availability is an important consideration. Collision with the receiving area airlock doors will reduce or stop processing until repairs are made.

Rail car derailment is again a medium-probability event with low radiological consequence potential. Occupational injury is a possibility for derailment, particularly in the restricted space surrounding the washdown and receiving area. Repository availability may also be affected because the receiving area represents the first single channel of repository material flow. The arrival area contains multiple flowpaths for material movement if a car should derail or other accident occur. However, in the receiving area a derailment would block one of the two available paths for unloading. This can affect repository throughput. Long-term effects are considered insignificant.

Equipment malfunctions in the washdown area should not create radiological hazards. The vehicle is neither opened nor the cask disturbed. Some potential exists for personnel hazards and equipment failure could certainly influence repository availability.

Shipping cask separation from the transport vehicle is again considered a low-probability event, with similar low radiological consequence severity (see Section 2.2.1). Personnel injury hazard and impact on repository availability for this event have somewhat higher probability consequences.

Failure of the receiving area airlock can have a range of consequences depending on what point in the waste processing failure occurred. Failure prior to initial sealing and vehicle opening would impact availability but present no significant radiological health hazards. Seal failure given air-

borne activity in the receiving area (after opening the vehicle and trying to sample the cask) would definitely create local radiological hazards, but this is an intermediate event in a sequence starting with release of airborne activity to the receiving area.

Once the vehicle is inside the closed receiving area, cask internal and external samples are taken to determine condition of the canisters and radiation/contamination levels. Rupture of the radwaste sampling line connected to the cask or improper connection leading to rupture are significant local radiological hazards and could, given additional air seal or HVAC failures, be potential site release sources.

When sampling of the internal cask environment and smear testing of the cask external area is complete, the inner airlock doors to the unloading area are opened and the transport vehicle moved inside. Malfunction of these airlock doors is treated similar to the receiving area airlock doors for both occurrence and consequence probabilities.

Several external events are capable of disrupting process flow in the receiving area. These are divided into natural and human-induced events.

Earthquakes and windstorms (tornados) are both low-probability natural events; however, radiological consequences could occur if ground motion or missile impact severes the radwaste line and opens the receiving area to the environment.

Potential human-caused events include aircraft crashes, fires, and explosions. The probability of these events occurring is considered low. Aircraft crash has the potential for energetic cask impact/crush, contributing to high radiological consequences in addition to personnel injury and loss of repository availability. Fire has a high potential for radiological release due to high internal cask pressures generated by heating. Explosion has a somewhat lower radiological release potential due to cask impact standards. Both events pose hazards to workers and can affect repository availability.

2.2.3 Unloading Area

Specific functions and processes performed in the unloading area are described in Section 2.1.4 and the relevant material flow diagram is provided in Fig. 2-9. A list of potential initiating events in this area is given in Table 2-5.

Cask vehicle movement accidents are again limited to lower energy collisions with stationary objects due to unidirectional process flow and the use of low-speed trackmobiles. Public and occupational radiological exposure is of medium severity; occupational injury and availability impact are considered high.

Rail car derailment consequences are similar to derailments in the receiving area. All radiological consequences are low but personnel injury and the potential for loss of availability are high.

Once the vehicle is in the unloading area, the airlock doors are required to be closed and sealed. Failure to close or maintain the seal could cause a radiological release to the receiving area only if airborne activity were already present. Seal failure is not an initiating event for radiological release. It is a concern for repository availability and possibly personnel injury if someone were caught in the closing doors. A jammed airlock would preclude any further waste processing until repairs were made.

Cask separation from the transport vehicle is a low-probability event due to vehicle qualification and inspection requirements; subsequent radiological consequences are considered low given cask strength and limited drop height.

The transport vehicle is automatically positioned under the hot cell access port, leveled, and clamped in place. These operations all use hydraulic systems. Hydraulic malfunction or operator error could tip the vehicle over; however, radiological exposure is unlikely as the cask shield cover is still in place during these operations. Personnel injury and reduction in repository availability are more serious concerns.

The next step requires entering the transport vehicle and rotating the cask to the upright position. The possibility of a leaking canister/cask escaping detection was already treated in the arrival area initiating events. The major concerns here are the potential for personnel injury and reduction in process availability if the cask is not rotated to the vertical position successfully.

The cask is connected to the hot cell by lowering a shield collar from the hot cell floor (unloading area roof) onto the mouth of the shipping cask and inflating a pneumatic seal. If either the collar moves, the seal fails, or the transport vehicle moves, the airtight integrity of the connection will be lost and airborne activity may escape to the unloading area. This assumes the hot cell and canister shield covers have been removed and canister removal is in progress. Prior to shield cover removal, the unload area personnel would normally be evacuated to minimize exposure if this incident occurred. Failure to evacuate personnel is an intermediate event associated with seal failure.

If the hot cell crane is unable to remove the hot cell and cask shielding covers, processing time is lost but no radiological hazards or personnel risks are created.

Hot cell crane operations required for canister removal from the shipping cask, smear testing, and placement in either the lag storage pit or the secondary area are lumped together in two initiating events. During all these operations, the hot cell status remains the same (cask connected at unload station, all other ports closed) and only one canister is in motion, implying that the same accident scenarios are required to generate the consequences of interest. Both mechanical crane failure and crane operator error are contributors and are thus treated separately. Radiological consequences are ranked medium severity, along with impact on repository availability. Personnel injury remains low as all operations are remotely controlled.

Preparation of the empty shipping cask poses no direct radiological or personnel injury hazards. No personnel are supposed to be in the unloading area during this time and the high-level radiological sources are now in the hot cell. Failure to restore the hot cell seal or failure to detect airborne activity in the unload area prior to readmitting personnel could contribute to radiological consequences, however. These events are low-probability occurrences and their estimated consequences are of medium severity for both local radiological release and repository availability. Other activities (shield collar movement, cask rotation, etc.) are addressed in previous events as they are merely a reversal of the initiating process. Exit airlock door failure and transportation accidents during removal of the empty cask vehicle can cause personnel injury and reduce availability but no credible mechanism exists to cause radiological hazards.

External events capable of disrupting the unloading area include earthquake, windstorm, and lightning strike. Other natural events are not of sufficient probability to pose credible threats. An earthquake during a cask unloading operation is a particularly sensitive event as the cask seal with the hot cell loading port would probably fail and the mechanism is also present for opening pathways to the atmosphere. Windstorm and lightning are treated the same as in the receiving area.

Human-induced external events such as fire, explosion, and aircraft crash are considered potential contributors to risk similar to previous areas.

2.2.4 Secondary Area

The function of the secondary area is to provide access to either transfer cask loading or canister overpacking in the process tank as determined from the canister smear tests and radiation level readings in the hot cell. Decontamination and smear test facilities are also available (see Section 2.1.5). A material flow diagram for the secondary area is given in Fig. 2-11.

Several crane drop accidents are treated separately in the secondary area due to different consequences. If the crane drops a canister in the lag storage or secondary storage pits on top of another canister, the potential exists for breaching two canisters simultaneously, producing up to twice the radiation contamination levels expected for one breached canister.

Crane drop of a single canister or canister collision with a wall/stationary object during transfer to and from the process tank, secondary storage, and the lag storage pool would release part or all of the contents of one canister. This is treated as a single initiating event with medium probability of local and boundary radiological hazards. Release would in turn affect repository availability.

Disruptions in the process tanks would definitely reduce waste processing rate and, given additional system failures, could also create radiological hazards. Specific steps required for overpack and sealing require movement, vacuum pressure, welding, and smear testing.

The process tank cover must be closed following canister/overpack insertion to evacuate the tank prior to laser welding of the overpack cap. Cover seal failure would result in either holdup of the process or a severely oxidized weld incapable of meeting pressure or ultrasonic testing requirements. In either event, availability is reduced, but no additional radiological hazards specific to seal failure are created.

If the laser welding equipment breaks or operator error occurs, the welding process could breach the canister, releasing radioactive gases and volatilizing fuel particles by heat addition. Releases would be contained in the hot cell and air filtration system unless additional failures occurred.

Following welding, the canister overpack assembly is tested to ensure a leak-proof seal has been made. Both helium sensing and ultrasonic weld examination are used. Helium sensing is performed by induction heating the pintle on the overpack, releasing helium inside the overpack from one of three cartridges provided. Ultrasonic testing is conducted by filling the tank with water and examining the weld area. Either process can affect availability and possibly long-term repository viability if a bad weld remains undetected.

Any release of gases, fuel fragments, or fission products to the hot cell environment creates the potential for occupational and public radiological exposure, and definitely decreases facility availability. Occurrences probability and consequence severity rankings are given for these events in Table 2-6.

External events capable of disrupting secondary area operations, given the constraints of the Hanford site, are similar to other process areas. Earthquake, windstorm, and lightning comprise the list of natural occurrences feasible for the Hanford site with sufficient energy potential to cause significant damage. Human-induced events external to the waste handling process considered credible are fires, explosions, and aircraft crashes. Probability rankings for the occurrences of these events and their consequences are also given in Table 2-6.

2.2.5 Hot Cell Discharge to Placement

Insertion of the canisters or canister/overpack assemblies into the transfer cask and transport to underground placement locations are the primary process steps considered in this section. A detailed description of these processing steps is given in Sections 2.1.6 and 2.1.7. A material flow diagram is provided in Fig. 2-16.

Loading of a canister overpack assembly into a transfer cask involves positioning the transfer cask under the hot cell outlet port and obtaining a seal (using the transfer cask crane), lifting a canister/overpack from either lag storage or a process tank and transfer to the outlet port (using the hot cell crane), and insertion of the assembly into the transfer cask. Transport

of canister/overpacks inside the hot cell was addressed in the previous section. Initiating events are included in Table 2-7 for canister insertion and transfer cask alignment failures.

As the transfer cask crane lowers the transfer cask from the hot cell port, an automatic cask shield door (spring-loaded) closes to seal the cask. Failure of this door mechanism will provide the environment surrounding the canister with a flowpath to the holding area airspace. The canister has been decontaminated in the hot cell area but may still possess residual activity or contamination from the lag storage pit. Failure of the door was considered to be of medium probability. Radiological consequence severity was estimated to be high and medium for local and site boundary exposures, respectively.

Transfer cask drops and collisions with holding area equipment are of minimal radiological release potential due to cask structural strength. Transfer casks are designed to withstand a drop test equivalent to a 7.3 m (24 ft) height on to an unyielding surface without loss of shielding protection. All heights in the holding area required for transfer cask movement are less than 7.3 m. Any cask breach is therefore considered unlikely.

The potential for undetected canister degradation by drop or collision is significant, however. While no immediate effects (other than unplanned reduction of repository availability) are observed by operations personnel from these incidents, possible denting, creasing, or distorting of the enclosed canister and likely shatter of some enclosed fuel rods can exert a long-term influence on repository viability.

Accidents involving waste cage operation include loading, cage travel, and unloading. Loading operations in the holding area as mentioned previously, never elevate the transfer cask to a height near the cask design limit of 7.3 m. From Fig. 2-13 the maximum height for transfer and loading is on the order of 5.2 m (17 ft) so cask breach from impact is not physically plausible. Puncture may be a potential mechanism, depending on crane path and area surrounding the waste cage. Drop/collision accidents are of medium occurrence frequency with low radiological and medium availability/occupational injury severities.

Cask separation from waste cage during lowering or cage hoist assembly failure could result in forces sufficient to breach the cask. If cask separation occurs, the cask will not fall directly down the shaft (due to Coriollis forces) but will repeatedly impact the concrete shaft walls for the duration of the fall. The function of a shock/impact absorber at the shaft bottom would probably be purely academic in this event as the cask may already be breached.

Failure of the cage hoist assembly and associated cage brake equipment will result in the waste cage/cask dropping to the shaft bottom and impact with the deformable shock absorber. Probability of cask puncture is high and impact breaching is also possible.

Either of these events combine a pathway (cask breach) with an energetic mechanism (impact) for the ejection of cask contents, particularly volatiles

and particulates. All consequences are considered of medium to high severity for these events. It is assumed that long-term repository viability will remain unaffected following either event as an unbreached cask/canister assembly would certainly be retrieved to the surface for inspection as a minimum safeguard.

Unloading of the waste cage at the underground transporter loading station is accomplished using a transfer cask crane identical to the surface loading crane. Lift heights are again insufficient to cause a cask breach directly from impact although puncture remains a possibility. Transporter movement during the loading operation could also puncture the cask.

Potential transporter-related initiating events during cask movement from the loading station to the placement area include collisions with stationary objects or other transporters, fuel tank ruptures, transporter breakdowns, and transfer cask fall from transporter. Cask breach due to impact is not physically reasonable and puncture is possible only in transporter/transporter collision. Local fire caused by fuel tank rupture or collision with stationary objects containing flammable material could cause cask rupture due to overpressure. Transporter breakdown would subject personnel to higher total integrated doses due to increased duration of time spent in proximity to cask but no radionuclide release mechanisms are plausible.

There are more external events capable of disrupting process stages for the hot cell discharge to emplacement sequence because of additional events that affect only subterranean activities. Natural events include earthquakes (surface and underground effects), windstorm, lightning, rock deformation (cave-in), and uplifting/subsidence of underground passages. Human-induced external events include uncontrolled subterranean flooding (seepage or groundwater changes), fire, explosion (both surface and underground), and aircraft crash (surface only). The estimated frequency ranking of these events and their expected consequence severity are given in Table 2-7.

Subterranean flooding is considered here due to the possibility of underground conditions causing the flood, not the surface conditions. The estimated flood plain for a 1000-yr flood near the Hanford site remains in the Columbia river basin and does not threaten repository operations. Mining experience, however, indicates that the presence of undetected groundwater, abnormal seepage, and sump pump failures have caused subterranean flooding in the past.

2.2.6 Waste Emplacement

Prior to arrival of the loaded transporter at the borehole site (placement area), a shield door assembly with a mating surface identical to the hot cell discharge port must be installed on the borehole opening and an automated canister transport dolly inserted into the borehole. This requisite step is only necessary for initially empty boreholes; partially filled sites will already have this equipment installed. Detailed descriptions of all processing steps required for emplacement are included in Section 2.1.8.1 and the related material flow diagram is given in Fig. 2-17.

Shield assembly and dolly installation have no credible offsite radiological release consequences but do present occupational injury hazards and could degrade repository availability if a borehole was blocked. Some occupational exposure could also be accumulated if the dolly/shield assembly becomes contaminated/activated.

Rotation of the cask and alignment with the borehole provide several possible initiating events. If the operator incorrectly locates the transporter or fails to lock the leveling jacks properly, the seal with the borehole could be lost during canister ejection, creating a local radiological exposure hazard from canister radiation fields but no released activity.

Inadvertent actuation of either the cask rotation or automatic alignment systems could puncture the canister during ejection, releasing radioactivity locally and possibly external to the repository. This accident will reduce repository availability.

Operator error during manual locking of the cask to the shield assembly or attachment of the power/control cable could lead to cask/borehole seal failure (local exposure) but not canister perforation. Radiological consequences and operational injury severities are low. Repository availability will be seriously affected due to unplanned downtime.

Borehole shielding door failures could cause local extended exposure to canister radiation fields but no release. Replacement/repair of the shield door would require more unplanned downtime (loss of availability).

Hydraulic ram ejection of the canister from the transfer cask has the potential for a canister breach and subsequent radioactivity release if control of ejection speed is lost. Impact between two canisters and subsequent higher release is also conceivable. The potential for loss of repository availability is also present, but a greater concern is the potential for canister degradation (due to impact) without immediate breach. The borehole filling operation might continue; the emplaced degraded canisters represent a potential for loss of long-term repository viability.

Another accident with identical physical consequences is loss of control of the dolly during canister insertion. Collision between canisters in the borehole is possible, with the above referenced consequences.

When a borehole is completely filled, the dolly and shield door assembly must be moved to a new borehole and a storage plug/retainer ring set installed on the full one. These activities present local radiological and occupational hazards, with an accompanying loss of repository availability.

Following completion of canister insertion, the cask/transporter must be prepared for travel, moved to the underground waste cage loading area, the cask loaded on the waste cage, returned to the surface, and moved to the hot cell transfer port to accept the next canister. The transporter may require refueling prior to transporting more canisters. All these activities pose no radiol-

ogical hazard but can create occupational injury hazards and potential loss of repository availability. Table 2-8 lists all initiating events for the placement area, along with their associated ranked occurrence and consequence severities.

This table also includes external events which could disrupt the placement operation. Natural events considered plausible are earthquakes, rock deformations, flooding, and subsidence/uplifting. Human-induced events external to waste processing activities include loss of sump pumps, fire, and explosions. Most of these events have the potential for compromising the long-term function of the repository due to undetected degradation of the emplaced canisters or inability to assure long-term retrievability (i.e., flooding).

2.2.7 Waste Retrieval

The option to retrieve waste may take place under three possible conditions: (1) immediate retrieval of waste following emplacement, with boreholes not yet backfilled; (2) retrieval without backfill material but borehole collapsed as a result of a mine cave-in; (3) retrieval after 20 to 30 years, boreholes backfilled, and room sealed with a bulkheaded panel. Detailed description of all processing steps required for retrieval of emplaced waste are discussed in Section 2.1.8.2. The related material flow diagram is given in Fig. 2-21. The potential initiating events identified for the retrieval operations are listed in Table 2-9.

Waste retrieval under normal conditions (borehole intact, no backfill) should require the same equipment used for cask transfer and emplacement. Personnel exposure hazards from retrieved waste will be influenced primarily by age of the waste, activity of surrounding borehole material, and whether or not the canister is intact.

Initiating events for portions of the retrieval process are the same as for emplacement due to use of the same equipment. Listing of these events in the retrieval section is repeated to allow treatment of the retrieval option separate from the rest of the analysis. Initiating event frequencies will be significantly different if immediate retrieval (within one to three years) is attempted. Since retrieval is an option and not a required step in the material flow process, separate quantification of retrieval risk is required.

Common events include shield door assembly not properly installed, operator error in transporter parking and jack leveling/locking, inadvertent actuation of cask rotation mechanism during canister retrieval, borehole shielding door malfunction, dolly/shield door assembly removal failure, and storage plug/retainer ring installation failure. Ranked occurrence frequencies and estimated consequence severities for these events are the same as given in Table 2-8 and described in Section 2.2.6.

These events address most of the handling operations needed for retrieval. There are, however, a few handling activities unique to the retrieval process (see Fig. 2-21). The grappling operation required to electromagnetically grip the steel canister with enough force for withdrawal is not a part of canister

placement. If the grapple is extended too far or too rapidly, the emplaced canister could be punctured. While this is considered a low-frequency equipment failure or operator error, consequences of local and offsite radiological release are possible in addition to loss of repository availability.

Loss of the electromagnetic couple between the canister and the grapple would leave the canister in the borehole with resulting impact on availability and possibly some increased personnel exposure. Canister jam during withdrawal would result in the same consequences. Grapple retraction force is not sufficient to damage the canister if a canister jam were to occur.

Once the canister is in the transfer cask, the activities associated with preparing the transporter for travel, transport to the waste cage, hoisting to the surface, and movement into the hot cell area are all identical to activities needed for emplacement. The potential initiating events (10) associated with these activities are again repeated in Table 2-9.

The events described above are adequate for estimating risk given immediate retrieval. If retrieval is desired at longer intervals following emplacement, the possibility of both borehole access and canister degradation must be considered.

Access to the borehole may be complicated by both natural causes and normal repository operations. Borehole collapse due to subsurface movement will bury the canisters and subject them to crush and puncture mechanisms. Normal repository operations may include backfilling of loaded boreholes with a mixture of crushed bentonite and basalt followed by bulkheading of entire panels (three rooms). Figure 2-21 provides a schematic of material flow for the three possibilities of open, backfilled, or collapsed boreholes in addition to bulkhead panels and deteriorated tunnel conditions. Immediate retrieval assumes no degradation of the repository subterranean environment and use of emplacement equipment (point A on Fig. 2-21). Further tasks anticipated for delayed retrieval include (1) remining of any tunnel or room areas that may have collapsed, (2) removal of bulkheads to provide panel access, (3) reestablishing active ventilation to remove mining dust, airborne activity, and exhaust gases from the area, (4) location of the desired borehole and assessment of its condition, and (5) overcoring of the borehole to remove either backfill material or collapsed borehole material around the canister to facilitate extraction.

Consideration of these additional operations in the retrieval process creates several new potential initiating events associated with equipment failure and operator errors. Assuming all operations indicated in Fig. 2-21 are needed, there will be additional mining activities required to remine tunnel areas and clear accumulated rubble. This has no radiological consequences but does affect personnel injury risk. Removal of panel bulkheads and establishing active ventilation present additional potential for operator injury but no radiological or availability concerns. Borehole retainer ring removal is addressed in the initiating events for immediate retrieval.

The unique activity required for delayed retrieval is removal of backfill material or collapsed borehole material using an overcoring machine. This machine must cut an annulus around the canister and remove the mined material to the end of the borehole (61 m) without striking the canister with the cutting head (see Section 2.1.8.2). Given the complexity of this task, breakdown of the overcore machine is an obvious possibility. In addition to loss of availability, some additional occupational exposure may result from extended operation/machine repair in the vicinity of activated mined material. Breakdown of the mined material removal machine is considered as part of the overcoring machine failure.

Contact between the rotating overcore cutting head and an emplaced canister would probably puncture the canister, releasing radioactive gases and possibly volatiles to the local subterranean environment. Contact due to machine error and contact due to operator error are treated separately because of the difference in expected frequency of occurrence. Due to the difficulty of this operation, machine error is given a medium frequency and operator error is given a high frequency.

These additional events are included in Table 2-9 under a separate heading of "delayed retrieval" to separate overcoring operations from normal retrieval.

External events capable of influencing retrieval are the same as those listed for hot cell discharge to placement (Table 2-7) and placement (Table 2-8). The justification for their selection is given in the later portions of Sections 2.2.5 and 2.2.6. These events are included in the retrieval events Table 2-9 to keep retrieval as a complete, separate option distinct from normal emplacement operations.

2.2.8 Service and Support Systems

The initiating events identified in previous sections occur either directly from nuclear waste processing activities or as a result of external events that disrupt those activities. The service systems of a repository facility represent another more subtle source of accident initiators. Service system failures also disrupt waste processing activities (similar to external events) but not only by catastrophic equipment damage. Instead, the equipment merely stops functioning. For service systems such as electrical power, it is commonplace to have one or more redundant backup systems to avoid loss of function. On the other hand, radwaste collection and treatment systems are usually single systems. Their failure can cause accumulation of waste processing by-products if operation is continued, eventually forcing shutdown for repair and creating occupational and offsite radiological exposure hazards. The initiating events identified for the service and support systems are shown in Table 2-10.

Service systems are those that serve more than a single building or facility, and those that occupy more than one facility or area. These include:

- Radwaste collection and treatment
- Bulk materials handling
- Personnel and material transport
- Electric power and lighting
- Subsurface ventilation.

The radiation waste collection and treatment system disposes of solid, liquid, and gaseous radioactive wastes generated on-site. The waste incinerator is used for burning of internally generated solid wastes. Rupture of the off-gas tank could lead to release of radioactive gases. The low-level of radioactivity involved and other mitigative measures within the system should make a radiological release to the public very unlikely; however, operator exposure is possible. Operator injury is also possible because of the high-temperature environment. Unless the rupture is a major one, overall repository availability will not be affected.

Liquid radioactive wastes generated from such sources as reverse osmosis concentrate, ion exchange resin sluicing water, cask flushing and venting water, radiation monitoring system drains, etc., are collected in a tank and then purified by ion-exchange resin treatment. The liquid radwaste line to the tank could rupture causing operator exposure to radioactive waste and impacting repository availability because of clean-up operations.

Bulk materials handling includes the following:

- handling of excavated basalt
- preparation and handling of aggregate from the mined basalt
- handling of cement, additives, and bentonite
- mixing and transfer of backfill mix to the repository horizon for placement holes and rooms.

All these activities pose occupational hazards and possible stoppage of mining operations, thus impacting the repository's availability to accept and emplace waste.

Personnel arrive at the NWRB by automobile or by bus; then walk to surface work stations. Materials and parts are hauled from the warehouse and the maintenance building to various destinations by flatbed trucks based at the maintenance building. A service shaft is provided for raising and lowering personnel, equipment, and materials between the surface and the mine area.

Personnel and material transport activities can lead to sequence of events that could cause occupational injury such as falls, entrapment, asphyxiation, etc., and could also impact repository availability.

The electric power system for the repository consists of three subsystems that provide power services to repository facilities: normal power supply,

standby power supply, and uninterrupted power supply. Mechanical and electrical failures of power supply components and external events such as fire could trigger loss of power supply to the repository which may lead to adverse radiological and nonradiological consequences if the mitigative system measures do not function.

The subsurface ventilation is designed to provide life support air and adequate cooling for subsurface personnel and equipment. Two independent ventilation systems are provided in the NWRB design: one to serve mine development, and the other to serve areas dedicated to nuclear waste handling and placement (confinement).

All underground repository operations would be affected by a loss of mine ventilation as activities would halt and the underground area would be evaluated. Working temperature and airborne activity constraints would require the removal of all personnel. The anticipated temperature rise is not sufficient to affect either the casks or the emplaced canisters. The only radiological consequences that could be expected would be lower level occupational hazards.

The same initiating event can take place in the mine development area but should lead mainly to occupational injury and reduction in repository availability.

In addition to the service directly related to the repository's primary function of accepting and storing radioactive wastes, initiating events occurring as a result of ongoing mine development during the active emplacement period should also be considered. These events primarily impact repository personnel safety and, to some extent, repository availability. Those initiating events that have been identified include failures of the rubber-tired haulage system and mine conveyor system, and excavation activities; in particular, the use of line explosives, which pose serious threats to mine personnel safety.

TABLE 2-2
EXTERNAL EVENTS TO BE CONSIDERED IN REPOSITORY RISK ASSESSMENTS

Natural Events	Human-Induced Events
chemical effects	air and spacecraft crashes
dissolution	explosion
earthquakes	dam failure
erosion	loss of offsite power
faulting	fire
forest fires	
inundation	
landslides	
lightning strike	
permeability change	
precipitation storms	
rock deformation	
rapid thaws	
sea or lake level changes	
sedimentation	
subsidence	
thermal change	
tidal waves (tsunamis, seiches)	
uplifting	
undetected features and processes	
volcanism	
windstorms (tornados, hurricanes, etc.)	

TABLE 2-3
POTENTIAL INITIATING EVENTS - ARRIVAL AREA AND STORAGE YARD

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
1. Train/truck collides with moving vehicles in the arrival area	M	1 2 3	M M H
2. Train/truck collides with stationary structures in the arrival area	H	1 2 3	M M H
3. Shipping cask falls off truck or rail car to ground	L	1 2 3	L L H
4. Rail car derailed	M	1 2 3	L L M
5. Breached canister/shipping cask undetected by radiation monitoring system	L	1 2 4	M H H
EXTERNAL EVENTS			
6. Earthquake damages arrival area; dumps rail cars/truck trailers	L	1 2 3 4	L L M M
7. Windstorms (tornados) strike arrival area	L	1 2 3 4	L L M M
8. Aircraft crash into arrival area (including railway intersection, portal monitoring area, suspect rail car/truck station, and vehicle inspection pits)	L	1 2 3 4	H H H M
9. Fire occurs in arrival area/storage yard	M	1 2 3 4	M M H M
10. Explosion occurs in arrival/storage area	L	1 2 3 4	L M H M

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure;
3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

TABLE 2-4
POTENTIAL INITIATING EVENTS - WASHDOWN AND RECEIVING AREA

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
11. Train/truck collides with stationary structures in the washdown and receiving area (including transport into and out of area)	M	1 2 3 4	M M H H
12. Rail car derailed	M	1 2 3 4	L L M M
13. Washdown cleaning area equipment malfunction	M	3 4	M M
14. Shipping cask falls off truck/rail car during area transport	L	1 2 3 4	L L M M
15. Receiving area airlock door failure	M	3 4	L M
16. Liquid radwaste sampling line rupture/improper connection	M	1 2 4	L M L
17. Unloading area airlock door failure to open	M	3 4	L M
EXTERNAL EVENTS			
18. Earthquake damages washdown/receiving area	L	1 2 3 4	M M H H
19. Windstorm damages washdown/receiving area	LL	1 2 3 4	M M H M
20. Lightning strike	L	1 2 3 4	L L M M
21. Aircraft crash onto washdown and receiving area	L	1 2 3 4	H H H H
22. Fire in the washdown and receiving area	L	1 2 3 4	M H H M
23. Explosion in the receiving area	L	1 2 3 4	M M M M

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure; 3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H = high; M = medium; L = low; LL = very low

TABLE 2-5
POTENTIAL INITIATING EVENTS - UNLOADING AREA

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
24. Train/truck collides with stationary structure in the unloading area	M	1 2 3 4	M M H H
25. Rail car derailed	M	1 2 3 4	L L M M
26. Airlock doors in the unloading area fail to close/seal	M	3 4	M M
27. Shipping cask falls from transport vehicle in unloading area	L	1 2 3 4	L L M M
28. Transport vehicle tipped over during positioning/leveling/ clamping operation	L	1 2 3 4	L L M M
29. Cask rotation hydraulic power supply failure/improper attachment	L	3 4	M L
30. Vehicle upper/lower hatches break/operator error	M	3 4	M L
31. Hot cell shielding collar/seal failure	M	1 2 4	L M M
32. Vehicle clamping/leveling system failure to maintain position for hot cell seal	M	1 2 4	L M M
33. Hot cell or cask shield covers stuck; hot cell crane unable to remove one or the other (removal and replacement)	L	4	M
34. Hot cell crane failure during cask unloading, canister smear test, or transfer to lag storage/secondary area	M	1 2 4	M M M
35. Operator error - hot cell crane failure, during cask unloading, canister smear test, or transfer to lag storage/secondary area	M	1 2 4	M M M
36. Hot cell crane smear attachment jams in shipping cask interior	L	4	M
37. Hot cell shield cover seal/seat failure	L	1 2 4	L M M
38. Operator error - failure to test unloading area environment prior to re-admitting personnel	L	2 4	M M

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure;
3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H = high; M = medium; L = low; LL = very low

TABLE 2-5
POTENTIAL INITIATING EVENTS, UNLOADING AREA
(continued)

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
39. Unloading area exit airlock doors fail to open	L	3	M
		4	M
40. Transportation accidents in movement of vehicle into and through cask preparation and shipping area	M	3	M
		4	M
EXTERNAL EVENTS			
41. Earthquake damages unloading area/affects cask/port seal	L	1	H
		2	H
		3	H
		4	M
42. Windstorm damages unloading area	LL	1	M
		2	H
		3	H
		4	H
43. Lightning strike	L	1	L
		2	L
		3	M
		4	M
44. Fire in the unloading area	L	1	M
		2	H
		3	H
		4	M
45. Explosion in unloading area	L	1	M
		2	M
		3	H
		4	M
46. Aircraft crash into unloading area	L	1	H
		2	H
		3	H
		4	H

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure; 3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

TABLE 2-6
POTENTIAL INITIATING EVENTS, SECONDARY AREA

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
47. Canister drop by crane onto another canistr in leg storage pit or secondary pit	M	1 2 4	M H M
48. Canister dro by crane during transfer to and from process tank, secondary storage, and transfer cask loading port	M	1 2 4	M M M
49. Crane breaches canister during automated smear testing	M	1 2 4	M M M
50. Process tank cover seal fails/cover jams	M	4	M
51. Welding process penetrates canister	L	1 2 4	M M M
52. Helium leak test/ultrasonic leak test failures	L	4 5	M M
53. Contaminated water leaks out of process tank	L	1 2 4	L M M
54. Radwaste piping rupture	L	1 2 4	L M M
55. Earthquake damages secondary area	L	1 2 3 4	M H H H
56. Windstorm damages secondary area	LL	1 2 3 4	M H H H
57. Lightning strikes secondary area	L	1 2 3 4	L L M M
58. Fire in the secondary area	L	1 2 3 4	M M M M
59. Explosion in the secondary area	L	1 2 3 4	M M M M
60. Aircraft crash into the secondary area	L	1 2 3 4	H H H H

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure;
3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

TABLE 2-7
POTENTIAL INITIATING EVENTS, HOT CELL DISCHARGE TO PLACEMENT

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE	
		TYPE*	SEVERITY**
61. Canister/overpack dropped, distorted, or jammed during transfer cask insertion	L	1	M
		2	M
		4	M
62. Transfer cask crane fails to maintain cask seal during canister insertion	L	1	M
		2	H
		4	M
63. Transfer cask shield door fails to automatically close/remain closed during transport	M	1	M
		2	H
		4	M
64. Transfer cask dropped to holding area floor	M	1	L
		2	L
		4	M
		5	M
65. Transfer cask/crane hits waste cage during transfer	M	1	L
		2	L
		4	M
		5	M
66. Tranfer cask falls off waste cage during waste cage travel	L	1	M
		2	H
		3	H
		4	H
67. Waste cage hoist assembly failure	L	1	M
		2	H
		3	H
		4	H
68. Underground transfer crane drops cask	M	1	M
		2	M
		3	H
		4	M
69. Transporter moves during cask loading operation	M	1	L
		2	M
		3	M
		4	M
70. Loss of mine ventilation during operations	M	2	L
		3	M
		4	M
71. Transporter collision with stationary object (no fire)	M	3	M
		4	L
72. Transporter collision with stationary object (fire from fuel tank rupture and/or flammable material)	L	1	M
		2	M
		3	H
		4	M
73. Transporter breakdown during cask movement	M	2	M
		3	M
		4	L

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure;
3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

TABLE 2-7
POTENTIAL INITIATING EVENTS, HOT CELL DISCHARGE TO PLACEMENT
(continued)

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
EXTERNAL EVENTS			
74. Earthquake damages holding area, shaft, or transporter load area	L	1 2 3 4	H H H M
75. Windstorm damages holding area, hoist surface equipment	LL	1 2 3 4	L L M M
76. Lightning strikes holding area, shaft head frame	L	1 2 3 4	L L M M
77. Rock deformation in transporter load area (cave-in)	L	1 2 3 4	M M H H
78. Uplifting/subsidence in transporter load area	L	1 2 3 4	M M H H
79. Uncontrolled subterranean flooding	L	1 2 3 4 5	L L M H H
80. Fire in holding area, shaft head frame, shaft, or transporter load area	L	1 2 3 4	L M H M
81. Explosion in holding area, shaft head frame, shaft, or transporter load area	L	1 2 3 4	M H H H
82. Aircraft carsh into holding area or mine shaft head frame	L	1 2 3 4	H H H H

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure; 3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L- low; LL - very low

TABLE 2-8
POTENTIAL INITIATING EVENTS, PLACEMENT AREA

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
83. Shield door assembly/transport dolly installation failures	L	2 3 4	M M L
84. Operator incorrectly locates transporter/locks leveling jacks; cask to borehole seal lost	M	2 3 4	M M M
85. Cask rotation/automatic alignment systems inadvertently actuated during canister ejection	L	1 2 4 5	M M M M
86. Operator incorrectly locks cask to borehole sleeve or incorrectly attached power/control cable	M	2 3 4	L L M
87. Borehole shielding door jammed closed/open	M	2 3 4	M M M
88. Hydraulic ram ejects canister too rapidly	L	1 2 3 4 5	M M L M H
89. Dolly control failure during canister placement	M	1 2 4 5	L L M H
90. Dolly/shield door assembly removal failure	L	2 3 4	M L M
91. Storage plus/retainer ring installation failure	L	2 3 4	M L M
92. Transporter accident during return (empty) trip	M	3 4	M M
93. Transfer crane drops/stalls during empty task movement	L	3 4	M M
94. Waste cage hoist assembly failure/cask drop during return to surface	L	3 4	H M
95. Surface transfer crane drops/stalls during empty cask movement to hot cell transfer port	L	3 4	M M
96. Diesel fuel drums for transporter dropped from material hoist	L	3 4	H M

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure;
3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

TABLE 2-8
POTENTIAL INITIATING EVENTS, PLACEMENT AREA
(continued)

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
97. Earthquake damages placement area	L	1	M
		2	H
		3	H
		4	H
		5	H
98. Rock deformation in placement area (cave-in)	L	1	M
		2	H
		3	H
		4	H
		5	H
99. Uplifting/subsidence in placement area	L	1	M
		2	H
		3	H
		4	H
		5	H
100. Flooding of mine area	L	1	L
		2	L
		3	H
		4	H
		5	H
101. Loss of sump pumps in mine area	L	1	L
		2	L
		3	M
		4	H
		5	H
102. Fire in placement area	L	1	M
		2	M
		3	H
		4	M
103. Explosion in placement area	L	1	M
		2	H
		3	H
		4	M
		5	M

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure;
3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

TABLE 2-9
POTENTIAL INITIATING EVENTS, RETRIEVAL

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
104. Transfer cask fails to go into retrieval position	L	4	L
105. Shield door assembly not properly installed	L	2 3 4	M M L
106. Operator incorrectly locates transporter/locks leveling jacks; cask to borehole seal lost	M	2 3 4	M M M
107. Cask rotation/automatic alignment systems inadvrtently actuated during canister retrieval	L	1 2 4	M M M
108. Operator incorrectly locks cask to borehole sleeve or incorrectly attaches power/control cable	M	2 3 4	L L M
109. Borehole shielding door jammed closed/open	M	2 3 4	M M M
110. Dolly/shield door assembly removal failure	L	1 2 3 4	L M L M
111. Storage plug/retainer ring installation failure	L	1 2 3 4	L M L M
112. Loss of grapple extension control/operator error	L	1 2 4	M M M
113. Loss of electromagnetic couple between grapple and canister	M	2 4	M M
114. Debris collection system failure	M	3 4	M M
115. Transporter accident enroute to borehole area (empty cask)	M	3 4	M L
116. Transporter with transfer cask/retrieved canister collides with stationary object (no fire)	M	1 2 3 4	M M M L
117. Transporter collision with stationary object (fire from fuel tank rupture and/or flammable material)	L	1 2 3 4	M M H M
118. Transporter breakdown during cask movement	M	2 3 4	M M L

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure; 3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

TABLE 2-9
POTENTIAL INITIATING EVENTS, RETRIEVAL
(continued)

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
119. Transporter moves during cask unloading operation	M	1 2 3 4	M M M M
120. Underground transfer crane drops cask	M	1 2 3 4	M M H M
121. Waste cage hoist assembly failure	L	1 2 3 4	M H H H
122. Transfer cask falls off waste cage during the cage travel	L	1 2 3 4	M H H H
123. Transfer cask/crane hits waste cage during surface transfer	M	1 2 4	L L M
124. Transfer cask dropped to holding area floor	M	1 2 4	L L M
125. Transfer cask shield door fails to close or automatically remains closed during transport	M	1 2 4	M H M
126. Transfer cask crane fails to maintain cask seal during canister removal	L	1 2 4	M H M
127. Canister/overpack dropped, distorted, or jammed during tranfer cask removal	L	1 2 4	M M M
DELAYED RETRIEVAL			
128. Overcore machine failure (including mined material removal)	M	2 3 4	L M H
129. Canister struck by overcome machine - control system error	M	1 2 4	M H M
130. Canister struck by overcome machine - operator error	H	1 2 4	M H M

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure;
3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

TABLE 2-9
POTENTIAL INITIATING EVENTS, RETRIEVAL
(continued)

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE TYPE*	CONSEQUENCE SEVERITY**
EXTERNAL EVENTS			
131. Earthquake damages placement area, transporter load area, shaft, or surface holding area	L	1 2 3 4 5	M H H H H
132. Rock deformation in placement area (cave-in) or transporter load area	L	1 2 3 4 5	M H H H H
133. Uplifting/subsidence in placement area	L	1 2 3 4 5	M H H H H
134. Flooding of mine area	L	1 2 3 4 5	L L H H H
135. Windstorm damages holding area, hoist surface equipment	LL	1 2 3 4	L L M M
136. Lightning strikes holding area, shaft head frame	L	1 2 3 4	L L M M
137. Aircraft crash into holding area or mine shaft head frame	L	1 2 3 4	H H H H
138. Loss of sump pumps in mine area	L	1 2 3 4 5	L L M H H
139. Fire in holding area, shaft head frame, shaft, transporter load area, or placement area	L	1 2 3 4	L M H M
140. Explosion in holding area, shaft head frame, shaft, or transporter load area	L	1 2 3 4	M H H H

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure; 3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

TABLE 2-10
POTENTIAL INITIATING EVENTS, SERVICE AND SUPPORT SYSTEMS

INITIATING EVENT	OCCURRENCE FREQUENCY LEVEL**	CONSEQUENCE	
		TYPE*	SEVERITY**
141. Loss of confinement ventilation during operations	M	2 3 4	L M M
142. Gaseous radwaste system failure	L	1 2 3 4	L M M L
143. Liquid radwaste system failure	L	1 2 3 4	L M L L
144. Bulk materials being hoisted to the surface falls off cage	L	3 4	H M
145. Dust and debris collection system (form mine development activities) failure	L	3 4	H M
146. Operator falls down the service shaft	L	3 4	H M
147. Loss of offsite electric power	M	1 2 3 4	L M M M
148. Substation transformer failures	M	1 2 3 4	L L M M
149. Electrical equipment causes operator injury	M	3	M
150. Loss of ventilation in the mine development area	L	3	H
151. Rubber-tired haulage system hits power lines, water lines, etc.	M	3 4	M M
152. Mine conveyor system fails	M	3	M
153. Excavation activities cause operator injury	M	3	M

*Types: 1 = public radiological exposure; 2 = occupational radiological exposure;
3 = occupational nonradiological consequence; 4 = impacts repository availability; 5 = long-term effects
**H - high; M - medium; L - low; LL - very low

2.3. ACCIDENT SCENARIO DEVELOPMENT

Initiating events identified in Section 2.2 (Tables 2-3 through 2-10) with a rating higher than L (low) in either event frequency or consequence severity are developed into a group of event tree models. An event tree consists of an initiating event coupled to all systems/operator actions (intermediate events) capable of influencing the end result (consequence). An accident scenario is a subset of a particular event tree, consisting of the initiating event and a unique path of assumed intermediate event successes and/or failures leading either to a consequence of interest or accident mitigation.

Development of event trees for all initiating events given in Section 2.2 is a significant task; however, many models are already available from previous studies of repository concepts and retrieval feasibility. These models are compatible with the event tree format and used with only minor modification for the basalt repository concept. Requirements for new logic model development are reduced significantly using this existing information.

A further reduction in the number of trees is possible if similar initiating events with similar projected consequences are grouped into a single initiating event for development into a single event tree. For example, rail/truck collisions with moving objects, collision with stationary objects, and derailments are listed as separate potential initiating events for the arrival area. All are transportation-related accidents with the potential for puncture, crush, or burst of the shipping cask. These similarities allow consideration of a composite initiating event, noting that the frequency data used for this initiating event should be the sum of the contributors. In this manner, the number of separate logic models required is reduced further.

Finally, independent event trees are not constructed for external events or events with only personnel injury consequences. External events can only act through existing plant system failures to cause the consequences of interest. Existing event trees representing these failure paths are examined for vulnerability to each external event of interest. Selected trees are used to construct additional accident sequences, using the external event frequency as the initiating event frequency and adjusting the failure probabilities of all intermediate events as necessary to reflect vulnerability to the external event.

Occupational injury to personnel does not require logic modeling as available statistics provide worker injury and fatality for activities similar to those required for repository operations. Data from rail handling, warehousing, and mining operations encompass most of the occupational health hazards expected during preclosure activities. Risk due to occupational injury can be quantified directly from the data.

The following sections describe the development of event trees for each major repository area. Retrieval is developed separately (acknowledging some overlap in the sequences) to enable future consideration of this step as an option rather than an integral part of the waste disposal process.

2.3.1 Arrival Area

Initiating events identified for the arrival area are given in Table 2-3. All transportation-related accidents are grouped into a single event tree (Fig. 2-22)[3] due to similar expected consequences. Derailment and collision with both fixed and moving objects are capable of breaching the cask by explosion, fire, or puncture mechanisms.

The other event tree developed for arrival area activities is the failure of the radiation monitoring system to detect a breached canister/shipping cask assembly (Fig. 2-23). Although most of the potential consequences of this event occur in other areas of the facility where the vehicle and cask are opened, the initiating event occurs in the arrival area. Consequently, the potential for a breached assembly reaching more sensitive areas is addressed here.

External events capable of causing a switchyard derailment or collision include earthquake, windstorm, aircraft crash, fire, and explosion. Aircraft crash will be developed only once as a separate event tree using the entire surface portion of the facility as the target area. The frequency of the event is expected to be low but the potential for radioisotope release is high, regardless of specific area or mitigative system. The event tree is given in Fig. 2-24. This approach groups the same initiating event in different areas into the same event tree. The arrival area collision/derailment event tree is the pertinent event tree for use in developing aircraft impact. No credit is taken for containment structures (other than the cask) or mitigative systems. Energetic high momentum impact is inherent, to the extent that containment or mitigation is not considered credible.

Earthquake (see Fig. 2-25) may cause derailment or overturning of shipping cask vehicles but will only slightly affect the probability of fire or explosion. Windstorm is considered even less likely to cause overturn/derailment but is included on a preliminary basis (see Fig. 2-26).

Human-induced events such as fire and explosion have a dual nature. They are justifiable intermediate events in accident scenarios where impact or crush may increase their occurrence probability (given presence of combustibles or explosives). In addition, fire and explosion have to be treated as initiating events, potentially capable of causing injury and radionuclide release by causing a series of system failures. Figures 2-27 and 2-28 are event trees for fire and explosion as initiating events in either the arrival/storage area or the washdown/receiving area. The assumption is made for these models that if fire as an intermediate event is not prevented or contained (given an explosion, Fig. 2-28), cask breach by overpressure burst will occur. In a similar manner, fire as an initiating event has to possess sufficient intensity and duration to compromise cask integrity (10CFR70).

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[3]Figs. 2-22 to 2-51 and Tables 2-11 to 2-16 are located at the end of Section 2.3.

2.3.2 Washdown and Receiving Area

Initiating events capable of causing accidents in the washdown and receiving area were identified in Table 2-4. Transportation-related events can again be lumped together. Furthermore, the event tree for collision/derailment in the washdown and receiving area has no new mitigative systems or protective features different from the same event in the arrival area. A single tree will adequately represent this event for both arrival and receiving areas (see Fig. 2-22).

Rupture or improper connection of the radwaste sample line is treated in Fig. 2-29. Personnel exposure will definitely occur given the initiating event; offsite release can be prevented by proper operation of the airlock seal and the secondary confinement exhaust system. This confinement system is made up of three 50% capacity exhaust fans and filter trains, one of which is in standby and powered by a standby electrical system. The filter trains consist of moisture separators, 90% NBS filters, a HEPA filter, and a final HEPA filter. It is assumed that manual operator action is required to actuate the standby system in the event of a normal exhaust fan failure.

External events capable of affecting washdown and receiving area activities are again similar to those identified for the arrival area. Aircraft crash has already been considered by lumping the projected impact area into the entire surface portion of the facility. Other events including earthquake, windstorm, fire, and explosion would cause the same consequences in the receiving area as in the arrival area. There are no additional mitigating systems or operator actions that could affect the consequences; thus, these events are combined with their counterparts in the arrival area to form a single set of event trees (Figs. 2-25, 2-26, 2-27, and 2-28).

2.3.3 Unloading Area

Activities in the unloading area will normally occur with the protection of one or two sets of airlock door seals (depending on specific activity) between the radiological source and the free environment. This increased level of protection is provided because the unload area is where the protective environment of the shipping cask is opened and the canisters removed. The strength of the shipping cask tends to limit the number of credible initiating events prior to this point in the disposal process.

Transportation-related events such as collision or derailment are assumed to occur with the unloading area airlock doors open and the receiving area and/or cask preparation area airlock doors open to allow trackmobile access. This provides a potential path for offsite release. Figure 2-22 also includes the event tree for this event in the unloading area.

Once the vehicle is positioned correctly, all sets of airlock doors (receiving, unload, and cask preparation) are closed, the cask is rotated to the vertical position, and the hot cell shielding collar lowered onto the cask mouth to form an airtight seal. The hot cell and cask shielding doors are then removed, directly exposing the cask interior for the first time.

At this point, several initiating events can occur that are capable of destroying the cask/hot cell seal and exposing the unload area to the cask internal environment. These events include (see Table 2-5) hot cell shield collar seal failure, vehicle clamping/leveling system failure to maintain reference position, or an earthquake. Failure of the hot cell shield door seal, shield door cover seal, and failure of the vehicle clamping leveling system are combined into a single event. Figure 2-30 is a diagram of the event tree for loss of seal between the hot cell floor and cask slip. It is assumed that the primary confinement exhaust system is operated at a slightly lower pressure (absolute) than the secondary confinement exhaust system. Either exhaust system is capable of preventing radiological hazard provided the unload area is isolated from the open environment (both the receiving area and cask preparation area airlock doors are closed).

Primary confinement exhaust ventilation systems A and B are grouped together as a single intermediate event because the B system is 100% redundant to A and is assumed to start automatically on loss of A (no operator action required). The secondary confinement system, however, consists of three 50% capacity exhaust trains, one in standby and assumed to require operator action to initiate. Thus, the operator action and the probability of successful backup system start are both modeled separately. In the event that either the receiving area and unload area airlock doors are open, or the cask preparation area airlock doors are open, HVAC system operation is irrelevant as a direct pathway exists to the environment.

The canisters are raised from the shipping cask into the hot cell and transferred to storage or processing by the hot cell crane. Failure of the hot cell crane or the crane operator during these operations could breach the canisters/fuel rods and release significant quantities of radioactive gases and possibly volatiles to the hot cell environment. The event tree for this event is given in Fig. 2-31 assuming the seals with the unload area and the holding area are intact. This event tree also includes the transfer of fuel canisters over other canisters in either the lag storage or secondary storage areas, where a canister drop could breach more than one canister, causing subsequently higher radioactive releases to the hot cell environment. The initiating and intermediate events are the same, only the potential release fraction is different.

The spectrum of external events capable of disrupting unloading area activities is similar to other areas; however, the mitigative systems and potential consequences, with the exception of aircraft crash, are not. Aircraft crash has already been addressed on a sitewide basis. Other external events requiring consideration are treated by listing the internal event tree (e.g., loss of seal between hot cell floor and canister lip) and all the external events capable of causing the initiating event and/or influencing the probabilities of the intermediate events or consequence types. In this manner, all systems, processes, and actions of importance to plant throughput, personnel and public safety, and repository viability are examined for

vulnerability to each type of external event. Credible accident sequences from these event trees are added to the list of internal event-generated accident sequences.

The external event tables are given by process area in accordance with each subsection. Table 2-11 lists the event trees developed for the unload area and the external events capable of influencing each tree. Loss of seal between the hot cell floor and the cask lip is almost guaranteed given an earthquake or an explosion in the unload area during unloading of the shipping cask. On the other hand, fire, windstorm, and lightning do not pass credible threats to the seal once it has been established. They can act in a more subtle way by causing a power failure, but this is addressed in the fault trees representing the various hydraulic and pneumatic systems involved.

The other event tree in Table 2-11 is canister drop during movement into the hot cell. An explosion or earthquake during this operation would enhance the likelihood of a drop. Fire could force a local evacuation of the operations room and cause an eventual drop. Windstorm and lightning are not considered credible as their only credible influence is loss of electrical power. Loss of power could affect operator response (not accounted for in a mechanical system fault tree) because of loss in lighting; but, provided normal design standards of battery-backed emergency lighting are followed, these events are not deemed credible.

2.3.4 Secondary Area

Potential initiating events identified for the secondary area are given in Table 2-6. Canister drop by the crane during transfer is included in the event tree of Fig. 2-31. This includes drops while the canister is suspended over another canister or when discharging canisters to the holding area.

Activities unique to the secondary area require separate event trees. Smear testing, canister inspection, overpack installation/welding, and leak checks all present potential hazards. Smear testing and welding incidents both puncture the canister inside the hot cell/secondary area environment, so they are grouped into a single event and developed into an event tree in Fig. 2-32. Contaminated water leakage and radwaste piping ruptures are assumed to leak contaminated water from a primary contaminated area to a secondary contaminated area, placing the burden of cleanup on the secondary confinement exhaust system. The event tree for this accident is given in Fig. 2-33.

External events for the secondary area are partially included in similar considerations for the hot cell portion of the unloading area where canister drop was considered for both areas. Further external event considerations for canister puncture (Fig. 2-32) are earthquake and explosion. Liquid radwaste leakage (Fig. 2-33) can be affected by earthquake, explosion, and fire. Table 2-12 lists the event trees developed for the secondary area and the external events considered for each.

2.3.5 Hot Cell Discharge to Placement

The process steps involved in discharging waste canisters to the transfer casks and transporting the casks to the emplacement area is described in Sections 2.1.6, 2.1.7, and 2.1.8. Potential initiating events for these activities are identified in Table 2-7.

Canisters prepared for emplacement and decontaminated in the secondary area are transferred out of the protected hot cell/secondary area through the fuel discharge port. A transfer cask (different from a fuel shipping cask) is mated with the lip on the discharge port, the port opened and a canister placed in the cask.

This process can be disrupted by a number of events capable of failing the seal between the hot cell discharge port and the transfer cask lip. These events are lumped together in the event tree given in Fig. 2-34. If the seal should be lost, the canister will be exposed to the local environment surrounding the waste shaft head frame. The canister must either be leaking or possess significant external contamination to create a radiological hazard. Neither condition is highly probable as the canister is subjected to weld inspection and smear-tested prior to loading. However, the accident that causes loss of seal could also affect canister integrity.

The primary confinement exhaust ventilation system serves the hot cell/secondary area. If this system is in operation, the air flow will be from the load area/waste shaft headframe environment into the hot cell. Another ventilation system serves the waste shaft and headframe area. If the primary confinement exhaust system is unavailable, this system would remove airborne activity from the immediate area surrounding the transfer cask. Both exhaust/filter systems would have to fail to create a release potential. Both are also assumed to include 100% redundant, automatically actuated backup systems.

All transport handling events during transfer to the waste cage, hoist operation, and underground transfer to the transporter are grouped together in Fig. 2-35. Only one canister/cask assembly is moved during any single operation from hot cell discharge to transporter loading. The air flow pathway is open for these operations and the ventilation source is the same (shaft exhaust system). Air flows from the subterranean level up the waste handling shaft to the shaft exhaust and filtering system. This commonality of event and consequence allows grouping of individual events associated with handling and transport accidents during waste cage loading and descent.

Transporter accidents during travel from the waste cage area to the placement area are represented in Fig. 2-36. During normal operations, two exhaust systems service the underground placement area. The waste shaft exhaust system removes air through the waste handling shaft. A separate exhaust system exiting through a separate shaft to the confinement exhaust building removes additional air. These two systems are dedicated to placement operations (the confinement area); three other ventilation systems are used for mining development. The confinement building exhaust system does not normally

have HEPA filters in line. When an alarm signal indicating radioactivity in a shaft is received, the filters are automatically valved in line with the exhaust blowers through a series of valve arrangements designed to prevent a temporary overpressure problem. In order to avoid an offsite release, the sensing system has to function, followed by the valve realignments.

An additional event tree (Fig. 2-37) is developed for the transporter breakdown initiating event. No offsite release hazard exists for this event, but extended personnel activity in the area immediately surrounding the cask may result in higher occupational exposure.

External events capable of disrupting process flow and generating the consequences of interest are divided into surface and subsurface contributors to include consideration of waste transport from the hot cell discharge to the subterranean placement area. Aircraft crash was already included for the entire surface portion of the facility. Subsurface damage due to these events is not considered credible. Table 2-13 lists the event trees developed for hot cell discharge to placement and the external events capable of influencing these activities.

Loss of seal between the hot cell secondary area floor and the transfer cask lip (Fig. 2-34) can be caused by earthquake or explosion. Fire, windstorm, or lightning could cause power losses, but this is not expected to affect the seal once the cask is locked in position.

Handling accidents with the transfer cask during movement from the hot cell to the subterranean transporter (Fig. 2-35) can be caused by fire, explosion, earthquake, or rock deformation (cave-in). Transporter collision hazards (Fig. 2-36) can be increased by earthquake, explosion, fire, rock deformation, uplift or subsidence, or subterranean flooding (including loss of sump pumps). Transporter breakdown (Fig. 2-37) is also possible due to these same external events. In this case, however, local evacuation would already be underway due to the event itself. The additional exposure would occur at a later date when reentry to the mine area is allowed and cleanup operations are in progress.

2.3.6. Placement Area

Placement activities are described in Section 2.1.8.1. Potential initiating events are identified in Table 2-8.

Several initiating events address placement activities associated with preparation of the borehole opening and cask alignment/sealing to the borehole. Shield door/transport dolly installation failure, operator error in operation/alignment of transporter, power control cable or borehole/cask lock failure or incorrect operation, and borehole shield door jam do not breach the canister but only subject personnel to extended exposure to local cask/canister environment. These events are grouped together in Fig. 2-38.

There are three initiating events capable of causing canister/fuel rod breach during insertion: inadvertent cask movement during insertion, overspeed

of the hydraulic ram ejection unit, and overspeed of canister placement dolly. These events are combined into a single canister breach event given in Fig. 2-39.

Two further initiating events can occur during removal of the cask from the borehole. These are the failure to remove the dolly shield door mechanism from a full borehole and installation failure of the storage plug/retainer ring. These events have the same local consequences as the preparation and cask alignment events. Consequently, they are grouped into the event tree shown in Fig. 2-38.

The influence of external events in the placement area is determined by examining the placement process event trees for vulnerability to each external event considered (see Table 2-14). Cask to borehole preparation, alignment, and removal error (Fig. 2-38) can be caused by all the external events under consideration including earthquake, rock deformation, flooding (including loss of sump pumps), fire or explosion. Canister breach during insertion into the borehole (Fig. 2-39) can also be caused by earthquake rock deformation, or explosion.

2.3.7 Retrieval

Retrieval is an option for repository operations; not an integral part of the waste processing activities. As such, it is treated separately in this analysis although many of the events and subsequent accident scenarios are the same. The objective is to provide a measure of relative risk strictly due to retrieval operations that would be considered in addition to emplacement risk. Retrieval activities are detailed in Section 2.1.8.2; potential initiating events are given in Table 2-9.

The first section of initiating events represent "normal" retrieval; that is, retrieval within a brief period (two years) following emplacement. For this option, retrieval is mostly a reversal of the emplacement process. Events and event trees will also be similar.

Events addressing borehole preparation, cask alignment, and cask removal are grouped together to form the single initiating event in Fig. 2-40. In this tree, the intermediate event "canister intact" refers to the survivability of the canister during storage, not damage due to the initiating event.

There are two events capable of breaching the canister during removal. These include inadvertent cask rotation and loss of grapple extension control. These events are treated as a single initiator in Fig. 2-41.

Transporter accidents during cask movement from the borehole to the waste shaft area similar to emplacement (see Fig. 2-42). Puncture is not considered a credible release mechanism as transporter speed is limited to less than 10 mph. An additional event tree given in Fig. 2-43 is necessary to represent occupational exposure hazard from transporter breakdown.

The canister can be breached by several types of transport accidents

during transporter unloading, movement to the surface, and transfer to the hot cell secondary area discharge port. These events include transporter movement during cask unloading, cask drop during underground or surface crane transfer (puncture only), waste cage hoist assembly failure or cask fall from waste cage, transfer cask/crane collision with waste cage during loading or unloading, and transfer cask shield door failures during transport. All events are grouped together due to common ventilation system requirements and potential consequences. Figure 2-44 is the event tree for all cask breach accidents associated with movement from transporter unloading at the subterranean level to cask unload and mate to the hot cell discharge port at the surface.

Figure 2-45 is the event tree representing loss of seal between the hot cell secondary area and the transfer cask lip. The event tree for canister damage during removal from the transfer cask to the secondary area by the hot cell/secondary area crane is given in Fig. 2-46.

Additional events in the retrieval process are for "delayed retrieval," where borehole accessibility is degraded due to either backfilling or borehole collapse. If either condition exists, a special overcoring machine will be required to cut an annulus around the embedded canister to free it. This is a delicate operation with a machine that has not yet been developed. Three possible initiating events are given in Table 2-9 to encompass the failure modes of this device. Overcore machine failure (including mined material removal) and coring tool contact with the canister due either to machine or operator error are grouped into a single initiating event expanded into an event tree in Fig. 2-47.

The inclusion of external events is similar to the preceding sections, except for aircraft crash which is developed separately for retrieval operations. This separation is to maintain the capability of evaluating incremental risk due to the retrieval option separately from emplacement activities. Figure 2-48 is the event tree for aircraft crash during retrieval operations. Table 2-15 provides a list of retrieval event trees and external events to be considered for each.

Failures and human error during borehole preparation, cask alignment and removal, and borehole plugging are susceptible to earthquake, rock deformation, and explosion (Fig. 2-40). Canister breach during retrieval can be caused by earthquake, cave-in, or local explosion also (see Fig. 2-41). Transporter collisions in Fig. 2-42 can be influenced by earthquake, rock deformation, fire, and explosion.

Transporter breakdown (Fig. 2-43) is vulnerable to explosion, rock deformation, flooding (including loss of sump pumps), fire, and explosion. Transfer cask damage during transport to the surface facilities can be caused by earthquake, rock deformation, fire, and explosion (refer to Fig. 2-44).

Loss of seal between the hot cell secondary floor and the transfer cask lip shown in Fig. 2-45 can be affected by earthquake or local explosion only. Canister handling accidents during transfer cask unload to the hot cell secondary area (see Fig. 2-46) can be caused by earthquake, fire, and explosion.

Overcore machine failure depicted in Fig. 2-47 can be increased by earthquake, uplift/subsidence, rock deformation, flooding (including loss of sump pumps), fire, and explosion.

All event trees noted require reevaluation using each identified external event as the initiating event. Intermediate event failure probabilities may also require adjustment to reflect a degraded state given the external event.

2.3.8 Service and Support Systems

Several service systems have appeared as intermediate events in the event trees developed for each process area. These same systems can also act as initiating events to various accident scenarios. For example, normal fuel processing in the hot cell implies the continued operation of the primary confinement exhaust ventilation system to provide a continual air flow and to keep the pressure differential in a direction from least to most contaminated. Failure of this system would cause seepage flow to reverse, possibly exposing less contaminated areas to higher concentrations of both particulate and noble gases. In addition, current plant design specifies that no charcoal filters will be used on the secondary confinement exhaust ventilation system. If reverse seepage of noble gases were to occur, they would be passed to the open environment.

Service system failures as accident initiators represent a more subtle accident possibility due to the number of other systems that could potentially be affected by a service system failure. Service systems identified from preliminary plant design were used to develop the list of initiating events given in Table 2-10.

A loss of confinement ventilation during repository operations would require an evacuation of the underground waste placement regions (not active mining areas, which are serviced by separate ventilation). The subterranean environment would rapidly increase in temperature to the point where manned activities are not possible. Buildup of combustion vapors, dust, and possible airborne contamination are additional hazards. Figure 2-49 is an event tree developed from this event. Note that the primary concerns are the removal of personnel from the subterranean environment as rapidly as possible and having alternate air supplies/circulation available. This event does not contribute directly to offsite release potential.

Failure of the gaseous and liquid radwaste systems to contain radioactive effluent from the fuel disposal process is potentially capable of causing both personnel and public exposure. In either accident, the primary confinement exhaust ventilation and filtering system must fail to create a release to the open environment as shown in the event trees of Figs. 2-50 and 2-51.

Several initiating events given in Table 2-10 are primarily hazardous to personnel (nonradiological hazards) and are therefore not developed into event trees due to the availability of data for these types of occurrences. These events include material hoist failures, operator injuries associated with the

service shaft, and mining ventilation system failures. These hazards are well-understood from mining experience.

The loss of offsite power does not directly cause any accident sequences to occur; however, it is influential in many existing event trees due to the reliance of system components on either normal or standby power systems. Many processes are dependent on electrical power but no repository systems involve the confinement or control of a highly energetic process; consequently, loss of site power is examined as a contributor to other system failures rather than a separate initiator.

External events capable of disrupting service and support systems are given in Table 2-16 according to the initiating event (and event tree) each will affect. Loss of confinement exhaust ventilation system can be caused by fire, explosion, earthquake, windstorm, or lightning. Gaseous and liquid radwaste system failures can be caused by fire, earthquake, or explosion.

It should be noted that although loss of site power is not treated explicitly as an initiating event for this analysis, it is a contributing event at the fault tree level for many systems (both normal and standby power). An external event can also affect the reliability of these systems. Fire, windstorm, lightning, flood, explosion, and earthquake are all capable of causing loss of offsite power.

Loss of standby diesel power is also not treated explicitly as an initiating event but as a contributing event at the fault tree level. Fire, explosion, and earthquake are capable of causing loss of diesel power.

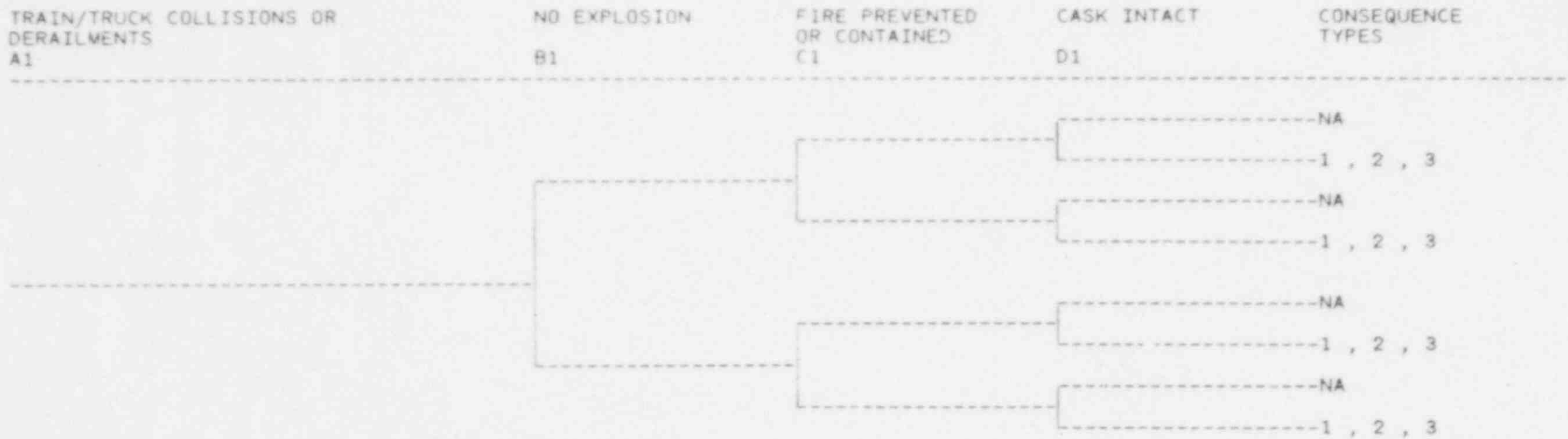

Fig. 2-22 Train/truck collisions/derailments in arrival, washdown and receiving, or the unload area.

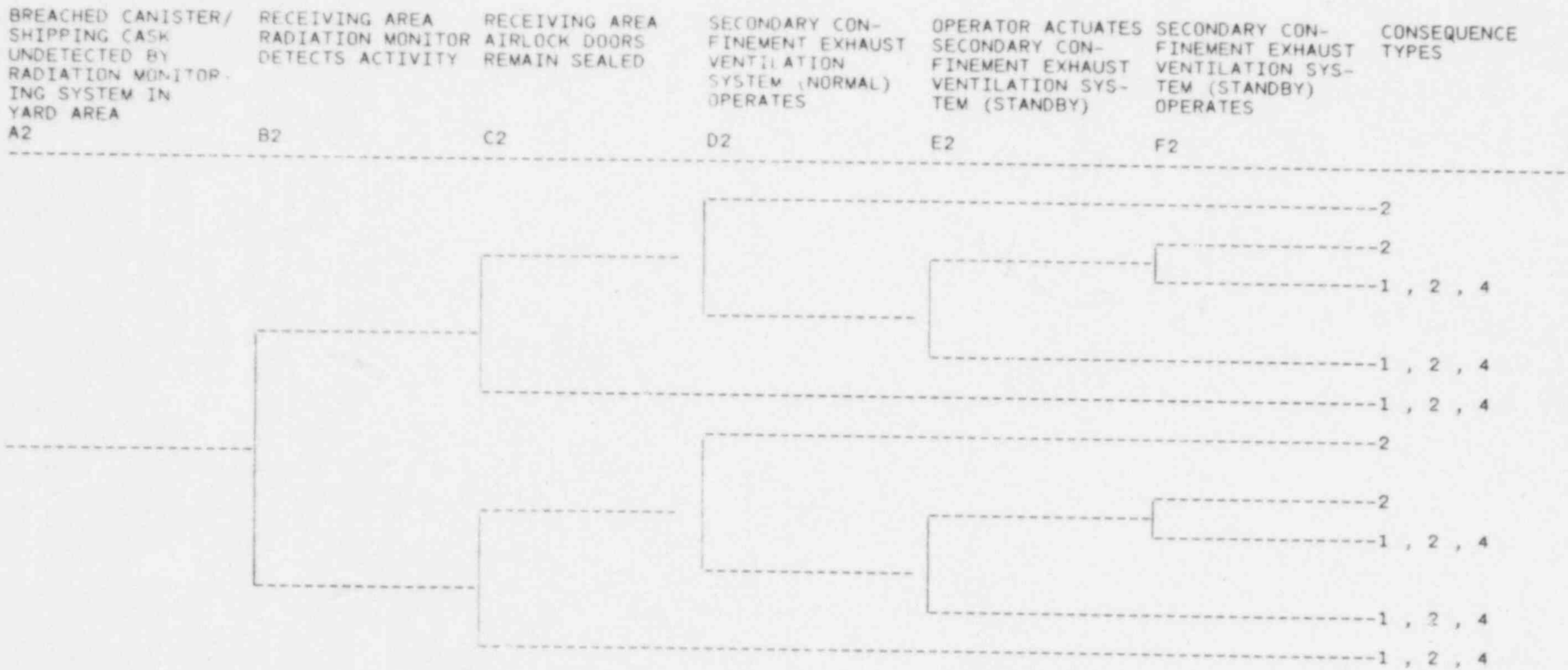

Fig. 2-23. Breached canister/shipping cask escapes detection by radiation monitoring system in arrival yard.

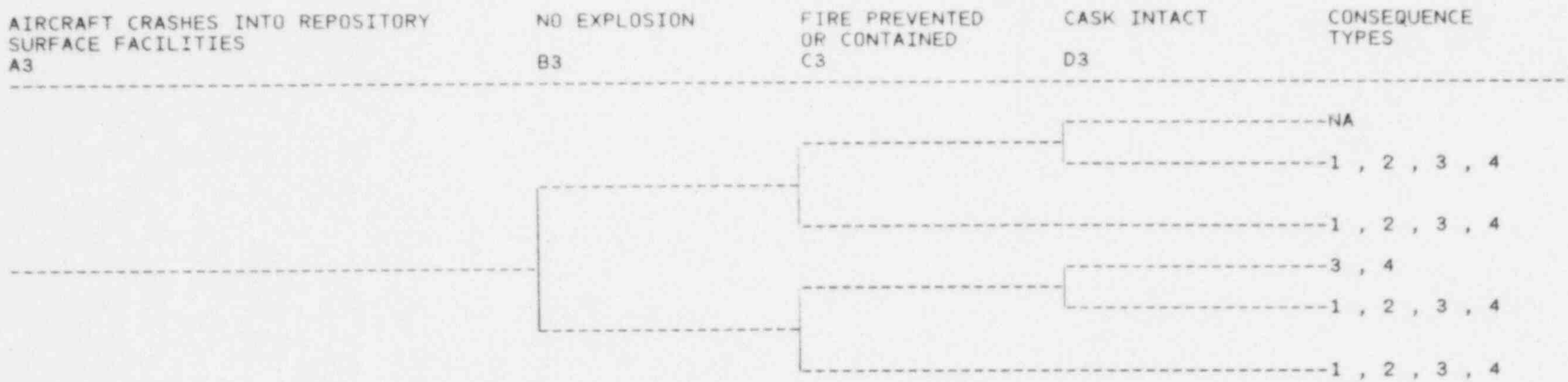

Fig. 2-24. Aircraft crash onto repository surface facilities.

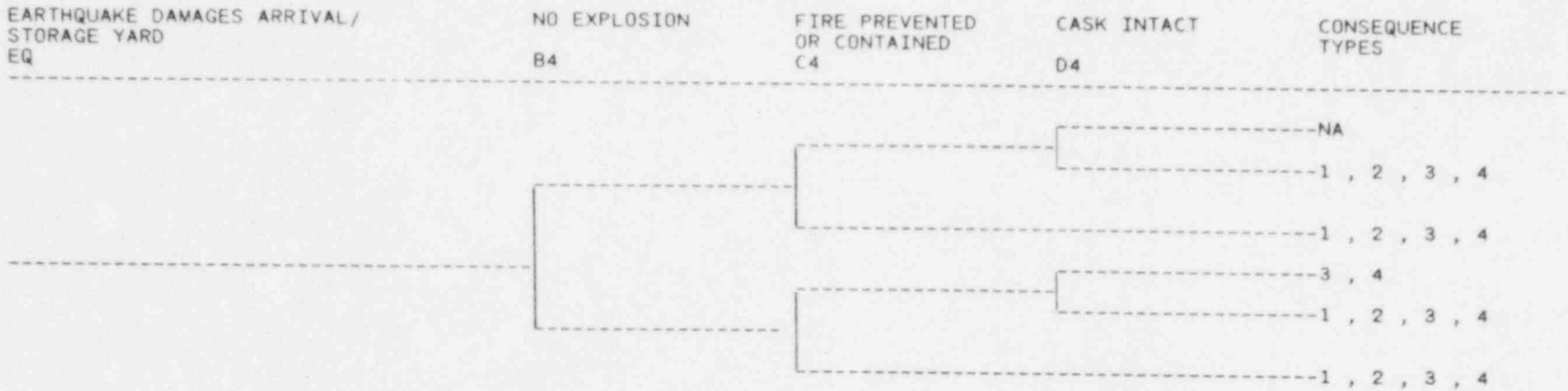

Fig. 2-25. Earthquake damages arrival/storage yard area or the washdown and receiving area.

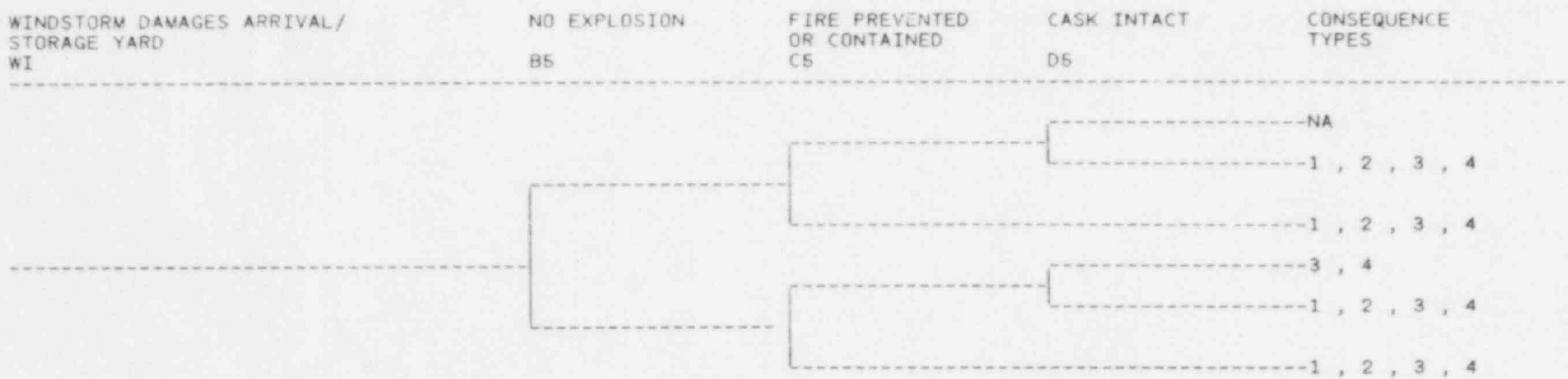

Fig. 2-26. Windstorm damages arrival/storage yard area or the washdown and receiving area.

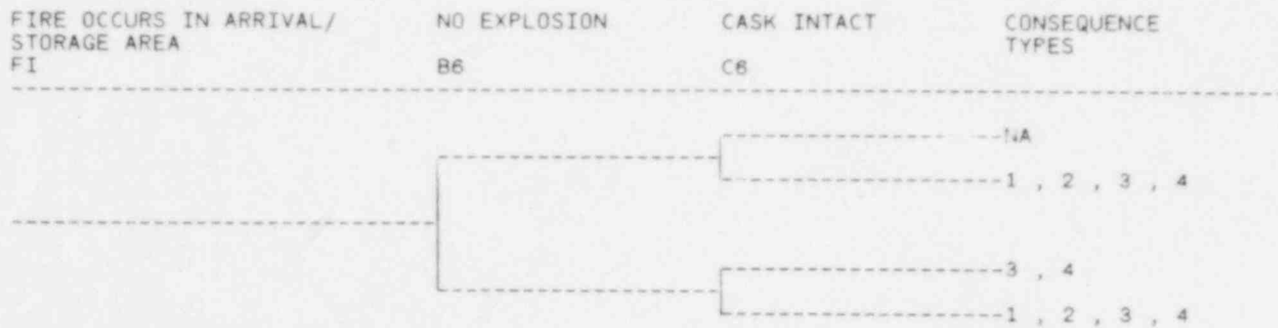

Fig. 2-27. Fire in the arrival/storage yard area or the washdown and receiving area.

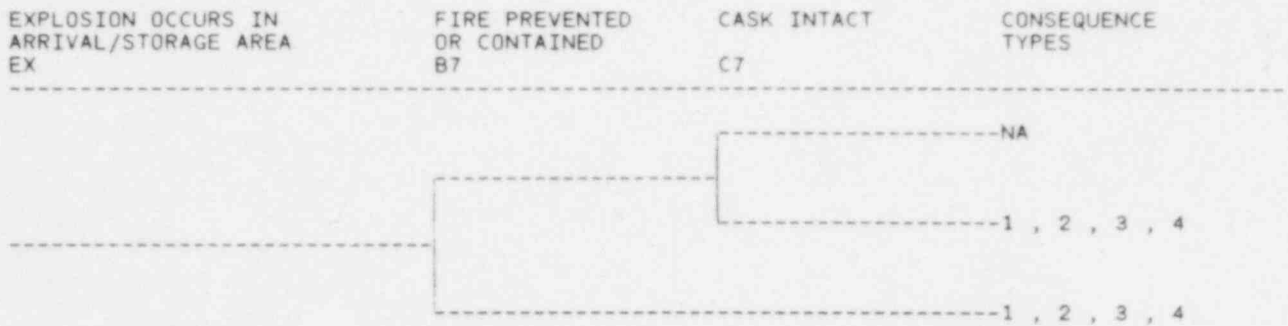

Fig. 2-28. Explosion in the arrival/storage yard area or the washdown and receiving area.

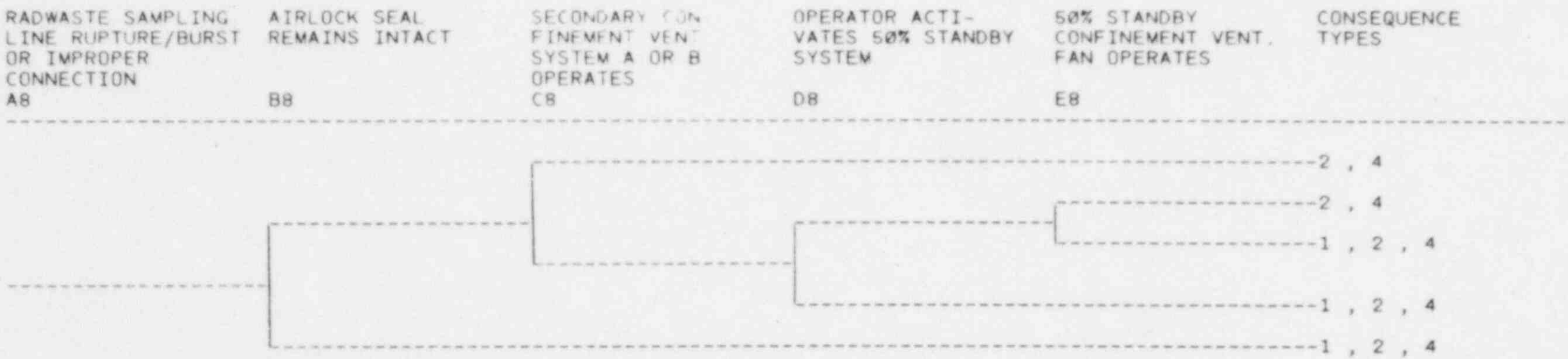

Fig. 2-29. Radwaste sampling line rupture or improper connect.

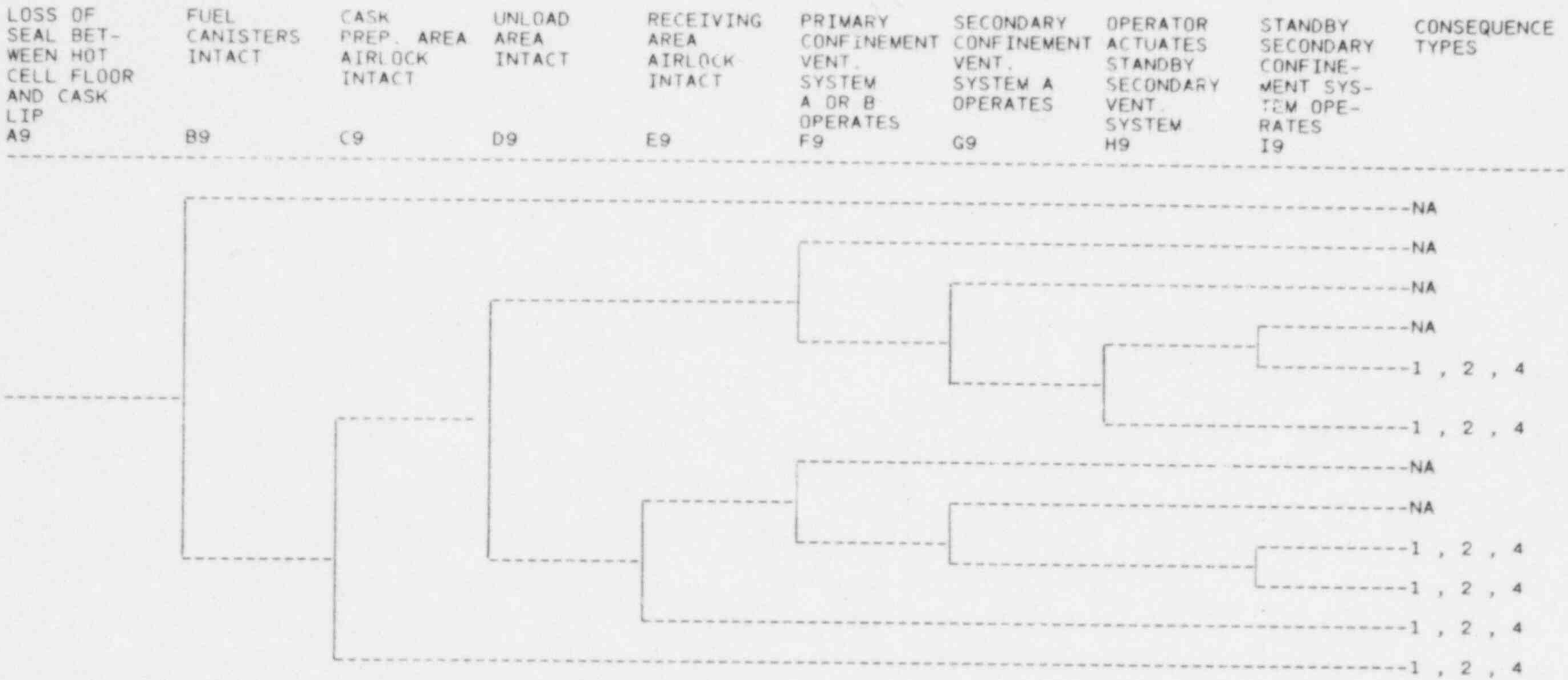

Fig. 2-30. Loss of seal between hot cell floor and cask lip during unloading operation.

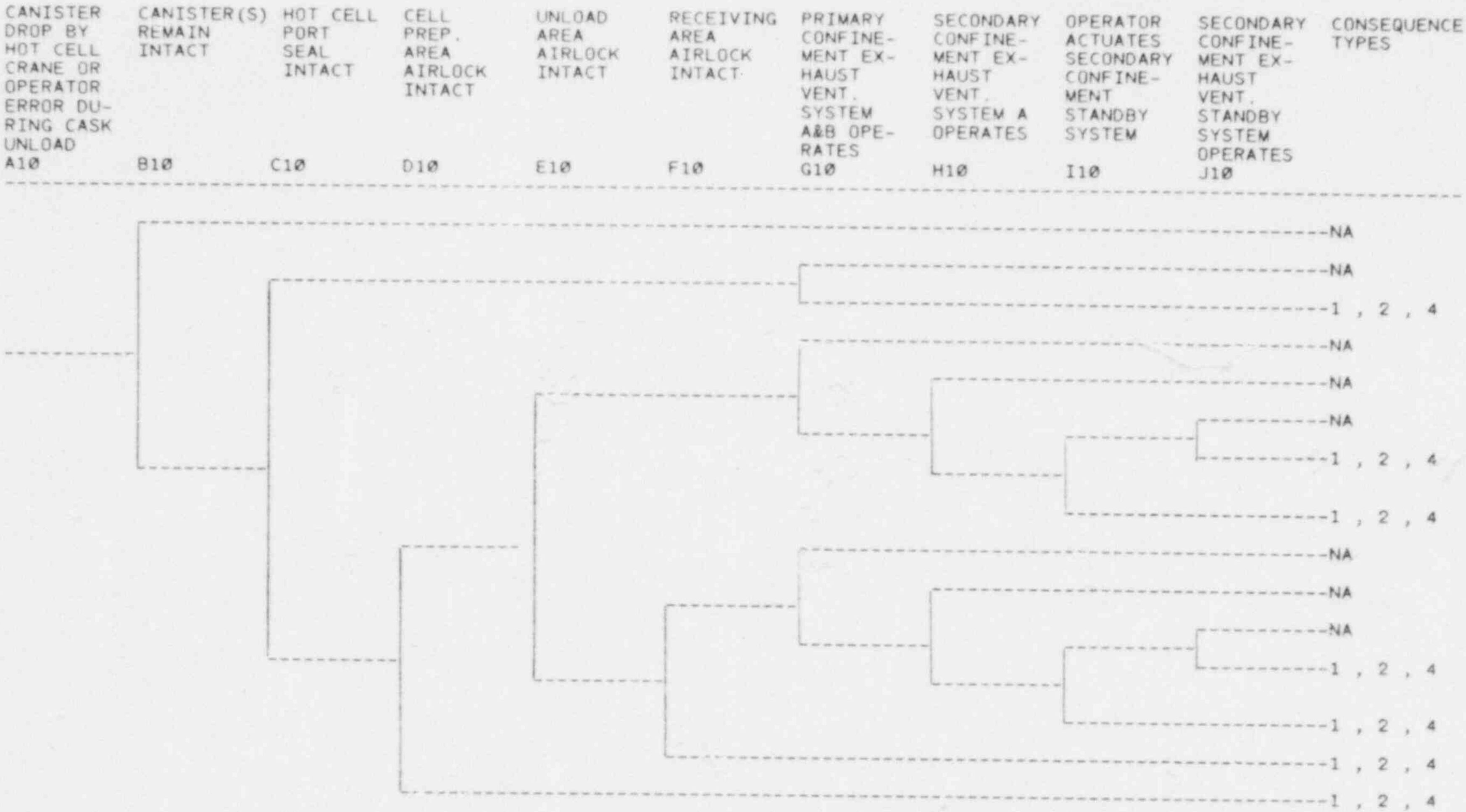

Fig. 2-31. Canister drop by hot cell crane or operator error during shipping cask unloading.

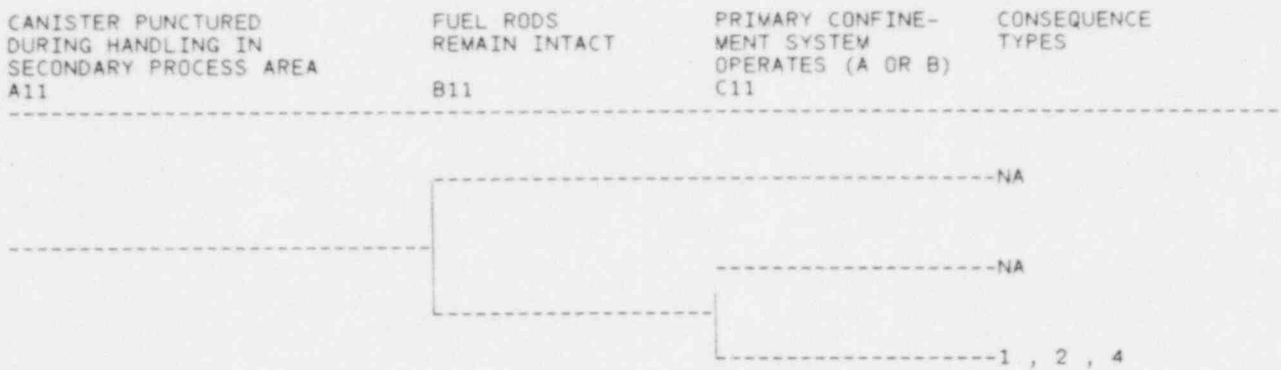

Fig. 2-32. Canister punctured during handling operations in the secondary process area.

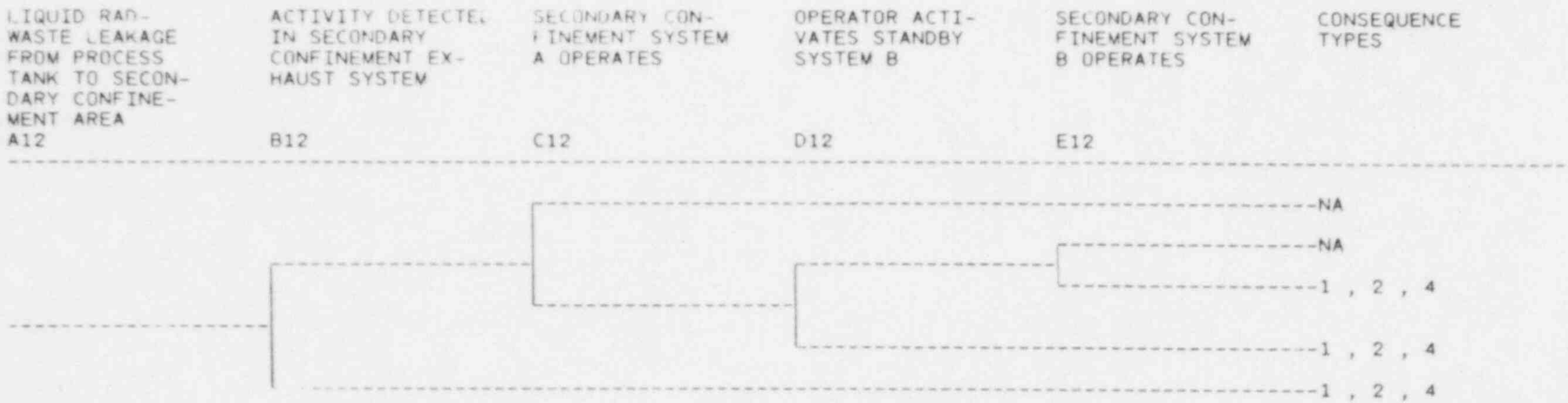

Fig. 2-33. Liquid radwaste leakage from process tanks to secondary confinement area.

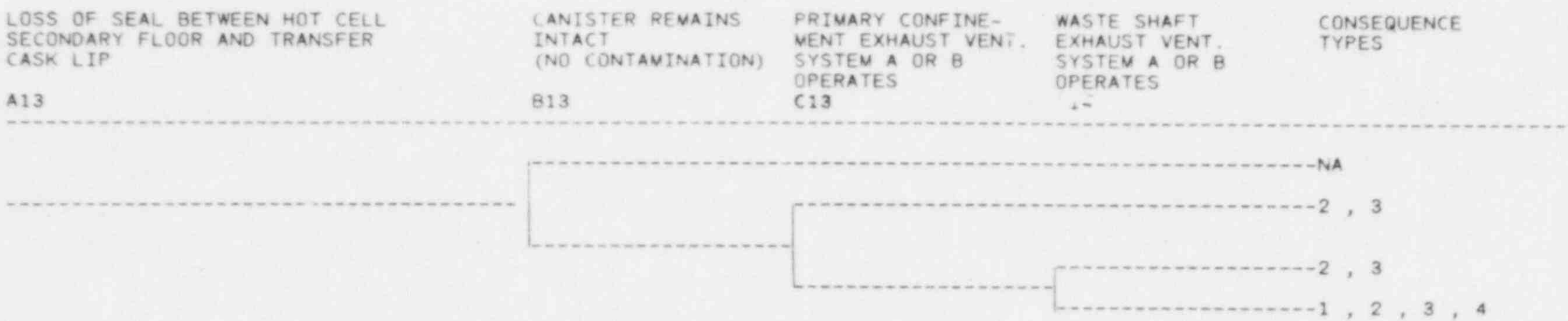

Fig. 2-34. Loss of seal between hot cell secondary area floor and transfer cask lip.

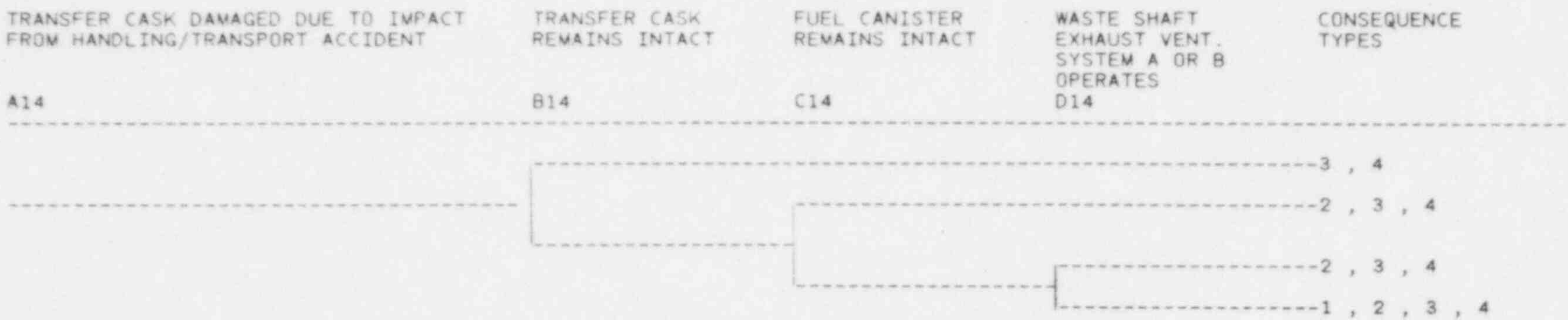

Fig. 2-35. Transfer cask damaged due to handling accident during transport from hot cell area to placement.

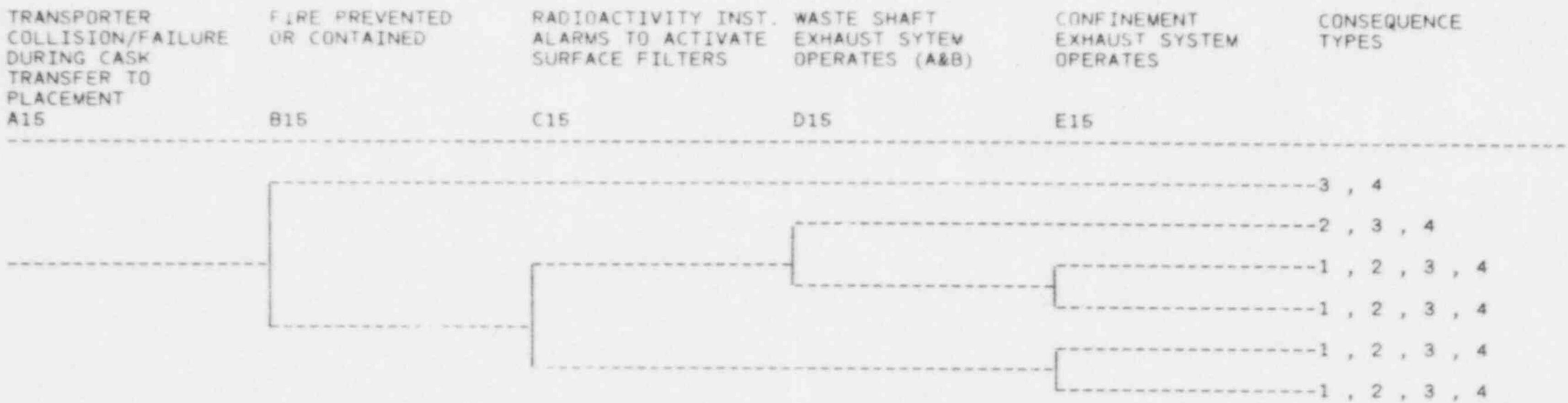

Fig. 2-36. Transporter collision during cask transport to placement.

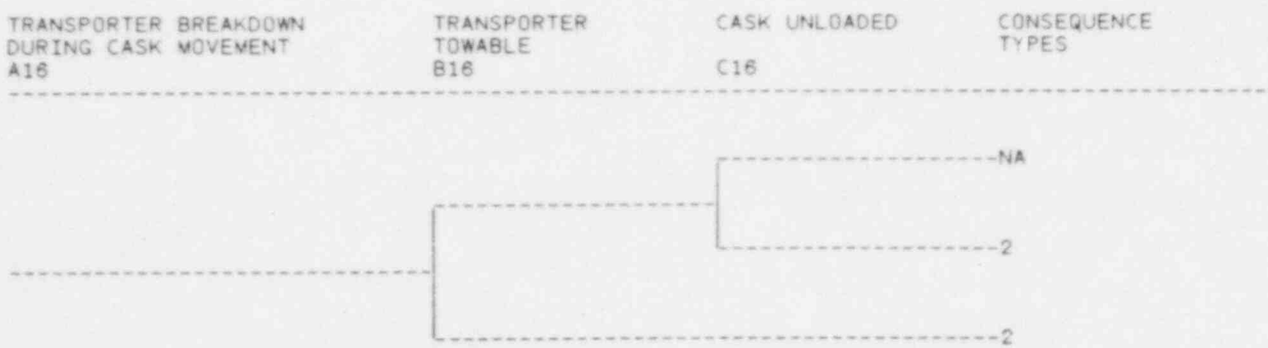

Fig. 2-37. Transporter breakdown during cask transport.

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CASK PREP AND ALIGNMENT          TRANSPORTER        CONSEQUENCE
TO BOREHOLE FAILURE OR ERROR     MOVABLE            TYPES
A17                              B17
----------------------------------------------------------------------

                                 --------------------NA
                                |
-----------------------------   |
                                |
                                 --------------------2
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Fig. 2-38. Preparation, alignment to borehole and removal failure or error.

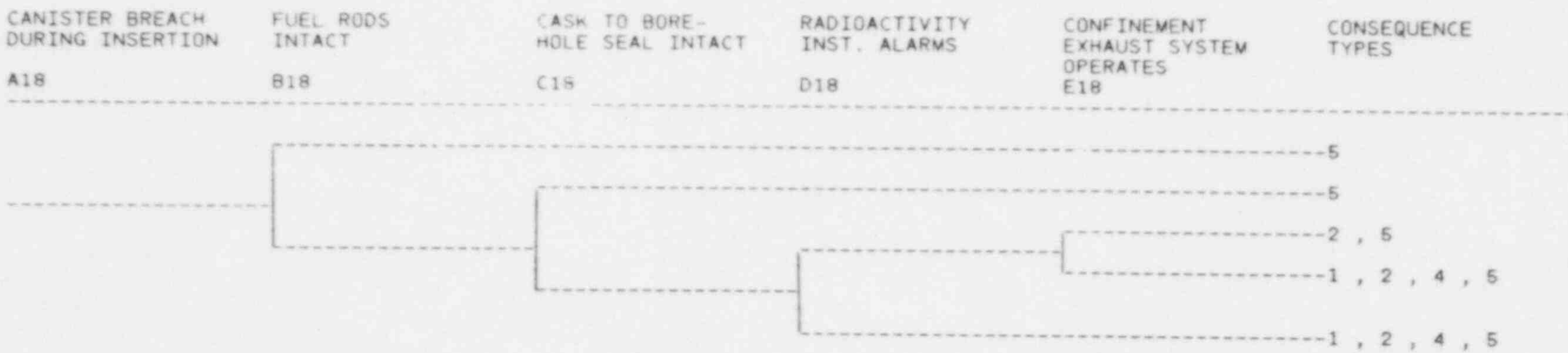


Fig. 2-39. Canister breach during insertion into borehole.

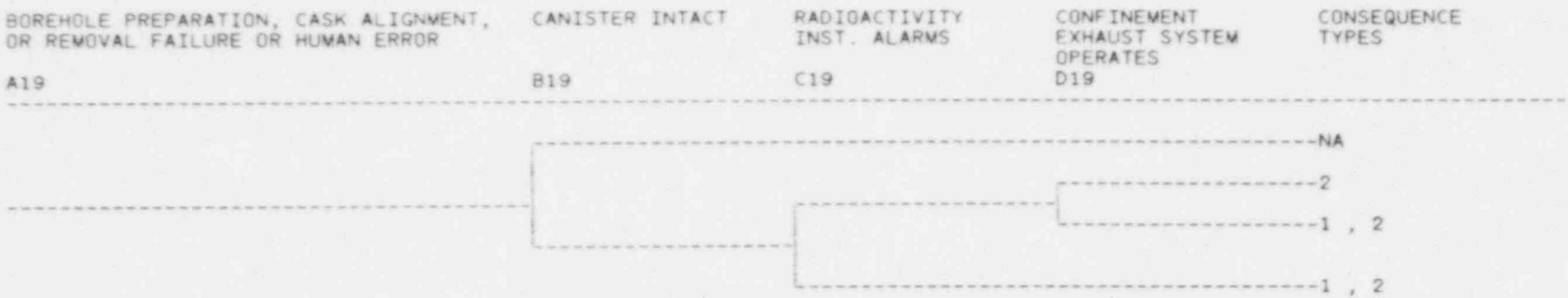


Fig. 2-40. Failure or error associated with borehole preparation, cask alignment, or removal for retrieval operation.

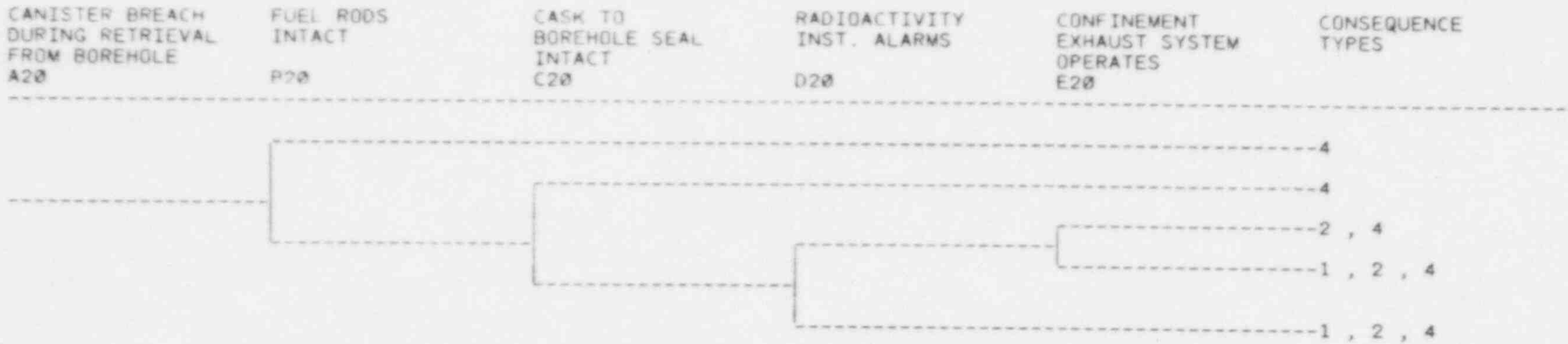


Fig. 2-41. Canister breach during retrieval from borehole.

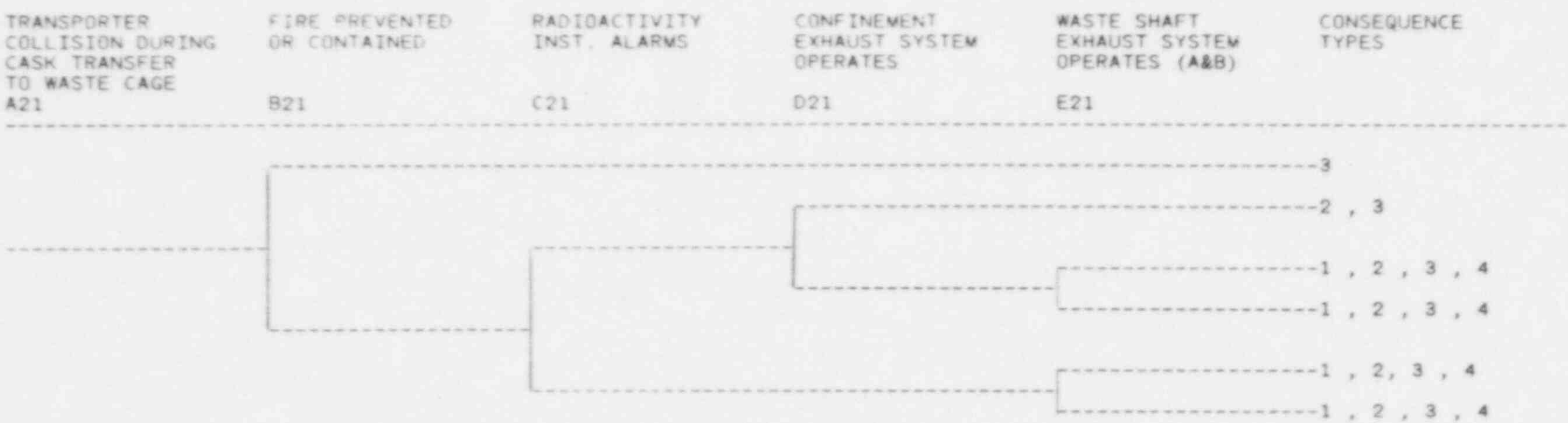


Fig. 2-42. Transporter collision during retrieval.

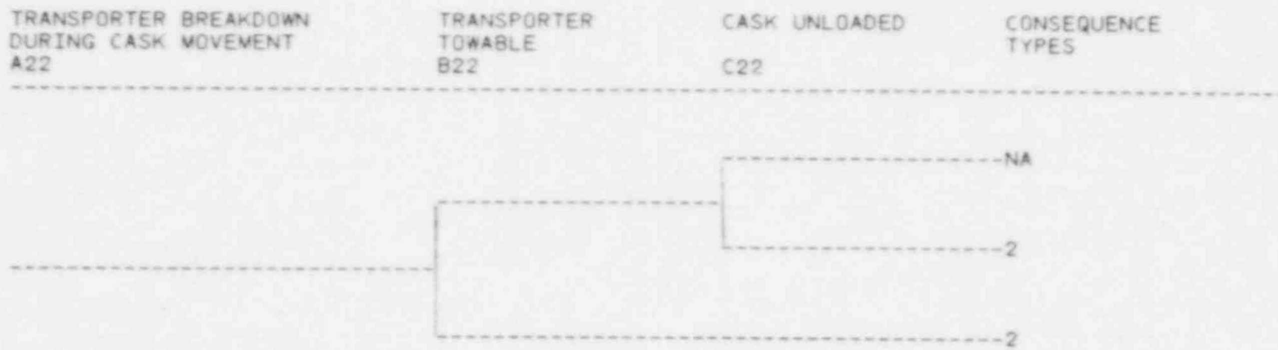


Fig. 2-43. Transporter breakdown during retrieval transport.

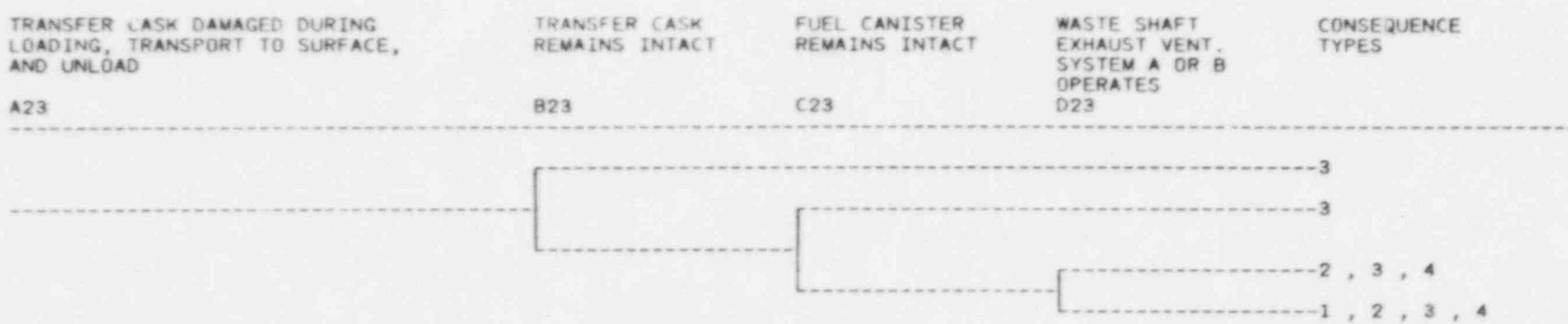


Fig. 2-44. Transfer cask damage during load, transport to surface, and unload.

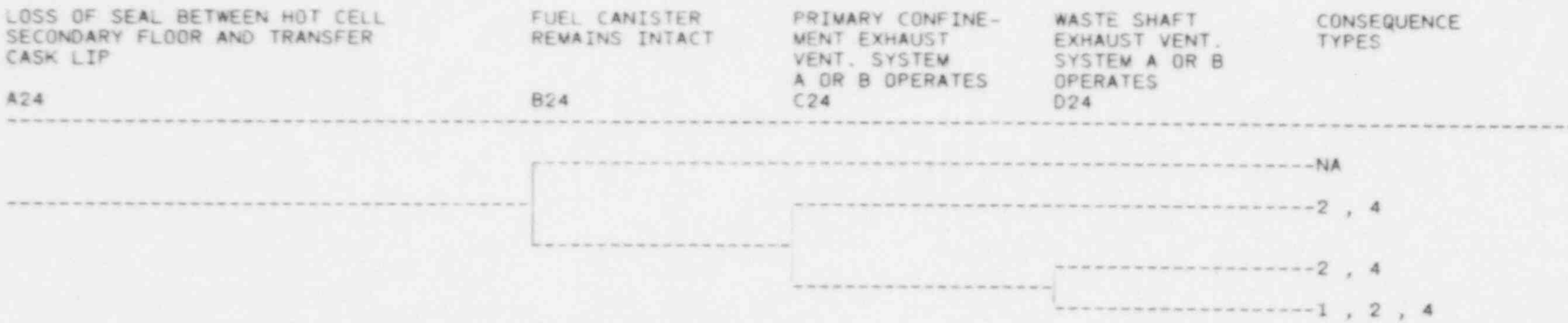


Fig. 2-45. Loss of seal between hot cell secondary floor and transfer cask lip - retrieval.

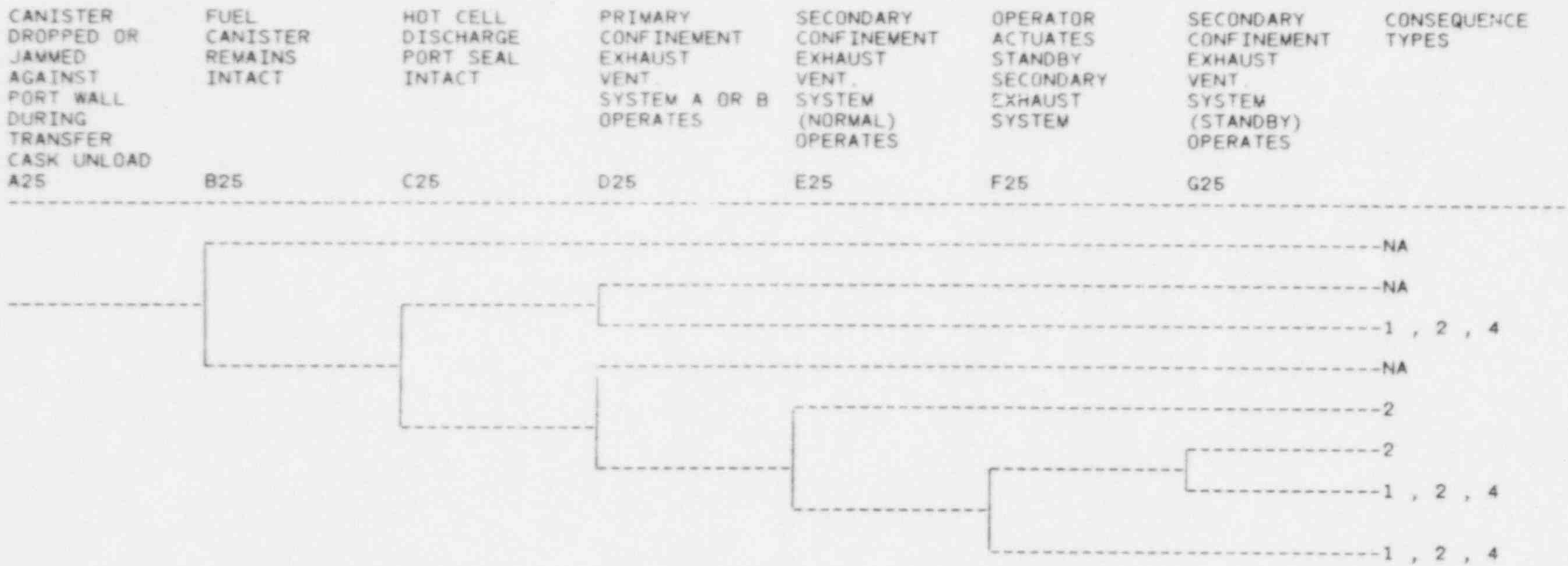


Fig. 2-46. Canister damaged during transfer cask unload.

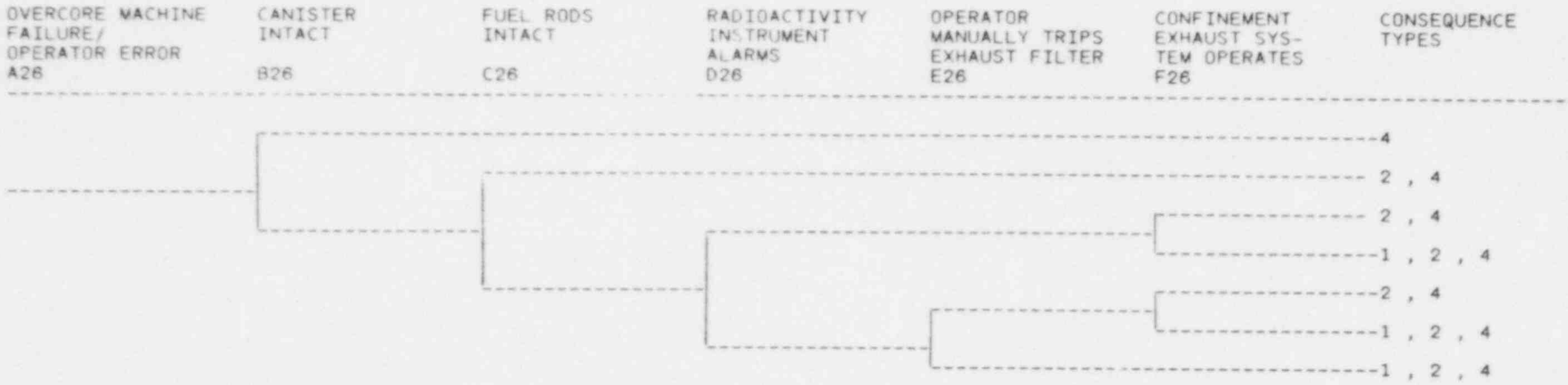


Fig. 2-47. Overcore machine failure during delayed retrieval.

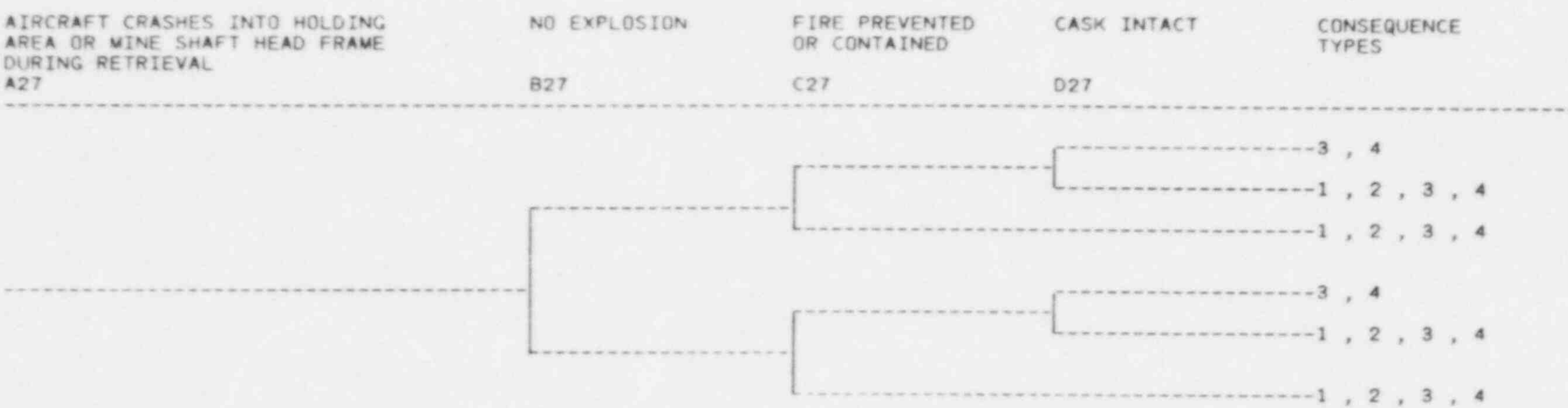


Fig. 2-48. Aircraft crash into holding area or mine shaft head frame from retrieval operation.

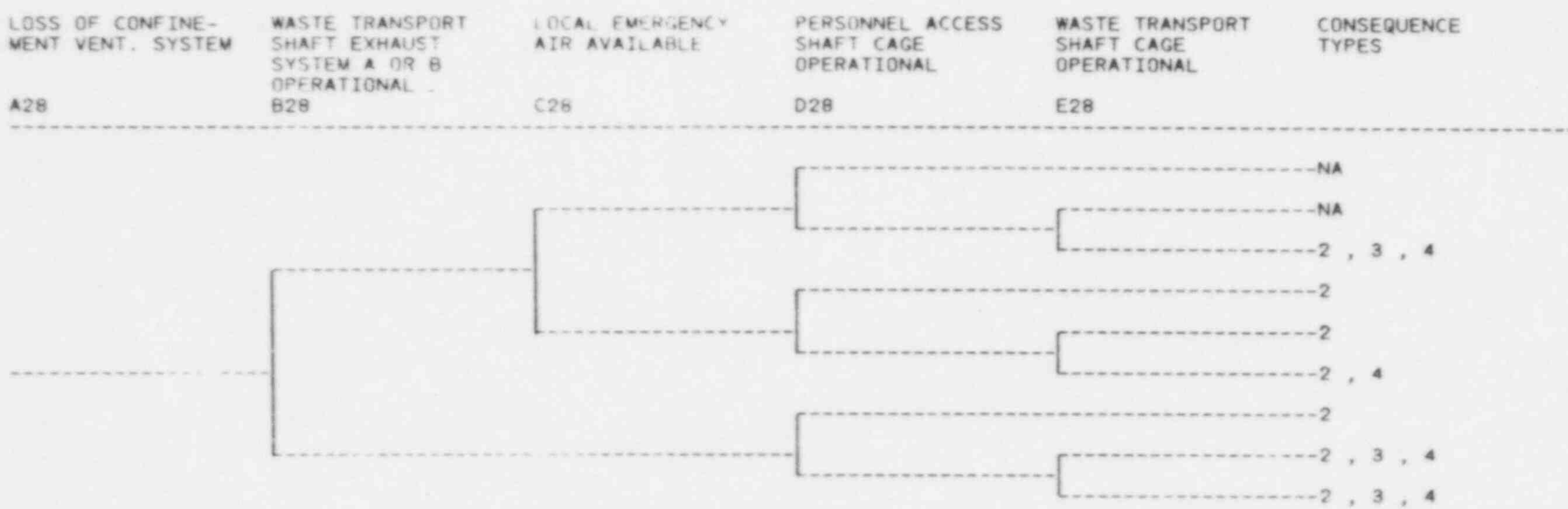


Fig. 2-49. Loss of confinement ventilation system during emplacement or retrieval operations.

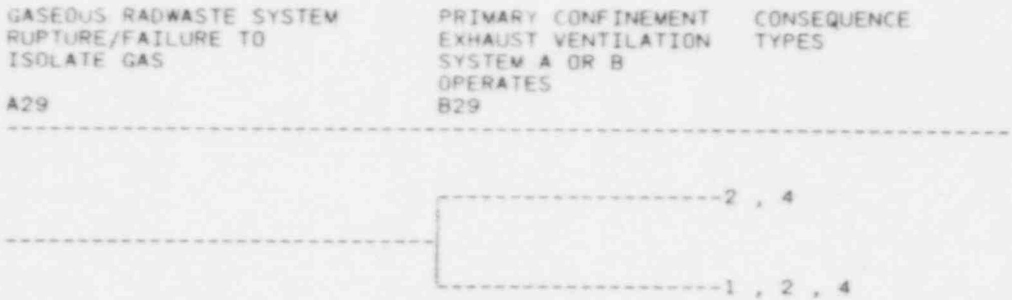


Fig. 2-50. Gaseous radwaste system rupture/failure to isolate gas.

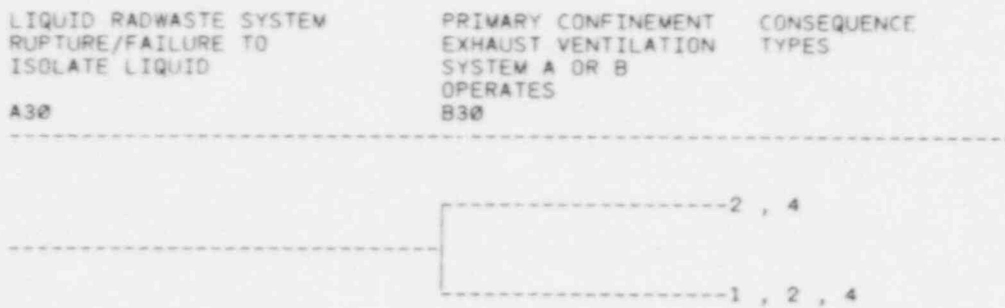


Fig. 2-51. Liquid radwaste system rupture/failure to isolate liquid.

TABLE 2-11
UNLOADING AREA EXTERNAL EVENT CONSIDERATIONS

Event Tree	External Event
Loss of Seal between Hot Cell Floor and Cask Lip	Earthquake Explosion in Unload Area
Canister Drop by Hot Cell Crane or Operator Error During Cask	Earthquake Explosion Fire

TABLE 2-12
SECONDARY AREA EXTERNAL EVENT CONSIDERATIONS

Event Tree	External Event
Canister Punctured During Handling in Secondary Process Area	Earthquake Explosion
Liquid Radwaste Leakage from Process Tank to Secondary Confinement Area	Earthquake Explosion Fire

TABLE 2-13
HOT CELL DISCHARGE TO PLACEMENT
EXTERNAL EVENT CONSIDERATIONS

Event Tree	External Event
Loss of seal between hot cell secondary floor discharge and transfer cask lip	Earthquake Explosion
Transfer cask damaged due to impact from handling/transport accident	Fire Explosion Earthquake Rock deformation (cave-in)
Transporter collision/failure during cask transfer to placement	Earthquake Explosion Fire Rock deformation Uplift or subsidence Subterranean flooding

TABLE 2-14
PLACEMENT AREA EXTERNAL EVENT CONSIDERATIONS

Event Tree	External Event
Cask preparation and alignment to borehole failure	Earthquake
	Rock deformation
	Flooding
	Fire
	Explosion
Canister breach during insertion	Earthquake
	Rock deformation
	Explosion

TABLE 2-15
RETRIEVAL - EXTERNAL EVENT CONSIDERATIONS

Event Tree	External Event
Borehole preparation, cask alignment and removal, borehole slugging	Earthquake Rock deformation Explosion
Canister breach during retrieval	Earthquake Rock deformation Explosion
Transporter collisions	Earthquake Rock deformation Fire Explosion
Transporter breakdown	Explosion Rock deformation Flooding Fire
Transfer cask damage during transport	Earthquake Rock deformation Fire Explosion
Loss of seal between hot cell secondary floor and transfer cask lip	Earthquake Explosion
Canister handling accident during transfer cask unload to hot cell secondary area	Earthquake Fire Explosion
Overcore machine failure	Earthquake Uplift/subsidence Rock deformation Flooding Fire Explosion

TABLE 2-16
SERVICE AND SUPPORT SYSTEM EXTERNAL EVENT CONSIDERATIONS

Event Tree	External Event
Loss of confinement exhaust ventilation system	Fire
	Explosion
	Earthquake
	Windstorm
	Lightning
Gaseous or liquid radwaste system rupture/failure to isolate	Fire
	Earthquake
	Explosion
Loss of offsite power*	Fire
	Explosion
	Earthquake
	Windstorm
	Lightning
	Flooding
Loss of standby diesel power*	Fire
	Explosion
	Earthquake

***These are contributing events.**

2.4 RESULTS OF ACCIDENT SCENARIO DEVELOPMENT

The results of the accident scenario development are grouped into accident sequences contributing to each of the five consequence types considered. In order to present the accident sequences in a concise and readable format, events were abbreviated as given in each tree. The appearance of an overbar (e.g., $\overline{B2}$) implies success of the system while the lack of an overbar denotes system failure.

Boolean algebraic reduction techniques have been used to further limit the sequences to only independent contributors. An independent accident sequence consists of the initiating event and the minimum number of additional intermediate event failures required to cause the consequence of interest. Many of the event trees contain multiple paths from the same initiating event to the same consequence type. If two accident sequences are the same (i.e., they contain the same intermediate event failures) except for one containing an additional failure, then this sequence is redundant and is discarded. Only sequences containing the minimum failures sufficient to generate the consequence type are retained. Accident sequences that contribute to each of the five consequence types are given in Tables 2-18 through 2-26. Abbreviations used for external event initiators are shown in Table 2-17. The sequences are divided into those that occur during normal emplacement operations and those that occur during retrieval. This will allow retrieval to be treated as an option in repository operations contributing an additional increment of risk.

Accident sequences generated from external events are included in the tables for each consequence type. It should be noted that:

1) Intermediate events following an external initiating event may have higher failure probabilities than those same intermediate events occurring in internal accident sequences. This is due to the fact that the occurrence of the external event may increase the failure probability of many key plant systems.

2) The data that will be used to quantify accident sequences generated by external events may indicate that some intermediate events have a failure probability of one (given the external event). This is not accounted for in the sequence listings.

3) Boolean reduction techniques have been applied to the accident sequences with the associated consequences as currently defined. Future work in phase II may subdivide a consequence type (e.g., radiological) into release categories (see Section 3). When these categories are explicitly defined, the accident sequence will require reassembly and additional reduction to adequately account for contributors to each level of release within a given consequence type. The independent accident sequence tables presented in this report may, therefore, be preliminary as each respective consequence type is treated as an entity in the reduction.

TABLE 2-17
ABBREVIATIONS USED FOR EXTERNAL EVENTS INITIATORS

EQ	EARTHQUAKE
EX	EXPLOSION
FI	FIRE
RD	ROCK DEFORMATION
US	UPLIFT/SUBSIDENCE
SF	SUBTERRANEAN FLOODING
WD	WINDSTORM
LI	LIGHTNING

TABLE 2-18
ACCIDENT SCENARIOS FOR PUBLIC RADIOLOGICAL EXPOSURE (CONSEQUENCE 1)
EMPLACEMENT

1.	A1	$\overline{B1}$	$\overline{C1}$	D1					
2.	A2	$\overline{B2}$	$\overline{C2}$	D2	$\overline{E2}$	F2			
3.	A2	$\overline{B2}$	$\overline{C2}$	D2	E2				
4.	A2	$\overline{B2}$	C2						
5.	A3	$\overline{B3}$	$\overline{C3}$	D3					
6.	A3	$\overline{B3}$	C3						
7.	EQ	$\overline{B4}$	$\overline{C4}$	D4					
8.	EQ	$\overline{B4}$	C4						
9.	WD	$\overline{B5}$	$\overline{C5}$	D5					
10.	WD	$\overline{B5}$	C5						
11.	FI	$\overline{B6}$	C6						
12.	EX	$\overline{B7}$	C7						
13.	EX	B7							
14.	A8	$\overline{B8}$	C8	$\overline{D8}$	E8				
15.	A8	$\overline{B8}$	C8	D8					
16.	A8	B8							
17.	A9	B9	$\overline{C9}$	$\overline{D9}$	F9	G9	$\overline{H9}$	I9	
18.	A9	B9	$\overline{C9}$	$\overline{D9}$	F9	G9	H9		
19.	A9	B9	$\overline{C9}$	D9	$\overline{E9}$	F9	G9	I9	
20.	A9	B9	$\overline{C9}$	D9	E9				
21.	A9	B9	C9						
22.	A10	B10	$\overline{C10}$	G10					
23.	A10	B10	C10	$\overline{D10}$	$\overline{E10}$	G10	H10	$\overline{I10}$	J10
24.	A10	B10	C10	$\overline{D10}$	$\overline{E10}$	G10	H10	I10	
25.	A10	B10	C10	$\overline{D10}$	E10	F10			
26.	A10	B10	C10	D10					
27.	A11	B11	C11						
28.	A12	$\overline{B12}$	C12	$\overline{D12}$	$\overline{E12}$				
29.	A12	$\overline{B12}$	C12	D12					
30.	A12	B12							

TABLE 2-18 (continued)

ACCIDENT SCENARIOS FOR PUBLIC RADIOLOGICAL EXPOSURE (CONSEQUENCE 1)

EMPLACEMENT

31.	A13	B13	C13	D13					
32.	A14	B14	C14	D14					
33.	A15	B15	$\overline{C15}$	D15	$\overline{E15}$				
34.	A15	B15	C15	$\overline{E15}$					
35.	A18	B18	C18	$\overline{D18}$	E18				
36.	A18	B18	C18	D18					
37.	A29	B29							
38.	A30	B30							
41.	EQ	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	$\overline{H9}$	I9
42.	EQ	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	H9	
43.	EQ	B9	$\overline{C9}$	D9	$\overline{E9}$	F9	G9		
44.	EQ	B9	$\overline{C9}$	D9	E9				
45.	EQ	B9	C9						
46.	EX	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	$\overline{H9}$	I9
47.	EX	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	H9	
48.	EX	B9	$\overline{C9}$	D9	$\overline{E9}$	F9	G9		
49.	EX	B9	$\overline{C9}$	D9	E9				
50.	EX	B9	C9						
51.	EQ	B10	$\overline{C10}$	D10					
52.	EQ	B10	C10	$\overline{D10}$	E10	F10			
53.	EQ	B10	C10	D10					
54.	EX	B10	C10	$\overline{D10}$	E10	F10			
55.	EX	B10	C10	D10					
57.	FI	B10	$\overline{C10}$	G10					
58.	FI	B10	C10	$\overline{D10}$	E10	F10			
59.	FI	B10	C10	D10					
60.	EQ	B11	C11						
61.	EX	B11	C11						
62.	EQ	$\overline{B12}$	C12	$\overline{D12}$	E12				
63.	EQ	$\overline{B12}$	C12	D12					
64.	EQ	B12							

TABLE 2-18 (continued)

ACCIDENT SCENARIOS FOR PUBLIC RADIOLOGICAL EXPOSURE (CONSEQUENCE 1)

EMPLACEMENT

65.	EX	$\overline{B12}$	C12	$\overline{D12}$	E12
66.	EX	$\overline{B12}$	C12	D12	
67.	EX	B12			
68.	FI	$\overline{B12}$	C12	$\overline{D12}$	E12
69.	FI	$\overline{B12}$	C12	D12	
70.	FI	B12			
71.	EQ	B13	C13	D13	
72.	EX	B13	C13	D13	
73.	EQ	B14	C14	D14	
74.	EX	B14	C14	D14	
75.	FI	B14	C14	D14	
76.	RD	B14	C14	D14	
77.	EQ	B15	$\overline{C15}$	D15	
78.	EQ	B15	C15		
79.	EX	B15	$\overline{C15}$	D15	
80.	EX	B15	C15		
81.	FI	B15	$\overline{C15}$	D15	
82.	FI	B15	C15		
83.	RD	B15	$\overline{C15}$	D15	
84.	RD	B15	C15		
85.	US	B15	$\overline{C15}$	D15	
86.	US	B15	C15		
87.	SF	B15	$\overline{C15}$	D15	
88.	SF	B15	C15		
89.	EQ	B18	C18	$\overline{D18}$	E18
90.	EQ	B18	C18	D18	
91.	EX	B18	C18	$\overline{D18}$	E18
92.	EX	B18	C18	D18	
93.	RD	B18	C18	$\overline{D18}$	E18
94.	RD	B18	C18	D18	
95.	EQ	B29			

TABLE 2-18 (continued)

ACCIDENT SCENARIOS FOR PUBLIC RADIOLOGICAL EXPOSURE (CONSEQUENCE 1)

EMPLACEMENT

96.	EX	B29
97.	FI	B29
98.	EQ	B30
99.	EX	B30
100.	FI	B30

TABLE 2-19

ACCIDENT SCENARIOS FOR PUBLIC RADIOLOGICAL EXPOSURE (CONSEQUENCE 1)

RETRIEVAL

1. A19 B19 $\overline{C19}$ D19
2. A19 B19 C19
3. A20 B20 C20 $\overline{D20}$ E20
4. A20 B20 C20 D20
5. A21 B21 C21 D21
6. A21 B21 C21
7. A23 B23 C23 D23
8. A23 B23 C23
9. A25 B25 $\overline{C25}$ D25
10. A25 B25 C25 D25 E25 $\overline{F25}$ G25
11. A25 B25 C25 D25 E25 F25
12. A26 B26 C26 $\overline{D26}$ $\overline{E26}$ F26
13. A26 B26 C26 D26 E26
14. A27 $\overline{B27}$ $\overline{C27}$ D27
15. A27 $\overline{B27}$ C27
16. A29 B29
17. A30 B30
18. EQ B19 $\overline{C19}$ D19
19. EQ B19 C19
20. EX B19 $\overline{C19}$ D19
21. EX B19 C19
22. RD B19 $\overline{C19}$ D19
23. RD B19 C19
24. EQ B20 C20 $\overline{D20}$ E20
25. EQ B20 C20 D20
26. EX B20 C20 $\overline{D20}$ E20
27. EX B20 C20 D20
28. RD B20 C20 $\overline{D20}$ E20
29. RD B20 C20 D20
30. EQ B21 $\overline{C21}$ D21

TABLE 2-19 (continued)

ACCIDENT SCENARIOS FOR PUBLIC RADIOLOGICAL EXPOSURE (CONSEQUENCE 1)

RETRIEVAL

31.	EQ	B21	C21				
32.	EX	B21	$\overline{\text{C21}}$	D21			
33.	EX	B21	C21				
34.	FI	B21	$\overline{\text{C21}}$	D21			
35.	FI	B21	C21				
36.	RD	B21	$\overline{\text{C21}}$	D21			
37.	RD	B21	C21				
38.	EQ	B23	C23	D23			
39.	EX	B23	C23	D23			
40.	FI	B23	C23	D23			
41.	RD	B23	C23	D23			
42.	EQ	B24	C24	D24			
43.	FI	B24	C24	D24			
44.	EQ	B25	C25	D25	E25	$\overline{\text{F25}}$	G25
45.	EQ	B25	C25	D25	E25	F25	
46.	EX	B25	C25	D25	E25	$\overline{\text{F25}}$	G25
47.	EX	B25	C25	D25	E25	F25	
48.	FI	B25	C25	D25	E25	$\overline{\text{F25}}$	G25
49.	FI	B25	C25	D25	E25	F25	
50.	EQ	B26	C26	$\overline{\text{D26}}$	$\overline{\text{E26}}$	F26	
51.	EQ	B26	C26	D26	E26		
52.	EX	B26	C26	$\overline{\text{D26}}$	$\overline{\text{E26}}$	F26	
53.	EX	B26	C26	D26	E26		
54.	FI	B26	C26	$\overline{\text{D26}}$	$\overline{\text{E26}}$	$\overline{\text{F26}}$	
55.	FI	B26	C26	D26	E26		
56.	RD	B26	C26	$\overline{\text{D26}}$	$\overline{\text{E26}}$	F26	
57.	RD	B26	C26	D26	E26		
58.	US	B26	C26	$\overline{\text{D26}}$	$\overline{\text{E26}}$	F26	
59.	US	B26	C26	D26	E26		
60.	SF	B26	C26	$\overline{\text{D26}}$	$\overline{\text{E26}}$	F26	
61.	SF	B26	C26	D26	E26		

TABLE 2-19 (continued)

ACCIDENT SCENARIOS FOR PUBLIC RADIOLOGICAL EXPOSURE (CONSEQUENCE 1)

RETRIEVAL

62.	EQ	B29
63.	EX	B29
64.	FI	B29
65.	EQ	B30
66.	EX	B30
67.	FI	B30

TABLE 2-20

ACCIDENT SCENARIOS FOR PERSONNEL RADIOLOGICAL EXPOSURE (CONSEQUENCE 2)

EMPLACEMENT

1.	A1	$\overline{B1}$	$\overline{C1}$	D1					
2.	A2								
3.	A3	$\overline{B3}$	$\overline{C3}$	D3					
4.	A3	$\overline{B3}$	C3						
5.	EQ	$\overline{B4}$	$\overline{C4}$	D4					
6.	EQ	$\overline{B4}$	C4						
7.	WD	$\overline{B5}$	$\overline{C5}$	D5					
8.	WD	$\overline{B5}$	C5						
9.	FI	$\overline{B6}$	C6						
10.	EX	$\overline{B7}$	C7						
11.	EX	B7							
12.	A8								
13.	A9	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	$\overline{H9}$	I9
14.	A9	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	H9	
15.	A9	B9	$\overline{C9}$	D9	$\overline{E9}$	F9	G9		
16.	A9	B9	$\overline{C9}$	D9	E9				
17.	A9	B9	C9						
18.	A10	B10	$\overline{C10}$	G10					
19.	A10	B10	C10	$\overline{D10}$	E10	F10			
20.	A10	B10	C10	D10					
21.	A11	B11	C11						
22.	A12	$\overline{B12}$	C12	$\overline{D12}$	E12				
23.	A12	$\overline{B12}$	C12	D12					
24.	A12	B12							
25.	A13	B13							
26.	A14	B14							
27.	A15	B15							
28.	A16	$\overline{B16}$	C16						
29.	A16	B16							

TABLE 2-20 (continued)

ACCIDENT SCENARIOS FOR PERSONNEL RADIOLOGICAL EXPOSURE (CONSEQUENCE 2)

EMPLACEMENT

30.	A17	B17							
31.	A18	B18	C18						
32.	A28	$\overline{B28}$	$\overline{C28}$	D28	E28				
33.	A28	$\overline{B28}$	C28						
34.	A28	B28							
35.	A29								
36.	A30								
37.	EQ	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	$\overline{H9}$	I9
38.	EQ	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	H9	
39.	EQ	B9	$\overline{C9}$	D9	$\overline{E9}$	F9	G9		
40.	EQ	B9	$\overline{C9}$	D9	E9				
41.	EQ	B9	C9						
42.	EX	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	$\overline{H9}$	I9
43.	EX	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	H9	
44.	EX	B9	$\overline{C9}$	D9	$\overline{E9}$	F9	G9		
45.	EX	B9	$\overline{C9}$	D9	E9				
46.	EX	B9	C9						
47.	EQ	B10	$\overline{C10}$	D10					
48.	EQ	B10	C10	$\overline{D10}$	E10	F10			
49.	EQ	B10	C10	D10					
50.	EX	B10	C10	$\overline{D10}$	E10	F10			
51.	EX	B10	C10	D10					
52.	EX	B10	C10	D10					
53.	FI	B10	$\overline{C10}$	G10					
54.	FI	B10	C10	$\overline{D10}$	E10	F10			
55.	FI	B10	C10	D10					
56.	EQ	B11	C11						
57.	EX	B11	C11						
58.	EQ	$\overline{B12}$	C12	$\overline{D12}$	E12				
59.	EQ	$\overline{B12}$	C12	D12					
60.	EQ	B12							

TABLE 2-20 (continued)

ACCIDENT SCENARIOS FOR PERSONNEL RADIOLOGICAL EXPOSURE (CONSEQUENCE 2)

EMPLACEMENT

61.	EX	$\overline{B12}$	C12	$\overline{D12}$	E12
62.	EX	$\overline{B12}$	C12	D12	
63.	EX	B12			
64.	FI	$\overline{B12}$	C12	$\overline{D12}$	E12
65.	FI	$\overline{B12}$	C12	D12	
66.	FI	B12			
67.	EQ	B13			
68.	EX	B13			
69.	EQ	B16			
70.	EX	B14			
71.	FI	B14			
72.	RD	B14			
73.	EQ	B15			
74.	EX	B15			
75.	FI	B15			
76.	RD	B15			
77.	US	B15			
78.	SF	B15			
79.	EQ	B17			
80.	EX	B17			
81.	FI	B17			
82.	RD	B17			
83.	SF	B17			
84.	EQ	B18	C18		
85.	EX	B18	C18		
86.	RD	B18	C18		
87.	EQ	$\overline{B28}$	$\overline{C28}$	D28	E28
88.	EQ	B28	C28		
89.	EQ	B28			
90.	EX	$\overline{B28}$	$\overline{C28}$	D28	E28
91.	EX	$\overline{B28}$	C28		

TABLE 2-20 (continued)

ACCIDENT SCENARIOS FOR PERSONNEL RADIOLOGICAL EXPOSURE (CONSEQUENCE 2)

EMPLACEMENT

92.	EX	B28			
93.	FI	$\overline{B28}$	$\overline{C28}$	D28	E28
94.	FI	$\overline{B28}$	C28		
95.	FI	B28			
96.	WD	$\overline{B28}$	$\overline{C28}$	D28	E28
97.	WD	$\overline{B28}$	C28		
98.	WD	B28			
99.	LI	$\overline{B28}$	$\overline{C28}$	D28	E28
100.	LI	$\overline{B28}$	C28		
101.	LI	B28			
102.	EQ	B29			
103.	EX	B29			
104.	FI	B29			
105.	EQ	B30			
106.	EX	B30			
107.	FI	B30			

TABLE 2-21

ACCIDENT SCENARIOS FOR PERSONNEL RADIOLOGICAL EXPOSURE (CONSEQUENCE 2)

RETRIEVAL

1. A19 B19
2. A20 B20 C20
3. A21 B21
4. A22 $\overline{B22}$ C22
5. A22 B22
6. A23 B23 C23
7. A24 B24
8. A25 B25 C25 D25
9. A26 B26
10. A27 $\overline{B27}$ $\overline{C27}$ D27
11. A27 $\overline{B27}$ C27
12. A28 C28
13. A28 B28 D28
14. EQ B19
15. EX B19
16. RD B19
17. EQ B20 C20
18. EX B20 C20
19. RD B20 C20
20. EQ B21
21. EX B21
22. FI B21
23. RD B21
24. EQ $\overline{B22}$ C22
25. EQ B22
26. EX $\overline{B22}$ C22
27. EX B22
28. FI $\overline{B22}$ C22
29. FI B22

TABLE 2-21

ACCIDENT SCENARIOS FOR PERSONNEL RADIOLOGICAL EXPOSURE (CONSEQUENCE 2)

RETRIEVAL

30.	SF	$\overline{B22}$	C22	
31.	SF	B22		
32.	RD	$\overline{B22}$	C22	
33.	RD	B22		
34.	EQ	B23	C23	
35.	EX	B23	C23	
36.	FI	B23	C23	
37.	RD	B23	C23	
38.	EQ	B24		
39.	EX	B24		
40.	EQ	B25	C25	D25
41.	EX	B25	C25	D25
42.	FI	B25	C25	D25
43.	EQ	B26		
44.	EX	B26		
45.	FI	B26		
46.	RD	B26		
47.	US	B26		
48.	SF	B26		
49.	EQ	C28		
50.	EQ	B28	D28	
51.	EX	C28		
52.	EX	B28	D28	
53.	FI	C28		
54.	FI	B28	D28	
55.	WD	C28		
56.	WD	B28	D28	
57.	LI	C28		
58.	LI	B28	D28	

TABLE 2-22

ACCIDENT SCENARIOS FOR PERSONNEL NONRADIOLOGICAL INJURY (CONSEQUENCE 3)

EMPLACEMENT

1.	A1	$\overline{B1}$	$\overline{C1}$	D1	
2.	A3				
3.	WD	$\overline{B5}$	$\overline{C5}$	D5	
4.	WD	$\overline{B5}$	C5		
5.	FI	$\overline{B6}$	C6		
6.	FI	B6	$\overline{C6}$		
7.	EX	$\overline{B7}$	C7		
8.	EX	B7			
9.	A13	B13			
10.	A14				
11.	A15				
12.	A28	$\overline{B28}$	$\overline{C28}$	D28	E28
13.	A28	B28	D28		
14.	EQ	B13			
15.	EX	B13			
16.	EQ				
17.	EX				
18.	FI				
19.	RD				

TABLE 2-23

ACCIDENT SCENARIOS FOR PERSONNEL NONRADIOLOGICAL INJURY (CONSEQUENCE 3)

RETRIEVAL

1. A21
2. A23
3. A27
4. EQ
5. EX
6. RD
7. FI B21

TABLE 2-24

ACCIDENT SCENARIOS FOR LOSS OF REPOSITORY AVAILABILITY (CONSEQUENCE 4)

EMPLACEMENT

1. A2 $\overline{B2}$ $\overline{C2}$ D2 $\overline{E2}$ F2
2. A2 $\overline{B2}$ $\overline{C2}$ D2 E2
3. A2 $\overline{B2}$ C2
4. A3
5. WD $\overline{B5}$ $\overline{C5}$ D5
6. WD $\overline{B5}$ C5
7. FI $\overline{B6}$ C6
8. FI B6 $\overline{C6}$
9. EX $\overline{B7}$ C7
10. EX B7
11. A8
12. A9 B9 $\overline{C9}$ $\overline{D9}$ $\overline{E9}$ F9 G9 $\overline{H9}$ I9
13. A9 B9 $\overline{C9}$ $\overline{D9}$ $\overline{E9}$ F9 G9 H9
14. A9 B9 $\overline{C9}$ D9 $\overline{E9}$ F9 G9
15. A9 B9 $\overline{C9}$ D9 E9
16. A9 B9 C9
17. A10 B10 $\overline{C10}$ G10
18. A10 B10 C10 $\overline{D10}$ E10 F10
19. A10 B10 C10 D10
20. A11 B11 C11
21. A12 $\overline{B12}$ C12 $\overline{D12}$ E12
22. A12 $\overline{B12}$ C12 D12
23. A12 B12
24. A13 B13 C13 D13
25. A14
26. A15
27. A18 B18 C18 $\overline{D18}$ E18
28. A18 B18 C18 D18
29. A28 $\overline{B28}$ $\overline{C28}$ D28 E28
30. A28 B28 D28

TABLE 2-24

ACCIDENT SCENARIOS FOR LOSS OF REPOSITORY AVAILABILITY (CONSEQUENCE 4)

EMPLACEMENT

31.	A29								
32.	A30								
33.	EQ	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	$\overline{H9}$	I9
34.	EQ	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	H9	
35.	EQ	B9	$\overline{C9}$	D9	$\overline{E9}$	F9	G9		
36.	EQ	B9	$\overline{C9}$	D9	E9				
37.	EQ	B9	C9						
38.	EX	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	$\overline{H9}$	I9
39.	EX	B9	$\overline{C9}$	$\overline{D9}$	$\overline{E9}$	F9	G9	H9	
40.	EX	B9	$\overline{C9}$	D9	$\overline{E9}$	F9	G9		
41.	EX	B9	$\overline{C9}$	D9	E9				
42.	EX	B9	C9						
43.	EQ	B10	$\overline{C10}$	D10					
44.	EQ	B10	C10	$\overline{D10}$	E10	F10			
45.	EQ	B10	C10	D10					
46.	EX	B10	C10	$\overline{D10}$	E10	F10			
47.	EX	B10	C10	D10					
48.	EX	B10	C10	D10					
49.	FI	B10	$\overline{C10}$	G10					
50.	FI	B10	C10	$\overline{D10}$	E10	F10			
51.	FI	B10	C10	D10					
52.	EQ	B11	C11						
53.	EX	B11	C11						
54.	EQ	$\overline{B12}$	C12	$\overline{D12}$	E12				
55.	EQ	$\overline{B12}$	C12	D12					
56.	EQ	B12							
57.	EX	$\overline{B12}$	C12	$\overline{D12}$	E12				
58.	EX	$\overline{B12}$	C12	D12					
59.	EX	B12							
60.	FI	$\overline{B12}$	C12	$\overline{D12}$	E12				
61.	FI	$\overline{B12}$	C12	D12					

TABLE 2-24 (continued)

ACCIDENT SCENARIOS FOR LOSS OF REPOSITORY AVAILABILITY (CONSEQUENCE 4)

EMPLACEMENT

62.	FI	B12			
63.	EQ	B13	C13	D13	
64.	EX	B13	C13	D13	
65.	EQ				
66.	EX				
67.	FI				
68.	RD				
69.	US				
70.	SF				
71.	EQ	$\overline{B28}$	$\overline{C28}$	D28	E28
72.	EQ	B28	D28		
73.	EX	$\overline{B28}$	$\overline{C28}$	D28	E28
74.	EX	B28	D28		
75.	FI	$\overline{B28}$	$\overline{C28}$	D28	E28
76.	FI	B28	D28		
77.	WD	$\overline{B28}$	$\overline{C28}$	D28	E28
78.	WD	B28	D28		
79.	LI	$\overline{B28}$	$\overline{C28}$	D28	E28
80.	LI	B28	D28		
81.	EQ	B29			
82.	EX	B29			
83.	FI	B29			
84.	EQ	B30			
85.	EX	B30			
86.	FI	B30			

TABLE 2-25

ACCIDENT SCENARIOS FOR LOSS OF REPOSITORY AVAILABILITY (CONSEQUENCE 4)

RETRIEVAL

1. A20 B20 C20
2. A21 B21 $\overline{C21}$ D21
3. A21 B21 C21
4. A23 B23 C23
5. A24 B24
6. A25 B25 C25 D25 E25 $\overline{F25}$ G25
7. A25 B25 C25 D25 E25 F25
8. A26
9. A27
10. A28 $\overline{B28}$ $\overline{C28}$ D28 E28
11. A28 B28 C28
12. A29 B29
13. A30 B30
14. EQ B20 C20
15. EX B20 C20
16. RD B20 C20
17. EQ B21 $\overline{C21}$ D21
18. EQ B21 C21
19. EX B21 $\overline{C21}$ D21
20. EX B21 C21
21. FI B21 $\overline{C21}$ D21
22. FI B21 C21
23. RD B21 $\overline{C21}$ D21
24. RD B21 C21
25. EQ B23 C23
26. EX B23 C23
27. FI B23 C23
28. RD B23 C23
29. EQ B24

TABLE 2-25 (continued)

ACCIDENT SCENARIOS FOR LOSS OF REPOSITORY AVAILABILITY (CONSEQUENCE 4)

RETRIEVAL

30.	EX	B24					
31.	EQ	B25	C25	D25	E25	$\overline{F25}$	G25
32.	EQ	B25	C25	D25	E25	F25	
33.	EX	B25	C25	D25	E25	$\overline{F25}$	GF25
34.	EX	B25	C25	D25	E25	F25	
35.	FI	B25	C25	D25	E25	$\overline{F25}$	G25
36.	FI	B25	C25	D25	E25	F25	
37.	EQ	$\overline{B28}$	$\overline{C28}$	D28	E28		
38.	EQ	B28	C28				
39.	EX	$\overline{B28}$	$\overline{C28}$	D28	E28		
40.	EX	B28	C28				
41.	FI	$\overline{B28}$	$\overline{C28}$	D28	E28		
42.	FI	B28	C28				
43.	WD	$\overline{B28}$	$\overline{C28}$	D28	E28		
44.	WD	B28	C28				
45.	LI	$\overline{B28}$	$\overline{C28}$	D28	E28		
46.	LI	B28	C28				
47.	EQ	B29					
48.	EX	B29					
49.	FI	B29					
50.	EQ	B30					
51.	EX	B30					
52.	FI	B30					

TABLE 2-26

ACCIDENT SCENARIOS FOR LONG-TERM EFFECTS (CONSEQUENCE 5)

1. A18 $\overline{B18}$
2. A18 B18 $\overline{C18}$
3. A18 B18 C18 $\overline{D18}$ $\overline{E18}$
4. A18 B18 C18 $\overline{D18}$ E18
5. A18 B18 C18 D18

3.0 CONSEQUENCE IDENTIFICATION

The identification and subsequent evaluation of radiological and nonradiological consequences resulting from abnormal operations or processes is a major element in the risk assessment of the preclosure phase of a geologic repository. The consequence can be expressed in terms of radiation dose, fatalities, dollars, etc. It is generally not a single value quantity but a distribution of values according to variations in weather, population, etc., and to uncertainties in the amount and conditions of radioactivity release.

There are several consequence types that are relevant to a repository's preclosure activities, namely:

1. radiological consequence to the public,
2. radiological consequence to the worker,
3. nonradiological consequence to the worker (i.e., occupational injury),
4. impact on repository availability,
5. compromise of a repository's ability for long-term geologic isolation of high-level waste (HLW), and
6. financial impact.

These consequence types may be grouped into two general categories: radiological and nonradiological consequences. The methods for evaluating each consequence category are quite different. Because of general lack of statistical data, radiological consequences usually require a numerical modeling approach to estimate the level of radionuclide release and the resulting radiation dose. On the other hand, nonradiological consequences (e.g., occupational injury, financial and availability impacts, etc.) are usually evaluated using statistical data based on past experience; however, nonradiological consequences can be radiation-induced and may be measured in monetary units. The different consequence types are interdependent, as illustrated in Fig. 3-1. Radiological consequences and occupational (nonradiological) injuries contribute to overall financial risk. Repository availability is affected by accidents and it in turn impacts financial risk.

The following subsections discuss phenomena for each of the consequence types in detail. This discussion is generally broader than the approach recommended for consequence quantification in the next study phase of this project. Each subsection contains a recommendation regarding the best approach for quantification of that particular consequence type. Subsection 3.6 presents the consequence types recommended for quantification in the next project phase.

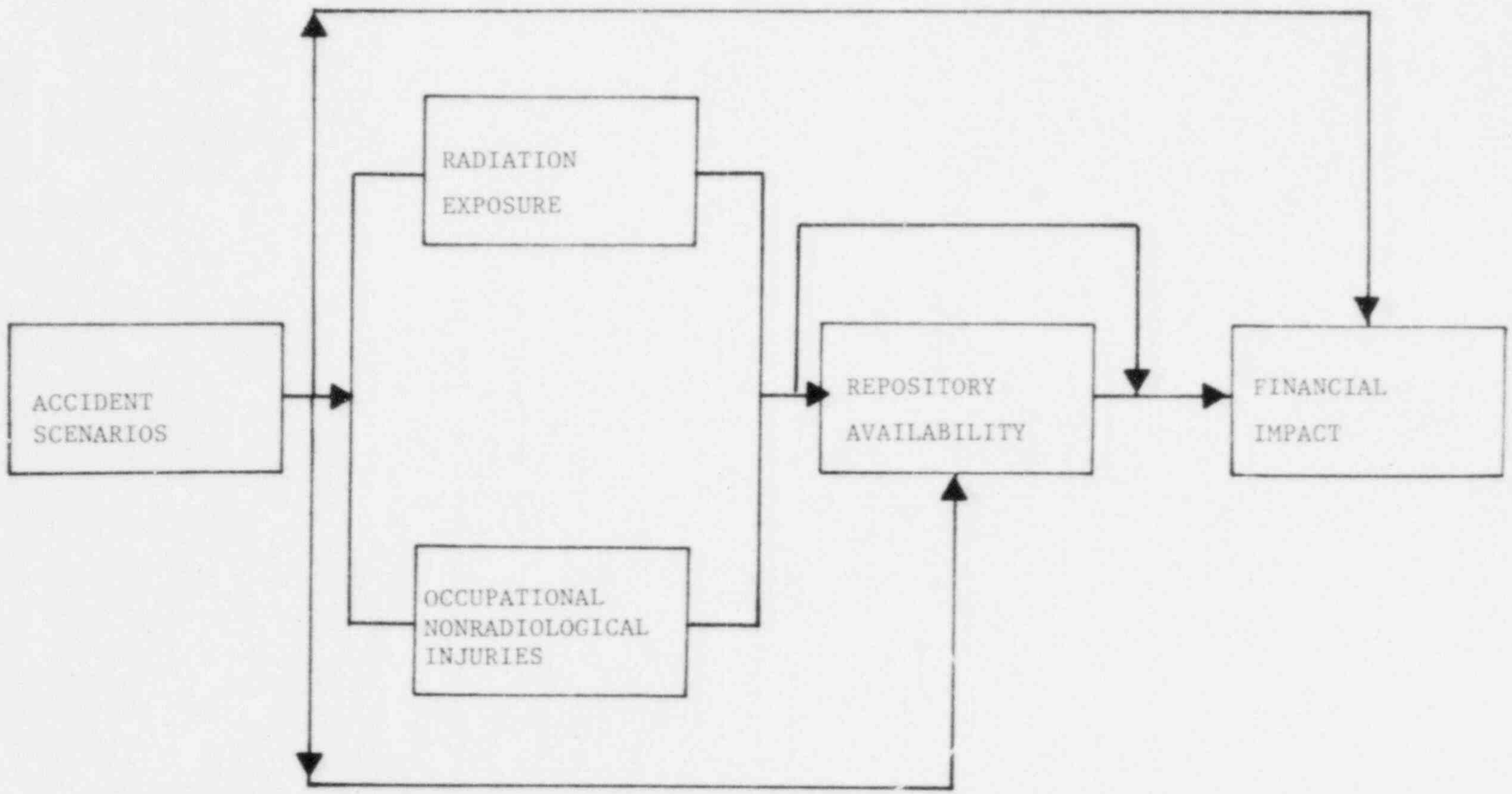


Fig. 3-1 Relationship of consequences applicable to preclosure repository activities.

3.1 RADIOLOGICAL CONSEQUENCE

Some accidents in a repository, whether initiated by external or internal events, can lead to the breach of containment barriers and release of radionuclides. The release may be airborne or transported via groundwater and the effects can be measured in terms of radiation dose or health effects (e.g., latent cancer deaths). The different waste forms in the repository (i.e., SURF, HLW, and TRU) have different levels of emissions and different radiological properties which affect the radionuclide release and transport mechanisms.

Spent fuel will be received at the basalt repository in a carbon steel canister. Each canister contains disassembled fuel elements of either 3 PWR or 7 BWR fuel assemblies. A PWR waste canister constitutes approximately 1.38 metric tons of heavy metal (MTHM) while a BWR waste container has approximately 1.32 MTHM.

The commercial HLW package is a stainless steel canister containing vitrified waste. It will be overpacked at the repository into a carbon steel cylinder. A HLW waste canister will contain approximately 2.28 MTHM.

The repository is designed to accept both spent fuel and HLW canisters (50/50 split) equivalent to 2370 MTHM/yr and 1,600 CHTRU drums/yr. Based on a 50/50 split between spent fuel and HLW, and a 20-yr schedule of receipt, 7400 BWR canisters (9,768 MTHM), 10,100 PWR canisters (13,938 MTHM), and 10,400 HLW canisters (23,712 MTHM) must be accommodated. Although the repository will store a large inventory of spent fuel assemblies and vitrified HLW (27,900 canisters), the radionuclide inventory per unit of stored waste is considerably less, and in a less releasable form than in a nuclear reactor. The bulk of biologically significant radioisotopes is in the form of solids or remain bound in the fuel matrix/vitrified HLW. The power density of the stored waste is also much lower than in a reactor. Therefore, reactivity considerations are unimportant. The time scale for heatup is relatively slower also because of the lower decay heat. In fact, heatup of the canisters due to decay heat alone is not physically possible. The term "heatup" denotes the raising of fuel temperatures to the point where release of activity from the canisters by one or more mechanisms described below becomes possible.

Although the source term per disruptive incident is smaller than in a reactor, we expect that more disruptive events could occur in the preclosure phase of the repository because of the many canister-handling operations involved over a long time interval. Understanding of the potential severity of radiological consequences in the preclosure phase of a repository is a necessary prerequisite to licensing, construction, and operation.

3.1.1 Radionuclide Classification and Characteristics

In order to effectively analyze the radiological consequences of repository accidents, it is convenient to classify the radionuclides on the basis of their physical forms at the time of the accident: gases, volatiles,

and solid particulates (nonvolatile fission products and neutron activation products). Table 3-1 lists the key radionuclides and their properties. A previous study performed at GA, (Seth, 1982) summarized the characteristics of the different radionuclides which are pertinent to the wastes that will be stored in the repository. They are given below.

3.1.2 Radioactive Gases and Volatiles

The principal radionuclides in this category are H-3, C-14, Kr-85, I-129, Ru-106, Cs-134, and Cs-137. These nuclides are present in significant quantities (except I-129) in the spent fuel, are soluble in water, and are biologically mobile. The quantities of these nuclides which can be released and their biological effects differ significantly. However, the first four of these radionuclides do constitute the primary sources of offsite doses due to minor accidents and routine venting of damaged fuel assemblies.

Gaseous tritium (H-3) and tritium oxide vapors released to the environment, mix rapidly with the ambient water and become part of the hydrologic cycle. C-14 is produced by the reactivation of nitrogen impurity in fuel. It will be released in the form of CO_2 and can either become incorporated in the plant material or washed out onto land and water surfaces. The noble gas Kr-85 present in the fuel pin void spaces (gaps) can leak out quickly from defective fuel elements. When released to the environment, it mixes rapidly throughout the atmosphere.

I-129 can be absorbed by plants, animals, and humans (thyroid), particularly in natural iodine-deficient locations. Although the I-129 inventory available for release is very small (for example, relative to Kr-85) its dose equivalent is many orders of magnitude greater. Of the other volatiles, Cs-137 dominates the overall inhalation and ingestion hazards. Ru-106 which forms the volatile oxide, RuO_4, is important during high-temperature oxidizing conditions. If RuO_4 is inhaled, it is deposited predominantly in the nasopharyngeal region. In a repository environment, the highly radioactive RuO4 will react with dust particles and reduce to RuO_2, which represents a much higher dose to the lungs following inhalation. However, the inventory of Ru-106 diminishes relatively rapidly as its half-life is only about one year.

3.1.3 Radioactive Particulates

Fuel dust (bearing fission products) could have a major impact on any radiation doses should it escape the storage facilities. Under large crushing forces, a fraction of the fuel can be expected to be ejected as an aerosol. Irradiated oxide fuel contains a variety of grain sizes (five to several hundred microns). The magnitude of dust formation and the particle size distribution are important since particles with Aerodynamic Equivalent Diameter (AED) of less than about 10 microns could be readily transported in the air flow and easily inhaled. The radiation hazard is dominated by the plutonium isotopes (particularly Pu-238) in the fuel and the fission product Sr-90. Radioactive products contained in the cladding and structural materials pose minor hazards as no significant fraction is likely to be released as respirable size particles.

TABLE 3-1

KEY RADIOACTIVE NUCLIDES

Radionuclide	Half-Life	Principal Exposure Mode
Gases		
H-3	2.3 y	Inhalation
C-14	5.73+3 y[a]	Inhalation
Kr-85	10.8 y	External
Volatiles		
I-129	1.60+7 y	Inhalation
Rn-106	1.0 y	Inhalation
Cs-134	2.1 y	Inhalation and external
Cs-137	30 y	Inhalation and external
Particulates		
Sr-90	29 y	Inhalation
Y-90	64 h	Inhalation
Pu-238	87.7 y	Inhalation
Pu-239	2.41+4 y	Inhalation
Pu-240	6.54+3 y	Inhalation
Pu-241	14.7 y	Inhalation
Am-241	432 y	Inhalation
Cm-244	18.1 y	Inhalation

(a) $5.73+3 = 5.73 \times 10^3$

3.1.4 Source Terms for Release

The radioactive source term available for release under accident conditions at a nuclear waste repository during preclosure activities depends on several general parameters:

- o total radionuclide activity,
- o physical properties of the release (e.g., volatiles, particles, liquids),
- o fraction of breached containers,
- o size of ruptures, and
- o fraction of aerosolized release.

The two types of waste forms of concern in this study are spent unreprocessed fuel (SURF) from commercial nuclear reactors and vitrified high-level waste (HLW).

In both cases, the radioactive material source of greatest concern consists of small particles liberated from the essentially monolithic waste matrix outside of the local canister containment and into the interior atmosphere of a repository or into the environment. Although these particles can be produced as a result of normal processes, the largest contribution is due to accidental loss. The particles of greatest health concern are the aerosolized particles having AED of 10 μm or less because they are regarded as "respirable." These particles require the most attention in calculating releases and transport during repository preclosure accidents. In addition, larger size particles deposited on the ground that can result in large doses of external exposure should also be considered (for example, Cs-137).

The liberation and migration of radionuclides from the waste form to the destination where they can cause health effects can be broadly considered to be a two-step process, involving the "release" and "transport." The two processes are related to some degree because certain types of transport are only possible for certain types of releases. Nevertheless, the separation is useful for the purpose of clarifying the presentation.

Under the conditions of the accident scenarios discussed in Section 2 of this report, the entire release process from either spent fuel or vitrified HLW can be regarded as a sequence of releases out of the inner containment (canister or assembly), out of the outer containment (cask), out of the repository and into the environment. The third step in the sequence is bypassed when the accident does not occur inside the repository. For occupational exposure, the greatest health concern is due to release inside confined repository volumes where operating personnel have access. For population exposure, direct releases into the environment are the most important, although releases through filter ventilation systems must also be evaluated.

The barriers, release mechanisms and pathways for both SURF and HLW are shown diagramatically in Fig. 3-2. The most important release mechanism is rupture which can be of essentially two types: impact and burst. Impact rupture is a mechanical disruption process caused by impact (fall, missile, etc.). Burst rupture occurs only when internal pressure (e.g., due to thermal environments) can build up to the point where the canister containment fails. Other release processes like diffusion, leaching, oxidation, and crud release tend to occur subsequent to a rupture and some are only possible following a rupture.

3.1.5 Radionuclide Release Models

The release models discussed here are specifically for particulate release. Release fractions for gases and volatiles have been characterized to a large extent in previous studies (Wilmot, 1980; Wilmot, 1981; Walker, 1978). Thus, no new modeling is required for this group of radionuclides and the data available in the literature will be used in consequence evaluations. Although release fractions have been estimated in previous waste management studies, large data uncertainties exist because of limited data and the accident-specific nature of the release.

The release of a particular radionuclide to the environment depends on such factors as the mechanistic nature of the event, the physical and chemical properties of the radionuclide, its plateout characteristics, and the ability of the ventilation exhaust system to contain the release.

The important canister/fuel element release models which require special computational attention are impact rupture, burst rupture, and diffusion. Leaching, oxidation, and crud release can be evaluated parametrically.

Impact rupture is the release of radioactive material out of a nonintact canister by the mechanical disruption of the cladding and subsequent depressurization of the fuel element. The mechanical force of an impact can cause a fuel element to bend or become punctured. As the fill gas and fission gases vent through a breach in the cladding, a driving force is produced which carries along other materials contained in the fuel-clad gap.

The rupture is a threshold process which occurs when a certain amount of mechanical energy is transferred to the waste container in the form of impact. The likelihood of a rupture depends on a number of properties including the material properties of the waste form and of the containers and the mode and spot of impact. As such, it is virtually impossible to express the impact rupture release through an all-inclusive mathematical expression. It is possible, however, to develop a relatively simple semiempirical expression to illustrate dependence on important parameters.

The particulate release through the break in the cladding or the canister can be calculated in the form of a mass release which can then be converted to activity based on isotopic composition.

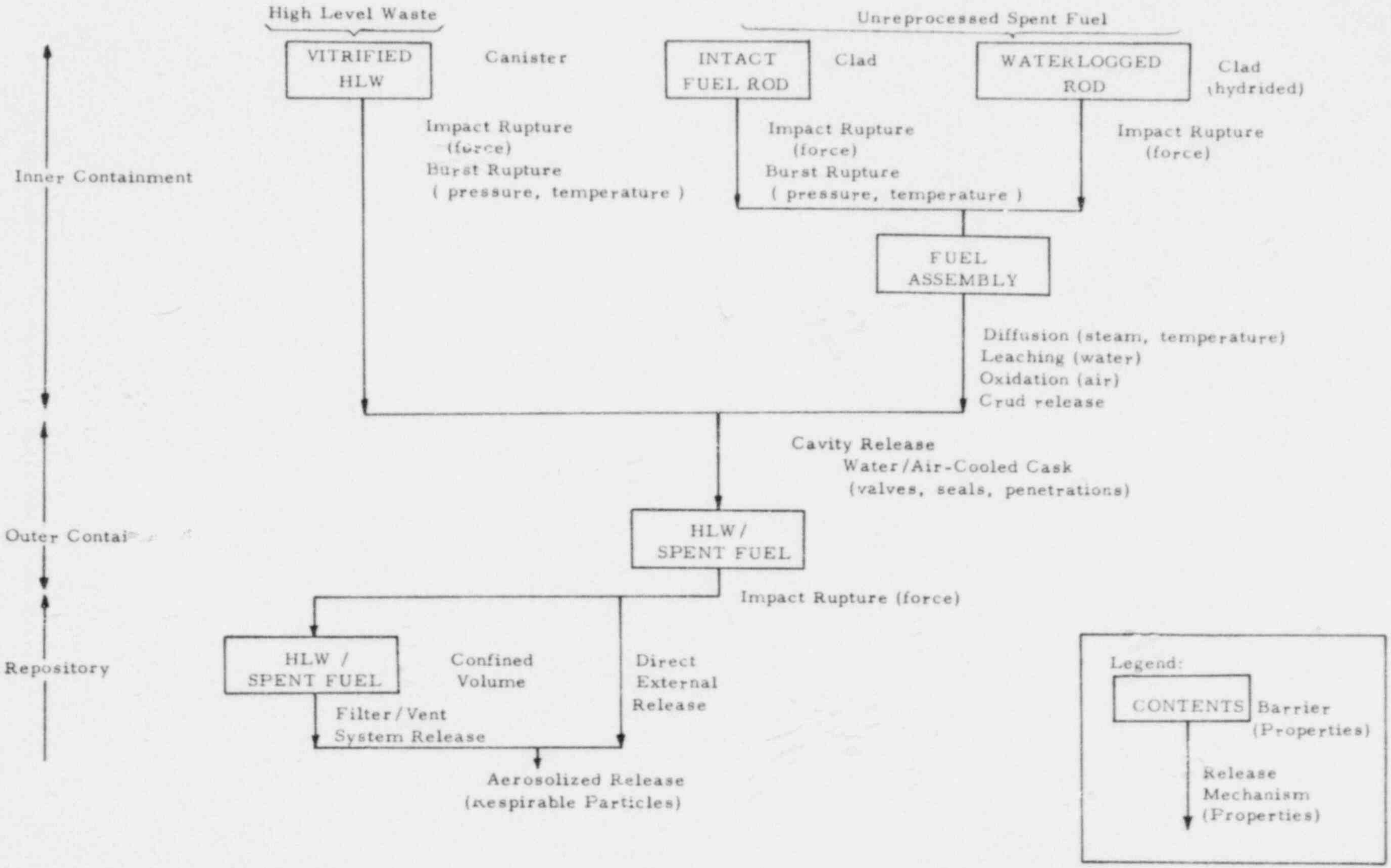


Figure 3-2 HLW/Spent fuel release mechanisms, barriers and pathways.

Thus,

$$M_I = \int_0^{t_o} \int_0^{r_o} \frac{M_a(r,t)}{f_d(r,t)} \, dr \quad f_{am}(t) \, dt \qquad (3\text{-}1)$$

where

M_I - mass release due to impact rupture

$M_a(r,t)$ - aerosolized mass within the opening for particle size (radius) r at time t

$f_d(r,t)$ - decontamination factor (within the opening) for particle of size (radius) r at time t

$f_{am}(t)$ - fraction of air mass released from the canister at time t

r_o - maximum size of aerosol particle corresponding to half of the diameter of the break

t_o - total time period for the release calculation.

$M_a(r,t)$, the aerosolized mass within the opening for particles of radius r at time t, can be expressed as the product of several quantities:

$$M_a(r,t) = M_t(t) \, . \, N(r) \, . \, f \, . \, L \, . \, a \qquad (3\text{-}2)$$

where

$M_t(t)$ - total fuel mass within the inner containment at time t

$N(r)$ - normalized particle distribution; the lognormal distribution is adequate

f - fraction of failed inner containers (equal to unity if calculation is performed for inner containment; e.g., fuel element)

L - fraction leaking from breached inner containment

a - aerosolized fraction.

The differential fraction of air mass being released from the canister at time t is

$$f_{am}(t) = \frac{dm_a(t)}{m_a(t)} = \frac{dP_c(t)}{P_c(t)} \qquad (3\text{-}3)$$

where

$m_a(t)$ - mass of air in canister at time t

$P_c(t)$ - internal canister pressure at time t.

The decontamination factor, $f_d(r,t)$, is the particle depletion factor due to diffusion during flow and it can be calculated from the following expression

$$\log f_d(r,t) = C_1 + C_2 \left\{ \frac{\mu d(r)\, \ell^2(t)}{\Delta P(t) D^4} \right\} \qquad (3\text{-}4)$$

where

μ - air viscosity

$d(r)$ - diffusion coefficient for particle of radius r

$\ell(t)$ - leak hole length at time t

$\Delta P(t)$ - drop in pressure along leak path

D - leak hole diameter

C_1 and C_2 are constants.

The burst rupture mechanism produces releases analogous to impact rupture but driven by internal pressure rather than external forces. It occurs in a severe thermal environment, whereas impact rupture occurs in a severe impact environment. As spent fuel is heated, internal pressure will increase until the cladding balloons and then bursts.

Once a spent fuel element is ruptured, vaporized fission products can diffuse into the fuel-clad gap and out of the rupture opening. Higher temperatures increase the likelihood of this diffusion mechanism.

The burst rupture release can be calculated according to the model of Lorenz, et. al. (Lorenz, 1980).

$$M_B = \alpha V_B (M_o/A)^a \exp - (c/T) \qquad (3\text{-}5)$$

where

M_B - mass released in a burst

V_B - volume of plenum gas vented

M_o - gap inventory of released element

A - internal containment (clad) area

T - temperature at rupture location, and

α, a and c - are fitted parameters based on experimental data.

The diffusion mechanism in steam is only important if steam is present; e.g., water-cooled cask in a fire accident.

The release for this mechanism can be evaluated according to a model by Lorenz, et. al. (Lorenz, 1980).

$$M_D = M_o \quad 1 - \exp - \quad [(R_o t/M_o)] \qquad (3\text{-}6)$$

where

M_D - mass released by diffusion

t - time interval at diffusion temperature

R_o - initial rate of release by diffusion.

The value of R_o can be calculated from the following expression

$$R_o = \delta (W/P) \, (M_o/A)^a \, \exp [-\gamma/T] \qquad (3\text{-}7)$$

where

W - radial gap width

P - system pressure, and

δ, γ, a - adjustable parameters based on experimental data.

The leaching process is important primarily under flooding conditions and only at high temperatures.

The crud release mechanism essentially releases no fission products because the fuel cladding does not have to rupture for a release to occur. Crud occurs as particulates in the water surrounding the fuel. Some fission products may deposit with the crud in small concentrations because they may be present in the reactor coolant or in solution in storage pools. However, the presence of Cobalt-60 in the crud is the major concern. This mechanism, therefore, assumes the release of only corrosion products (crud) either by impact, vibration, abrasion, or severe rapid thermal transients. Crud release depends on shock and temperature.

Oxidation is significant only for temperature conditions well in excess of 430°C.

Release fractions for leaching, crud release, and oxidation are given in the transportation accident scenarics report by Wilmot (Wilmot, 1981).

Radionuclides in vitrified HLW can be released as a result of canister impact. Physical impact can cause rupture of a cask and canister. The depressurization of the cask and canister could drive respirable particulates or volatiles from the cavity interior. However, volatilization is not a viable release mechanism for HLW as this waste form is almost completely devoid of gaseous radionuclides.

3.1.6 Exposure Pathways and Indivdual Doses

A generalized illustration of the pathways leading to radiation exposure of humans due to accidental release of radioactivity from the repository is given in Fig. 3-3. Important pathways for exposure to gasborne activity include direct (both external and inhalation) radiation from the passing cloud or plume as well as direct and indirect radiation from radioisotopes deposited on the ground (contamination). Indirect exposure refers to ingestion of foods which contain radionuclides and milk produced by consumption of contaminated pasture grass. Response or control measures can be taken to reduce both direct and indirect exposure, depending on the severity and timing of the accident. Typical measures for direct exposure control are sheltering, evacuation and iodine tablets. Measures for indirect exposure control include impounding of local milk and crops.

Important liquid pathways for human exposure due to groundwater contamination include: (a) direct radiation from consumption of fish and water, (b) direct radiation from radionuclides deposited on stream banks and sediments, and (c) indirect exposure from ingestion of crops irrigated with contaminated water. Of these, item (b) is often found to be the most significant. It includes exposure during recreational activities such as boating, swimming, and fishing. Liquid pathway exposures can be mitigated also by response or control measures.

3.1.6.1 Gasborne Pathways. The term "gasborne pathway" is used here to designate the release and transport of gases or particulates in the air. Affected are repository worker exposures in confined volumes of the repository as well as offsite exposure of the public in the vicinity of the passing radioactive plume. The plume can contain the following forms of radioactivity:

a) Noncondensible fission or activation product gases such as Kr-85 and tritium.

b) Volatile nongaseous radionuclides, which can be in gaseous compounds such as CO_2 or RuO_4, or condensible vapor form (at elevated plume

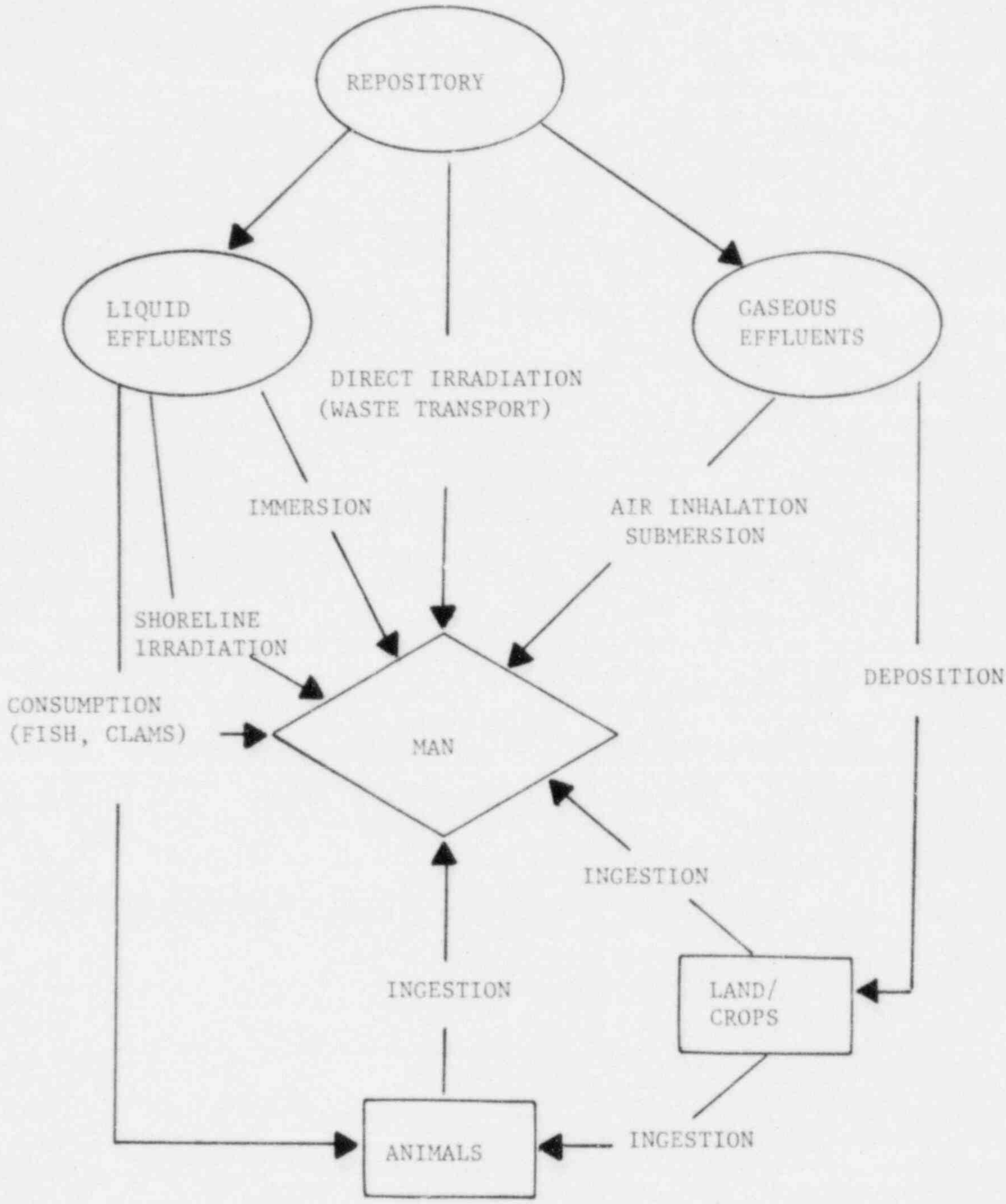


Figure 3-3 Generalized exposure pathways for man.

temperatures), or in condensed particulate form. Included in this group are the hazardous fission product nuclides of Cs, Sr, and I (Table 3-1).

c) Aerosol particles from fuel dust, in particular, particles of respirable size (less than 10 μm AED). This group includes plutonium radionuclides.

Within confined volumes, the following types of models are generally applicable for describing the airborne concentration of the radionuclides and the release to the atmosphere:

1) Aerosol trajectory models, predicting the movement of individual particles.

2) Indoor Air Quality (IAQ) models, predicting the global behavior of gases and particulates in a simple manner.

3) Nuclear reactor accident aerosol models, predicting the global behavior of particulates in a more complex manner.

The aerosol behavior models developed for nuclear reactor accidents are mainly used for one or more large, well-mixed compartments. Most of these models account for a wide variety of behaviors for the aerosol cloud as a whole as opposed to individual particles considered in the trajectory models. The following aerosol phenomena are considered in these models:

- Brownian coagulation
- turbulent coagulation
- gravitational coagulation
- gravity settling
- diffusion to the walls
- leakage or ventilation
- air cleaning.

The applicable computer codes for the different aerosol models are discussed in Section 3.1.8. Their use may, however, be unnecessarily complex for conceptual repository configurations where a simpler approach may suffice.

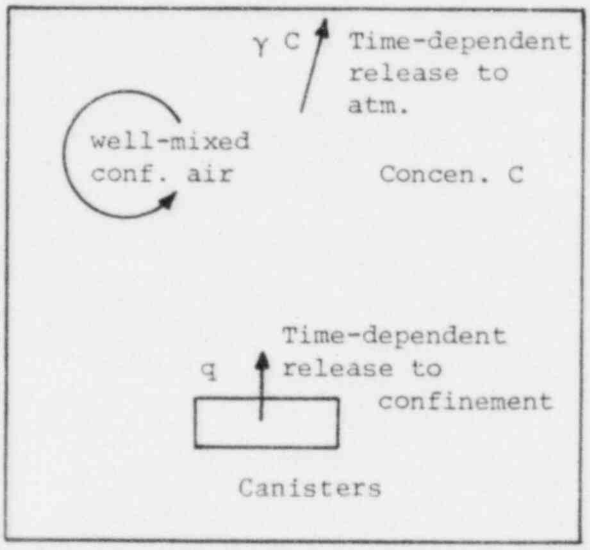
Indoor air quality models do not generally treat the coagulation processes explicitly or differentiate between deposition on walls or the floor. Instead, overall deposition parameters based on empirical data are used which implicitly account for coagulation by the appropriateness of the test simulations.

For releases inside the repository, the indoor air quality (IAQ) model is reasonably adequate for the purpose of the present study since final design information on repository compartmentalization is not yet available.

In Section 3.1.5 (Eqs. 3-1, 3-5, and 3-6), the mass of gas and fuel dust released to the confinement air (gasborne pathway) is given in terms of the mechanisms of burst release (M_B), impact release (M_I), diffusion release (M_D),

and oxidation release (leaching mechanisms pertain to liquid pathways). The mass release can be converted to an activity release for a specific radionuclide by knowing the relative concentration of that nuclide in the canister gas or on the fuel dust particles.

In the Indoor Air Quality Model, for purposes of calculating doses, some of the gasborne release mechanisms may be taken to be essentially instantaneous, burst and impact release for example. Others, such as diffusion and oxidation release, may occur in a time-dependent manner. The simplified sketch below illustrates the IAQ model.



An initial instantaneous release at t_o of radionuclide n produces an initial concentration $C_{n,o}$ of that nuclide in the confinement air, which is assumed to be perfectly mixed. Thereafter, a time- dependent release occurs which is instantaneously mixed in the confinement air. In many cases, the rate of time-dependent release can be described by an exponential function (see Eq. 3-6) of the form

$$q = q_o \ e^{-A(t-t_o)} \tag{3-8}$$

where q_o is the initial radionuclide inventory remaining after the burst release at time t_o, and A is a removal constant.

Within the confinement air, the concentration of the radionuclide may be reduced by the following confinement processes included in the IAQ model:

- o radioactive decay per exponential constant λ_r.
- o natural deposition or fallout per exponential constant λ_p.
- o leakage rate to the atmosphere per linear constant γ.
- o recirculation filter cleanup per exponential constant λ_c.

For convenience, an overall removal constant is defined by:

$$B = \lambda_r + \lambda_p + \lambda_c + \gamma \qquad (3\text{-}9)$$

The differential equation for time-dependent containment activity c is

$$\underbrace{\frac{dc}{dt}}_{\substack{\text{rate of}\\ \text{conc.}\\ \text{change}}} = \underbrace{q}_{\substack{\text{rate of}\\ \text{activity}\\ \text{added}}} - \underbrace{BC}_{\substack{\text{rate of}\\ \text{activity}\\ \text{removed}}} \qquad (3\text{-}10)$$

with the boundary condition that $C = C_o$ at $t = t_o$ (after burst release). For constant parameters A and B, the solution over an interval of time is

$$c = \frac{q_o}{B-A}\left[e^{-A(t-t_o)} - e^{-B(t-t_o)}\right] + C_o\, e^{-B(t-t_o)} \qquad (3\text{-}11)$$

Depending on the relative importance of the terms, this equation predicts monotonically increasing, monotonically decreasing, or peak concentration in the time interval.

Based on this concentration, the dose to workers in the confinement can be calculated, using the methods for reactor containments specified in 10CFR20. In 10CFR20, maximum permissible concentrations (MPC) of specific radionuclides in air are given, corresponding to a dose rate of 2.5 mrem/hr (occupational dose limit). For a given mixture of radionuclide concentrations C1, C2,... C_N, the actual overall dose rate can be obtained from

$$\text{Dose rate} = \sum_{n=1}^{N} \frac{C_n}{(MPC)_n} \quad \frac{2.5}{V} \qquad (3\text{-}12)$$

where V is the confinement-free volume. Implicit in this formulation is the assumption of immersion in a semi-infinite medium, which can be conservative for certain nuclides.

Over the same interval, the integrated release from the confinement to the atmosphere is

$$Q = \int_{t_o}^{t} \gamma\, C\, dt \qquad (3\text{-}13)$$

$$Q = \frac{q_o\gamma}{B-A}\left\{ \frac{1}{A}\left[1-e^{-A(t-t_o)}\right] - \frac{1}{B}\left[1-e^{-B(t-t_o)}\right]\right\} + \frac{C_o}{B}\left[1-e^{-B(t-t_o)}\right] \tag{3-14}$$

This is a monotonically increasing function (in units of curies) that levels off with time.

The foregoing equations express the fission product concentration and release of a single nuclide to the atmosphere in a specific time interval in which the parameters such as removal constants (λ) and leak rates (γ) remain constant. The accident is broken up into intervals. The first interval extends from time t_o until the parameters change significantly. One can use as many intervals (denoted by k) as needed to simulate the accident history.

If n is a subscript denoting each of the N nuclides listed in Table 3-1 and k is a subscript denoting each of the K time intervals, the total integrated release to the atmosphere for each nuclide is

$$\sum_{n=1}^{N} \sum_{K=1}^{K} Q_{n,k}$$

The external whole body dose to a person at distance z from the confinement can be calculated in a simple manner by the equation

$$D_{WB\gamma} = \frac{1}{4}\,(X/Q) \sum_{n=1}^{N} \sum_{k=1}^{K} Q_{n,k}\, E_n \tag{3-15}$$

where X/Q is the atmospheric dispersion factor at distance z and E_n is the effectivity factor for nuclide n, converting from curies to whole body dose. Similarly, the inhalation thyroid, lung, and bone organ dose equations are, respectively,

$$D_{thy} = b(X/Q) \sum_{n=1}^{N} \sum_{k=1}^{K} Q_{n,k}\, E_n \ , \tag{3-16}$$

$$D_{lung} = b(X/Q) \sum_{n=1}^{N} \sum_{k=1}^{K} Q_{n,k}\, E_n \ , \tag{3-17}$$

$$D_{bone} = b(X/Q) \sum_{n=1}^{N} \sum_{k=1}^{K} Q_{n,k}\, E_n \ , \tag{3-18}$$

where the E_n are dose effectivity factors for the different organs and b is the breathing rate.

In this formulation, the X/Q values pertain to the atmospheric dispersion factors in the Gaussian plume dispersion model. This model is used extensively in reactor accident analysis and is considered adequate for the environmental transport of gases and dry, small airborne aerosols where fallout en route can be neglected.

More complete dose calculations may be made using computer programs which simulate the plume dispersion and transport, as described in Section 3.1.8. Some of these also calculate ground deposition and fallout of particulates, direct exposure from ground deposition, food and milk pathway exposures, and the interactions with the populations, including mitigating actions of sheltering, evacuation, etc. The only computer code able to analyze all these effects is the CRAC2 code, which is recommended to be used in the next study phase to determine dose consequences of offsite releases.

3.1.6.2 Liquid Pathways. A brief discussion of liquid pathway exposure due to crud or leaching release is given here for completeness. Differential equations describing the one-dimensional groundwater transport of activity subject to retardation and hydrodynamic dispersion are given in ONWI-121 (Bechtel, 1981). Data requirements for the simulation include:

- Retardation or sedimentation coefficients
- Groundwater speed
- Rate of containment flow from the repository
- Dispersion coefficient

At the Hanford site, there are no nearby wells or usage of groundwater for crop irrigation. The most significant exposure pathway is likely to be contamination flow and dilution in the nearby Columbia River, with subsequent deposition and sedimentation along the downstream shores of the Tri-Cities area.

Doses can be estimated knowing the groundwater (aquifer) flow rate, the river flow rate, irrigation uptake rate along the river, and so-called usage terms or consumption rates. The latter account for (mass or volume of water/yr) human consumption, if any, of potable water, fish, and food produced by irrigation agriculture. They also account for the use of shoreline recreation facilities. Also required for ingestion pathways are the dose conversion factors for internal exposure, based on a 50-year dose commitment of a radionuclide n introduced into the human body.

A typical dose equation for liquid pathways is

$$D_j = \sum_P \sum_n C_{n,P} \cdot F_{j,n,P} \cdot U_P \qquad (3\text{-}19)$$

Where D_j = dose to the organ j for all pathways P and all nuclides n.

$C_{n,P}$ = activity concentration of nuclide n in the media consumed for pathway P (PCi/liter)

$F_{j,n,P}$ = dose factor for radionuclide n, pathway P and organ j.

U_P = usage term for pathway P, liters/yr.

The dose factors $F_{j,n,P}$ are well-known and are given in NRC publications. The activity concentrations can be estimated from the repository release calculation and gross approximations of the groundwater and river dilution. Usage terms can be appropriately derived from an analysis of the area agriculture and recreational history.

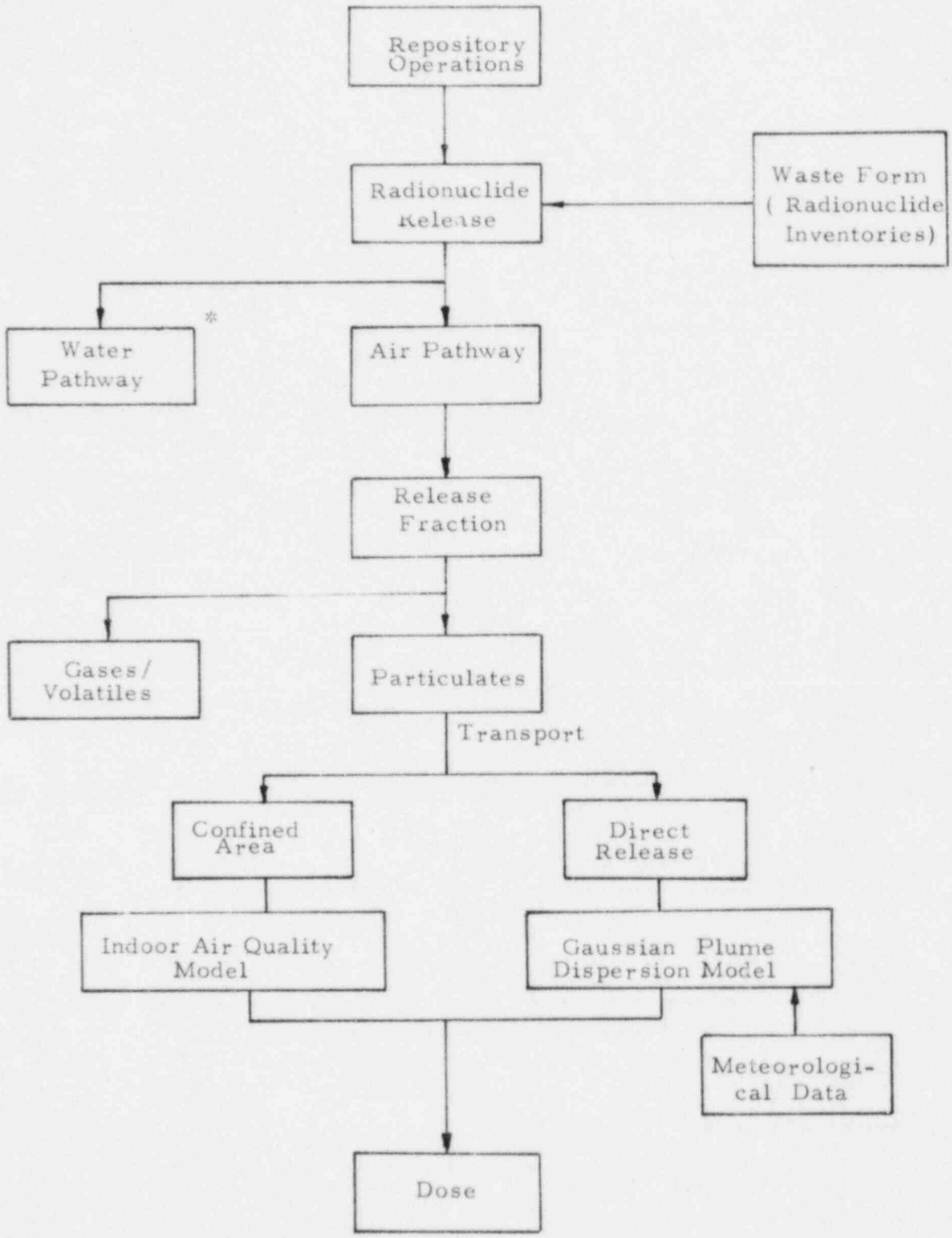
3.1.7 Recommended Approach

Figure 3-4 illustrates the proposed approach for the quantitative evaluation of the radiological consequences of repository accidents. Note that the evaluation will only address airborne releases since this is the major release pathway in the preclosure period. Except for flood, radionuclide release via water pathway is a slow process which is not likely to affect the preclosure safety of a repository.

As noted in Sections 3.1.4 and 3.1.5, the transport and behavior of radionuclides play a key role in the consequence evaluation. The fractional release of radionuclides from breached canisters and fuel pins is used directly in dose calculations. However, providing reliable estimates of release fractions (source terms) is difficult, largely because of the accident-specific nature of the release and the lack of adequate experimental data to support postulated release assumptions.

The release fraction for gases and volatiles given in previous studies (Wilmot, 1980; Wilmot, 1981; Walker, 1978) will be used in this project. The models discussed in Section 3.1.5 will be applied to the extent permitted by required data to determine the release fraction of particulates having AED of 10 m or less. For transport of these particulates in confined areas, the

Rather than calculate radiological consequences for each and every accident sequence, consequences will be evaluated for a short list of release categories, representing discrete levels of release ranging from fuel-cladding gap release only to significant release of radionuclides under fire or explosion conditions. That is, each release category corresponds to a certain set of accident conditions and corresponding set of release fractions of radionuclide groups. The list of release categories is so chosen such that the complete range of accidents and activity releases to be analyzed are covered. Offsite release consequences, in the form of doses at a given distance, will be calculated for each release category by the CRAC2 computer code, accounting for site-characteristic weather variations and human actions. The statistical sampling process in the CRAC2 code for weather and human response produces a statistical statement of consequences. This is called a risk curve, given



* Will not be considered in the analysis.

1. Use release fractions data available from the literature.
2. Apply release models discussed in Sec. 3.1.5 .

Fig. 3-4. Approach to radiological consequence evaluation.

(conditional upon) the release category. Each release category (or conditional) risk curve is expressed as the frequency of exceeding a certain dose as a function of dose. This is basically the same procedure as used in all nuclear plant PRAs since WASH-1400.

Each accident sequence in the event trees will be assigned a release category based on the physical conditions of the sequence and on predetermined criteria which define the release categories. Table 3-2 shows a proposed set of release categories for repository accidents, and the typical events and release mechanisms corresponding to each category. The category criteria will be closely based on these typical events and release mechanisms. A set of release fractions for the radionuclide groups will be defined quantitatively in each release category.

To derive the radiological risk, the summed frequency of all sequences leading to each release category will be derived. By combining the summed frequency with the CRAC2 conditional risk curve for that release category, one obtains an absolute risk curve for the release category. The overall repository risk is obtained by the standard combination of the absolute release category risk curves (i.e., frequencies are summed at each consequence level).

An advantage of this approach is that the consequence assessment (conditional risk curves) can proceed in parallel with the accident event trees. But the release category definitions and criteria (leading to quantitative release definition) must be made early on from preliminary accident scenario analysis.

It is recognized that there are large uncertainties involved in the characterization of release fractions. To permit the identification of dominant radiological risk contributors, uncertainties of release fractions in different accident scenarios should be considered on an individual (scenario dependent) basis.

3.1.8 Available Computer Programs

Of the computer programs applicable to the HLW-PSSA listed in the Task 1 Literature Review report (Ligon, 1984) the following programs are directly applicable for the consequence evaluation of preclosure activities in a repository: ORIGEN-S, CRAC2, PADLOC, XOQDOQ, and GASPAR. The extent of their applicability is summarized in Table 3-3. These computer programs such as CRAC2, XOQDOQ, and GASPAR are adequate for offsite consequence calculations only.

ORIGEN-S is a zero-dimensional depletion code which solves the Bateman equations for radioactive growth and decay of a large number of isotopes (Mills, 1983). It is an enhanced version of ORIGEN, the ORNL Isotope Generation and Depletion Code. ORIGEN-S can be used to estimate the radionuclide inventories in a HLW repository and to estimate the heat and radiation source terms from radioactive waste packages. Because radionuclide inventory information (based on ORIGEN runs) are available in the literature, it is not necessary to run this code specifically for this project.

TABLE 3-2
QUALITATIVE RADIONUCLIDE RELEASE CATEGORY DEFINITION

Release Category	Release Mechanism(s)	Radionuclide Release*	Typical Events
I	Heatup of failed fuel during fire or explosion driving off volatile and nonvolatile radionuclides	All noble gases, most volatiles, significant fraction of nonvolatiles from amount of failed fuel involved in the accident	Severe impact or release coupled with fire or explosion
II	Limited heated fuel release (<1400°C) or severe aerosol release	All noble gases, significant fraction of volatiles, small to moderate fraction of nonvolatiles	Temporary loss of cooling to canisters; severe impact and aerosol formation without fire, explosion
III	Failed fuel gap release + aerosol release - no fuel heatup	A significant fraction of noble gases, small fraction of volatiles, very small fraction of nonvolatiles on fuel aerosols	Mild impact and aerosol release, no fire or explosion
IV	Failed fuel gap release only - no fuel heatup	A significant fraction of noble gases, very small fraction of iodine and other volatiles	Mild impact or other canister failure, no aerosol formation or release

*Release shown is qualitative; quantitative values will be defined in the later phase of the study.

TABLE 3-3
COMPUTER CODES FOR RADIOLOGICAL CONSEQUENCE EVALUATION

Code Name	Radionuclide Inventory	Radionuclide Release	Atmospheric Dispersion/ Deposition Model	Dose Calculation	Availability
ORIGEN-S	X			**	
CRAC2			X	X	Sandia
PADLOC*		X			GA
XOQDOQ			X		NRC
GASPAR				X	NRC

*PADLOC can be used to calculate plateout on confinement side ventilation walls and within filters.

**Radionuclide inventory results from ORIGEN runs are readily available in the literature.

CRAC2 (Ritchie, 1983) is a newer version of CRAC (Calculation of Reactor Accident Consequences), a computer code that has been used to study the effects of varying population density, weather conditions, and radionuclide release inventories on various consequences of atmospheric releases in the event of a nuclear power plant accident. Improvements in CRAC2 include the modification of the atmospheric dispersion model, the introduction of a new meteorological sampling technique, a new evacuation model, new output capabilities, and a new thyroid damage model.

The major inputs to CRAC2 are:

- meteorological data
- population distribution
- radionuclide inventories
- decay half-lives of radionuclides
- breathing rate
- dose conversion factors.

PADLOC (Hudritsch, 1977) is a one-dimensional mass transfer computer code developed at GA to analyze steady-state and time-dependent plateout of fission products in an arbitrary network of pipes. It was developed as a general model to simulate radionuclide transport and plateout in applications such as the primary circuit of a high-temperature gas-cooled reactor (HTGR). It can also be applied to radionuclide release during hypothetical accidents in the repository, including plateout on confinement-side walls and plateout within filters positioned at the top of upcast shaft. Major input data types include: material properties, ventwork geometry, air flow rate, and temperature.

The application of PADLOC to calculate release fractions of radionuclides transported within the repository was demonstrated in NUREG/CR-1931 (Pepping, 1981). The PADLOC results indicated that plateout and deposition within the repository will have a modest impact on reducing airborne activity. In each instance where significant radionuclide release was predicted, the receiving medium was air.

XOQDOQ (Mills, 1983) is a computer code used by the NRC in its meteorological evaluation of routine releases from commercial nuclear power reactors. This code, which uses the steady state Gaussian plume assumptions, can be used to estimate ground-level radionuclide concentrations and deposition amounts associated with atmospheric releases from waste repository operations. Meteorological data, decay half-lives and source parameters are the major types of inputs to XOQDOQ. Outputs from XOQDOQ such as atmospheric dispersion and deposition factors can be used as inputs to GASPAR. XOQDOQ runs on IBM 360/370 computers.

GASPAR performs air-release dose calculation of noble gases and of radioactive particulate emissions, including doses for both the entire population and the individual. It currently runs on CDC computers. The present documentation of the code is quite poor, requiring the user to study the program listing before running the code.

Figure 3-5 illustrates how each body of information will be utilized to calculate the radiation dose and the computer codes to be used. An ORIGEN-S run need not be performed. There is considerable information available from previous studies of radionuclide inventory data for spent fuel and HLW that were generated using this code. PADLOC will be used to calculate the plateout factor as required in the evaluation of a given accident scenario. Release fraction and filter decontamination factor data will be taken from the literature, making sure that the number selected applies closely to the scenario in question.

For the following reasons, the use of CRAC2 is recommended to quantify aerosol release to the environment in terms of radiation dose:

- The code is readily available at GA Technologies and Sandia National Laboratories.
- Sandia and GA Technologies have extensive experience in using the code.
- Its predecessor, CRAC, has been used in a previous preclosure risk analysis of a repository for spent unreprocessed fuel (Pepping, 1981).
- It is uniquely qualified to perform the statistical weather variation required to derive the risk curves.

For calculating radiation dose as the consequence of interest, the CRAC2 code would be run in a simplified mode; e.g., without population or evacuation models, no health effect calculations, etc. The code running time should be suitably short.

As discussed earlier (Section 3.1.6) the nuclear accident aerosol models are quite complex and appear unnecessary for the project. They are addressed here to give the reader some idea on which codes may be applicable to the repository.

The more complex computer programs are essentially of three types: monodispersed, fixed-distribution, and discrete distribution types. CORRAL, MADCA, and ETHERDEMO are of the first type and employ mass balance equations that are similar to those of the IAQ model. TRAP, TRAP-MELT, HAA-3, and HAA-4A are of the second type and utilize time-dependent continuous distributions of aerosol size. NAUA-4, MATADOR, PARDISEKO, CRAB, MAEROS, and QUICK belong to the third category and tend to be generally time consuming. In this category, QUICK and MAEROS appear to have the best blend of efficiency and accuracy for simple applications.

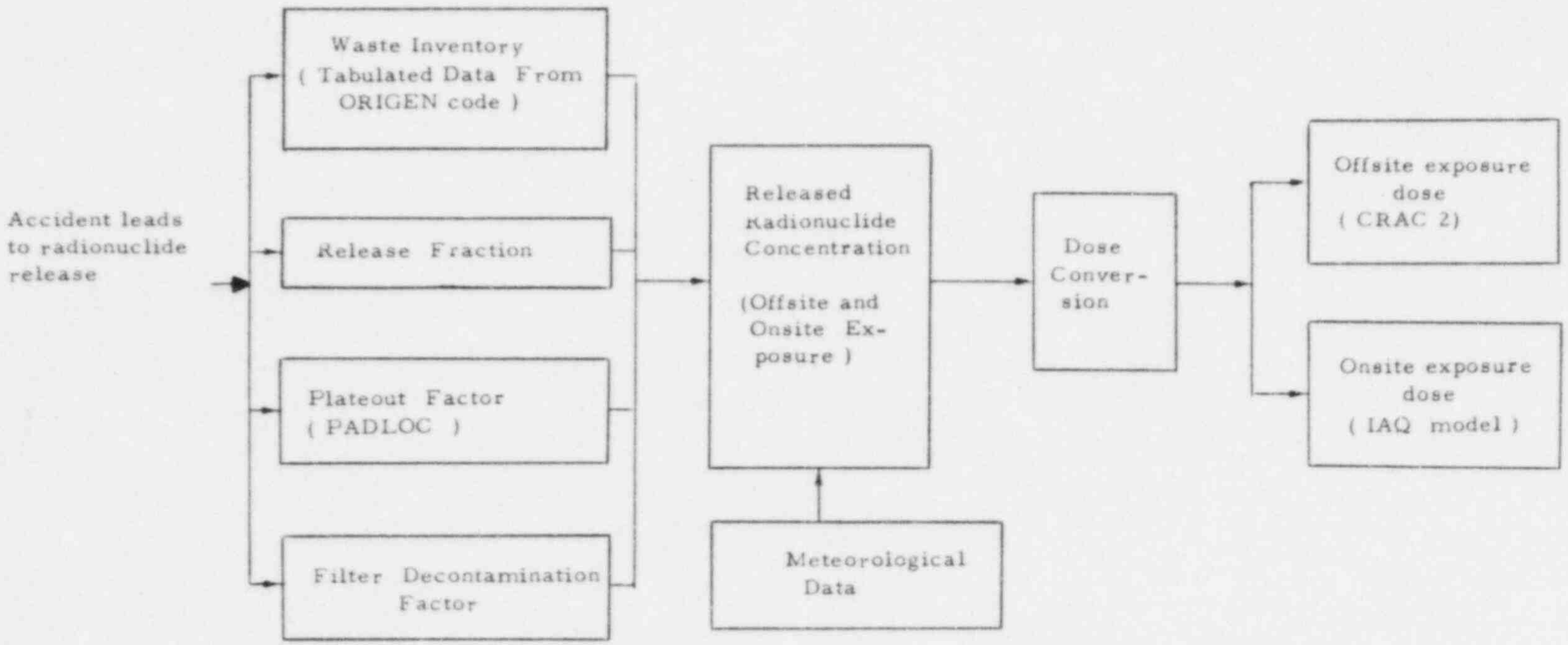


Figure 3-5 Radiation dose evaluation

3.2 NONRADIOLOGICAL OCCUPATIONAL CONSEQUENCE

Many activities within the repository require operator actions which expose workers to industrial accidents in addition to possible radiation exposure. A worker hit by a falling cask may be killed or injured, regardless of whether or not the cask is breached. Hence, the evaluation of nonradiological occupational consequences (i.e., personal injury or death) requires a much simpler approach than that required for a radiological consequence evaluation.

3.2.1 Recommended Approach

In this project, we propose to do the following regarding nonradiological consequences:

1. Categorize the initiating events identified in Section 2 by repository activity (e.g., surface activities, hoisting, emplacement, etc.)

2. Identify the probable causes of injury (e.g., fire, entrapment, accidental falls, etc.)

3. Determine frequency and risk of personnel injury/death for each repository activity from industry data.

Categorization of initiating events according to repository activity and identification of those leading to operator injury has been presented in Section 2. Accident frequency and occupational injury data are addressed in Section 7.

3.3 IMPACT ON REPOSITORY AVAILABILITY

The availability of a subsystem, equipment, or component can be expressed in terms of its mean "uptime" or mean-time-between-failures (MTBF) and its mean "downtime" of mean-time-to-restore (MTTR) as:

$$A = \frac{MTBF}{MTBF + MTTR}$$

As was shown in Fig. 3-1, accidents leading to both radiological consequences and nonradiological occupational consequences affect repository availability. In addition, we must consider repository downtime due to clean-up and equipment repair (even though the waste package is not breached or no operator is injured), and downtime due to scheduled maintenance of equipment.

3.3.1 Evaluation of Repository Availability

The general approach to evaluating availability consists of three sequential steps: (1) qualitative analysis, (2) system modeling, and (3) quantification.

The qualitative analysis generally performed by reliability engineers is the Failure Modes and Effects Analysis (FMEA). An FMEA is a method for identifying all the failure modes of a plant, system, or process; determining the effect(s) of each one; and determining the mechanisms for the failure. In the FMEA process, the analyst postulates a failure mode, then considers the effect of this failure alone throughout the system. Failure modes consisting of multiple or dependent failures may also be postulated.

An effective way of ensuring that no significant failure modes are omitted in the FMEA, is the development of system symbolic logic diagrams (SSLDs) which subdivide the facility into successively smaller portions (e.g., systems and components) proceeding one level at a time. This is helpful because it restricts the number of items that must be considered in each subportion, thus reducing the likelihood of omissions. The FMEA can then be based directly on the SSLDs. It is usually desirable to prepare the FMEA at the lowest level of assembly possible because this allows more specific analysis of the effect(s) of each failure mode and the ways to mitigate the effect(s).

System modeling involves the use of reliability block diagrams (RBDs), fault trees, and/or event trees to estimate system availability and/or reliability.

Once a system is adequately modeled using one or a combination of the above techniques, system availability is estimated. The quantification process includes obtaining the failure rate and repair time data, defining the data uncertainties and modeling uncertainties, and obtaining the system reliability or availability.

3.3.2 Recommended Approach

In evaluating the impact of repository availability, there are three major areas to consider:

1. Repository downtime resulting from accidents which lead to personnel and/or public exposure to radiation.

2. Repository downtime due to work days lost from operator injury or death following an accident whether or not radiation is released.

3. Repository downtime for cleanup and equipment repair even if the accident scenario does not lead to a radionuclide release or operator injury.

Items 1 and 2 can be calculated by simply translating the consequences discussed in Sections 3.1 and 3.2 in terms of repository downtime. To bound the problem, the radiological and nonradiological consequences can be grouped into three or four categories and an upper and lower estimate of downtime assigned to each category.

Item 3 can be quantified by translating system availability given a certain initiating event in terms of system downtime.

System modeling techniques such as event trees and fault trees are used to calculate the frequency of each accident scenario. The product of accident frequency and the associated downtime summed over all accident scenarios gives the overall unplanned repository downtime.

3.4 FINANCIAL CONSEQUENCE

The financial impact of repository-related accidents represents the offsite and onsite costs related to public and plant personnel health effects, waste storage replacement, cleanup, decontamination, repair, etc.

Offsite financial consequences estimate the monetary loss to the government resulting primarily from the release of radioactivity to the environment. The cost factors that should be considered include: (1) the monetary value of health effects (i.e., the expenditure society is willing to make to prevent loss of life and the cost of providing health care to the affected population); and (2) property damage (e.g., lost public and private property, interdicted land and farm crop costs, lost wages, decontamination costs, etc.).

Onsite financial consequences pertain to the monetary loss affecting the repository facility and its personnel because of accidents which may or may not lead to a radionuclide release, and of maintenance and repair operations which reduces the repository's availability to accept and dispose of nuclear wastes. The cost factors include: (1) plant personnel health effects, (2) replacement storage costs, (3) cleanup costs, (4) capital costs, (5) litigation costs, and (6) indirect costs (e.g., shutdown of reactors because of spent fuel storage problems, loss of industrial capacity, loss of jobs, etc.).

3.4.1 Discounting Costs

In the economic consequence evaluation of an adverse condition which could occur at any time during the life of the repository, there are two major variables: money and time. The time value of money makes it unrealistic to directly compare monetary amounts unless they occur at the same point in time. Discounting techniques have been proposed (EPRI, 1982 and Strip, 1982) to determine the present value of money. For repository applications, the discounting formulas based on continuous discounting used in (Strip, 1982) appear adequate. They are given below.

1. For calculations of early health effects and offsite property damage, the present value of a cost C_o which occurs with frequency f:

$$\int_{t_i}^{t_f} e^{-rt} C_o f \, dt = C_o f \left[\frac{e^{-rt_i} \quad e^{-rt_f}}{r} \right] \tag{3-20}$$

where

r = effective discount rate

f = frequency of accident costing C_o

t_i = time of onset of risk of accident

t_f = time of end of risk of accident.

2. For calculating the present value of an expense C_o which recurs for M number of years (e.g., cleanup expense):

$$\int_{t_i}^{t_f} f \int_{t}^{t+m} C_o e^{-rt'} \, dt' dt = \frac{C_o f e^{-rti}}{r^2} \left[1-e^{-r(t_f-t_i)} \right] 1-e^{-rM} \tag{3-21}$$

3. For calculating the present value of an expense C_o that will recur until a fixed date, rather than a fixed number of years (e.g., replacement storage costs).

$$\int_{t_i}^{t_f} f \int_{t}^{t_f} C_o e^{-rt'} \, dt' \, dt = \frac{C_o f}{r} \left[\frac{e^{-rt_i} - e^{-rt_f}}{r} - e^{-rt_f} \left(t_f - t_i \right) \right] \tag{3-22}$$

3.4.2 Available Methods to Calculate Financial Consequences

Offsite financial consequences of nuclear power plant accidents have been calculated using the CRAC code (NRC, 1975). The work reported in (Strip, 1982) used CRAC2 (Ritchie, 1983) to estimate offsite financial consequences. Economic effects taken into consideration in CRAC2 include lost wages, relocation expenses of the evacuated population, decontamination costs, lost public and

private property, and interdicted land and farm crop costs. They are all calculated on the basis of statewide land use and land value data, and the population distribution surrounding the specific site. Economic consequences which are not included are the cost of providing health care to the affected population, all onsite costs, litigation costs, and indirect costs.

An EPRI report (Stamatelatos, 1982) describes a value-impact methodology that goes beyond the calculation of financial impact since it also considers attributes (values and impacts) not directly measurable in monetary units. However, the method used for calculating discounted costs resulting from postulated accidents follows that described in EPRI's Technical Assessment Guide (EPRI, 1982).

3.4.3 Recommended Approach

If quantitative evaluation of economic consequences of repository-related activities is considered, the following steps are recommended.

1. From the set of accident scenarios identified as dominant risk contributors in the preliminary screening process (discussed in Section 2, Scenario Enumeration and Selection), group the severity of each consequence type into three or four categories.

2. Develop a list of assumptions regarding DOE's and NRC's actions in response to a repository accident.

3. Based on these assumptions, identify all the important cost factors and determine the cost data necessary for realistic cost calculations.

4. Calculate an upper and lower bound cost estimate for each consequence category in each consequence type using the discounting formulas presented in Section 3.4.1.

3.5 EFFECT OF PRECLOSURE OPERATIONS ON LONG-TERM REPOSITORY FUNCTION

The design of a repository follows the "engineered barrier system" approach. The engineered barrier system includes the waste packages and the underground facility. A waste package, as defined in 10CFR60, is composed of the waste form and any containers, shielding, packing, and absorbent materials immediately surrounding an individual waste container. The underground facility refers to the mine structure, including openings and backfill materials. Such a system would ensure substantially complete containment of HLW over a long period of time. However, processes (either natural or manmade) which occur during the preclosure phase could compromise this multiple barrier concept and, consequently, prevent the repository from ensuring the safe and permanent disposal of radioactive waste for thousands of years.

This type of consequence is not a risk contributor in the preclosure phase; therefore, no attempt will be made to quantify any contribution to risk. Instead, the sequences/operations that are potentially capable of generating this consequence will be identified to facilitate consideration of preventive

procedures and operations at an early stage in the repository design/licensing process. Potential initiating events that could lead to this type of consequence were presented in Section 2.

3.6 CONSEQUENCE TYPES RECOMMENDED FOR FURTHER EVALUATION

An important goal of this project is to use the results of the risk evaluation in the licensing of the construction and operation of a nuclear repository. To achieve this goal and at the same time effectively use the available resources, we recommend that radiological (public and worker) and nonradiological occupational consequences be evaluated in more detail in order to demonstrate the usefulness of the HLW-PSSA methodology.

Relevant initiating events and their corresponding consequences have been discussed in great detail in Section 2. As quantification of radiological consequences will be performed in the next study phase of this project, those accident scenarios leading to radiological consequences could then be further categorized according to the four release categories shown in Table 3-2. It is expected that a major portion of release fraction modeling effort (using the models discussed earlier) will concern aerosol release.

While repository availability and financial impacts are also important considerations, particularly in a comprehensive risk evaluation, their not being addressed at this time will not preclude our ability to demonstrate the methodology.

4.0 FAULT TREE DEVELOPMENT

Fault tree development requirements for this project were based on the systems identified as intermediate events in the event tree accident scenarios. Wherever possible, portions of fault trees developed in previous repository safety analyses were used to avoid duplication of effort. Systems not explicitly described in the "Conceptual System Design Description, Nuclear Waste Repository in Basalt" (SD-BWI-SD-005) were not modeled due to lack of information. Instead, these systems have been assigned failure probabilities from similar systems currently operating for other applications (see Section 7, Data Base). Future phases of this project may include fault tree modeling of these systems as design information becomes available.

The systems modeled were primarily the various building and subterranean confinement exhaust ventilation/filtration systems. In order for a radiological incident to occur, the barriers to local and environmental radionuclide release must be defeated. For the repository concept, those barriers consist mostly of the air circulation/cleanup systems. The availability of electrical power for these systems is a key issue in any safety analysis.

Adjacent to the repository power substation is the standby diesel generator building. There are two 13.8 kV diesel generator units that automatically start (using compressed air) and pick up all standby power loads in the event of loss of offsite power. Individual load centers further reduce the service voltage to 4.16 kV for motors >200 hp and 480 V for smaller motors and normal utilities. A third power source, designated as the uninterruptible power source, is also described in the conceptual design; however, no loads are identified for this system and most battery backed sources are at a voltage insufficient to power ventilation loads. Thus, only two power sources with their associated load centers are considered for waste handling exhaust ventilation systems.

The fault trees developed for the waste handling building primary and secondary confinement ventilation systems are given in Figs. 4-1 and 4-2. Passive failures such as duct collapse are lumped together into a single event. Multiple failure modes of active components are considered (e.g., filter collapse in addition to loss of filtration function). Human error is included at the component level where applicable.

Faults occurring with both the intake supply and the exhaust were considered to cause system failure. It was assumed that if a supply is interrupted, the preferred pressure differential would be lost between the confinement systems. This in turn defeats the basic confinement concept and was thus considered as a mode of confinement failure.

There are filter trains on both the intake and exhaust blowers. On the intake side, there is a bird screen, tornado dampers, and a combination of 30% and 90% National Bureau of Standards (NBS) particulate filters. These inlet barriers are designed to prevent damage to the intake circulating machinery and

to limit the circulation of particulate matter from the external environment.

Exhaust filtration immediately upstream of both primary and secondary confinement exhaust systems include a moisture separator, a 90% NBS filter, and two HEPA filters. In addition, the primary confinement exhaust system contains a charcoal filter for limited gaseous absorption. Maintenance errors allowing any of these filters to become plugged were lumped together into a single human error. Primary failures of each filter component were modeled separately.

It should be noted that the exhaust of the primary and secondary confinement (and the waste transport shaft exhaust system) is sent via a common header to a single stack. This is a potential dependent failure, causing both systems to fail, particularly if an earthquake is considered as an initiating event. Many of the accident scenarios require the failure of both primary and secondary systems to generate an offsite release; this intersystem dependency is one way these conditions can be satisfied.

The normal and standby power systems are identified in the primary and secondary confinement ventilation fault trees. There is an implicit assumption in the fault tree modeling concerning standby power in that the backup system utilizing standby power is assumed (by fault tree structure) to be available immediately to pick up the function of the disabled normally-operating system. This is not strictly accurate because on a loss of offsite power, either diesel has to start, achieve operating speed, and sequentially pick up the required standby loads. The implicit assumption is that the time interval between loss of normal power to a ventilation system and successful operation of the standby ventilation system (upon successful diesel start and run) is insufficient to allow an appreciable amount of airborne contamination to escape to the outside environment. Other ventilation systems using back-up capability/standby power also have this assumption inherent in their fault tree structure.

4.1 WASTE HANDLING BUILDING CONFINEMENT VENTILATION SYSTEM

The philosophy of waste handling building confinement system operation outlined in the referenced basalt conceptual design description is different for the various systems considered. The waste handling building has both a primary and a secondary confinement exhaust ventilation system. The primary exhaust ventilation services the hot cell and secondary areas. These areas are the ones expected to have the highest contamination levels during normal operation. The primary system consists of two 100% redundant exhaust fan and filter trains. It was assumed for this analysis that the backup primary system started automatically upon the functional loss of the normal system.

The secondary confinement exhaust ventilation system consists of three 50% capacity fan and filter train assemblies. In the event of a failure of either of the running systems, it was assumed that initiation of the standby system was by manual operator action. The normal direction of leakage (determined by pressure control of these exhaust ventilation systems) is in the direction of higher contamination; that is, from the personnel areas to the secondary confinement areas, and finally, to the primary confinement areas.

There are two sources of power available to the waste handling building confinement ventilation systems. Normal power for all site loads is delivered to the repository using two 138 kV transmission lines. Power is converted to 13.8 kV and split into the various load centers at the receiving substation. Standby power comes from two diesel generators.

4.2 WASTE TRANSPORT SHAFT VENTILATION SYSTEM

The head frame of the waste transport shaft is located in the waste handling building outside the hot cell/secondary area. This ventilation system required modeling as an intermediate event because a transport accident occurring during hoist cage loading, transport to the subterranean level, or hoist cage unloading would be limited to a local release potential, given proper shaft ventilation/filtration system operation (see Fig. 4-3).

The exhaust fan and filter train for the waste transport shaft are located in the exhaust filter fan house section of the waste handling building adjacent to the secondary confinement exhaust fan/filter train assemblies. As noted previously, this system also exhausts to a tunnel and stack common to both the primary and secondary confinement exhaust systems. This is not a crucial commonality because the waste shaft ventilation cannot serve as a redundant backup to primary and secondary confinement ventilation (in most cases).

The waste shaft exhaust system is comprised of two fan/filter train systems. Both are required to run continuously for satisfactory performance so loss of either train constitutes system failure. The two trains have common inlet and outlet headers but use parallel fans and filter assemblies. They can also be cross-tied between the outlets of the filters and the inlets of the fans.

The exhaust portion of the waste shaft ventilation system is modeled similar to the waste handling building primary and secondary confinement exhaust systems previously discussed. The fans are smaller than the other systems, but the filter assemblies are identical to the secondary ventilation system. Human error at the component level, multiple component failure modes, and loss of electric power are the same as fault trees discussed above.

Loss of supply air to the waste transport shaft is the same as loss of supply air to subterranean confinement systems and is treated in detail in the next section. However, it should be emphasized here that this commonality creates another dependency between two redundant systems for some of the accident sequences.

4.3 SUBTERRANEAN CONFINEMENT SYSTEMS

The confinement exhaust ventilation system designed for the underground portion of the repository is contained in a separate building enclosing a dedicated vertical shaft for air flow. The purpose of this system is primarily to remove the heat from the waste panels. In the event of a subterranean radionuclide release, filtration of the exhaust air is also possible. Proper operation of the filtration would eliminate any offsite environmental release

(see Fig. 4-4). The system as described in the conceptual design document has five fan/filter train assemblies. Three are required for normal operation leaving two of the parallel units as installed backup capacity. These backups are assumed to operate on standby power while the normally operating units are supplied from the normal power system.

The flow of air is upward through the confinement exhaust shaft to a common plenum at the surface. Each of the five units is connected to the plenum. Normally, the air then flows through a separate filter bypass duct for each of the units, through the fan and out to the outside environment.

In the event that subterranean radioactivity monitors detect airborne contamination, a series of damper movements is required for each operating assembly, followed by start-up of a second stage for each fan, in order to align the respective filter trains with the exhaust flow prior to the arrival of the contamination. The following equipment actions must take place for each operating assembly to successfully remove the contamination:

1. inlet and outlet dampers on filter train open

2. second stage of fan starts and achieves operating speed

3. inlet or outlet dampers on bypass duct close (either is assumed to successfully block the flow).

This complex series of operations is necessitated by the conflicting requirements of minimum power consumption (minimum fan hp) during normal operation but full high-efficiency filtration capability during an accidental release situation. The alternatives of full-time operation of the filter systems require the additional power consumption needed for second stage fan operation. This is considerable for the air flow requirements defined in the system design description.

The receipt of a subterranean radioactivity alarm was assumed to trigger the realignment of the damper/filter assemblies automatically. An additional intermediate event was included in the appropriate event trees for manual operator initiation probability given automated system failure. This occurrence of human interaction at the event tree level represents the potential for an operator to mitigate an accident in addition to contributing to one.

Loss of supply air to the subterranean environment was considered to be a confinement system failure mode as either insufficient air flow or additional load on the exhaust fans would eventually result in loss of filter function. Confinement air intake is located adjacent to the refrigeration building which, together, enclose the confinement air intake shaft. The building houses five 150 hp axial flow fans, three of which are normally operational. Each fan assembly is capable of 74,000 ft#/min flow. This subterranean air supply system and the fault tree section (see Subterranean Confinement Air Ventilation System Fault Tree (Fig. 4-4), Transfer M1) are common to both the subterranean confinement air ventilation and waste transport shaft ventilation systems,

adding interdependency to accident sequences where both systems are capable of furnishing ventilation.

From the conceptual design description, the inlet of the supply fans appears to be through a common tornado damper/bird screen assembly; however, the text describes the units as separate. The system was modelled assuming separation. Intake filter trains upstream of the fan consist of the tornado damper/bird screen assembly and a particulate air filter to keep external dust from being circulated.

Power supplies to the fan motors were assumed to be normal for running fan units and standby power for the two installed spares. Each fan motor is rated at 150 hp, so it is assumed the power will be from a 480V load center. Motors (200 hp and over) will usually be supplied by 4.16 kV load centers (ref. system design description).

Power to operate the exhaust filter dampers and bypass duct dampers was assumed to be from the standby source (diesel generator backup). From the cursory description of the uninterruptible power source (battery backed) the only loads assumed to be tied to this system were the radiation monitoring systems, along with the switching logic necessary to realign the confinement exhaust from a bypass to a filtration configuration.

4.4 OTHER SYSTEMS

It is expected that more detailed system description will be available for the facilities addressed here in the later stages of design/construction than is currently available. Systems such as radiation monitoring, onsite power distribution, and compressed/breathing air systems are usually modelled to accurately represent the particular facility.

The conceptual design description for a repository in basalt does not provide this level of detail; consequently, fault trees have not been developed for these systems. Instead, data (industry-wide) are available (see Section 7) depicting the overall probable behavior of these types of systems. These data have been adopted in lieu of a more sophisticated modeling approach for this analysis.

Lack of sophisticated detail is not necessarily a drawback at this stage in the design process. The objective of performing this safety analysis is to develop a methodology capable of identifying and prioritizing significant contributors to accidents. Use of data for system behavior at the conceptual stage implies the system that is finally constructed is at <u>least</u> as reliable and safe as current generation systems of similar function. Using these numbers to identify design weaknesses and proritize corrective action is justifiable in view of current quality control measures on existing systems. An additional feature inherent in this approach is the ability to examine a potential change intended to address one problem in light of other problems that may be created.

Systems such as the radiation monitoring systems (both surface and subterranean) have been quantified as subsystems. Normal and standby power have been modeled very simply to provide a breakdown of potential contributing subsystems. No further subdivision of these systems was considered justifiable given the current status of system design. As the design matures, additional effort should be expended on level of detail in the systems modeling.

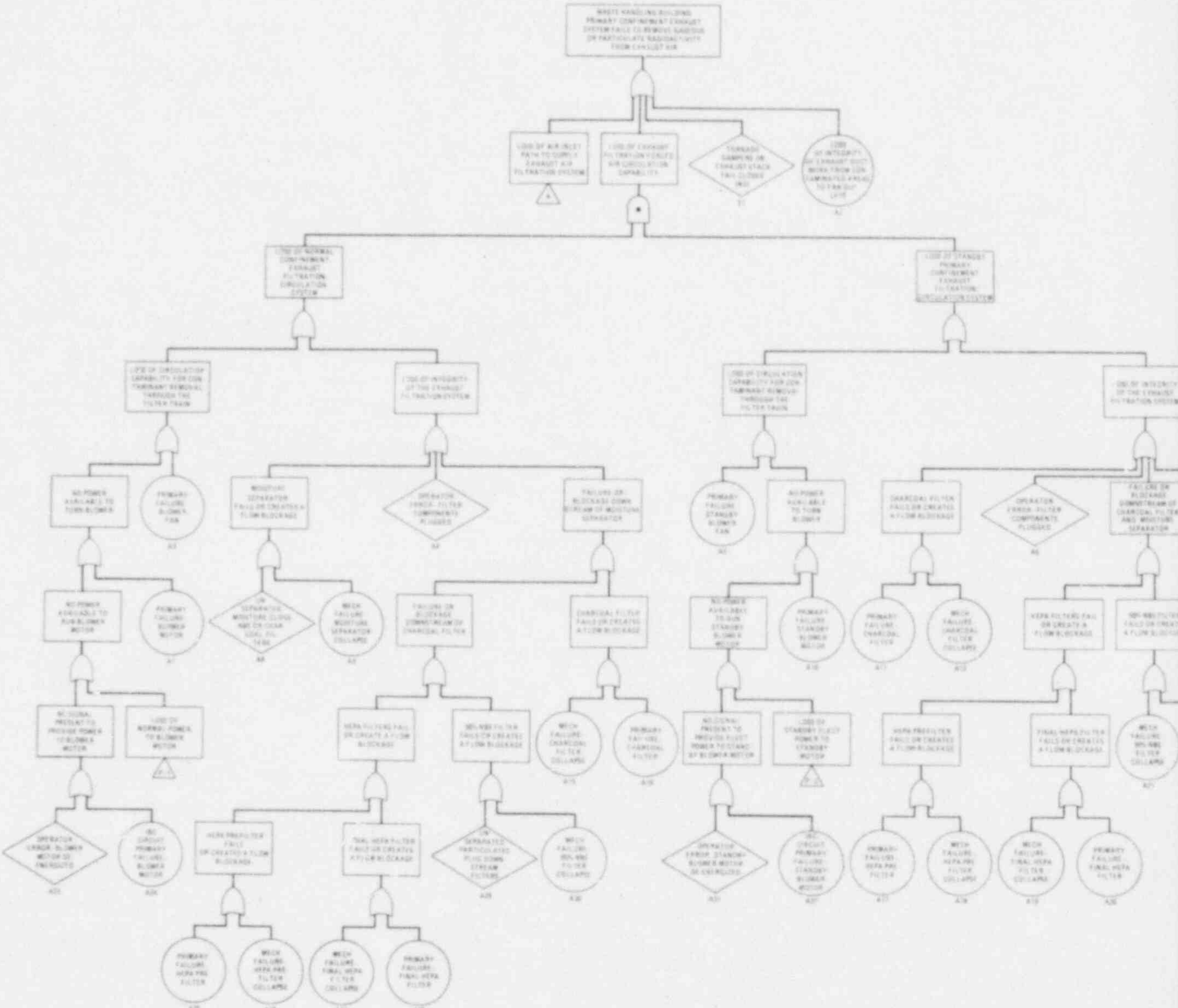
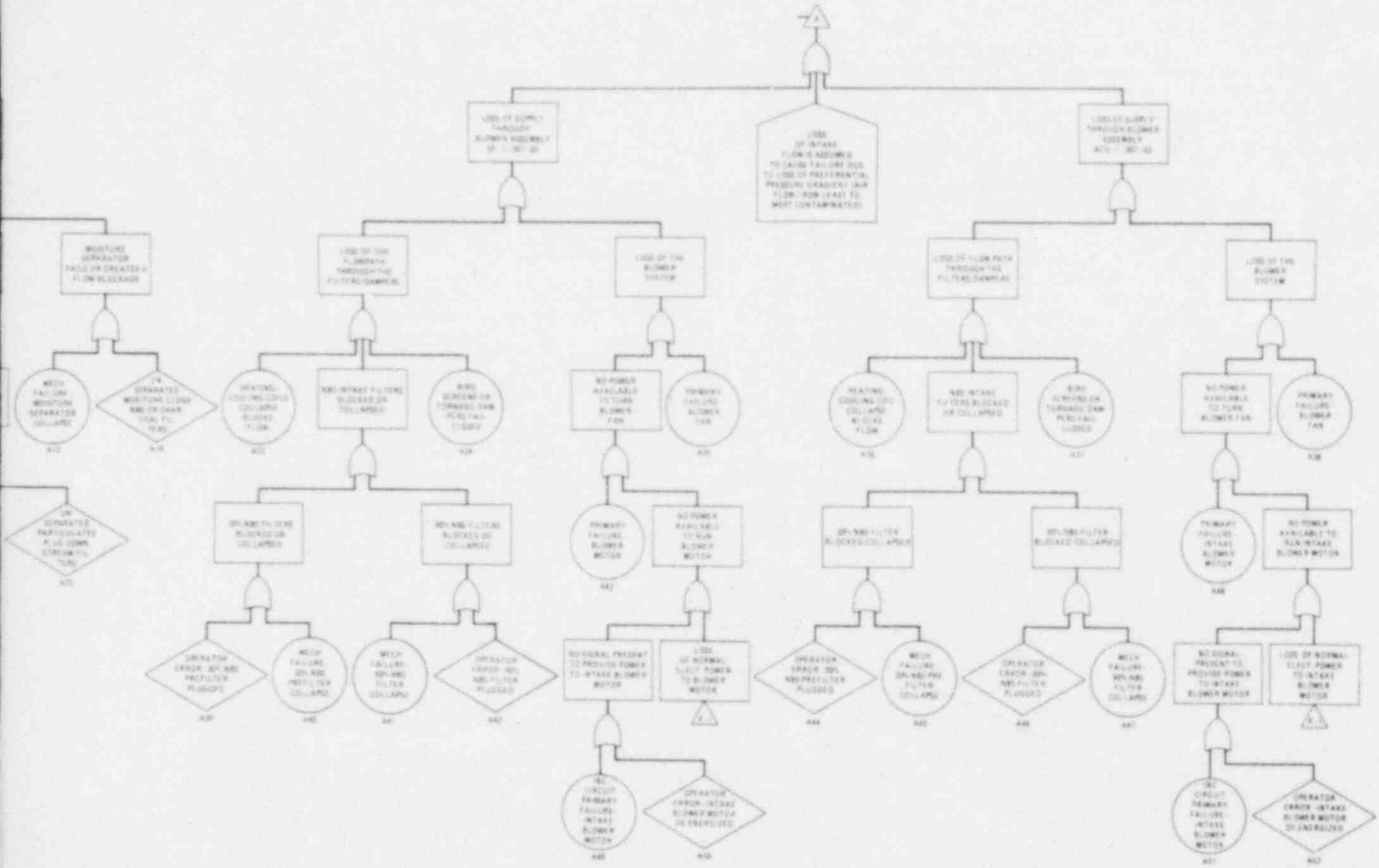


Fig. 4-1. Waste handling building primary confinement exhaust system

(Sheet 1 of 2)



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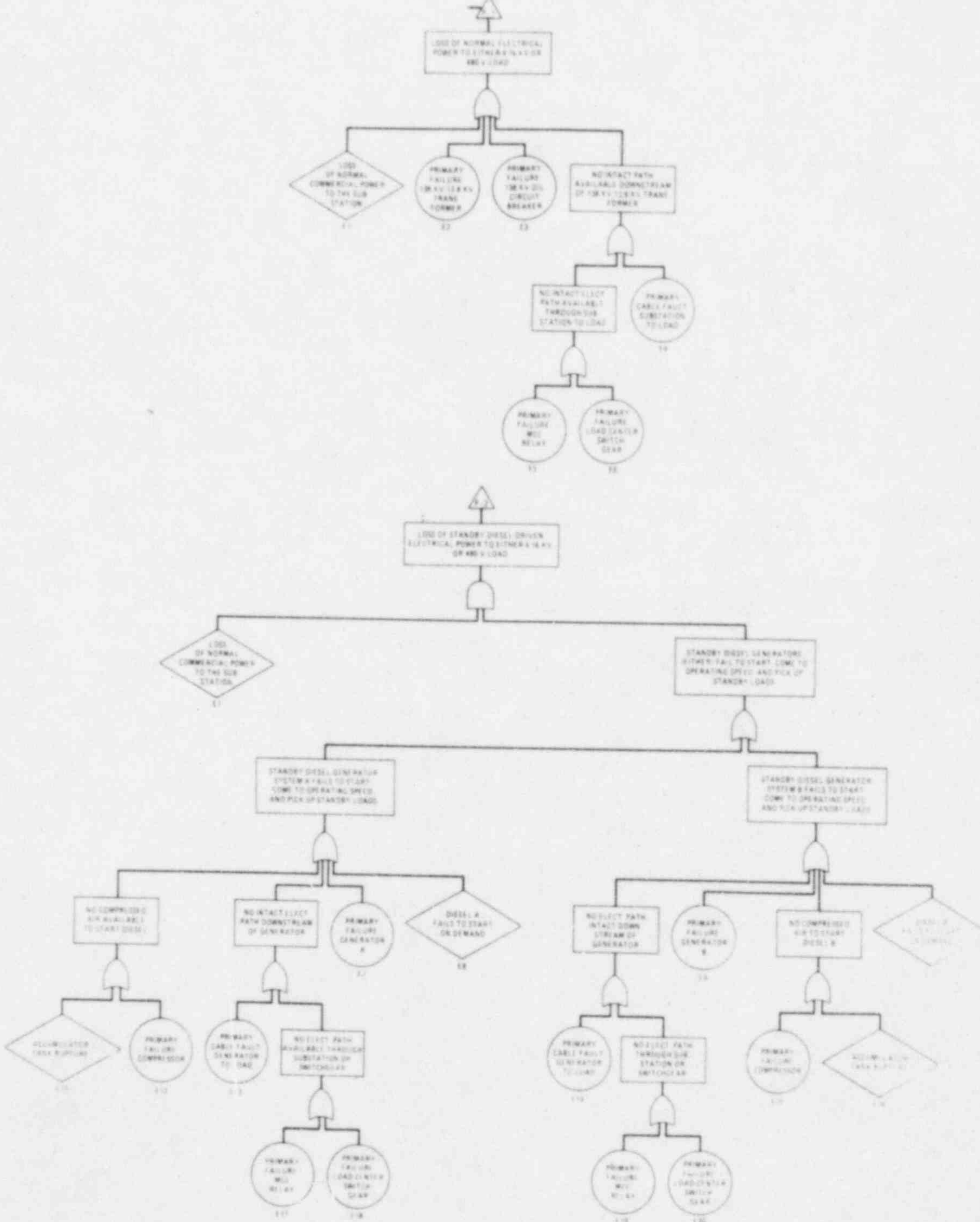

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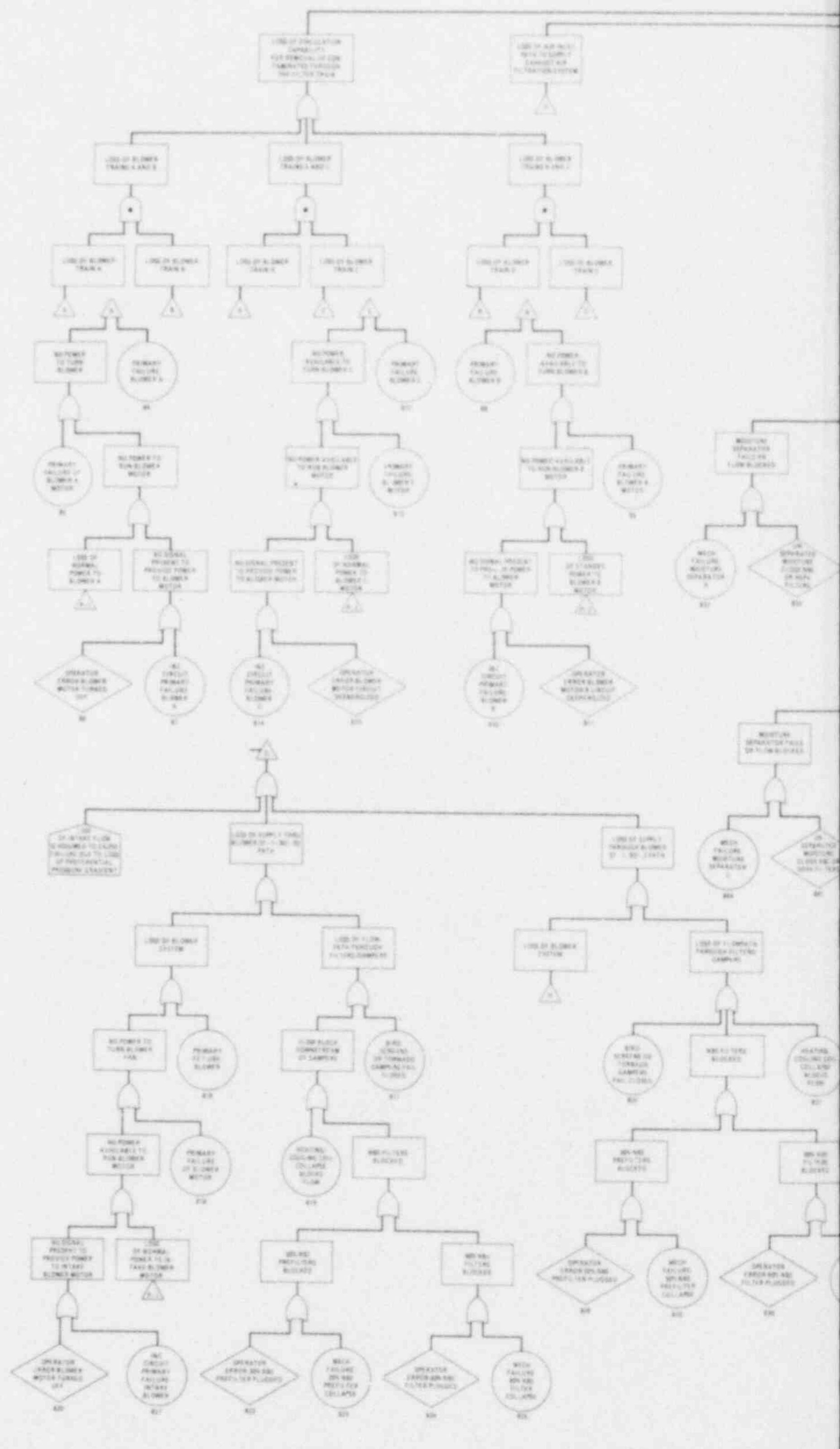
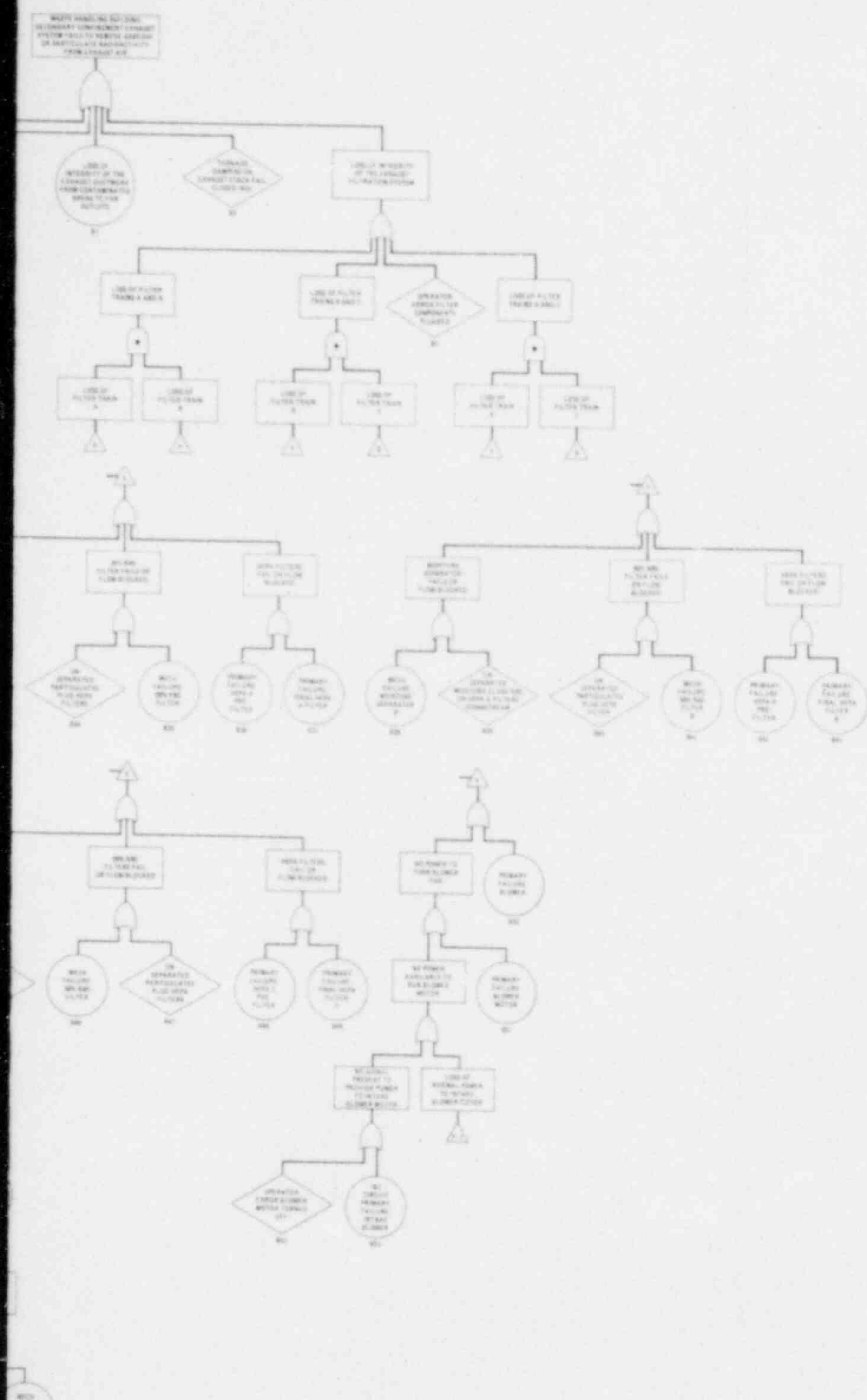


Fig. 4-2. Waste handling building secondary confinement exha



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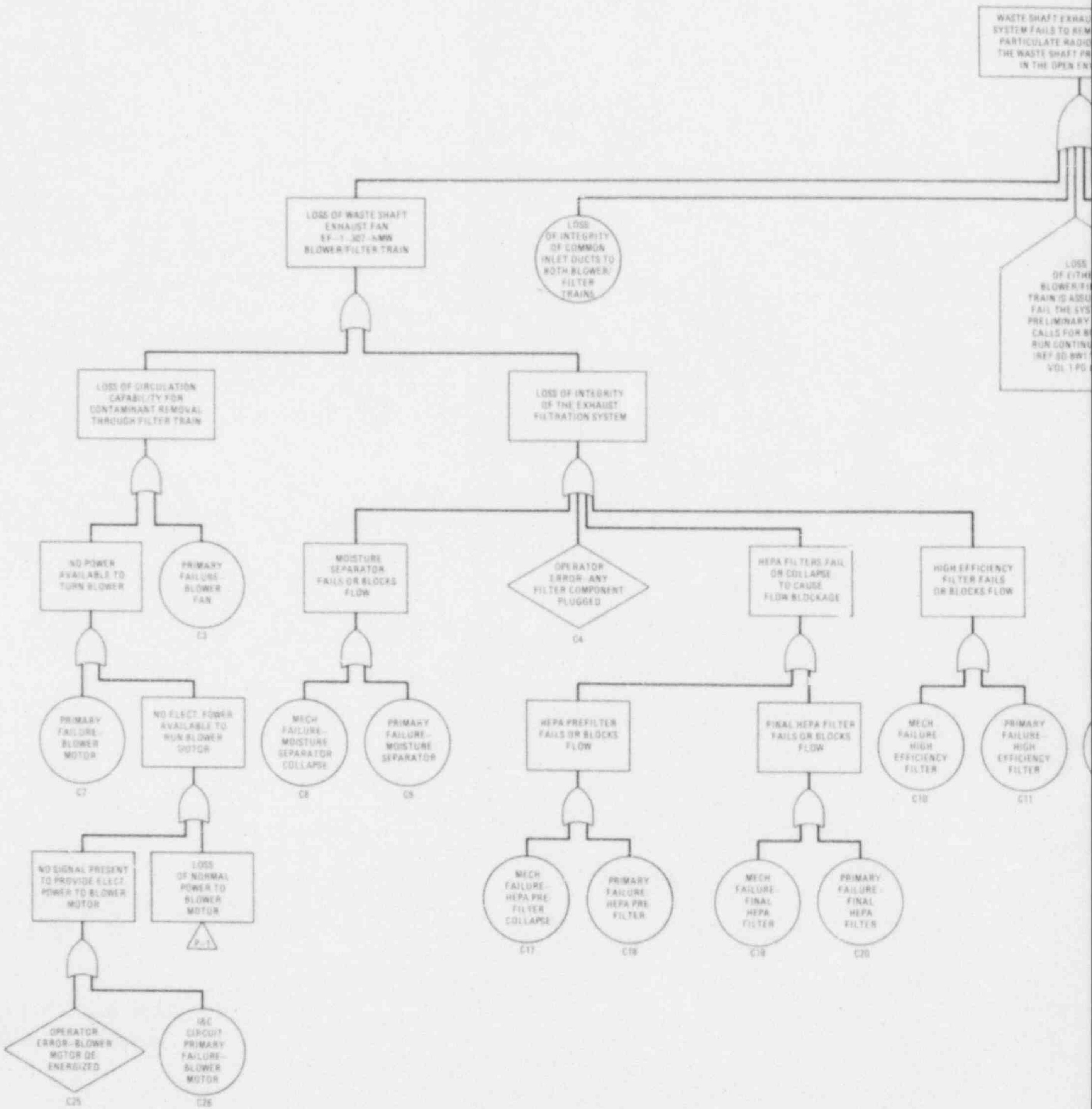
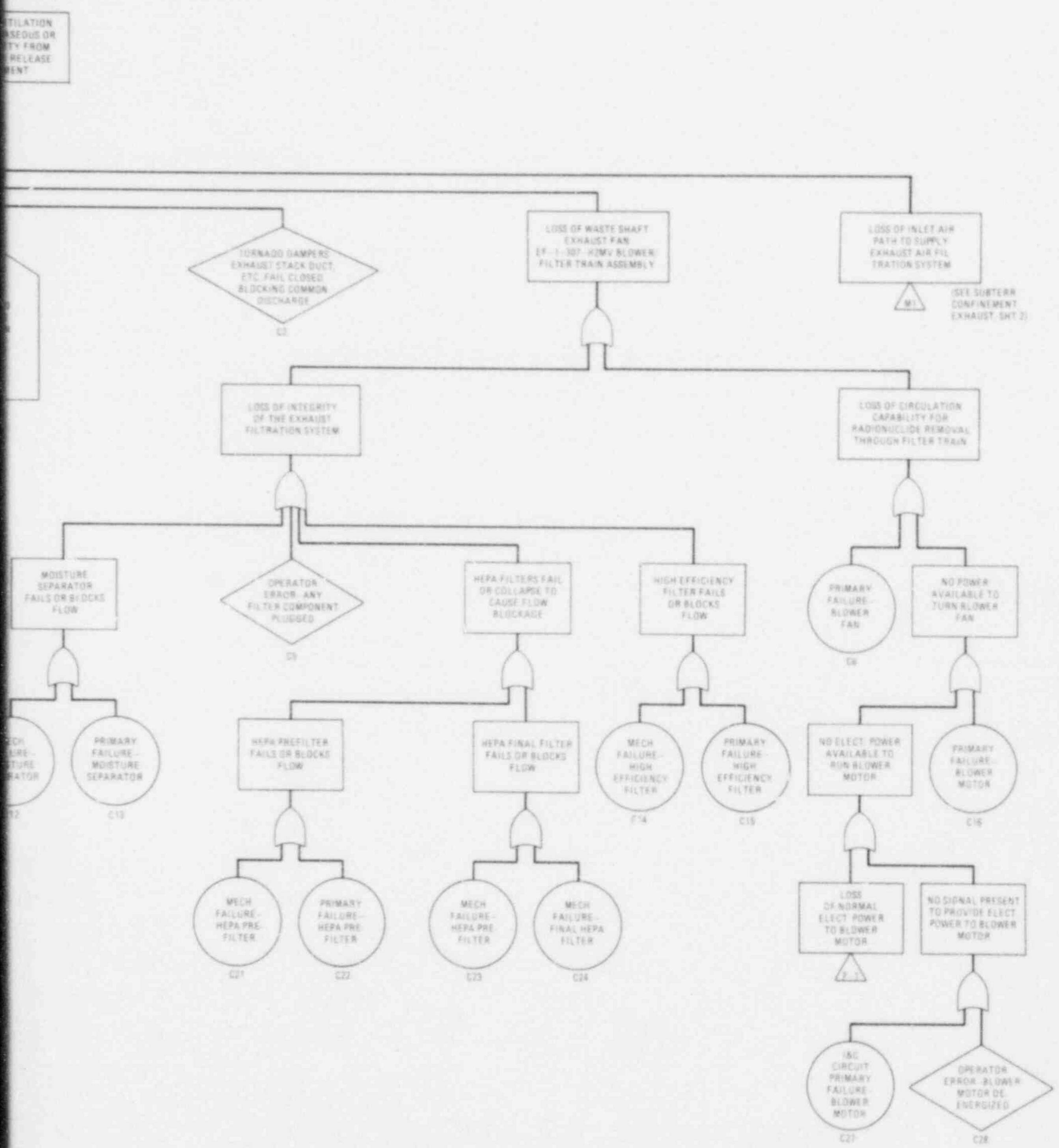


Fig. 4-3. Waste Shaft exhaust system.



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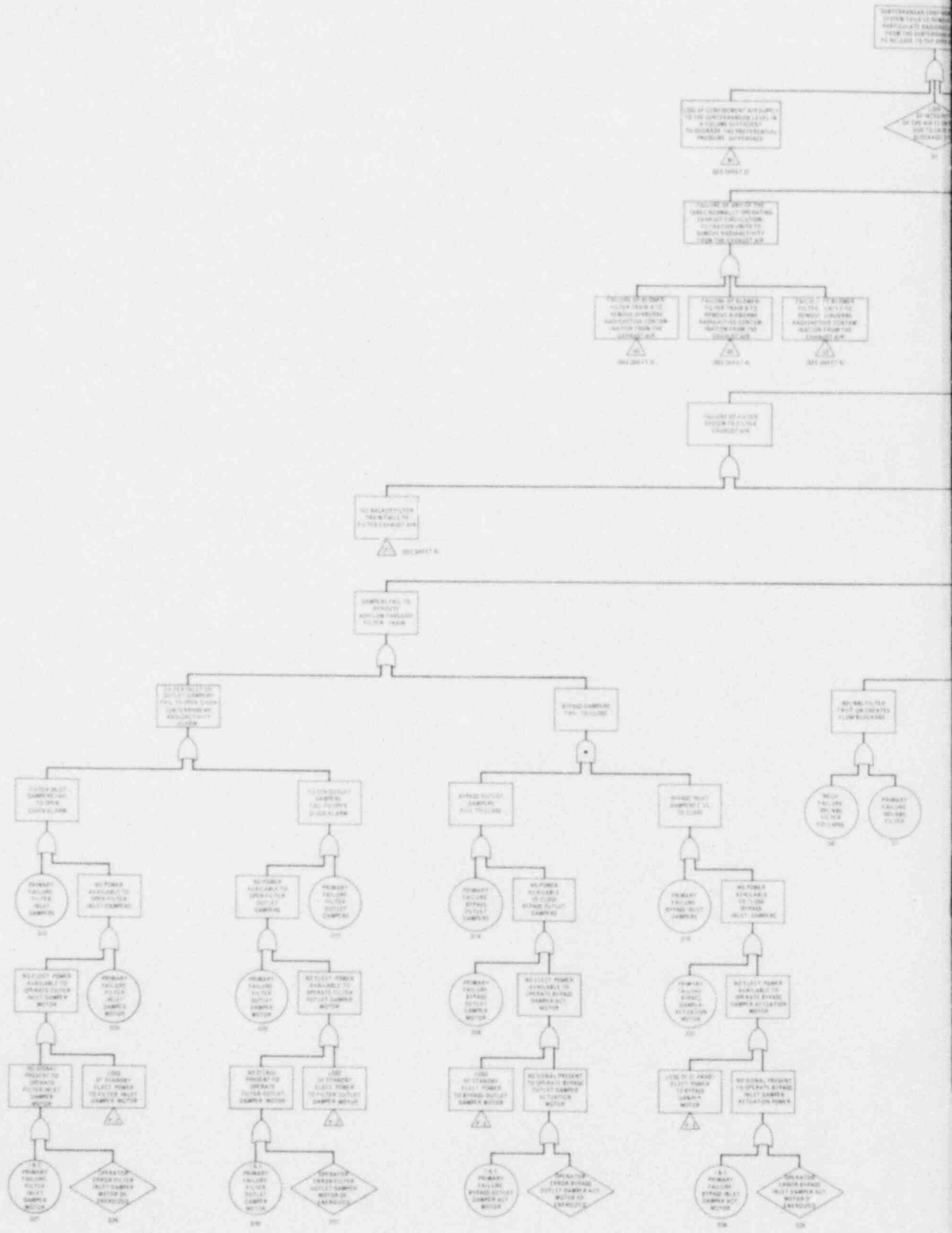
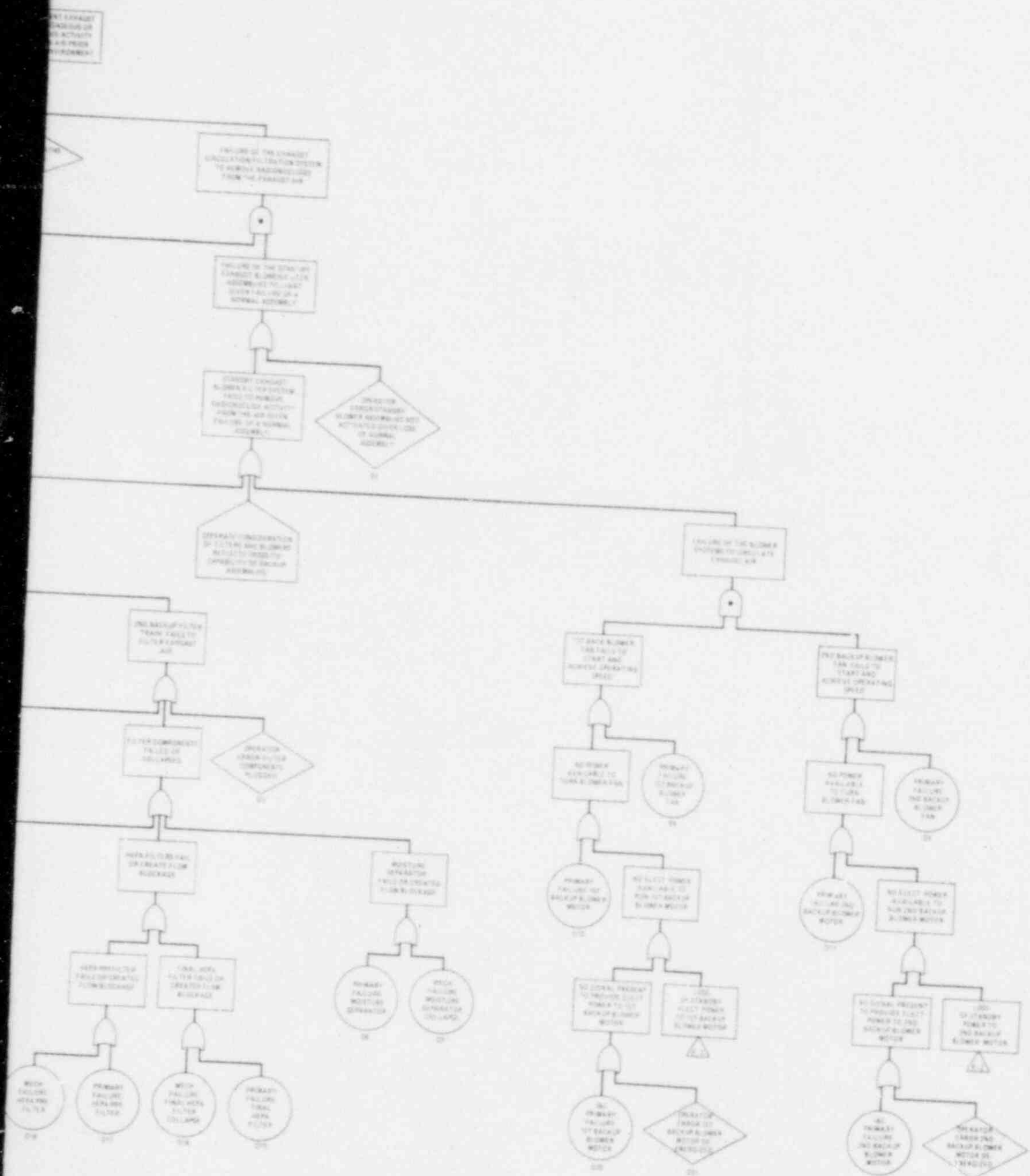


Fig. 4-4. Subterranean confinement exhaust system (Sheet 1 of



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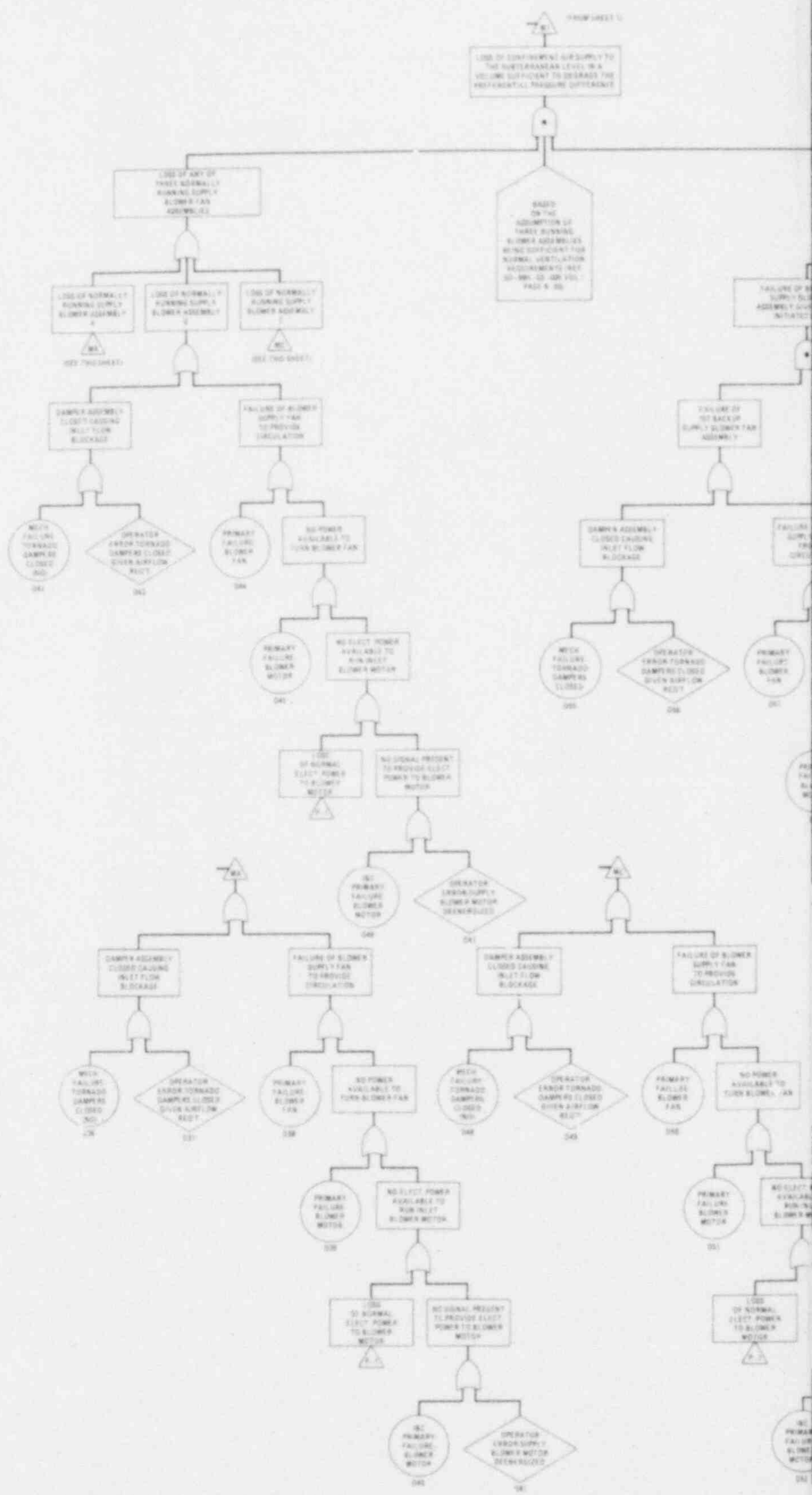

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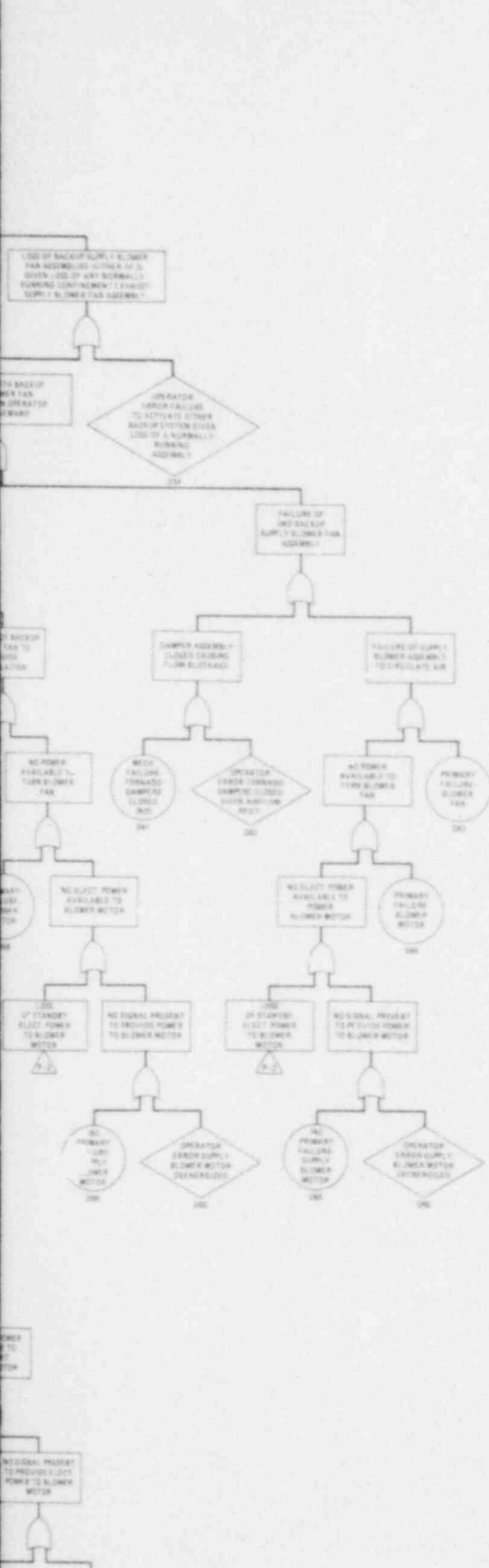

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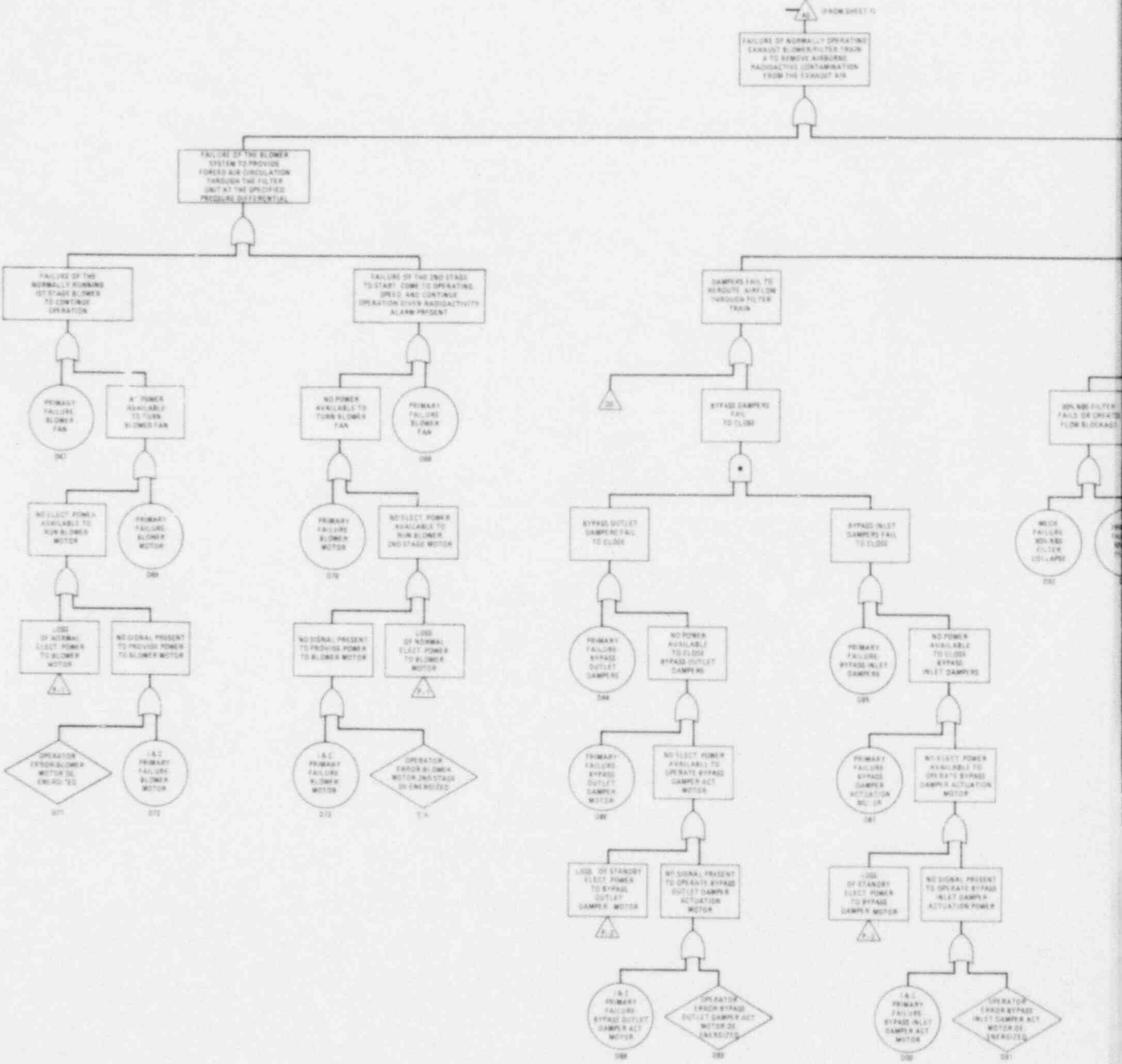

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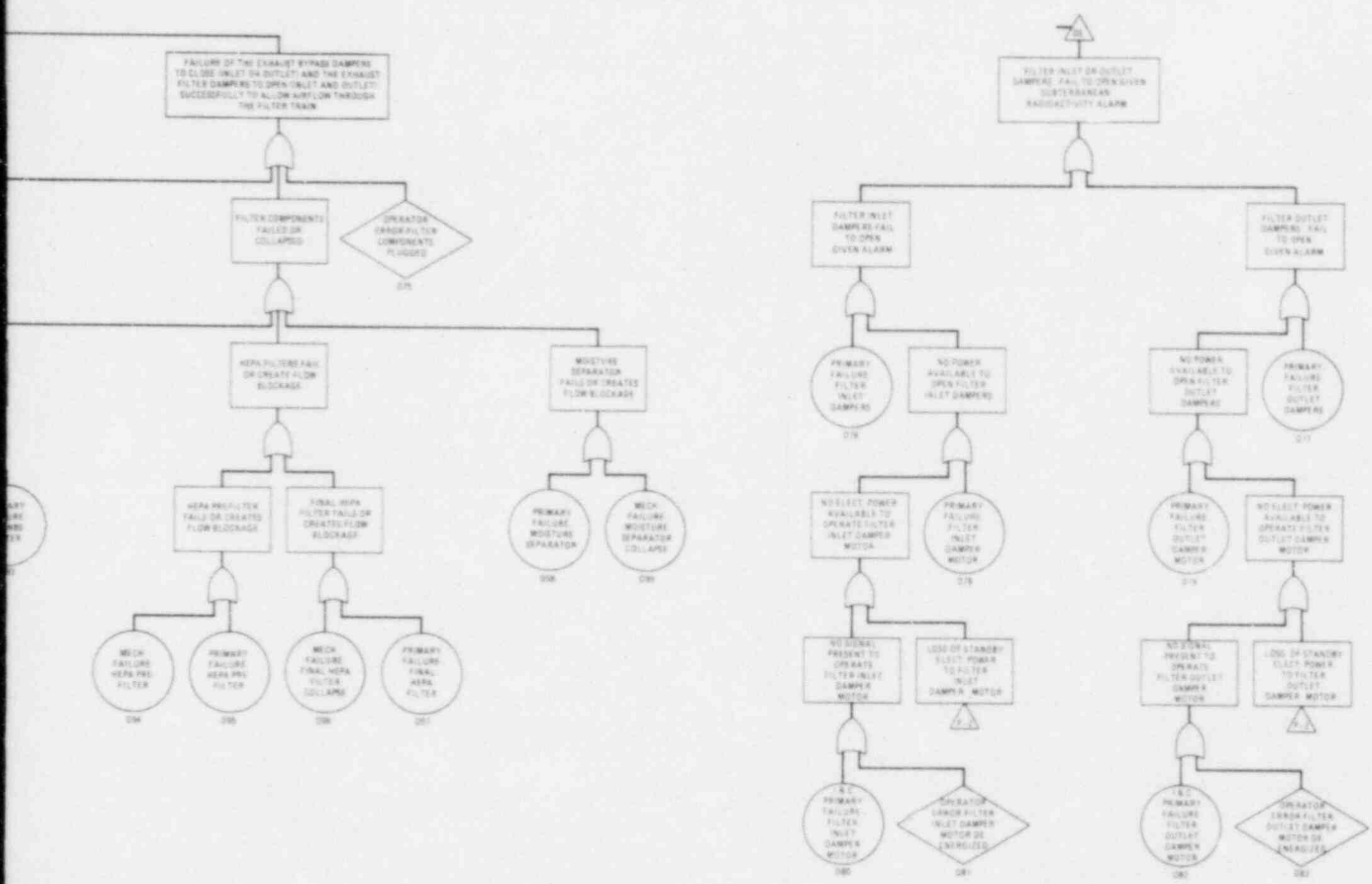

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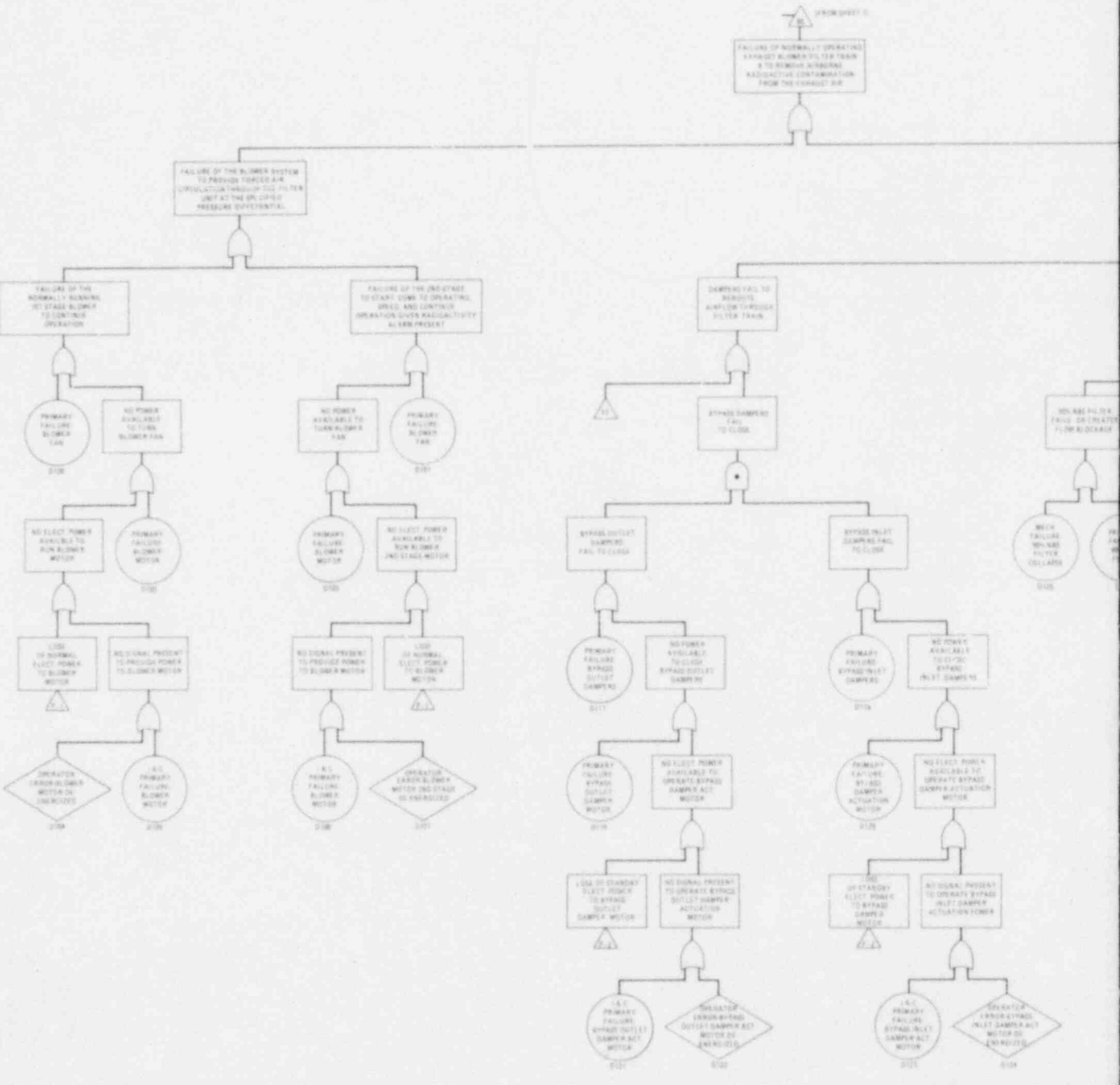

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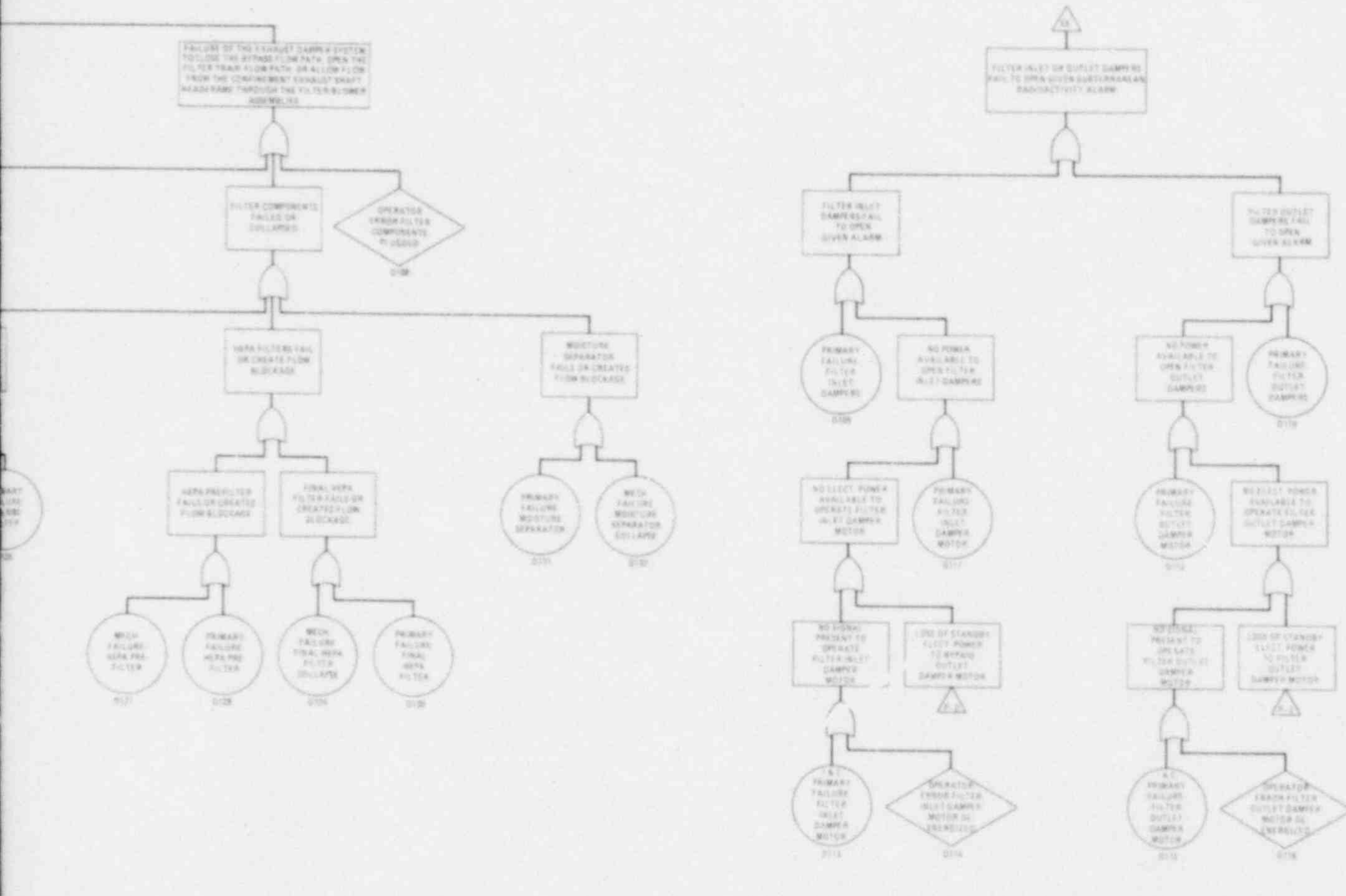

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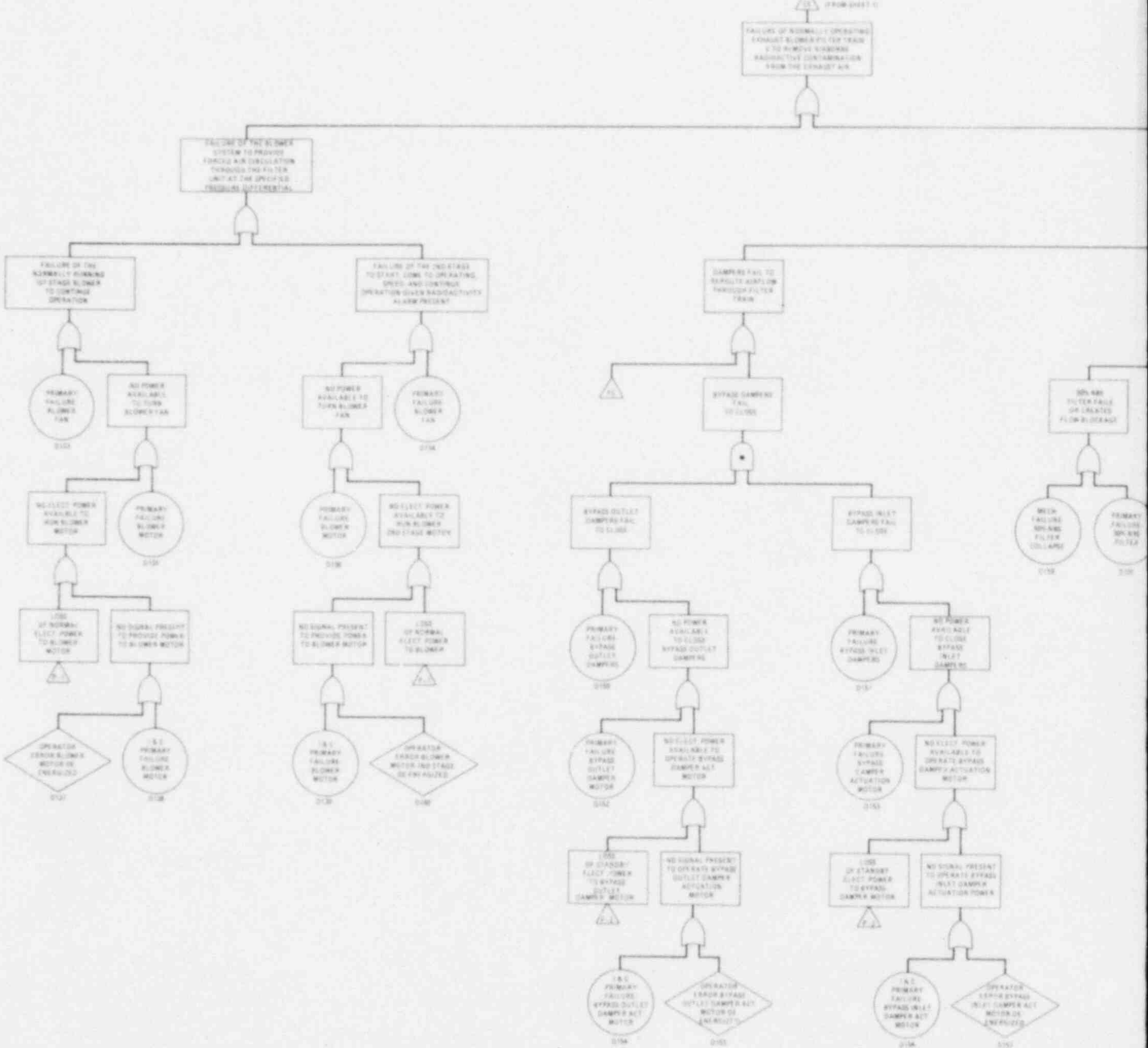

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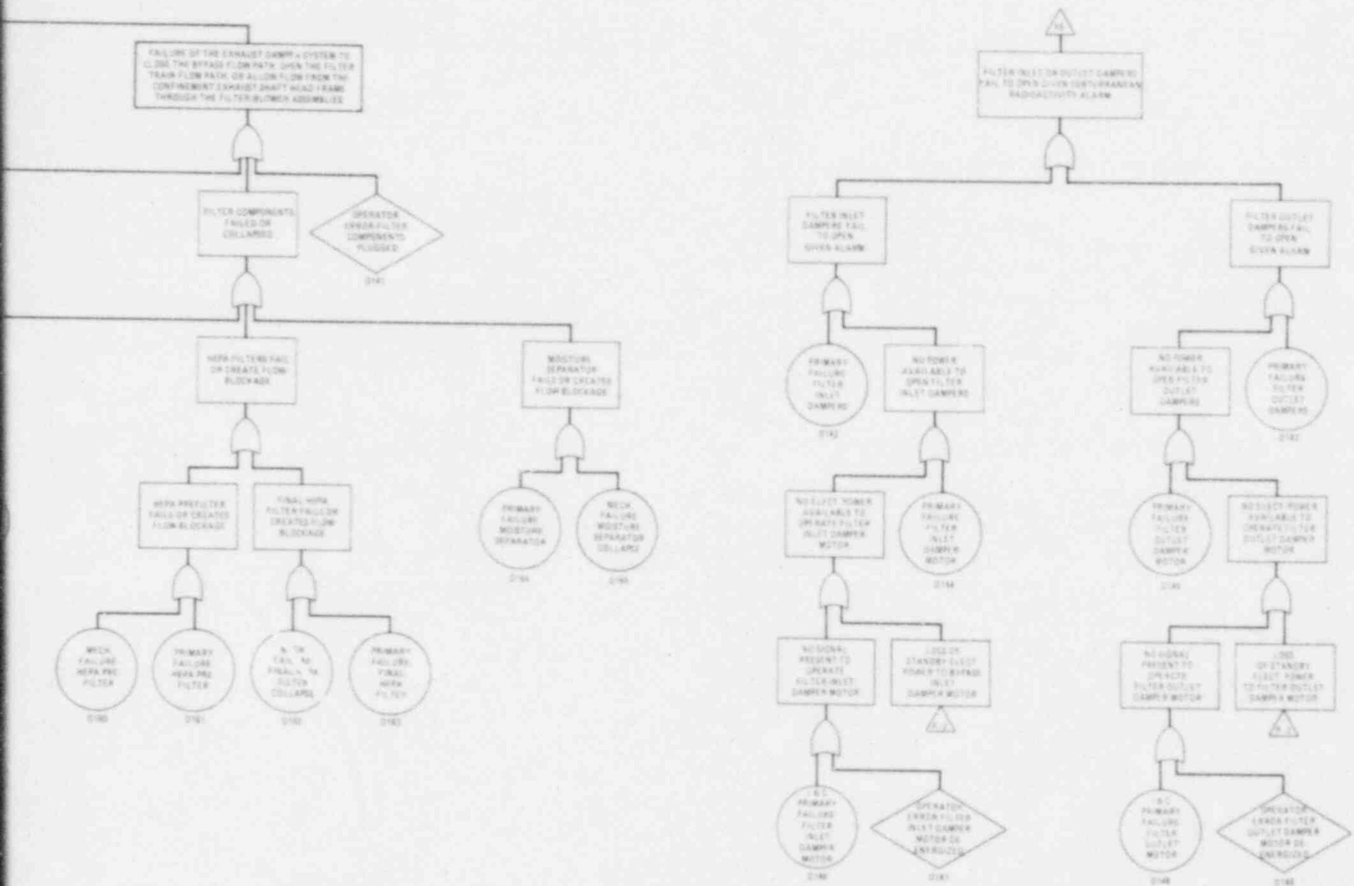

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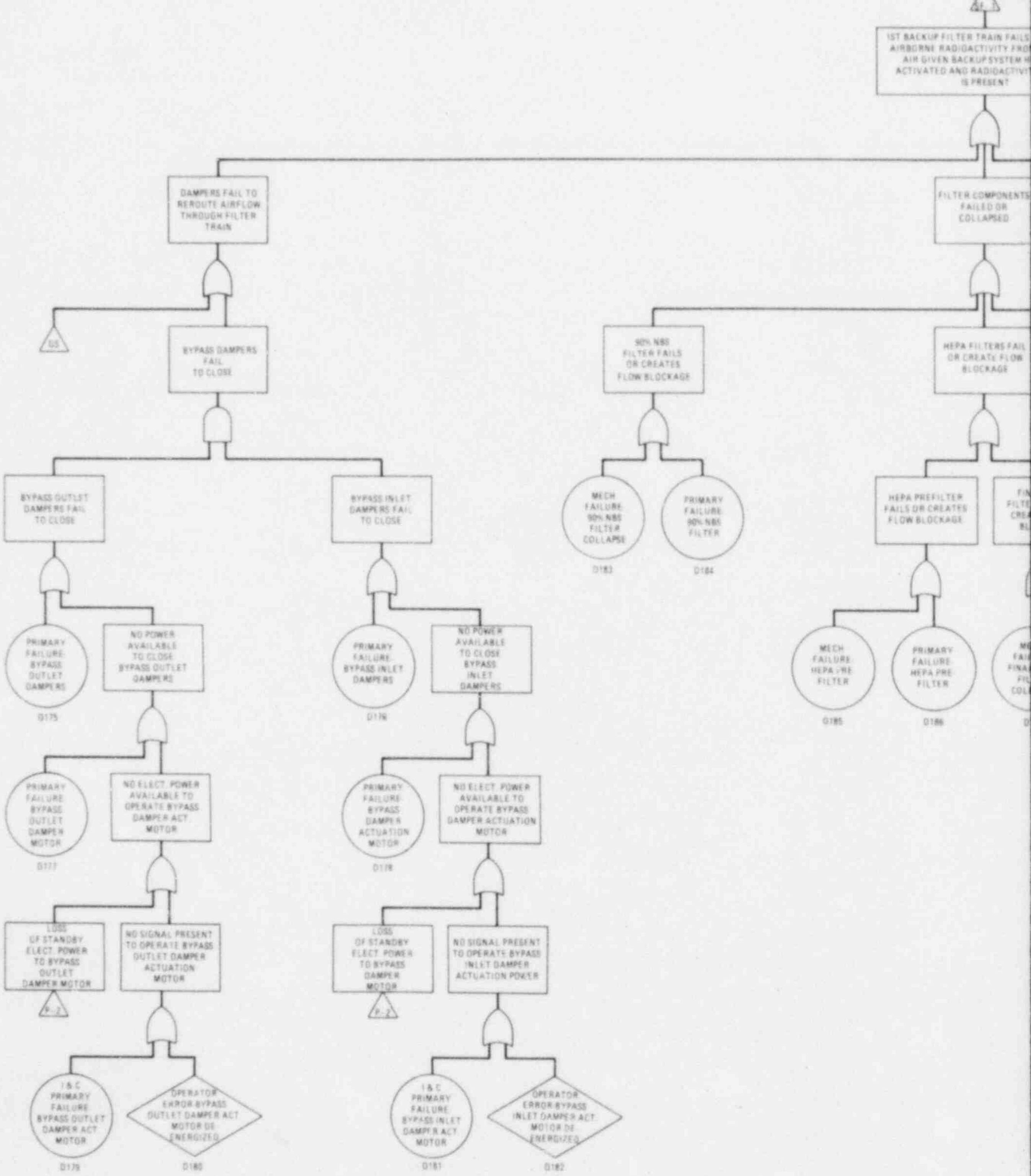

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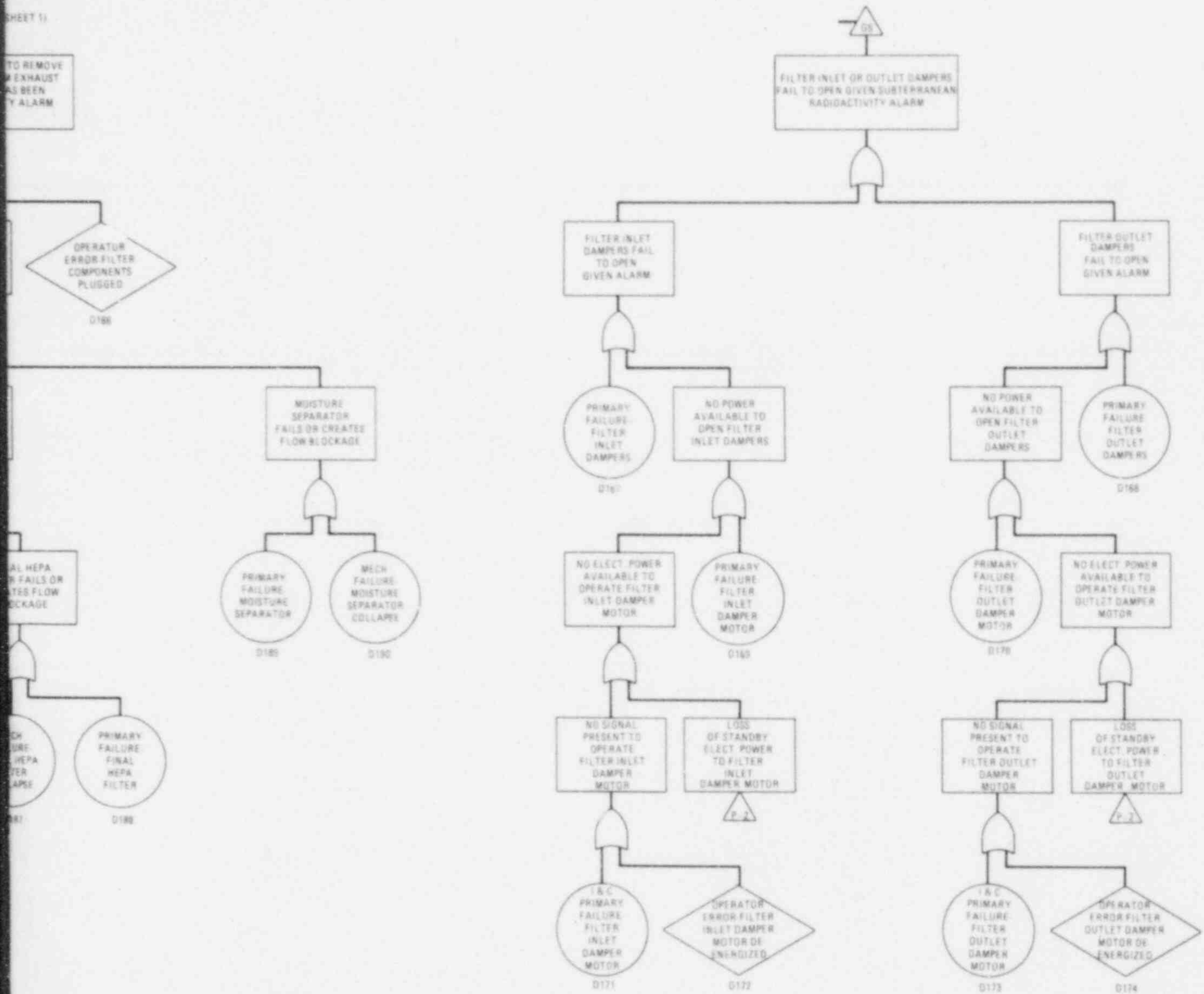

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5.0 SAMPLE PROBLEM FOR HLW-PSSA METHODOLOGY DEMONSTRATION

The sample problem selected for this analysis is a small subset of the complete family of repository accident scenarios leading to public radiological exposure (Ref. Section 2.3). The primary objectives in solving a simplified problem are to (1) demonstrate the methodology, in particular the ranking process; and (2) to provide a test case for the computer programs used in determining risk values and importance ranking. This test case must be simple enough to allow hand calculations of key results to validate computer methods.

The list of accident scenarios chosen for the sample problem is given in Table 5-1. The identifiers used for initiating and intermediate events are the same as those used in Table 2-18. A bar across the top of an intermediate event implies successful operation; lack of a bar implies failure. Each intermediate event can be identified by referring to the indicated event tree (by number) and matching the appropriate intermediate event designator (by letter).

The eleven accident sequences contained in Table 5-1 represent all major waste processing areas identified in Table 2-1. They also require the fault trees developed in Section 4 as intermediate events. Two of the sequences are external events (earthquake and windstorm). External events are included to demonstrate how they are incorporated into the overall risk estimation. Human error is included at both the component and system interaction levels.

The use of all existing fault trees and a reduced number of event tree sequences is optimal for checking computer program calculations of risk and importance ranking. At the fault tree level, no verification is required. The SETS program for fault tree reduction is an established analysis tool in use for many years. The VALUE computer program (see literature review report for this project - Ligon, 1984) quantifies the minimum cut set expressions developed in SETS. VALUE has been verified previously, including the importance ranking of components using a modified (normalized to one) Fussell-Vesely importance measure.

Accident sequences from the event trees also require reduction, quantification and importance ranking of intermediate events. To facilitate the event tree error checking process the number of accident sequences used in the sample problem has been restricted to a size amenable to hand calculations.

It should be emphasized that the results of the sample problem cannot be interpreted as a preliminary boundary estimate of repository pre-closure risk. Restriction of the problem to consideration of only selected accident sequences makes the results only useful as an exercise in applying the analytical methods. Meaningful results must necessarily be generated from the entire family of accident sequences generated for a specific consequence type.

TABLE 5-1

SAMPLE PROBLEM ACCIDENT SCENARIOS

Accident Sequence	Initiating Event Description
A1 $\overline{B1}$ $\overline{C1}$ D1	Train/Truck Collision or Derailment
A2 $\overline{B2}$ $\overline{C2}$ D2 E2	Breached Shipping Cask Undetected by Radiation Monitoring System in Yard Area.
WD $\overline{B5}$ C5	Windstorm Damages Arrival/Storage Yard.
A8 $\overline{B8}$ C8 $\overline{D8}$ E8	Radwaste Sampling Line Rupture or Improper Connect in Receiving Area
A9 B9 $\overline{C9}$ $\overline{D9}$ F9 G9 H9	Loss of Seal between Hot Cell Floor and Shipping Cask Lip.
A12 $\overline{B12}$ C12 $\overline{D12}$ E12	Liquid Radwaste Leak from Process Tank to Secondary Area.
A13 B13 C13 D13	Loss of Seal between Hot Cell Floor and Transfer Cask Lip.
A14 B14 C14 D14	Transfer Cask Rupture during Handling Accident - Subterranean Transport.
A15 B15 $\overline{C15}$ D15 $\overline{E15}$	Transporter Collision during Transport to Placement.
A18 B18 C18 $\overline{D18}$ E18	Canister Breach during Borehole Insertion.
EQ B9 $\overline{C9}$ D9 E9	Earthquake Causes Loss of Seal between Hot Cell Floor and Shipping Cask Lip.

6.0 IMPORTANCE RANKING

A major objective of the High-Level Waste Preclosure Systems Safety Analysis (HLW-PSSA) project, is the development of a systematic methodology to identify and quantitatively prioritize the structures, components, systems, and operations which are important to safety during the preclosure phase of the HLW repository. Although the repository is still at the conceptual stage, analysis at this time is directed mainly at identifying the components or systems that are dominant risk-contributors in order that design modifications can be made (if needed) before construction and operation.

The selection of importance measure(s) for use in this project has to consider several factors relevant to the repository such as the extent of the availability of repository-specific data, compatibility with analytical tools that will be used in the preclosure risk assessment, the ease of applicability to repository situations, etc.

Several importance ranking measures that have been used in the nuclear industry to study systems performance were evaluated for the purpose of identifying one or two measures that would be used in risk evaluations of preclosure activities at proposed high-level-waste repositories. They are: the Birnbaum measure, structural measure, criticality measure of basic event importance, upgrading function, Fussell-Vesely measure of basic event and minimal cut set importance, Barlow-Proschan measure of basic event and minimal cut set importance, the sequential contributory measure of basic event importance, the significance indices, and the risk importance measures. Details of the study, including the discussion of the different importance measures are presented in the appendix.

6.1 EVALUATION

A set of criteria was developed to identify those measures which are applicable to the safety and risk evaluation of a nuclear waste repository. These criteria are:

1. The importance measure should use readily available repository data (i.e., component failure rates and repair times) and not require a prohibitive level of data detail.

2. Useful insights on a repository system performance should be provided by the importance measure.

3. The measure should be easily applicable to repository situations and it should yield scrutable results.

4. Numerical importance ranking should be provided by the measure.

5. Uniform ranking of all components (or events) represented in the fault tree model should be possible.

6. The measure should be applicable to repairable and non-repairable components.

7. A computer program should be available to calculate the rankings.

6.1.1 Computer Programs

Of the computer programs that are currently available for ranking basic events and cut sets in their order of importance to system failure (i.e., top event occurs) we have considered the following ones because of their ability to accept as input the minimal cut sets generated from the SETS (Worrell, 1978) fault-tree reduction code:

IMPORTANCE (Lambert, 1981) allows calculation of various importance measures such as the structural measure, the Birnbaum measure, the criticality measure, the Fussell-Vesely (F-V) measure, the Barlow-Proschan (B-P) measure, and the sequential contributory (S-C) measure.

In addition to minimal cut sets (obtained from SETS), this code requires as input the failure rates and repair times of all basic events contained in the minimal cut sets. The failure and repair distributions are assumed to be exponential. All measures are computed assuming statistical independence of basic events.

The SEP (Set Evaluation Program) computer code (Olman, 1982) provides a means of measuring the contribution of each basic event to system failure by taking the product of the Birnbaum measure and the event probability. When normalized by the system failure (top event) probability, this expression essentially becomes the Fussell-Vesely measure. SEP calculates these measures separately for the noncomplemented and complemented occurrence of each event. These measures are calculated assuming that the event and its complement are independent events. The sum of these two measures yields the true importance measure for the event.

VALUE (Harris, 1982) ranks the basic event and minimal cut set contribution to top event probability using the F-V equation, modified to yield importance rankings which sum to unity.

The computer code STADIC-2 (Koch, 1983), which is a general purpose Monte Carlo simulation code, can also be used to calculate importance measures such as the F-V measure and to propagate data uncertainties, thus, providing importance ranking factors in the form of distributions.

6.2 PRELIMINARY SCREENING

The significance index is directly related to the upgrading function. The risk reduction worth and risk appreciation worth measures do not seem to offer any advantage over the other basic importance measures that were considered (See the Appendix). Hence, these measures are not addressed explicitly in the preliminary screening results shown in Table 6-1.

Based on the results of the study, the following measures have been screened out as inadequate in meeting project objectives:

- o Birnbaum measure
- o Structural measure
- o Upgrading Function
- o Barlow-Proschan measure

The remaining importance measures, namely, criticality, Fussell-Vesely (F-V) and Sequential Contributory (SC) were subjected to more rigorous analysis to determine whether they rank components in a consistent manner and to determine how the ranking behaves as a function of various factors such as repair time, top event probability, minimal cut set order, etc.

6.3 RECOMMENDED MEASURE

From the results of the analysis, the F-V measure was found to be most suitable for ranking component importance under a variety of conditions (i.e., as a function of system unavailability, unreliability, or both). Furthermore, the F-V method provides ranking for both basic events and minimal cut sets. In general, the use of the F-V measure will provide importance rankings that would be both useful and meaningful in assessing the safety of repository systems and operations.

Although none of the computer codes for calculating importance measures included an analysis of the uncertainty in the failure rate and repair data, the inherent straightfowardness of the F-V equation easily permits the calculation of probability distributions for the importance measure. STADIC-2 is further recommended to perform the uncertainty distribution calculations for the HLW-PSSA project. This program is a validated tool for combining distributions according to an input algorithm. The F-V measure for both components and systems can then be evaluated in the form of distributions.

TABLE 6-1
IMPORTANCE MEASURE SELECTION CRITERIA

CRITERIA	BIRNBAUM'S	STRUCTURAL	CRITICALITY	UPGRADING FUNCTION	FUSSELL-VESELY[a]	BARLOW-PROSCHAN	SEQUENTIAL CONTRIBUTORY
1. Uses readily available repository data	yes	yes	yes	yes	yes	yes (to some extent)	yes (to some extent)
2. Provides useful insights on a repository system performance	yes	yes	yes	yes	yes	yes	yes
3. Easily applied to repository situations and provides scrutable results	yes	yes	yes	yes	yes	yes (to some extent)	yes (to some extent)
4. Provides numerical ranking	yes	yes	yes	yes	yes	yes	yes
5. Uniformly ranks all components (or events) represented in the fault tree model	no	no	yes	yes	yes	Initiating events only	Enabling events only
6. Applicable to repairable and non-repairable components	yes	yes	yes	Non-repairable only	yes	Repairable only	yes
7. Computer program available to calculate the rankings	yes	yes	yes	yes	yes	yes	yes

7.0 DATA BASE

This section presents a summary of data derived from the literature and government/industry data banks pertaining to the scenarios, fault trees and consequences discussed in Sections 2 to 4. Emphasis is given to those scenarios selected for quantification in the next study phase of the HLW-PSSA project (see Section 5). As noted in Section 1, the data gathering task will be continued in the next phase. Hence, the data presented here are not complete, particularly in the areas of human reliability and operator injuries resulting from industrial accidents (i.e., non-radiological). The data are grouped into the following categories:

(1) Initiating event frequency data (Sec. 7.1)
(2) Intermediate event probability data (Sec.7.2)
(3) Basic event failure data (Sec. 7.3)
(4) Human error rates (Sec. 7.4)
(5) Radiological data (Sec. 7.5)
(6) Occupational injury data (Sec. 7.6)

7.1 INITIATING EVENT FREQUENCY DATA

Data for the initiating events identified in Section 2 are presented in Table 7-1.[1] For some initiating events representing a system failure (e.g., radiation monitoring system, hot cell collar seal, hydraulic clamping/locking system, etc.), where not enough design detail on the specific system is currently available, generic data for primary components of the system are given. In subsequent work involving quantification of _all_ accident scenarios, these systems will have to be modeled in more detail in order to include more realistic data. No uncertainty bands are reported for the data shown. In the next study phase, which includes quantification of selected scenarios to demonstrate the methodology, a more detailed uncertainty analysis will be performed.

7.2 INTERMEDIATE EVENT PROBABILITY DATA

Mean-value probability data for intermediate events identified in the event trees presented in Section 2 are shown in Table 7-2. Many are conditional probability values which were derived from previous studies and involved significant engineering judgement. Uncertainty analyses will be performed in the next study phase to establish uncertainty bands for these values.

7.3 BASIC EVENT FAILURE DATA

Failure data for the events identified in the fault trees in Section 4 are presented in Table 7-3. The mean values and the variance about the means are shown, assuming a lognormal distribution. The basis for the use of lognormal distribution for basic event failure rate is discussed below.

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[1]All tables are presented at the end of the section.

7.3.1 Data Uncertainty

Estimates of failure rate and repair times derived from various data sources are subject to uncertainties. It is not always specified what failure modes are represented, what environment is applicable, and what is the total population. For some events, event description matching is not possible; therefore events operating under conditions that are similar to the event under consideration are selected as representative. This ambiguity results in uncertainty that is reflected in the spread of the probability distribution for the event.

For the events listed in Table 7-3, the assumption of a lognormal distribution for the failure rates seems to be adequate. The lognormal distribution was explicitly used in WASH-1400 (NRC, 1975) and other PRA studies of nuclear power plants because of its mathematical behavior. The format of the data contained in various sources differ as to the distribution parameters provided; hence, a consistent approach for presenting the data is given here using the properties of a lognormal function as shown in the following equations.

$$\text{Mean} = \quad = e^{(\mu + \sigma^2/2)} \qquad (7\text{-}1)$$

$$\text{Median} = \lambda_{50} = e^{\mu} \qquad (7\text{-}2)$$

$$\text{Variance} = \beta^2 = e^{(2\mu + \sigma^2 \quad)} (e^{\sigma^2} - 1) \qquad (7\text{-}3)$$

$$\text{Range Factor} = RF = \frac{\lambda_{95}}{\lambda_{50}} = \frac{\lambda_{50}}{\lambda_{05}} = \sqrt{\frac{\lambda_{95}}{\lambda_{05}}} \qquad (7\text{-}4)$$

$$\lambda_{95} = e^{(\mu + 1.645 \quad)} \qquad (7\text{-}5)$$

$$\lambda_{05} = e^{(\mu - 1.645 \quad)} \qquad (7\text{-}6)$$

$$\mu = \ln \lambda_{50} \qquad (7\text{-}7)$$

$$\mu = \frac{\ln RF}{1.645} \qquad (7\text{-}8)$$

Using these relations it is possible to calculate the median, range factor, mean, variance, 95% confidence level and 5% confidence level, given that any two of these variables are known.

7.4 HUMAN ERROR RATES

Quantification of human error rates is not within the scope of the present work. The types of human errors identified in the event tree and fault tree levels requiring quantification are listed in Table 7-4.

7.5 RADIOLOGICAL DATA

The calculation of radiological consequence in terms of radiation doses requires information on the types and amounts of radionuclides present in the waste form under consideration (i.e., SURF and HLW), the release fraction resulting from different release mechanisms, and the dose conversion factors necessary to convert the radioactivity released to radiation dose to man.

7.5.1 Radionuclide Inventories

The radionuclide compositions of spent fuel and HLW were chosen to be consistent with other widely accepted waste management studies. The reference studies all utilize the ORIGEN code to obtain the radionuclide compositions. The results, however, maybe somewhat different because assumptions regarding fuel-cycle options, fuel exposure, plant capacity factor, etc. vary when defining the discharged fuel composition.

The radionuclides present in PWR, BWR, and HLW canisters are given in Tables 7-5, 7-6, and 7-7, respectively. Note that the spent fuel radionuclide inventories are shown as per metric tonne heavy metal. Since the Basalt repository design assumes a waste canister containing 3 PWR assemblies (1.38 MTHM) or 7 BWR assemblies (1.32 MTHM), the data in Tables 7-5 and 7-6 will have to be adjusted for the higher metal content per canister.

7.5.2 Release Fractions

A number of accident scenarios which could lead to a breach of containment barriers and subsequent radionuclide release were identified in Section 2. The mechanisms for the release and the form of release (i.e., gases, volatiles and particulates) were discussed in Section 3.

A key issue in consequence calculations is the determination of the amount of radionuclides released from failed canisters or fuel pins. The insufficiency of data and the accident - specific nature of the release make it quite difficult to provide highly reliable values for the release fractions. Release fractions and decontamination factors are available in a number of reports (Bechtel, 1981; NRC, 1979; DOE/ET-0028, 1978; Wilmot, 1980; Wilmot, 1981; Walker, 1978). Data provided by Walker (Walker; 1978) seem to be the most comprehensive because the author provides the ranges of

values from experiments and current practices as well as his recommended values, when appropriate.

Release fractions from Walker (Walker, 1978) for release mechanisms applicable to this study are shown in Table 7-8. Data for other parameters such as filter efficiency, resuspension factors, and plateout are shown in Table 7-9.

Another source of useful data is provided by Wilmot (Wilmot, 1981). Release fractions for specific radionuclides in a spent fuel are given for various release mechanisms relevant to transportation accidents (Table 7-10). Caution must be exercised when using these data since the conditions assumed in the derivation of these values are considered more extreme than those that could occur in a repository. They can at best be considered to represent an upper bound when applied to a repository facility.

7.5.3 On-site Exposure Data

For on-site exposure calculations, the following types of data are required:

- o Plateout rates - use available data or derive them using PADLOC.
- o Release fraction - use available data or perform more specific calculations.
- o Volume of building.
- o Exhaust rates (ventilation).
- o Filter efficiency - use design - specific data.
- o Natural deposition rates - calculate as a function of particle density, particle size and air viscosity.
- o Maximum permissible concentrations as given in 10CFR20 Appendix B.

7.5.4 Meteorological Data For CRAC-2

Sufficient meteorological information is required to characterize transport processes out to a distance of 50 miles (~80,000 meters) from the facility. The primary source of meteorological information for the Hanford site will be the National Climatic Center (Asheville, North Carolina). The following meteorological data for Hanford, Washington will be obtained:

- o Windspeed (hourly values)
- o Wind direction (hourly values)
- o Atmospheric stability
- o Mixing height
- o Precepitation

7.5.5 Dose Factors

The translation of radionuclide releases to observable doses requires (1) the computation of the concentrations of radioactivity in the air and on the ground and (2) the computation of doses that could accrue from both external and internal irradiation.

Dose factors for external irradiation from a uniformly contaminated ground plane are given in Table 7-11 (NRC, 1977). These factors apply to surface contamination via deposition of airborne effluents on ground surfaces or liquid effluents on shoreline sediments. These factors are derived from a consideration of the dose rate to air 1 meter above the ground plane and the penetration of radiation into the body.

Dose factors for internal exposure are provided in Table 7-12 (NRC, 1977). These dose factors are appropriate for continuous intake over a one-year period and include the dose commitment over a 50-year period.

7.6 OCCUPATIONAL INJURY DATA

Statistical data on expected injuries resulting from mine-related activities were derived from an initial study performed by Engineers International specifically for this project (Engineers International, 1984) and summarized in Table 7-13. These data will be augmented in follow-on activities.

The primary mining systems that will most likely be used during repository development and which could lead to the types of accidents shown in Table 7-13 are:

- o rail haulage
- o rubber-tired haulage
- o conveyor
- o hoisting
- o drill and blast, or continuous miner
- o roof support.

The basis for the accident statistical analysis is the Mine Injury and Worktime Quarterly Report published by the U.S. Department of Labor, Mine Safety and Health Administration (MSHA). This document contains accident statistics for metal, non-metal, stone, and coal operations in the United States. The time period studied was January through September, 1983. This time period was chosen because it corresponds to the most current information available and, for the first time, contractor accident experience statistics are separated from mining experience. This is important because contractor work may have been highly specialized and dangerous, and contractor personnel may have been unfamiliar with the mine or surroundings and therefore prone to accidents.

This analysis attempts to incorporate mining experience for use in waste repository construction and operation. That is, the number of accidents in work-hours in the mining industry was related to job classification and work-hours during peak development of the repository during the first 9 years. This process assumes that the injury incidence rate (IR) ((number of injuries ÷ number of employee hours) x 200,000) for mining is analogous to repository operations. Another assumption is that the working environments of the two industries are equal. However one major assumption of this analysis is that personnel in mines or repositories are equally responsible. This assumption has limitations.

Repository personnel may be generally more educated and better trained in the job before actually performing work. The QA/QC requirements for repository design, construction, and operation will undoubtedly upgrade the operation level above that for a mine, but it is not likely to approach that of reactor construction experience.

A repository operation, for many reasons, cannot afford major accidents due to the nature of the operation. Schedules will be made and kept if possible, but incentives may not exist, and production is not as economic related as the mining industry counterpart. Consequently, the rush for "extra footage" or a push to meet the "standard" advance rate for the week may not exist for the repository personnel.

7.6.1 Accident Statistics

Table 7-14 from MSHA data summarizes 1983 accident experience. For the purpose of this study, only underground metal, non-metal, and stone mines were examined, as their mining equipment and methods are more analogous to repository construction in hard rock. Fatal, non-fatal days lost (NFDL), and no days lost (NDL) accidents were studied. These kinds of accidents were classified by cause, and the following classifications were studied:

- electrical
- explosives and breaking agents
- fall of rib or face
- fall of roof
- fire
- haulage trucks
- other powered haulage
- hoisting
- ignition of gas
- drilling.

7.6.2 Analysis

Personnel requirements by job classification and number per shift were used to develop Table 7-15. Table 7-15 shows how development crew requirements change within the first 9 years of repository development. Preliminary documents assume 350 work days per year. Actual working hours at the face are assumed to be 6 hours per shift. Consequently, for the purposes of this study, the development crew (as defined in Table 7-15) will spend 1,247,000 work-hours during the first 9 years. Trammers and electricians are considered separately since many accidents occur within these job classifications. The total underground personnel work-hours during the first 9 years is 4,706,100. These figures are to be related to mining accident experience for underground personnel during the first three- quarters of 1983, where 25,602,195 work-hours were expended.

7.6.2.1 Example. Roof fall injuries in 1983 include 2 fatalities, 73 NFDL, and 54 NDL accidents. These numbers represent 25.0%, 9.8%, and 11.3% of the total accidents in the respective class of injuries. The problem is to relate the 2 fatalities of 25,602,195 work-hours to the 1,247,400 work-hours of the face development crew during the first 9 years of repository construction. Other job classifications are not included for this category because roof falls primarily occur at or near the face during development.

All fatalities in 1983 yield an incidence rate of 0.06 for 200,000 work-hours. Roof fall fatalities are responsible for 25% of 0.06. This product, when multiplied by the number of 200,000 work-hour units of the development crew in the first 9 years (6.237) yields a product of 0.094, the expected injury rate for roof fall fatalities during construction. This number is very small, but realistic in the sense that many underground hard rock mines have operated for a long time without a roof fall fatality. Similar calculations were performed for NFDL and NDL accidents.

A tabulation of calculations as described above appears in Table 7-13. The injury numbers are estimated, but not unreasonable given the facts that repository personnel will be more highly trained, will be working with new equipment and will use better-maintained equipment than their industrial counterparts. Roof fall incidents may be greatly reduced due to the much lower extraction ratios that will help keep roof stresses to a minimum.

Several classifications result in zero reported fatalities. This means that the expected fatalities are extremely low for all classifications. It may be safe to say that NFDL and NDL accidents will occur in all classifications.

One limitation of this data analysis is that MSHA data do not differentiate the method of mining in presenting statistics; that is, the data include all types of mining techniques, not just room-and-pillar which will be practiced during repository construction. This fact is important because other development methods such as raising, or draw point development, can be much more hazardous due to working in closer quarters than with driving main drifts and crosscuts.

Injuries resulting from ignition and fires are not common occurences, and cannot be adequately identified by studying statistics of one year. Table 7-16 illustrates the causes of fires in metal and non-metal mines from 1968 to 1979. Electrical fires are the most numerous, representing over 42% of the total.

7.6.3 Accident Categories

Fatal accidents are, by definition, those which result in death. Nonfatal days lost (NFDL) accidents are those that require days away from work or days of restricted activity. No days lost (NDL) accidents are those resulting in medical treatment other than first aid, but do not result in days away from work.

The severity for these accidents categories is difficult to determine because much depends on workers' attitudes. Some miners are happy to get time off while others cannot be kept away from work. Fatalities and nonfatal days lost are extremely important to repository operations because high incidence could lead to public concerns as to repository safety and consequent radionuclide isolation.

A fatality is considered equal to a loss of 6,000 work-hours (National Safety Council, 1974). MSHA data do not indicate what injury resulted from any particular accident classification, but they contain total days lost within that classification. Unfortunately these data are not yet available for 1983, so the figures in Table 7-17 are taken from 1981 injury experience in underground metal mining. The reader is reminded that these figures include contractor accident experience. The table illustrates that severity is related to the accident classification, that is, roof falls are inherently more severe than falls of face and rib judging from the days of partial permanent disability. Explosives accidents are also more severe in NFDL, than the average of all accidents.

Table 7-1
INITIATING EVENT FREQUENCY DATA

	Initiating Event	Frequency	Reference	Comments
1.	Train collision/derailment	12.6-6[a] per train mile	Bechtel, 1981	
	Severe with fire	1.5-9 per mile	DOE, 1979	
	Moderate with fire	8.0-8 per mile	DOE, 1979	
2.	Truck accident	1.44-4 per mile	Bechtel, 1981	
	Severe with fire	8.2-9 per mile	DOE, 1979	
	Moderate with fire	3.1-7 per mile	DOE, 1979	
3.	Radiation Monitoring System	1.1-4 per hour	SRENCO, 1978	
	a) Sensor/detector/indicator	6.81-5 per hour	SRI, 1981	
	Sensor/detector/indicator	1.098-5 per hour	IEEE, 1984	
	b) Alarms	2.33-6 per hour	IEEE, 1984	
4.	Aircraft crash			
	a) Fatal accidents	6.17-8 per mile	FAA, 1981	These values will be used to determine aircraft crash frequency following model described in (Pepping, 1981).
	b) Aircraft movements (Spokane, Wash)	18,339 per year	FAA, 1981	
5.	Earthquake	1.0-6 per year	Stottlemyre, 1979	For a moderately sized earthquake (~6.7 Richter scale) in any 100 km^2 area - Hanford.

[a] $12.6\text{-}6 = 12.6 \times 10^{-6}$

Table 7-1 (Continued)

	Initiating Event	Frequency	Reference	Comments
6.	Tornado activity	6.824-6 per year[b]	Stone, 1983	Within 100 miles of Hanford site.
		1.0-6 per year	Davis, 1983	Within 100 miles of Hanford site
7.	Fire in rail switching and truck warehousing depots	-	-	Data may be available from the National Fire Protection Agency, New York
8.	Explosion in rail switching and truck warehousing depots	-	-	Same
9.	Radwaste sampling line rupture/burst	3.973-6 per hour	FARADA, 1972	Corrugated stainless steel wire braid 400 psig
	(Stainless steel flex line mechanical failure)	3.940-6 per hour[b]	Bhaskaran, 1979	Flex metal hose includ-ing pipe connection.
10.	Hot cell collar seal failures			
	a) Rotating lip gasket and seal	<2.23-7 per hour	GIDEP, 1981	
	b) Pressure door (hydraulic)	3.765-5 per hour	FARADA, 1972	

(b) Recommended value.

Table 7-1
(Continued)

Initiating Event	Frequency	Reference	Comments
11. Vehicle hydraulic clamping/locking system failure			
a) Interlock system	1.2-3 per year	Bechtel, 1981	
b) Hydraulic piston	5.508-6 per hour	Bhaskaran, 1979	
c) Orientation control system	2.088-5 per hour	Bhaskaran, 1979	
d) Hydraulic piping system	3.4095-5 per hour	Bhaskaran, 1979	
e) Mechanical pin connections	3.250-6 per hour	Bhaskaran, 1979	
f) Pneumatic actuator	2.010-6 per hour	IEEE, 1984	
g) Magnetic jack latch drive mechanism	7.80-7 per hour	IEEE, 1984	
h) Hydraulic actuator	2.90-7 per hour	RAC, 1978	
12. Hot cell shield door seal (pneumatic seals)			
a) Pressure door (hydraulic)	3.765-5 per hour	FARADA, 1972	
b) Electromagnetic radiation shields	5.0847-5 per hour	GIDEP, 1975	
c) Pressure door (mechanical)	4.455-5 per hour	FARADA, 1972	
13. Hot cell crane drop			
a) Cranes (bridge, composite)	1.001-4 per hour	IEEE, 1984	
b) Cranes	5.00-5 per hour	Bechtel, 1981	
c) Structural failure of cranes	1.00-5 per hour	SRENCO, 1978	
d) Crane brake fails	3.00-4 per hour	SRENCO, 1978	

TABLE 7-1
(Continued)

Initiating Event	Frequency	Reference	Comments
14. Canisters punctured during handling			
Hydraulic manipulator (actuator)	2.90-7 per hour	RAC, 1978	
Sufficient stress to breach canister (cask lid falls)	1.00-4 per year	SRENCO, 1978	
15. Radwaste piping rupture	1.00-8 per year	SRENCO, 1978	
16. Transfer cask crane fails to hold position			
Crane brake fails	3.00-4 per year	SRENCO, 1978	
Crane cable fails	1.00-5 per year	SRENCO, 1978	
Crane (composite)	1.001-4 per hour	IEEE, 1984	
17. Transfer cask crane mechanical failure resulting in cask drop or canister puncture by sharp object	Same as 16.	Same as 16.	
18. Waste transport cage mechanical failure			
Chain - door interlock fails (relays)	1.00-4 per year	SRENCO, 1978	
Cage catch gear fails	1.00-2 per year	SRENCO, 1978	
Cage structural degradation	1.00-7 per year	SRENCO, 1978	
Tie-down fails (structural)	1.00-5 per year	SRENCO, 1978	
Cable-to-cage attachment fails	1.00-5 per year	SRENCO, 1978	
19. a) Transporter mechanical failure during transport results in transporter collision with moving or fixed objects			
Collision with moving object	1.00-4 per hour	Bechtel, 1981	
Collision with fixed object	4.80-5 per hour	Bechtel, 1981	

TABLE 7-1
(Continued)

Initiating Event	Frequency	Reference	Comments
b) Operator inadvertently moves underground transporter	1.00-3 per year	SRENCO, 1978	
c) Power steering assembly fails	1.044-4 per hour	Bhaskaran, 1979	
20. Borehole shield door failure during installation of canister			
a) Hatch closure latch fails	1.00-5 per year	SRENCO, 1978	
b) Interlock failure	1.20-4 per operation	Bechtel, 1981	
21. Transport dolly failure			
Relay failure	3.00-7 per hour	Bechtel, 1981	
Relay failure	2.16-6 per hour	SRI, 1981	
22. Power/control cable failure			
Power cable	4.84-6 per hour	IEEE, 1984	
Control cable	4.79-6 per hour	IEEE, 1984	
23. Borehole/cask mechanical lock (interlock) fails	1.20-4 per operation	Bechtel, 1981	
24. Borehole shield door jams hydraulic pressure door	3.24-5 per hour	FARADA, 1972	
Mechanical door - personnel, cargo	1.62-5 per hour	FARADA, 1972	
Door mechanism fails	1.00-6 per year	SRENCO, 1978	
25. Storage plug/retainer ring fails			
a) Fasteners, clip, or clamp	<5.54-5 per hour	GIDEP, 1975	
b) Retaining rings (safety coupling)	1.145-6 per hour	FARADA, 1972	

TABLE 7-1
(Continued)

	Initiating Event	Frequency	Reference	Comments
26.	Inadvertent cask movement during canister insertion			
	a) Accidental transporter motion			
	Relay failure	3.00-7 per hour	Bechtel, 1981	
	Relays/circuits fail	2.16-6 per hour	SRI, 1981	
	b) Hydraulic autopositioner inadvertent movement			
	Servo controls (alignment)	6.87-4 per hour	GIDEP, 1981	
	Position/angle indicator (motion)	7.00-5 per hour	GIDEP, 1981	
	Automatic guidance control system	2.088-6 per hour	Bhaskaran, 1979	
	Sleeve engagement mechanism (oper.)	1.308-4 per hour	Bhaskaran, 1979	
	Sleeve engagement mechanism (static)	7.688-5 per hour	Bhaskaran, 1979	
27.	a) Overspeeding of hydraulic ram ejector unit			
	Sleeve engagement mechanism (oper.)	1.308-4 per hour	Bhaskaran, 1979	
	Brakes fail	1.00-5 per year	SRENCO, 1978	
	Speed control device fails	1.00-4 per year	SRENCO, 1978	
	b) Overspeed of placement dolly			
	Brake mechanism fails	1.00-5 per year	SRENCO, 1978	
	Speed control device fails	1.00-4 per year	SRENCO, 1978	

TABLE 7-2
INTERMEDIATE EVENTS FAILURE DATA

Intermediate Event	Failure Rate or Probability of Occurrence	Reference	Comments
1. Receiving area airlock door failure	2.90 - 4 per hour[a]	Graham, 1971	
2. Receiving area radiation monitoring system failure	1.10 -4 per hour	SRENCO, 1978	
3. Explosion given derailment or truck/rail collision in switchyard activities (excluding shipment of explosives)	TBD[b]		Data may be available from the National Fire Protection Agency in New York.
4. Fire given derailment or truck/rail collision in switchyard	.016	Clarke, 1976	
5. Cask remains intack given fire or explosion	.999	SRENCO, 1978	
6. Explosion given aircraft crash	.5[c]	Engineering judgment	Assumes aircraft explodes upon impact.
7. Fire given aircraft crash	.34	Clarke, 1976	
8. Explosion given earthquake	.01[c]	Engineering judjment	

(a) 2.90-4 = 2.90×10^{-4}

(b) TBD = To be determined.

(c) The probability distribution to be developed for these values will reflect the uncertainty in assigning the most reasonable value in lieu of the absence of more specific data.

TABLE 7-2 (Continued)

Intermediate Event	Failure Rate or Probability of Occurrence	Reference	Comments
9. Fire given earthquake	.10[c]	Engineering judgment	Assumes earthquake severe enough to damage power lines, resulting in fire.
10. Explosion given windstorm	.01[c]	Engineering judgment	
11. Fire given windstorm	.10[c]	Engineering judgment	Assumes wind velocity strong enough to damage power lines, resulting in fire.
12. Explosion given fire in switch yard area	TBD		Data may be available from the National Fire Protection Agency in New York.
13. Cask preparation area airlock door failure	2.90-4 per hour	Graham, 1971	
14. Unload area airlock door	2.90-4 per hour	Graham, 1971	
15. Hot cell port seal (pneumatic seal) failure	1.0-5 per year	SRENCO, 1978	
16. Radiation monitoring alarm for secondary confinement exhaust system failure	1.10-4 per hour	SRENCO, 1978	

[c] The probability distribution to be developed for these values will reflect the uncertainty in assigning the most reasonable value in lieu of the absence of more specific data.

TABLE 7-2 (Continued)

Intermediate Event	Failure Rate or Probability of Occurrence	Reference	Comments
17. Cask to borehole seal intact given canister breach due to cask motion	TBD		Need more design information to predict seal failure given this condition.
18. Personnel access shaft cage failure	1.0-5 per year	NRC, 1975	Structural failure
19. Waste transport shaft cage failure	1.0-5 per year	NRC, 1975	Structural failure

TABLE 7-3
BASIC EVENT FAILURE DATA

Event ID	Event Description	Failure Rate H (D)[a] Mean	Failure Rate H (D)[a] Variance	Repair Time H Mean	Repair Time H Variance	Reference
A7,A10,A43, A48,C7,C16, D10,D11,D39, D45,D51,D52, D58,D64,D69, D70,D102,D103, D135,D136	Exhaust blower motor (300-500 HP)	1.689-5[b]	1.603-10	2.100+1	7.100+2	NCSR, 1979 (failure rate) GIDEP, 1984 (repair)
B5,B9,B13, B18,B51	Exhaust blower motor (3000 HP)	3.001-5	5.490-9	1.070+2	7.170+2	SRI, 1981 and Westinghouse, 1978 (failure rate)[c] SRI, 1978 (repair)
D24,D25,D26, D27,D78,D79, D86,D87,D111, D112,D119,D120, D144,D145,D152, D153,D169,D170, D177,D178	Damper motor (<1HP)	4.242-6	1.011-11	4.0	9.0	SRI, 1981 and RAC, 1978 (failure rate)[c] SRI, 1980 (repair)

(a) H = hrs   D = Demand

(b) 1.689-5 = 1.689×10^{-5}

(c) Composite data derived from the sources listed.

TABLE 7-3 (Continued)

Event ID	Event Description	Failure Rate H (D)[a] Mean	Failure Rate H (D)[a] Variance	Repair Time H Mean	Repair Time H Variance	Reference
A3,A5,A35, A38,B4,B8, B12,B16,B50, C3,C6,D4,D5, D38,D44,D50, D57,D67,D68, D100,D101, D133,D134	Exhaust blower fan	5.361-5	1.615-9	1.800+1	6.300+1	GD, 1963; Graham, 1971; NCSR, 1979; EPRI, 1982; EPRI, 1981; SRI, 1981 (failure rate)[c] EPRI, 1982; EPRI, 1981 (repair)[c]
A24,A49,A51, B7,B10,B14,B21, B53,C26,C27, D28,D30,D32, D34,D40,D46, D52,D59,D65, D72,D73,D80	Instrumentation and controls circuit failure	4.234-6	1.007-11	8.00	3.600+1	SRI, 1981 (failure rate) Engineering judgment (repair)
A11,A16,A22, A29,B34,B40, B47, D7,D93, D126,D159, D184,C11,C15, A17,A25,B36, B42,B48,C18, C22,D17,D95, D128,D161,D186, A20,A28,B37, B43,B49,C20, C24,D19,D97, D130,D163,D188	Air filters (NBS, HEPA, charcoal) primary failure (rupture, out of limits)	3.154-6	6.740-11	0.3	0.144	GIDEP, 1981; IEEE, 1984; EPRI, 1982; SRI, 1981 (failure rate)[c] GIDEP, 1983 (repair)

(a) H = hrs D = Demand

(c) Composite data derived from the sources listed.

TABLE 7-3 (Continued)

Event ID	Event Description	Failure Rate H (D)[a] Mean	Failure Rate H (D)[a] Variance	Repair Time H Mean	Repair Time H Variance	Reference
A12,A15,A18, A26,C17,C21, D16,D94,D127, D160,D185,A19, A27,C19,C23, D18,D96,D129, D162,D187,A21, A30,A41,A47, B25,B31,B35, B41,B46,D6,D92, D125,D158,D183, A40,A45,B23,B29, C10,C14	Air filters (NBS, HEPA, charcoal) Mechanical failure (collapse)	3.505-7	7.488-12	0.3	0.144	Same as above
A1,A34,A37,B2, B17,B26,C2, D36,D42,D48, D55,D61	Tornado dampers fail closed	4.260-6	2.910-11	8.00	56.2	Duphily, 1982 (failure rate and repair)
D12,D13,D14, D15,D76,D77, D84,D85,D109, D110,D117,D118, D142,D143,D150, D151,D167,D168, D175,D176	Filter dampers fail	5.590-6	1.756-11	13.0	32.0	EPRI. 1981 (failure rate) Duphily, 1982 (repair)

(a) H = hrs D = Demand

TABLE 7-3 (Continued)

Event ID	Event Description	Failure Rate H (D)[a] Mean	Variance	Repair Time H Mean	Variance	Reference
A2,B1,C1,D1	Loss of integrity of exhust duct	2.660-6	4.320-11	12.5	87.8	Duphily, 1982 (failure rate and repair)
A33,A36,B19, B27	Heating/cooling coils collapse and block flow	1.900-5	2.202-9	8.0	36.0	EPRI, 1981 (failure rate) Engineering judgment (repair)
A8,A14,B33, B39,B45,C9, C13,D8,D98, D131,D164, D189	Moisture Separator primary failure	4.748-4	1.408-7	1.0	1.60	FARADA, 1973 (failure rate) Engineering judgment (repair)
A9,A13,B32, B38,B44,C8, C12,D9,D99, D132,D165, D190	Moisture separator collapse - mech. failure	5.276-5	1.564-8	1.0	1.60	FARADA. 1973 (failure rate) Engineering judgment (repair)
E4,E13,E14	Primary cable fault	3.220-6[d]	5.828-12	22.5	285	IEEE, 1984 (failure rate) Duphily, 1982 (repair)
E8,E10	Diesel generator fails to start on demand	(4.000-2)	1.000-3	--	--	NRC, 1975
E7,E10	Diesel generator fails while running (primary failure)	8.000-3	3.890-4	26.0	387.0	NRC, 1975

(a) H = hrs D = Demand

(d) Per 1000 circuit feet

TABLE 7-3 (Continued)

Event ID	Event Description	Failure Rate H (D)[a] Mean	Variance	Repair Time H Mean	Variance	Reference
E11,E16	Accumulator tank rupture	9.259-6	4.819-11	85.0	452.0	SRI, 1981 (failure rate and repair)
E12,E15	Compressor - primary failure	2.498-5	3.508-10	1.0	1.60	SRI, 1981 (failure rate) Engineering judgment (repair)
E1	Loss of normal commercial power	1.594-5	1.429-10	1.0	1.60	EPRI, 1982 (failure rate and repair)
E2	138 KV/13.8 KV transformer fails	6.900-7	8.820-14	61.2	2110.0	IEEE, 1984 (failure rate and repair)
E3	138 KV oil circuit breaker fails to open	1.776-7	1.922-13	10.0	19.4	IEEE, 1984 (failure rate and repair)
E5,E17,E19	MCC relay fails to energize	(1.250-4) 3.749-7	8.768-9 7.891-14	-- 0.41	-- 0.094	NRC, 1975 (failure rate) NRC, 1975 (failure rate) IEEE, 1984 (repair)
E6,E18,E20	Load center switchgear failure	2.752-6	4.615-11	66.0	846.0	IEEE, 1974 and SRI, 1981 (failure rate)[c] IEEE, 1974 (repair)

[a] H = hrs D = Demand

[c] Composite data derived from the sources listed.

Table 7-4

HUMAN ERROR RATES REQUIRING QUANTIFICATION

A. Initiating Events - Emplacement

1. Improper connection of radwaste sample line

    a. low to moderate stress
    b. checklist required
    c. supervisor check required

2. Crane operator error - canister drop

    a. low to moderate stress
    b. heavy equipment operator training required

3. Operator error during remote welding process

    a. low to moderate stress
    b. checklist required
    c. supervisor check required
    d. QA signoff required

4. Valve misalignment of radwaste piping, operator error

    a. low to moderate stress
    b. checklist required

5. Transfer cask crane operator error of commision - cask inadvertently moved during canister insertion

    a. low to moderate stress
    b. checklist required
    c. supervisor check required

Table 7-4

HUMAN ERROR RATES REQUIRING QUANTIFICATION (Contd)

6. Transfer cask crane operator error of commission - cask drop on sharp object resulting in canister puncture

   a. low to moderate stress
   b. heavy equipment operator training required

7. Operator error during shield door installation on borehole

   a. low to moderate stress
   b. checklist required
   c. QA signoff required

8. Operator error during transport dolly installation and removal

   a. low to moderate stress
   b. checklist required
   c. QA signoff required

9. Operator error during alignment of transporter to borehole

   a. low to moderate stress
   b. checklist required with specified position tolerances

10. Operator error during lock of cask to borehole mouth

    a. low to moderate stress
    b. checklist required

Table 7-4

HUMAN ERROR RATES REQUIRING QUANTIFICATION (Contd)

11. Operator error of commissions - cask inadvertently rotated during canister insertion

    a. low to moderate stress
    b. checklist required
    c. QA signoff required for disable of rotation mechanisms during insertion

12. Operator error of commissions - transporter moved during canister insertion

    a. low to moderate stress
    b. checklist required
    d. QA check required for disable of transporter drive during insertion

13. Operator error of commission, gaseous radwaste system inadvertently isotated

    a. low to moderate stress
    b. panel alarms
    c. checklist required
    d. supervisor concurence required

14. Operator error of commission liquid radwaste system inadvertently isolated

    a. low to moderate stress
    b. panel alarms
    c. checklist required
    d. supervisor concurence required

Table 7-4

HUMAN ERROR RATES REQUIRING QUANTIFICATION (Contd)

B. Initiating Events - Retrieval
(Note: only errors not listed for emplacement are included here)

1. Operator error during overcore machine operation causes canister breach

   a. moderate stress (difficult task)
   b. checklist required

   See emplacement for other applicable HERs

C. Human Error Rates Occurring As Intermediate Events

1. Operator fails to actuate standby secondary confinement exhaust system given failure of normal system

   a. medium to high stress
   b. checklist required
   c. emergency training required
   d. supervised activity
   e. alarm activated

2. Operator fails to manually activate subteranean confinement exhaust filtration system given radioactivity detection and failure of auto system

   a. medium to high stress
   b. emergency training required
   c. supervised activity
   d. alarm activated

Table 7-4

HUMAN ERROR RATES REQUIRING QUANTIFICATION (Contd)

D. Human Error Modeled in Fault Trees (Figs 4-1 to 4-4)

1. Operator fails to activate backup blower system (Events D2 and D54)

2. Operator fails to detect plugged filter components during maintenance (Events A4, A6, A39, A42, B3, C4, C5, D3, D75, D108, D141, D166, A44, A46, B22, B24, B28, B30)

3. Operator fails to open tornado damper following maintenance (D37, D43, D49, D56, D62)

4. Operator de-energizes filter damper motor or blower motor (D29, D31, D33, D35, A23, A31, A50, A52, B6, B11, B15, B20, B52, C25, C28, D21, D23, D41, D47, D53, D60, D66, D71, D74, D81, D83, D89, D91, D104, D107, D114, D116, D122, D124, D137, D140, D147, D149, D155, D157, D172, D174, D180, D182)

Table 7-5

PWR Canister Inventory, Ci/MTHM

A total HM content of 461.4 kg is assumed.

	Years After Receipt at Repository					
Isotope	1	5	10	20	50	100
H-3	3.09E+02	2.47E+02	1.86E+02	1.06E+02	1.95E+01	1.17E+00
C-14	2.78E+04	2.78E+04	2.78E+04	2.78E+04	2.76E+04	2.74E+04
Mn-54	1.27E-01	4.48E-03	6.87E-05	1.62E-08	2.10E-19	1.51E-37
Fe-55	4.96E+02	1.71E+02	4.50E+01	3.13E+00	1.05E-03	1.71E-09
Co-60	2.87E+01	1.70E+01	8.77E+00	2.35E+00	4.51E-02	6.20E-05
Ni-59	3.37E+00	3.37E+00	3.36E+00	3.36E+00	3.36E+00	3.36E+00
Ni-63	4.86E+02	4.72E+02	4.54E+02	4.21E+02	3.36E+02	2.31E+02
Zn-65	4.45E-09	7.12E-11	4.06E-13	1.32E-17	4.54E-31	1.65E-53
Se-79	4.08E-01	4.08E-01	4.08E-01	4.08E-01	4.08E-01	4.07E-01
Kr-81	2.34E-06	2.34E-06	2.34E-06	2.34E-06	2.34E-06	2.34E-06
Kr-85	5.44E+03	4.20E+03	3.04E+03	1.60E+03	2.30E+02	9.09E+00
Rb-87	2.43E-05	2.43E-05	2.43E-05	2.43E-05	2.43E-05	2.43E-05
Sr-90	6.42E+04	5.82E+04	5.14E+04	4.02E+04	1.92E+04	5.59E+03
Y-90	6.42E+04	5.82E+04	5.15E+04	4.02E+04	1.92E+04	5.59E+03
Zr-93	3.25E+00	3.25E+00	3.25E+00	3.25E+00	3.25E+00	3.25E+00
Nb-93m	1.59E+00	1.91E+00	2.21E+00	2.60E+00	3.01E+00	3.10E+00
Nb-94	2.25E-01	2.25E-01	2.25E-01	2.25E-01	2.25E-01	2.24E-01
Mo-93	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.27E-02
Tc-99	1.44E+01	1.44E+01	1.44E+01	1.44E+01	1.44E+01	1.44E+01
Ru-106	2.60E+02	1.67E+01	5.42E-01	5.67E-04	6.52E-13	8.22E-28
Rh-106	2.60E+02	1.67E+01	5.42E-01	5.67E-04	6.52E-13	8.22E-28
Pd-107	1.03E-01	1.03E-01	1.03E-01	1.03E-01	1.03E-01	1.03E-01
Ag-108	2.06E-06	2.02E-06	1.96E-06	1.86E-06	1.59E-06	1.21E-06
Ag-108m	2.67E-05	2.62E-05	2.55E-05	2.42E-05	2.06E-05	1.58E-05
Ag-109m	4.24E-08	4.55E-09	2.80E-10	1.05E-12	5.66E-20	4.32E-32
Ag-110	8.67E-04	1.56E-05	1.02E-07	4.43E-12	3.59E-25	5.45E-47
Ag-110m	6.19E-02	1.11E-03	7.32E-06	3.17E-10	2.57E-23	3.89E-45
Cd-109	4.24E-08	4.55E-09	2.80E-10	1.05E-12	5.66E-20	4.32E-32
Cd-113m	1.98E+01	1.64E+01	1.29E+01	8.03E+00	1.93E+00	1.79E-01
In-115	8.36E-12	8.36E-12	8.36E-12	8.36E-12	8.36E-12	8.36E-12
Sn-119m	1.50E-03	2.41E-05	1.37E-07	4.46E-12	1.53E-25	5.57E-48

Table 7-5 (Continued)

PWR Canister Inventory, Ci/MTHM

A total HM content of 461.4 kg is assumed.

	Years After Receipt at Repository					
Isotope	1	5	10	20	50	100
Sn-121m	6.84E-01	6.48E-01	6.04E-01	5.26E-01	3.47E-01	1.73E-01
Sn-123	1.98E-06	7.73E-10	4.25E-14	1.28E-22	3.53E-48	8.88E-01
Sn-126	5.63E-01	5.63E-01	5.63E-01	5.63E-01	5.63E-01	5.63E-01
Sb-125	5.93E+02	2.15E+02	6.04E+01	4.76E+00	2.34E-03	7.18E-09
Sb-126	7.88E-02	7.88E-02	7.88E-02	7.88E-02	7.88E-02	7.88E-02
Sb-126m	5.63E-01	5.63E-01	5.63E-01	5.63E-01	5.63E-01	5.63E-01
Te-123	1.03E-13	1.03E-13	1.03E-13	1.03E-13	1.03E-13	1.03E-13
Te-125m	1.45E+02	5.25E+01	1.47E+01	1.16E+00	5.72E-04	1.75E-09
Te-127	1.13E-07	1.05E-11	9.47E-17	7.76E-27	4.26E-57	0
Te-127m	1.16E-07	1.07E-11	9.67E-17	7.92E-27	4.35E-57	0
I-129	3.27E-02	3.27E-02	3.27E-02	3.27E-02	3.27E-02	3.27E-02
Cs-134	6.70E+03	1.75E+03	3.24E+02	1.12E+01	4.64E-04	2.29E-11
Cs-135	3.43E-01	3.43E-01	3.43E-01	3.43E-01	3.43E-01	3.43E-01
Cs-137	8.65E+04	7.89E+04	7.03E+04	5.59E+04	2.80E+04	8.85E+03
Ba-137m	8.19E+04	7.47E+04	6.65E+04	5.29E+04	2.65E+04	8.38E+03
Ce-142	2.86E-05	2.86E-05	2.86E-05	2.86E-05	2.86E-05	2.86E-05
Ce-144	6.86E+01	1.95E+00	2.27E-02	3.09E-06	7.78E-18	3.62E-37
Pr-144	6.86E+01	1.95E+00	2.27E-02	3.09E-06	7.78E-18	3.62E-37
Pr-144m	8.24E-01	2.34E-02	2.73E-04	3.71E-08	9.34E-20	4.35E-39
Nd-144	1.65E-09	1.65E-09	1.65E-09	1.65E-09	1.65E-09	1.65E-09
Pm-147	6.14E+03	2.13E+03	5.69E+02	4.05E+01	1.46E-02	2.67E-08
Sm-147	3.71E-06	3.81E-06	3.85E-06	3.86E-06	3.86E-06	3.86E-06
Sm-148	6.28E-12	6.28E-12	6.28E-12	6.28E-12	6.28E-12	6.28E-12
Sm-149	9.53E-13	9.53E-13	9.53E-13	9.53E-13	9.53E-13	9.53E-13
Sm-151	8.04E+02	7.81E+02	7.52E+02	6.98E+02	5.58E+02	3.84E+02
Eu-152	4.36E+00	3.52E+00	2.70E+00	1.58E+00	3.19E-01	2.22E-02
Eu-154	4.94E+03	3.58E+03	2.39E+03	1.07E+03	9.48E+01	1.68E+00
Eu-155	5.18E+02	2.91E+02	1.41E+02	3.33E+01	4.36E-01	3.18E-04
Gd-152	2.65E-12	2.67E-12	2.70E-12	2.74E-12	2.78E-12	2.79E-12
Gd-153	2.95E-04	4.40E-06	2.29E-08	6.21E-13	1.24E-26	1.81E-49

Table 7-5 (Continued)

PWR Canister Inventory, Ci/MTHM

A total HM content of 461.4 kg is assumed.

	Years After Receipt at Repository					
Isotope	1	5	10	20	50	100
Ho-166m	6.98E-04	6.96E-04	6.94E-04	6.90E-04	6.78E-04	6.59E-04
TL-207	8.67E-06	1.12E-05	1.42E-05	1.96E-05	3.34E-05	5.27E-05
TL-208	6.37E-03	6.70E-03	6.65E-03	6.14E-03	4.61E-03	2.85E-03
TL-209	2.02E-09	2.58E-09	3.35E-09	5.14E-09	1.25E-08	3.20E-08
Pb-209	9.20E-08	1.17E-07	1.52E-07	2.34E-07	5.69E-07	1.46E-06
Pb-210	3.84E-08	7.97E-08	1.62E-07	4.60E-07	2.86E-06	1.40E-05
Pb-211	8.69E-06	1.12E-05	1.42E-05	1.97E-05	3.35E-05	5.29E-05
Pb-212	1.77E-02	1.86E-02	1.85E-02	1.71E-02	1.28E-02	7.91E-03
Pb-214	2.79E-07	4.70E-07	7.83E-07	1.67E-06	6.66E-06	2.39E-05
Bi-210	3.83E-08	7.95E-08	1.62E-07	4.60E-07	2.85E-06	1.40E-05
Bi-211	8.69E-06	1.12E-05	1.42E-05	1.97E-05	3.35E-05	5.29E-05
Bi-212	1.77E-02	1.86E-02	1.85E-02	1.71E-02	1.28E-02	7.91E-03
Bi-213	9.20E-08	1.17E-07	1.52E-07	2.34E-07	5.69E-07	1.46E-06
Bi-214	2.79E-07	4.70E-07	7.83E-07	1.67E-06	6.66E-06	2.39E-05
Po-210	3.46E-08	7.29E-08	1.51E-07	4.38E-07	2.79E-06	1.38E-05
Po-211	2.61E-08	3.37E-08	4.27E-08	5.91E-08	1.01E-07	1.59E-07
Po-212	1.13E-02	1.19E-02	1.18E-02	1.09E-02	8.20E-03	5.07E-03
Po-213	9.00E-08	1.15E-07	1.49E-07	2.28E-07	5.56E-07	1.42E-06
Po-214	2.79E-07	4.70E-07	7.83E-07	1.67E-06	6.66E-06	2.39E-05
Po-215	8.69E-06	1.12E-05	1.42E-05	1.97E-05	3.35E-05	5.29E-05
Po-216	1.77E-02	1.86E-02	1.85E-02	1.71E-02	1.28E-02	7.91E-03
Po-218	2.79E-07	4.70E-07	7.83E-07	1.67E-06	6.66E-06	2.39E-05
At-217	9.20E-08	1.17E-07	1.52E-07	2.34E-07	5.69E-07	1.46E-06
Rn-219	8.69E-06	1.12E-05	1.42E-05	1.97E-05	3.35E-05	5.29E-05
Rn-220	1.77E-02	1.86E-02	1.85E-02	1.71E-02	1.28E-02	7.91E-03
Rn-222	2.79E-07	4.70E-07	7.83E-07	1.67E-06	6.66E-06	2.39E-05
Fr-221	9.20E-08	1.17E-07	1.52E-07	2.34E-07	5.69E-07	1.46E-06
Fr-223	1.23E-07	1.58E-07	2.00E-07	2.77E-07	4.70E-07	7.41E-07
Ra-223	8.69E-06	1.12E-05	1.42E-05	1.97E-05	3.35E-05	5.29E-05
Ra-224	1.77E-02	1.86E-02	1.85E-02	1.71E-02	1.28E-02	7.91E-03

Table 7-5 (Continued)

PWR Canister Inventory, Ci/MTHM

A total HM content of 461.4 kg is assumed.

Isotope	Years After Receipt at Repository					
	1	5	10	20	50	100
Ra-225	9.23E-08	1.18E-07	1.53E-07	2.34E-07	5.69E-07	1.46E-06
Ra-226	2.80E-07	4.71E-07	7.84E-07	1.67E-06	6.67E-06	2.39E-05
Ra-228	1.72E-11	8.34E-11	1.62E-10	3.12E-10	7.44E-10	1.46E-09
Ac-225	9.20E-08	1.17E-07	1.52E-07	2.34E-07	5.69E-07	1.46E-06
Ac-227	8.77E-06	1.13E-05	1.43E-05	1.98E-05	3.36E-05	5.29E-05
Ac-228	1.72E-11	8.34E-11	1.62E-10	3.12E-10	7.44E-10	1.46E-09
Th-227	8.60E-06	1.11E-05	1.41E-05	1.94E-05	3.31E-05	5.21E-05
Th-228	1.77E-02	1.86E-02	1.85E-02	1.71E-02	1.28E-02	7.91E-03
Th-229	9.26E-08	1.18E-07	1.53E-07	2.35E-07	5.70E-07	1.46E-06
Th-230	9.52E-05	1.26E-04	1.65E-04	2.49E-04	5.35E-04	1.10E-03
Th-231	1.67E-02	1.67E-02	1.67E-02	1.68E-02	1.68E-02	1.68E-02
Th-232	1.82E-10	2.39E-10	3.11E-10	4.54E-10	8.83E-10	1.60E-09
Th-234	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01
Pa-231	2.91E-05	3.05E-05	3.23E-05	3.59E-05	4.66E-05	6.44E-05
Pa-233	3.58E-01	3.60E-01	3.63E-01	3.72E-01	4.02E-01	4.56E-01
Pa-234	3.14E-04	3.14E-04	3.14E-04	3.14E-04	3.14E-04	3.14E-04
Pa-234m	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01
U-232	1.89E-02	1.88E-02	1.82E-02	1.66E-02	1.25E-02	7.70E-03
U-233	6.41E-05	7.02E-05	7.80E-05	9.37E-05	1.43E-04	2.35E-04
U-234	8.62E-01	8.94E-01	9.32E-01	1.00E+00	1.19E+00	1.42E+00
U-235	1.67E-02	1.67E-02	1.67E-02	1.68E-02	1.68E-02	1.68E-02
U-236	2.91E-01	2.91E-01	2.91E-01	2.91E-01	2.91E-01	2.92E-01
U-237	1.56E+00	1.29E+00	1.02E+00	6.35E-01	1.53E-01	1.42E-02
U-238	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01	3.14E-01
U-240	8.80E-14	1.19E-13	1.58E-13	2.37E-13	4.72E-13	8.63E-13
Np-237	3.58E-01	3.60E-01	3.63E-01	3.72E-01	4.02E-01	4.56E-01
Np-239	1.98E+01	1.98E+01	1.97E+01	1.97E+01	1.97E+01	1.96E+01
Np-240m	8.80E-14	1.19E-13	1.58E-13	2.37E-13	4.72E-13	8.63E-13
Pu-236	2.66E-02	1.01E-02	2.98E-03	2.62E-04	1.78E-07	9.30E-13
Pu-238	2.86E+03	2.78E+03	2.67E+03	2.47E+03	1.96E+03	1.33E+03

Table 7-5 (Continued)

PWR Canister Inventory, Ci/MTHM

A total HM content of 461.4 kg is assumed.

Isotope	Years After Receipt at Repository 1	5	10	20	50	100
Pu-239	3.31E+02	3.31E+02	3.31E+02	3.31E+02	3.31E+02	3.30E+02
Pu-240	4.86E+02	4.87E+02	4.87E+02	4.88E+02	4.88E+02	4.86E+02
Pu-241	6.51E+04	5.39E+04	4.25E+04	2.64E+04	6.36E+03	5.93E+02
Pu-242	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00	1.45E+00
Pu-243	3.14E-07	3.14E-07	3.14E-07	3.14E-07	3.14E-07	3.14E-07
Pu-244	8.81E-14	1.19E-13	1.59E-13	2.37E-13	4.72E-13	8.64E-13
Am-241	1.57E+03	1.94E+03	2.31E+03	2.81E+03	3.33E+03	3.26E+03
Am-242	8.66E+00	8.50E+00	8.31E+00	7.94E+00	6.93E+00	5.51E+00
Am-242m	8.66E+00	8.50E+00	8.31E+00	7.94E+00	6.93E+00	5.51E+00
Am-243	1.98E+01	1.98E+01	1.97E+01	1.97E+01	1.97E+01	1.96E+01
Am-245	1.04E-11	4.13E-13	7.33E-15	2.31E-18	7.22E-29	2.24E-46
Cm-242	7.12E+00	6.99E+00	6.84E+00	6.53E+00	5.70E+00	4.53E+00
Cm-243	3.15E+00	2.89E+00	2.59E+00	2.09E+00	1.09E+00	3.69E-01
Cm-244	1.82E+03	1.56E+03	1.29E+03	8.79E+02	2.79E+02	4.11E+01
Cm-245	4.01E-01	4.01E-01	4.01E-01	4.00E-01	3.99E-01	3.98E-01
Cm-246	8.16E-02	8.16E-02	8.15E-02	8.14E-02	8.11E-02	8.05E-02
Cm-247	3.14E-07	3.14E-07	3.14E-07	3.14E-07	3.14E-07	3.14E-07
Cm-248	1.02E-06	1.02E-06	1.02E-06	1.02E-06	1.02E-06	1.02E-06
Cm-250	1.59E-13	1.59E-13	1.59E-13	1.59E-13	1.59E-13	1.58E-13
Bk-249	6.92E-07	2.75E-08	4.88E-10	1.54E-13	4.81E-24	1.49E-41
Bk-250	1.43E-13	1.43E-13	1.43E-13	1.43E-13	1.43E-13	1.43E-13
Cf-249	1.30E-05	1.29E-05	1.28E-05	1.25E-05	1.18E-05	1.07E-05
Cf-250	2.87E-05	2.32E-05	1.78E-05	1.05E-05	2.14E-06	1.51E-07
Cf-251	3.76E-07	3.75E-07	3.73E-07	3.70E-07	3.62E-07	3.48E-07
Cf-252	4.00E-06	1.40E-06	3.79E-07	2.76E-08	1.07E-11	2.18E-17
Total	3.96E+05	3.45E+05	2.97E+05	2.27E+05	1.07E+05	3.52E+04

Table 7-6

BWR Canister Inventory, Ci/MTHM

A total HM content of 366.6 kg is assumed.

Isotope	Years After Receipt at Repository 1	5	10	20	50	100
H-3	1.55E+02	1.24E+02	9.34E+01	5.31E+01	9.80E+00	5.85E-01
C-1	2.05E+04	2.05E+04	2.05E+04	2.05E+04	2.04E+04	2.03E+04
Mn-54	7.64E-02	2.70E-03	4.14E-05	9.74E-09	1.27E-19	9.10E-38
Fe-55	3.97E+02	1.37E+02	3.60E+01	2.50E+00	8.41E-04	1.37E-09
Co-60	1.34E+01	7.93E+00	4.11E+00	1.10E+00	2.11E-02	2.91E-05
Ni-59	8.79E-01	8.79E-01	8.79E-01	8.79E-01	8.79E-01	8.78E-01
Ni-63	1.26E+02	1.23E+02	1.18E+02	1.10E+02	8.74E+01	6.00E+01
Se-79	2.10E-01	2.10E-01	2.10E-01	2.10E-01	2.10E-01	2.10E-01
Kr-81	1.13E-06	1.13E-06	1.13E-06	1.13E-06	1.13E-06	1.13E-06
Kr-85	2.64E+03	2.04E+03	1.48E+03	7.73E+02	1.11E+02	4.41E+00
Rb-87	1.23E-05	1.23E-05	1.23E-05	1.23E-05	1.23E-05	1.23E-05
Sr-90	3.22E+04	2.91E+04	2.58E+04	2.01E+04	9.60E+03	2.80E+03
Y-90	3.22E+04	2.91E+04	2.58E+04	2.01E+04	9.61E+03	2.80E+03
Zr-93	1.69E+00	1.69E+00	1.69E+00	1.69E+00	1.69E+00	1.69E+00
Nb-93m	8.59E-01	1.01E+00	1.16E+00	1.36E+00	1.56E+00	1.60E+00
Nb-94	4.16E-10	4.16E-10	4.16E-10	4.16E-10	4.15E-10	4.14E-10
Tc-99	7.56E+00	7.56E+00	7.56E+00	7.56E+00	7.56E+00	7.56E+00
Ru-106	9.60E+01	6.17E+00	2.00E-01	2.09E-04	2.41E-13	3.03E-28
Rh-106	9.60E+01	6.17E+00	2.00E-01	2.09E-04	2.41E-13	3.03E-28
Pd-107	5.58E-02	5.58E-02	5.58E-02	5.58E-02	5.58E-02	5.58E-02
Ag-108	1.11E-06	1.08E-06	1.06E-06	1.00E-06	8.52E-07	6.53E-07
Ag-108m	1.44E-05	1.41E-05	1.37E-05	1.30E-05	1.11E-05	8.48E-06
Ag-109m	1.56E-08	1.68E-09	1.03E-10	3.89E-13	2.09E-20	1.60E-32
Ag-110	2.75E-04	4.94E-06	3.25E-08	1.41E-12	1.14E-25	1.73E-47
Ag-110m	1.96E-02	3.53E-04	2.32E-06	1.00E-10	8.14E-24	1.24E-45
Cd-109	1.56E-08	1.68E-09	1.03E-10	3.89E-13	2.09E-20	1.60E-32
Cd-113m	9.72E+00	8.04E+00	6.34E+00	3.94E+00	9.48E-01	8.81E-02
In-115	4.80E-12	4.80E-12	4.80E-12	4.80E-12	4.80E-12	4.80E-12
Sn-119m	4.89E-04	7.83E-06	4.47E-08	1.45E-12	4.99E-26	1.81E-48
Sn-121m	3.95E-01	3.74E-01	3.49E-01	3.04E-01	2.00E-01	1.00E-01
Sn-123	5.28E-07	2.06E-10	1.13E-14	3.43E-23	9.44E-49	2.37E-91

Table 7-6 (Continued)

BWR Canister Inventory, Ci/MTHM

A total HM content of 366.6 kg is assumed.

	Years After Receipt at Repository					
Isotope	1	5	10	20	50	100
Sn-126	2.98E-01	2.98E-01	2.98E-01	2.98E-01	2.98E-01	2.97E-01
Sb-125	2.65E+02	9.61E+01	2.70E+01	2.13E+00	1.05E-03	3.21E-09
Sb-126	4.17E-02	4.17E-02	4.17E-02	4.17E-02	4.17E-02	4.16E-02
Sb-126m	2.98E-01	2.98E-01	2.98E-01	2.98E-01	2.98E-01	2.97E-01
Te-123	4.13E-14	4.13E-14	4.13E-14	4.13E-14	4.13E-14	4.13E-14
Te-125m	6.48E+01	2.35E+01	6.59E+00	5.21E-01	2.56E-04	7.85E-10
Te-127	2.95E-08	2.72E-12	2.46E-17	2.02E-27	1.11E-57	0
Te-127m	3.01E-08	2.78E-12	2.51E-17	2.06E-27	1.13E-57	0
I-129	1.72E-02	1.72E-02	1.72E-02	1.72E-02	1.72E-02	1.72E-02
Cs-134	2.58E+03	6.71E+02	1.25E+02	4.31E+00	1.78E-04	8.80E-12
Cs-135	2.21E-01	2.21E-01	2.21E-01	2.21E-01	2.21E-01	2.21E-01
Cs-137	4.41E+04	4.03E+04	3.59E+04	2.85E+04	1.43E+04	4.52E+03
Ba-137m	4.18E+04	3.81E+04	3.39E+04	2.70E+04	1.35E+04	4.27E+03
Ce-142	1.48E-05	1.48E-05	1.48E-05	1.48E-05	1.48E-05	1.48E-05
Ce-144	2.19E+01	6.22E-01	7.26E-03	9.87E-07	2.48E-18	1.16E-37
Pr-144	2.19E+01	6.22E-01	7.26E-03	9.87E-07	2.48E-18	1.16E-37
Pr-144m	2.63E-01	7.47E-03	8.71E-05	1.18E-08	2.98E-20	1.39E-39
Nd-144	8.25E-10	8.25E-10	8.25E-10	8.25E-10	8.25E-10	8.25E-10
Pm-147	3.08E+03	1.07E+03	2.85E+02	2.03E+01	7.33E-03	1.34E-08
Sm-147	2.23E-06	2.28E-06	2.30E-06	2.31E-06	2.31E-06	2.31E-06
Sm-148	3.63E-12	3.63E-12	3.63E-12	3.63E-12	3.63E-12	3.63E-12
Sm-149	4.99E-13	4.99E-13	4.99E-13	4.99E-13	4.99E-13	4.99E-13
Sm-151	4.64E+02	4.50E+02	4.34E+02	4.02E+02	3.22E+02	2.22E+02
Eu-152	5.33E+00	4.30E+00	3.30E+00	1.93E+00	3.90E-01	2.71E-02
Eu-154	2.27E+03	1.65E+03	1.10E+03	4.91E+02	4.37E+01	7.75E-01
Eu-155	2.45E+02	1.38E+02	6.68E+01	1.57E+01	2.06E-01	1.50E-04
Gd-152	2.08E-12	2.11E-12	2.15E-12	2.19E-12	2.24E-12	2.26E-12
Gd-153	1.15E-04	1.71E-06	8.89E-09	2.41E-13	4.79E-27	7.01E-50
Ho-166m	2.75E-04	2.75E-04	2.74E-04	2.72E-04	2.68E-04	2.60E-04
Tl-207	7.55E-06	9.58E-06	1.19E-05	1.61E-05	2.62E-05	3.92E-05

Table 7-6 (Continued)

BWR Canister Inventory, Ci/MTHM

A total HM content of 366.6 kg is assumed.

	Years After Receipt at Repository					
Isotope	1	5	10	20	50	100
Tl-208	5.64E-03	5.88E-03	5.81E-03	5.36E-03	4.02E-03	2.49E-03
Tl-209	1.72E-09	2.16E-09	2.77E-09	4.21E-09	1.03E-08	2.66E-08
Pb-209	7.82E-08	9.81E-08	1.26E-07	1.91E-07	4.67E-07	1.21E-06
Pb-210	4.00E-08	7.93E-08	1.56E-07	4.26E-07	2.54E-06	1.22E-05
Pb-211	7.57E-06	9.61E-06	1.20E-05	1.62E-05	2.62E-05	3.93E-05
Pb-212	1.57E-02	1.63E-02	1.61E-02	1.49E-02	1.12E-02	6.90E-03
Pb-214	2.73E-07	4.46E-07	7.26E-07	1.51E-06	5.89E-06	2.09E-05
Bi-210	3.98E-08	7.90E-08	1.56E-07	4.25E-07	2.54E-06	1.22E-05
Bi-211	7.57E-06	9.61E-06	1.20E-05	1.62E-05	2.62E-05	3.93E-05
Bi-212	1.57E-02	1.63E-02	1.61E-02	1.49E-02	1.12E-02	6.90E-03
Bi-213	7.82E-08	9.81E-08	1.26E-07	1.91E-07	4.67E-07	1.21E-06
Bi-214	2.73E-07	4.46E-07	7.26E-07	1.51E-06	5.89E-06	2.09E-05
Po-210	3.63E-08	7.28E-08	1.46E-07	4.06E-07	2.48E-06	1.21E-05
Po-211	2.27E-08	2.88E-08	3.59E-08	4.86E-08	7.87E-08	1.18E-07
Po-212	1.00E-02	1.04E-02	1.03E-02	9.53E-03	7.15E-03	4.42E-03
Po-213	7.65E-08	9.60E-08	1.23E-07	1.87E-07	4.57E-07	1.18E-06
Po-214	2.73E-07	4.46E-07	7.26E-07	1.51E-06	5.89E-06	2.09E-05
Po-215	7.57E-06	9.61E-06	1.20E-05	1.62E-05	2.62E-05	3.93E-05
Po-216	1.57E-02	1.63E-02	1.61E-02	1.49E-02	1.12E-02	6.90E-03
Po-218	2.73E-07	4.46E-07	7.26E-07	1.51E-06	5.89E-06	2.09E-05
At-217	7.82E-08	9.81E-08	1.26E-07	1.91E-07	4.67E-07	1.21E-06
Rn-219	7.57E-06	9.61E-06	1.20E-05	1.62E-05	2.62E-05	3.93E-05
Rn-220	1.57E-02	1.63E-02	1.61E-02	1.49E-02	1.12E-02	6.90E-03
Rn-222	2.73E-07	4.46E-07	7.26E-07	1.51E-06	5.89E-06	2.09E-05
Fr-221	7.82E-08	9.81E-08	1.26E-07	1.91E-07	4.67E-07	1.21E-06
Fr-223	1.07E-07	1.35E-07	1.68E-07	2.27E-07	3.68E-07	5.51E-07
Ra-223	7.57E-06	9.61E-06	1.20E-05	1.62E-05	2.62E-05	3.93E-05
Ra-224	1.57E-02	1.63E-02	1.61E-02	1.49E-02	1.12E-02	6.90E-03
Ra-225	7.84E-08	9.83E-08	1.26E-07	1.92E-07	4.68E-07	1.21E-06
Ra-226	2.74E-07	4.47E-07	7.27E-07	1.52E-06	5.89E-06	2.09E-05

Table 7-6 (Continued)

BWR Canister Inventory, Ci/MTHM

A total HM content of 366.6 kg is assumed.

	Years After Receipt at Repository					
Isotope	1	5	10	20	50	100
Ra-228	1.65E-11	7.87E-11	1.50E-10	2.83E-10	6.57E-10	1.27E-09
Ac-225	7.82E-08	9.81E-08	1.26E-07	1.91E-07	4.67E-07	1.21E-06
Ac-227	7.63E-06	9.67E-06	1.20E-05	1.62E-05	2.63E-05	3.94E-05
Ac-228	1.65E-11	7.87E-11	1.50E-10	2.83E-10	6.57E-10	1.27E-09
Th-227	7.49E-06	9.50E-06	1.18E-05	1.60E-05	2.59E-05	3.88E-05
Th-228	1.57E-02	1.63E-02	1.61E-02	1.49E-02	1.12E-02	6.90E-03
Th-229	7.87E-08	9.86E-08	1.26E-07	1.92E-07	4.68E-07	1.21E-06
Th-230	8.71E-05	1.13E-04	1.47E-04	2.20E-04	4.67E-04	9.61E-04
Th-231	1.09E-02	1.09E-02	1.09E-02	1.09E-02	1.09E-02	1.09E-02
Th-232	1.74E-10	2.23E-10	2.84E-10	4.07E-10	7.76E-10	1.39E-09
Th-234	2.01E-01	2.01E-01	2.01E-01	2.01E-01	2.01E-01	2.01E-01
Pa-231	2.41E-05	2.50E-05	2.61E-05	2.85E-05	3.54E-05	4.69E-05
Pa-233	3.18E-01	3.19E-01	3.21E-01	3.26E-01	3.43E-01	3.73E-01
Pa-234	2.01E-04	2.01E-04	2.01E-04	2.01E-04	2.01E-04	2.01E-04
Pa-234m	2.01E-01	2.01E-01	2.01E-01	2.01E-01	2.01E-01	2.01E-01
U-232	1.66E-02	1.64E-02	1.59E-02	1.45E-02	1.09E-02	6.72E-03
U-233	4.99E-05	5.54E-05	6.22E-05	7.61E-05	1.19E-04	1.96E-04
U-234	7.41E-01	7.69E-01	8.03E-01	8.67E-01	1.03E+00	1.24E+00
U-235	1.09E-02	1.09E-02	1.09E-02	1.09E-02	1.09E-02	1.09E-02
U-236	2.50E-01	2.50E-01	2.50E-01	2.50E-01	2.50E-01	2.51E-01
U-237	8.54E-01	7.06E-01	5.57E-01	3.47E-01	8.34E-02	7.77E-03
U-238	2.01E-01	2.01E-01	2.01E-01	2.01E-01	2.01E-01	2.01E-01
U-240	2.12E-14	2.86E-14	3.79E-14	5.64E-14	1.12E-13	2.04E-13
Np-237	3.18E-01	3.19E-01	3.21E-01	3.26E-01	3.43E-01	3.73E-01
Np-239	8.43E+00	8.43E+00	8.43E+00	8.42E+00	8.40E+00	8.36E+00
Np-240m	2.12E-14	2.86E-14	3.79E-14	5.64E-14	1.12E-13	2.04E-13
Pu-236	2.09E-02	7.91E-03	2.35E-03	2.06E-04	1.40E-07	7.31E-13
Pu-238	2.56E+03	2.48E+03	2.39E+03	2.21E+03	1.75E+03	1.19E+03
Pu-239	2.10E+02	2.10E+02	2.10E+02	2.10E+02	2.10E+02	2.09E+02
Pu-240	2.89E+02	2.89E+02	2.89E+02	2.89E+02	2.89E+02	2.88E+02

Table 7-6 (Continued)

BWR Canister Inventory, C/MTHM

A total HM content of 366.6 kg is assumed.

	Years After Receipt of Repository					
Isotope	1	5	10	20	50	100
Pu-241	3.55E+04	2.94E+04	2.32E+04	1.44E+04	3.47E+03	3.23E+02
Pu-242	6.98E-01	6.98E-01	6.98E-01	6.98E-01	6.98E-01	6.99E-01
Pu-243	8.33E-08	8.33E-08	8.33E-08	8.33E-08	8.33E-08	8.33E-08
Pu-244	2.13E-14	2.87E-14	3.79E-14	5.64E-14	1.12E-13	2.04E-13
Am-241	9.04E+02	1.11E+03	1.31E+03	1.58E+03	1.86E+03	1.82E+03
Am-242	8.41E+00	8.25E+00	8.07E+00	7.71E+00	6.72E+00	5.35E+00
Am-242m	8.41E+00	8.25E+00	8.07E+00	7.71E+00	6.72E+00	5.35E+00
Am-243	8.43E+00	8.43E+00	8.43E+00	8.42E+00	8.40E+00	8.36E+00
Am-245	1.98E-12	7.88E-14	1.40E-15	4.41E-19	1.38E-29	4.28E-47
Cm-242	6.91E+00	6.79E+00	6.64E+00	6.34E+00	5.53E+00	4.40E+00
Cm-243	1.40E+00	1.28E+00	1.15E+00	9.27E-01	4.84E-01	1.64E-01
Cm-244	6.64E+02	5.70E+02	4.70E+02	3.21E+02	1.02E+02	1.50E+01
Cm-245	1.35E-01	1.35E-01	1.35E-01	1.35E-01	1.35E-01	1.34E-01
Cm-246	2.41E-02	2.41E-02	2.41E-02	2.40E-02	2.39E-02	2.38E-02
Cm-247	8.33E-08	8.33E-08	8.33E-08	8.33E-08	8.33E-08	8.33E-08
Cm-248	2.41E-07	2.41E-07	2.41E-07	2.41E-07	2.41E-07	2.41E-07
Bk-249	1.32E-07	5.25E-09	9.32E-11	2.94E-14	9.19E-25	2.85E-42
Cf-249	2.78E-06	2.76E-06	2.73E-06	2.68E-06	2.53E-06	2.29E-06
Cf-250	5.35E-06	4.33E-06	3.32E-06	1.95E-06	3.99E-07	2.82E-08
Cf-251	6.83E-08	6.81E-08	6.78E-08	6.73E-08	6.58E-08	6.33E-08
Cf-252	6.42E-07	2.25E-07	6.08E-08	4.42E-09	1.71E-12	3.50E-18
Total	2.03E+05	1.77E+05	1.53E+05	1.17E+05	5.53E+04	1.86E+04

TABLE 7-7a
ACTIVITIES OF FISSION PRODUCTS IN HLW VERSUS DECAY TIMES

Isotope	Ci/MTHM for Various Decay Periods[a]									
	0.5 yr	.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{3}H [b]	4.4×10^{2}	4.[illegible] $\times 10^{2}$	3.7×10^{2}	3.1×10^{2}	2.5×10^{2}	1.6				
^{14}C [b,c,d]	7.4×10^{-1}	7.[illegible] $\times 10^{-1}$	7.4×10^{-1}	7.4×10^{-1}	7.4×10^{-1}	7.3×10^{-1}	6.5×10^{-1}	2.2×10^{-1}	4.2×10^{-6}	
^{14}C [b,c,e]	6.2×10^{-1}	6.[illegible] $\times 10^{-1}$	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	5.5×10^{-1}	1.9×10^{-1}	3.5×10^{-6}	
^{79}Se	3.5×10^{-1}	3.[illegible] $\times 10^{-1}$	3.5×10^{-1}	3.5×10^{-1}	3.5×10^{-1}	3.5×10^{-1}	3.5×10^{-1}	3.2×10^{-1}	1.2×10^{-1}	8.2×10^{-6}
^{85}Kr [b]	9.5×10^{3}	8.[illegible] $\times 10^{3}$	7.9×10^{3}	6.5×10^{3}	5.0×10^{3}	1.6×10^{1}				
^{87}Rb	1.7×10^{-5}	1.[illegible] $\times 10^{-5}$	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}
^{89}Sr	5.5×10^{4}	4.[illegible] $\times 10^{2}$	2.5×10^{-2}	1.1×10^{-8}						
^{90}Sr	6.7×10^{4}	6.[illegible] $\times 10^{4}$	6.3×10^{4}	5.8×10^{4}	5.2×10^{4}	5.7×10^{3}	1.3×10^{-6}			
^{90}Y	6.7×10^{4}	6.[illegible] $\times 10^{4}$	6.3×10^{4}	5.8×10^{4}	5.2×10^{4}	5.7×10^{3}	1.3×10^{-6}			
^{91}Y	9.5×10^{4}	1.[illegible] $\times 10^{3}$	2.4×10^{-1}	5.8×10^{-7}						
^{93}Zr	1.7	1.[illegible]	1.7	1.7	1.7	1.7	1.7	1.7	1.6	1.1
^{95}Zr	1.7×10^{5}	3.[illegible] $\times 10^{3}$	1.4	1.2×10^{-5}						
^{93m}Nb	1.7×10^{-1}	2.[illegible] $\times 10^{-1}$	3.9×10^{-1}	5.7×10^{-1}	7.7×10^{-1}	1.7	1.7	1.7	1.6	1.1
^{95m}Nb	3.6×10^{3}	7.[illegible] $\times 10^{1}$	3.0×10^{-2}	2.6×10^{-7}						
^{95}Nb	3.3×10^{5}	7.[illegible] $\times 10^{3}$	3.2	2.7×10^{-5}						
^{99}Tc	1.3×10^{1}	1.[illegible] $\times 10^{1}$	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.3×10^{1}	1.2×10^{1}	9.2	4.9×10^{-1}
^{103}Ru	4.3×10^{4}	7.2×10^{1}	2.0×10^{-4}	9.5×10^{-13}						
^{106}Ru	3.4×10^{5}	1.7×10^{5}	4.2×10^{4}	5.3×10^{3}	3.4×10^{2}					
^{103m}Rh	4.3×10^{4}	7.2×10^{1}	2.0×10^{-4}	9.5×10^{-13}						
^{106}Rh	3.4×10^{5}	1.7×10^{5}	4.2×10^{4}	5.3×10^{3}	3.4×10^{2}					
^{107}Pd	9.2×10^{-2}	9.9×10^{-2}	9.8×10^{-2}	9.8×10^{-2}	9.9×10^{-2}	9.9×10^{-2}	9.9×10^{-2}	9.9×10^{-2}	9.8×10^{-2}	9.0×10^{-2}
^{110m}Ag	1.8×10^{3}	6.6×10^{2}	8.8×10^{1}	4.4	8.1×10^{-2}					
^{110}Ag	2.3×10^{2}	8.6×10^{1}	1.1×10^{1}	5.7×10^{-1}	1.0×10^{-2}					
^{113m}Cd	1.2×10^{1}	1.1×10^{1}	7.2	6.2	7.0	8.2×10^{-2}				
^{119m}Sn	8.6	3.1	4.1×10^{-1}	2.0×10^{-2}	3.4×10^{-4}					
^{121m}Sn	4.6×10^{-4}	4.6×10^{-4}	4.4×10^{-4}	4.3×10^{-4}	4.2×10^{-4}	1.9×10^{-4}	5.1×10^{-8}			
^{123}Sn	2.7×10^{3}	3.6×10^{2}	6.3	1.4×10^{-2}	4.4×10^{-6}					
^{126}Sn	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.5×10^{-1}	2.4×10^{-1}	4.7×10^{-4}

a. Periods are measured from reactor discharge.
b. Multiply by 0.08, 0.0, 0.0 for H-3, C-14, and Kr-85, respectively (see text).
c. Not a fission product, ^{14}C is formed by neutron activation of ^{14}N impurity in fuel.
d. Once-through.
e. Uranium-only recycle.

TABLE 7-7a(Continued)

Isotope	0.5 yr	1.5 yr	3.5 yr	6.5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
	Ci/MTHM for Various Decay Periods[a]									
^{124}Sb	4.3×10^{1}	6.4×10^{-1}	1.4×10^{-4}	4.4×10^{-10}						
^{125}Sb	6.8×10^{3}	5.3×10^{3}	3.1×10^{3}	1.4×10^{3}	5.3×10^{2}	4.9×10^{-8}				
^{126m}Sb	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.5×10^{-1}	2.4×10^{-1}	4.7×10^{-4}
^{126}Sb	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.8×10^{-1}	4.5×10^{-1}	2.4×10^{-1}	4.7×10^{-4}
^{123m}Te	1.6×10^{-1}	1.8×10^{-2}	2.3×10^{-4}	3.5×10^{-7}						
^{125m}Te	2.8×10^{3}	2.2×10^{3}	1.3×10^{3}	6.0×10^{2}	2.2×10^{2}	2.0×10^{-8}				
^{127m}Te	4.2×10^{3}	4.2×10^{2}	4.0	3.8×10^{-3}	3.5×10^{-7}					
^{127}Te	4.2×10^{3}	4.1×10^{2}	3.9	3.7×10^{-3}	3.5×10^{-7}					
^{129}I[b]	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}
^{134}Cs	1.7×10^{5}	1.2×10^{5}	6.0×10^{4}	2.2×10^{4}	5.7×10^{3}					
^{135}Cs	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.7×10^{-1}	2.6×10^{-1}	2.1×10^{-1}
^{137}Cs	9.4×10^{4}	9.2×10^{4}	8.8×10^{4}	8.2×10^{4}	7.5×10^{4}	9.4×10^{3}	8.8×10^{-6}			
^{137m}Ba	8.8×10^{4}	8.6×10^{4}	8.2×10^{4}	7.8×10^{4}	7.0×10^{4}	8.8×10^{3}	8.2×10^{-6}			
^{141}Ce	2.4×10^{4}	9.7	1.6×10^{-6}	1.1×10^{-16}						
^{144}Ce	6.1×10^{5}	2.5×10^{5}	4.2×10^{4}	2.9×10^{3}	8.2×10^{1}					
^{144}Pr	6.1×10^{5}	2.4×10^{5}	4.2×10^{4}	2.9×10^{3}	8.2×10^{1}					
^{147}Pm	9.0×10^{4}	6.9×10^{4}	4.1×10^{4}	1.9×10^{4}	6.4×10^{3}	2.9×10^{-7}				
^{148m}Pm	1.7×10^{3}	4.1	2.4×10^{-5}	3.4×10^{-13}						
^{148}Pm	1.4×10^{2}	3.3×10^{-1}	1.9×10^{-6}	2.7×10^{-14}						
^{151}Sm	1.2×10^{3}	1.1×10^{3}	1.1×10^{3}	1.1×10^{3}	1.1×10^{3}	5.2×10^{2}	4.0×10^{-1}			
^{152}Eu	1.3×10^{1}	1.2×10^{1}	1.1×10^{1}	9.0	7.1	3.9×10^{-2}				
^{154}Eu	5.8×10^{3}	5.5×10^{3}	5.0×10^{3}	4.4×10^{3}	3.7×10^{3}	7.6×10^{1}				
^{155}Eu	5.7×10^{3}	3.9×10^{3}	1.8×10^{3}	5.7×10^{2}	1.2×10^{2}					
^{153}Gd	1.9×10^{1}	6.6	8.8×10^{-1}	3.5×10^{-2}						
^{160}Tb	1.8×10^{2}	5.2	4.6×10^{-3}	1.2×10^{-7}						
^{166m}Ho	5.4×10^{-4}	5.4×10^{-4}	5.2×10^{-4}	5.2×10^{-4}	5.4×10^{-4}	5.1×10^{-4}	3.0×10^{-4}	1.7×10^{-6}		
Total	3.3×10^{6}	1.4×10^{6}	5.9×10^{5}	3.5×10^{5}	2.7×10^{5}	3.0×10^{4}	1.9×10^{1}	1.8×10^{1}	1.4×10^{1}	2.9
Total Thermal Watts	1.5×10^{4}	6.3×10^{3}	2.5×10^{3}	1.2×10^{3}	8.9×10^{2}	9.2×10^{1}	2.0×10^{-2}	1.9×10^{-2}	1.2×10^{-2}	7.8×10^{-4}

a. Periods are measured from reactor discharge.
b. Multiply by 0.005 for I-129 (see text).

TABLE 7-7b

ACTIVITIES OF ACTINIDES AND DAUGHTERS IN REFERENCE HIGH-LEVEL WASTE AS A FUNCTION OF DECAY TIME, URANIUM-ONLY RECYCLE, PLUTONIUM STORED SEPARATELY

Isotope	Ci/MTHM for Various Decay Periods(a) 0 yr	2 yr	5 yr	10^1 yr	10^2 yr	10^3 yr	10^4 yr	10^5 yr	10^6 yr
^{210}Pb(b)				4.57×10^{-10}	4.77×10^{-8}	2.03×10^{-5}	1.09×10^{-3}	8.89×10^{-3}	3.00×10^{-3}
^{226}Ra(c)				3.12×10^{-9}	8.33×10^{-8}	2.03×10^{-5}	1.09×10^{-3}	8.89×10^{-3}	3.00×10^{-3}
^{227}Ac(d)					2.41×10^{-6}	4.13×10^{-6}	2.04×10^{-5}	2.03×10^{-4}	2.81×10^{-4}
^{228}Th(e)	3.99×10^{-3}	1.96×10^{-3}	7.23×10^{-4}	2.08×10^{-4}	5.00×10^{-5}	8.80×10^{-9}	1.13×10^{-9}	1.48×10^{-8}	1.52×10^{-7}
^{229}Th(f)				9.72×10^{-9}	9.55×10^{-7}	9.75×10^{-5}	8.00×10^{-3}	1.70×10^{-1}	4.21×10^{-1}
^{230}Th					4.20×10^{-6}	1.22×10^{-4}	1.41×10^{-3}	8.79×10^{-3}	3.00×10^{-3}
^{231}Th	1.69×10^{-2}	8.47×10^{-5}	8.47×10^{-5}	8.47×10^{-5}	8.49×10^{-5}	8.63×10^{-5}	1.10×10^{-4}	2.66×10^{-4}	2.81×10^{-4}
^{232}Th(g)				7.86×10^{-13}	7.89×10^{-12}	8.22×10^{-11}	1.07×10^{-9}	1.48×10^{-8}	1.52×10^{-7}
^{234}Th	3.15×10^{-1}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}
^{231}Pa	2.37×10^{-6}	2.37×10^{-6}	2.38×10^{-6}	2.39×10^{-6}	2.55×10^{-6}	4.12×10^{-6}	2.04×10^{-5}	2.03×10^{-4}	2.81×10^{-4}
^{233}Pa	4.67×10^{-1}	4.67×10^{-1}	4.67×10^{-1}	4.68×10^{-1}	4.78×10^{-1}	5.29×10^{-1}	5.43×10^{-1}	5.28×10^{-1}	3.94×10^{-1}
^{234m}Pa	3.15×10^{-1}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}
^{234}Pa	3.15×10^{-4}	1.58×10^{-6}	1.58×10^{-3}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}	1.58×10^{-6}
^{232}U	5.25×10^{-5}	7.94×10^{-5}	9.95×10^{-5}	1.09×10^{-4}	4.87×10^{-5}	8.42×10^{-9}			
^{233}U	1.38×10^{-7}	4.56×10^{-6}	1.12×10^{-5}	2.23×10^{-5}	2.04×10^{-4}	2.16×10^{-3}	2.26×10^{-2}	1.85×10^{-1}	4.18×10^{-1}
^{234}U	1.78×10^{-4}	3.33×10^{-4}	6.06×10^{-4}	1.05×10^{-3}	7.26×10^{-3}	1.73×10^{-2}	1.71×10^{-2}	1.36×10^{-2}	2.54×10^{-3}
^{235}U	8.47×10^{-5}	8.47×10^{-5}	8.47×10^{-5}	8.47×10^{-5}	8.49×10^{-5}	8.63×10^{-5}	1.10×10^{-4}	2.66×10^{-4}	2.81×10^{-4}

a. Years after chemical separation; 1.5 yr elapse between reactor discharge and chemical separation; chemical separation assumed to remove 99.5% of U and Pu, but no other activities.

b. Activities of ^{210}Bi and ^{210}Po are the same as ^{210}Pb.

c. Activities of ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po are the same as ^{226}Ra.

d. Activities of ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl are the same as ^{227}Ac.

e. Activities of ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi are the same as ^{228}Th. ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

f. Activities of ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb are the same as ^{229}Th. ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.

g. Activities of ^{228}Ra and ^{228}Ac are the same as ^{232}Th.

TABLE 7-7b(Continued)

Isotope	Ci/MTHM for Various Decay Periods[a] 0 yr	2 yr	5 yr	10^{1} yr	10^{2} yr	10^{3} yr	10^{4} yr	10^{5} yr	10^{6} yr
^{236}U	1.60×10^{-3}	1.60×10^{-3}	1.60×10^{-3}	1.60×10^{-3}	1.61×10^{-3}	1.75×10^{-3}	2.60×10^{-3}	3.15×10^{-3}	3.07×10^{-3}
^{237}U	1.31×10^{-2}	1.19×10^{-2}	1.04×10^{-2}	8.22×10^{-3}	1.26×10^{-4}	3.95×10^{-6}	1.86×10^{-6}	9.80×10^{-10}	
^{238}U	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-2}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}	1.58×10^{-3}
^{237}Np	4.67×10^{-1}	4.67×10^{-1}	4.67×10^{-1}	4.68×10^{-1}	4.78×10^{-1}	5.29×10^{-1}	5.43×10^{-1}	5.28×10^{-1}	3.94×10^{-1}
^{239}Np	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.35×10^{1}	1.34×10^{1}	1.24×10^{1}	5.48	1.58×10^{-3}	1.11×10^{-7}
^{236}Pu	1.80×10^{-3}	1.11×10^{-3}	5.35×10^{-4}	1.58×10^{-4}					
^{238}Pu	1.57×10^{1}	3.20×10^{1}	3.22×10^{1}	3.13×10^{1}	1.89×10^{1}	2.26×10^{-1}	3.35×10^{-19}		
^{239}Pu	1.46	1.46	1.46	1.46	1.50	1.79	3.26	4.46×10^{-1}	1.11×10^{-7}
^{240}Pu	2.26	2.50	2.61	3.29	5.39	4.98	1.98	1.95×10^{-4}	1.17×10^{-9}
^{241}Pu	5.24×10^{2}	4.77×10^{2}	4.55×10^{2}	3.28×10^{2}	5.02	1.58×10^{-1}	7.42×10^{-2}	3.92×10^{-5}	
^{242}Pu	7.91×10^{-3}	7.92×10^{-3}	7.93×10^{-3}	7.95×10^{-3}	8.20×10^{-3}	8.71×10^{-3}	8.83×10^{-3}	7.57×10^{-3}	1.46×10^{-3}
^{241}Am	3.65×10^{2}	3.66×10^{2}	3.66×10^{2}	3.66×10^{2}	3.27×10^{2}	7.77×10^{1}	7.43×10^{-2}	3.92×10^{-5}	
^{242m}Am	1.07×10^{1}	1.06×10^{1}	1.04×10^{1}	1.02×10^{1}	6.76	1.12×10^{-1}	1.69×10^{-19}		
^{242}Am	1.07×10^{1}	1.06×10^{1}	1.04×10^{1}	1.02×10^{1}	6.76	1.12×10^{-1}	1.69×10^{-19}		
^{243}Am	1.35×10^{1}	1.35×10^{1}	1.[illegible] $\times 10^{1}$	1.35×10^{1}	1.34×10^{1}	1.24×10^{1}	5.48	1.58×10^{-3}	1.11×10^{-7}
^{242}Cm	3.48×10^{3}	1.64×10^{2}	1.00×10^{1}	8.37	5.55	9.16×10^{-2}	1.39×10^{-19}		
^{243}Cm	3.82	3.66	3.43	3.07	4.38×10^{-1}	1.50×10^{-9}			
^{244}Cm	1.21 × 10[illegible]	1.13×10^{3}	1.00×10^{3}	8.28×10^{2}	2.64×10^{1}	3.21×10^{-14}	3.47×10^{-14}	3.18×10^{-13}	1.52×10^{-12}
^{245}Cm	1.71×10^{-1}	1.71×10^{-1}	1.71×10^{-1}	1.71×10^{-1}	1.70×10^{-1}	1.58×10^{-1}	7.41×10^{-2}	3.91×10^{-5}	
^{246}Cm	3.27×10^{-2}	3.27×10^{-2}	3.27×10^{-2}	3.27×10^{-2}	3.23×10^{-2}	2.83×10^{-2}	7.52×10^{-3}	1.34×10^{-8}	
^{249}Bk	4.63×10^{-4}	9.24×10^{-5}	8.23×10^{-6}	1.46×10^{-7}					
^{249}Cf	3.18×10^{-6}	4.07×10^{-6}	4.25×10^{-6}	4.23×10^{-6}	3.54×10^{-6}	6.02×10^{-7}	1.21×10^{-14}		
^{250}Cf	1.48×10^{-5}	1.33×10^{-5}	1.14×10^{-5}	8.73×10^{-6}	7.42×10^{-8}	3.5×10^{-14}	2.45×10^{-14}	6.79×10^{-16}	
^{252}Cf	1.40×10^{-5}	8.27×10^{-6}	3.77×10^{-6}	1.02×10^{-6}					
Total	5.66×10^{3}	2.22×10^{3}	1.88×10^{3}	1.62×10^{3}	4.32×10^{2}	1.11×10^{2}	1.76×10^{1}	3.17	4.62
Total W/MTHM	1.84×10^{2}	5.95×10^{1}	4.96×10^{1}	4.34×10^{1}	1.34×10^{1}	3.31	3.95×10^{-1}	7.25×10^{-2}	1.10×10^{-1}

a. Years after chemical separation; 1.5 yr elapse between reactor discharge and chemical separation; chemical separation assumed to remove 99.5% of U and Pu, but no other activities.

TABLE 7-8
RELEASE FRACTIONS DERIVED (FROM WALKER, 1978)

Release Mechanism	Safety Analysis Parameter	Release Fraction: Range of Observations[a]	Release Fraction: Current Practice[b]	Release Fraction: Recommended Values[c]
A. Failed fuel - gap release	1. Noble gas	0.015 - 0.34	0.018 - 0.10	0.10
	Kr-85	--	0.30	0.30
	2. Halogens	0.025 - 0.49	0.0032 - 0.10	0.10
	I-129	--	0.30	0.30
	3. Volatiles (Cs,Rb,Ru)	$<4.0\times10^{-6}$ - 0.80	--	0.01
	4. Non-Volatiles	$<2.0\times10^{-6}$ - 8×10^{-4}	$<10^{-6}$ - 0.05	0.01
B. Heated fuel release T = 1,000-1,400°C	1. Noble gas	0.12 - 0.86	0.90	0.90
	2. Halogens	0.12 - 0.95	0.90	0.25
	3. Volatiles	2×10^{-4} - 0.999	--	0.01
	4. Non-Volatiles	$<10^{-5}$ - 7×10^{-3}	--	0.01
C. Fire release (Release fraction except as noted)	1. Noble gas	--	0.90 - 1.00	1.00
	2. Halogens	0.65 - 0.84	1.00	1.00
	3. Volatiles	$\sim3\times10^{-6}$ - 0.01	0.01 - 0.90	0.01
	4. Non-Volatiles	$\sim5\times10^{-4}$ - 0.20	0.01 - 0.05	0.01
	5. Fly ash	$\sim5\times10^{-4}$ - 0.20	0.01 - 0.05	0.01
	6. Airborne Particle size (μ)	<0.1 - 10	<5	<5

(a) From actual data.

(b) Values used in various studies reviewed in (Walker 1978).

(c) Values recommended by Walker.

TABLE 7-8 (Continued)

Release Mechanism	Safety Analysis Parameter	Release Fraction: Range of Observations[a]	Current Practice[b]	Recommended Values[c]
D. Explosions (Fraction released except as noted)	1. Noble gas	--	1.00	1.00
	2. Halogens	--	1.00	1.00
	3. Volatiles	--	0.001	0.01
	4. Non-Volatiles	9×10^{-5} - 0.14	0.01	0.01
	5. Airborne material (time >100 sec)	1.0 - 71 mg/m^3	10 - 100 mg/m^3	100 mg/m^3 [d]
	6. Airborne particle size (μ)	--	<10 - <30	<10
E. Impact stress - airborne fraction	Vitrified material (solid):			
	1. Impact velocity ~40 m/s	4×10^{-4} - 3×10^{-3}	5×10^{-3} - 2×10^{-2}	NR[e]
	2. Impact velocity ~20 m/s	2×10^{-5} - 3×10^{-4}	5×10^{-5} - 3×10^{-3}	NR

(a) From actual data.

(b) Values used in various studies reviewed in (Walker 1978).

(c) Values recommended by Walker .

(d) Applicable to particulate material only, not to gas or volatile material release.

(e) NR = No recommendation.

TABLE 7-9
FILTER EFFICIENCY AND RESUSPENSION FACTOR

Parameter	Range of Observations[a]	Current Practice[b]	Recommended Values[c]
Particulate Filters (% Efficiency)			
HEPA - 1st Stage	96.0 - 99.999	99.0 - 99.99	99.9
- 2nd Stage	99.976 - 99.992	99.0 - 99.9	99.0
- 3rd Stage	99.49 - 99.99+	99.8 - 94.0	99.0
- 4th Stage	--	83.0	83.0
Noble Gas Traps (% Efficiency)			
1. Refrigerant	75.0 - 99.99+	--	NR[d]
2. Cryogenic (CO_2)	90.0 - 99.993	--	NR
Halogen Filters (% Efficiency for bed depths $\geq$ 2.0 inches)			
Or Activated charcoal:			
I_2 @ RH < 70%[e]	81.9 - 99.999	95.0 - 99.99	99.0
I_2 @ RH > 70%	> 90.0 - 99.9997	90.0	90.0
CH_3I @ RH < 70%	50.25 - 99.999+	85.0 - 99.0	99.0
CH_3I @ RH > 70%	8.77 - 99.99	30.0 - 98.0	30.0

(a) From actual data.

(b) Values used in various studies reviewed in (Walker 1978).

(c) Values recommended by Walker.

(d) NR - No recommendation.

(e) RH - Relative humidity.

TABLE 7-9 (Continued)

Parameter	Range of Observations[a]	Current Practice[b]	Recommended Values[c]
Resuspension Factor (m^{-1})			
Indoors	1×10^{-2} - 3×10^{-10}	1×10^{-3} - 1×10^{-6}	1×10^{-4}
Outdoors	1×10^{-3} - 2×10^{-11}	1×10^{-6}	1×10^{-6}
Plateout			
Iodine (time <2 hr)	DF = 10-100[f]	DF = 2 - 10	NR[d]
Particulate diameter > 10	DF = 1 - 4×10^{5}	DF = 1.0-5.5	NR

(a) From actual data.

(b) Values used in various studies reviewed in (Walker 1978).

(c) Values recommended by Walker.

(d) NR - No recommendation.

(f) DF - Decontamination factor = initial airborne activity/average airborne activity.

TABLE 7-10
RELEASE FRACTIONS FOR SPENT FUEL RELEASE MECHANISMS
(WILMOT, 1981)

Radionuclide	Impact Rupture	Leaching Waterlogged	Other	Crud	Burst Rupture[a]	Diffusion (Steam)	Oxidation (Air)
Noble gases	0.2	-	-	-	0.2	0.02	0.02
Cs-134	2.0-6[b]	3.0-3	2.0-4	-	4.0-3	5.0-4	0.03
Cs-137	2.0-6	3.0-3	2.0-4	-	4.0-3	5.0-4	0.03
I-129	2.0-6	3.0-3	2.0-4	-	7.0-3	4.0-3	0.08
Sr-90	2.0-6	2.0-4	3.0-5	-	2.0-5	-	-
Ru-106	2.0-6	-	-	-	2.0-5	-	8.0-4
Actinides	2.0-6	2.0-4	2.0-5	-	2.0-5	-	-
Co-60	-	-	-	0.25	-	-	-

[a] Diffusion or oxidation releases must be added to burst releases when the burst rupture mechanism occurs.

[b] $2.0\text{-}6 = 2.0 \times 10^{-6}$

Table 7-11

EXTERNAL DOSE FACTORS FOR STANDING ON CONTAMINATED GROUND

(mrem/hr per pCi/m^2)

Element	Total Body	Skin
H-3	0.0	0.0
C-14	0.0	0.0
NA-24	2.50E-08	2.90E-08
P-32	0.0	0.0
Cr-51	2.20E-10	2.60E-10
Mn-54	5.80E-09	6.80E-09
Mn-56	1.10E-08	1.30E-08
Fe-55	0.0	0.0
Fe-59	8.00E-09	9.40E-09
Co-58	7.00E-09	8.20E-09
Co-60	1.70E-08	2.00E-08
Ni-63	0.0	0.0
Nr-65	3.70E-09	4.30E-09
Cu-64	1.50E-09	1.70E-09
Zn-65	4.00E-09	4.60E-09
Zn-69	0.0	0.0
Br-83	6.40E-11	9.30E-11
Br-84	1.20E-08	1.40E-08
Br-85	0.0	0.0
Rb-86	6.30E-10	7.20E-10
Rb-88	3.50E-09	4.00E-09
Rb-89	1.50E-08	1.80E-08
Sr-89	5.60E-13	6.50E-13
Sr-91	7.10E-09	8.30E-09
Sr-92	9.00E-09	1.00E-08
Y-90	2.20E-12	2.60E-12
Y-91M	3.80E-09	4.40E-09
Y-91	2.40E-11	2.70E-11
Y-92	1.60E-09	1.90E-09
Y-93	5.70E-10	7.80E-10
Zr-95	5.00E-09	5.80E-09
Zr-97	5.50E-09	6.40E-09
Nb-95	5.10E-09	6.00E-09
Mo-99	1.90E-09	2.20E-09
Tc-99M	9.60E-10	1.10E-09
Tc-101	2.70E-09	3.00E-09
Ru-103	3.60E-09	4.20E-09
Ru-105	4.50E-09	5.10E-09
Ru-106	1.50E-09	1.80E-09
Ag-110M	1.80E-08	2.10E-08
Te-125M	3.50E-11	4.80E-11
Te-127M	1.10E-12	1.30E-12
Te-127	1.00E-11	1.10E-11
Te-129M	7.70E-10	9.00E-10
Te-129	7.10E-10	8.40E-10
Te-131M	8.40E-09	9.90E-09
Te-131	2.20E-09	2.60E-06
Te-132	1.70E-09	2.00E-09
I-130	1.40E-08	1.70E-08
I-131	2.80E-09	3.40E-09
I-132	1.70E-08	2.00E-08
I-133	3.70E-09	4.50E-09
I-134	1.60E-08	1.90E-08
I-135	1.20E-08	1.40E-08

Table 7-11 (Continued)

Element	Total Body	Skin
Cs-134	1.20E-08	1.40E-08
Cs-136	1.50E-08	1.70E-08
Cs-137	4.20E-09	4.90E-09
Cs-138	2.10E-08	2.40E-08
Ba-139	2.40E-09	2.70E-09
Ba-140	2.10E-09	2.40E-09
Ba-141	4.30E-09	4.90E-09
Ba-142	7.90E-09	9.00E-09
La-140	1.50E-08	1.70E-08
La-142	1.50E-08	1.80E-08
Ce-141	5.50E-10	6.20E-10
Ce-143	2.20E-09	2.50E-09
Ce-144	3.20E-10	3.70E-10
Pr-143	0.0	0.0
Pr-144	2.00E-10	2.30E-10
Nd-147	1.00E-09	1.20E-09
W-187	3.10E-09	3.60E-09
Np-239	9.50E-10	1.10E-09

Table 7-12

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INHALATION DOSE FACTORS FOR ADULTS
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	1.58E-07	1.58E-07	1.58E-07	1.58E-07	1.58E-07	1.58E-07
C 14	2.27E-06	4.26E-07	4.26E-07	4.26E-07	4.26E-07	4.26E-07	4.26E-07
NA 24	1.28E-06	1.28E-06	1.28E-06	1.28E-06	1.28E-06	1.28E-06	1.28E-06
P 32	1.65E-04	9.64E-06	6.26E-06	NO DATA	NO DATA	NO DATA	1.08E-05
CR 51	NO DATA	NO DATA	1.25E-08	7.44E-09	2.85E-09	1.80E-06	4.15E-07
MN 54	NO DATA	4.95E-06	7.87E-07	NO DATA	1.23E-06	1.75E-04	9.67E-06
MN 56	NO DATA	1.55E-10	2.29E-11	NO DATA	1.63E-10	1.18E-06	2.53E-06
FE 55	3.07E-06	2.12E-06	4.93E-07	NO DATA	NO DATA	9.01E-06	7.54E-07
FE 59	1.47E-06	3.47E-05	1.32E-06	NO DATA	NO DATA	1.27E-04	2.35E-05
CO 58	NO DATA	1.98E-07	2.59E-07	NO DATA	NO DATA	1.16E-04	1.33E-05
CO 60	NO DATA	1.44E-06	1.85E-06	NO DATA	NO DATA	7.46E-04	3.56E-05
NI 63	5.40E-05	3.93E-06	1.81E-06	NO DATA	NO DATA	2.23E-05	1.67E-06
NI 65	1.92E-10	2.62E-11	1.14E-11	NO DATA	NO DATA	7.00E-07	1.54E-06
CU 64	NO DATA	1.83E-10	7.69E-11	NO DATA	5.78E-10	8.48E-07	6.12E-06
ZN 65	4.05E-06	1.29E-05	5.82E-06	NO DATA	8.62E-06	1.08E-04	6.68E-06
ZN 69	4.23E-12	8.14E-12	5.65E-13	NO DATA	5.27E-12	1.15E-07	2.04E-09
BR 83	NO DATA	NO DATA	3.01E-08	NO DATA	NO DATA	NO DATA	2.90E-08
BR 84	NO DATA	NO DATA	3.91E-08	NO DATA	NO DATA	NO DATA	2.05E-13
BR 85	NO DATA	NO DATA	1.60E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	1.69E-05	7.37E-06	NO DATA	NO DATA	NO DATA	2.08E-06
RB 88	NO DATA	4.84E-08	2.41E-08	NO DATA	NO DATA	NO DATA	4.18E-19
RB 89	NO DATA	3.20E-08	2.12E-08	NO DATA	NO DATA	NO DATA	1.16E-21
SR 89	3.80E-05	NO DATA	1.09E-06	NO DATA	NO DATA	1.75E-04	4.37E-05
SR 90	1.24E-02	NO DATA	7.62E-04	NO DATA	NO DATA	1.20E-03	9.02E-05
SR 91	7.74E-09	NO DATA	3.13E-10	NO DATA	NO DATA	4.56E-06	2.39E-05
SR 92	8.43E-10	NO DATA	3.64E-11	NO DATA	NO DATA	2.06E-06	5.38E-06
Y 90	2.61E-07	NO DATA	7.01E-09	NO DATA	NO DATA	2.12E-05	6.32E-05
Y 91M	3.26E-11	NO DATA	1.27E-12	NO DATA	NO DATA	2.40E-07	1.66E-10
Y 91	5.78E-05	NO DATA	1.55E-06	NO DATA	NO DATA	2.13E-04	4.81E-05
Y 92	1.29E-09	NO DATA	3.77E-11	NO DATA	NO DATA	1.96E-06	9.19E-06

Table 7-12

INHALATION DOSE FACTORS FOR ADULTS
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	1.18E-05	NO DATA	3.26E-10	NO DATA	NO DATA	6.06E-06	5.27E-05
ZR 95	1.34E-05	4.30E-06	2.91E-06	NO DATA	6.77E-06	2.21E-04	1.38E-05
ZR 97	1.21E-08	2.45E-09	1.13E-09	NO DATA	3.71E-09	9.84E-05	6.54E-05
NB 95	1.76E-06	9.77E-07	5.26E-07	NO DATA	9.67E-07	6.31E-05	1.30E-05
MO 99	NO DATA	1.51E-08	2.87E-09	NO DATA	3.64E-08	1.14E-05	3.10E-05
TC 99M	1.29E-13	3.64E-13	4.63E-12	NO DATA	5.52E-12	9.55E-08	5.20E-07
TC101	5.22E-15	7.52E-15	7.38E-14	NO DATA	1.35E-13	4.99E-08	1.36E-21
RU103	1.91E-07	NO DATA	8.23E-08	NO DATA	7.29E-07	6.31E-05	1.38E-05
RU105	9.88E-11	NO DATA	3.89E-11	NO DATA	1.27E-10	1.37E-06	6.02E-06
RU106	8.64E-06	NO DATA	1.09E-06	NO DATA	1.67E-05	1.17E-03	1.14E-04
AG110M	1.35E-06	1.25E-06	7.43E-07	NO DATA	2.46E-06	5.79E-04	3.78E-05
TE125M	4.27E-07	1.98E-07	5.84E-08	1.31E-07	1.55E-06	3.92E-05	8.83E-06
TE127M	1.58E-06	7.21E-07	1.96E-07	4.11E-07	5.72E-06	1.20E-04	1.87E-05
TE127	1.75E-10	8.03E-11	3.87E-11	1.32E-10	6.37E-10	8.14E-07	7.17E-06
TE129M	1.22E-06	5.84E-07	1.98E-07	4.30E-07	4.57E-06	1.45E-04	4.79E-05
TE129	6.22E-12	2.99E-12	1.55E-12	4.87E-12	2.34E-11	2.42E-07	1.96E-08
TE131M	8.74E-09	5.45E-09	3.63E-09	6.88E-09	3.86E-08	1.82E-05	6.95E-05
TE131	1.39E-12	7.44E-13	4.49E-13	1.17E-12	5.46E-12	1.74E-07	2.30E-09
TE132	3.25E-08	2.69E-08	2.02E-08	2.37E-08	1.82E-07	3.60E-05	6.37E-05
I 130	5.72E-07	1.68E-06	6.60E-07	1.42E-04	2.61E-06	NO DATA	9.61E-07
I 131	3.15E-06	4.47E-06	2.56E-06	1.49E-03	7.66E-06	NO DATA	7.85E-07
I 132	1.45E-07	4.07E-07	1.45E-07	1.43E-05	6.48E-07	NO DATA	5.08E-08
I 133	1.08E-06	1.85E-06	5.65E-07	2.69E-04	3.23E-06	NO DATA	1.11E-06
I 134	8.05E-08	2.16E-07	7.69E-08	3.73E-06	3.44E-07	NO DATA	1.26E-10
I 135	3.35E-07	8.73E-07	3.21E-07	5.60E-05	1.39E-06	NO DATA	6.56E-07
CS134	4.66E-05	1.06E-04	9.10E-05	NO DATA	3.59E-05	1.22E-05	1.30E-06
CS136	4.88E-06	1.83E-05	1.38E-05	NO DATA	1.07E-05	1.50E-06	1.46E-06
CS137	5.98E-05	7.76E-05	5.35E-05	NO DATA	2.78E-05	9.40E-06	1.05E-06
CS138	4.14E-08	7.76E-08	4.05E-08	NO DATA	6.00E-09	6.07E-09	2.33E-13
BA139	1.17E-10	8.32E-14	3.42E-12	NO DATA	7.78E-14	4.70E-07	1.12E-07

Table 7-12

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INHALATION DOSE FACTORS FOR ADULTS
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	4.88E-06	6.13E-09	3.21E-07	NO DATA	2.09E-09	1.59E-04	2.73E-05
BA141	1.25E-11	9.41E-15	4.20E-13	NO DATA	8.75E-15	2.42E-07	1.45E-17
BA142	3.29E-12	3.38E-15	2.07E-13	NO DATA	2.96E-15	1.49E-07	1.96E-26
LA140	4.30E-08	2.17E-08	5.73E-09	NO DATA	NO DATA	1.70E-05	5.73E-05
LA142	8.54E-11	3.88E-11	9.65E-12	NO DATA	NO DATA	7.91E-07	2.64E-07
CE141	2.49E-06	1.69E-06	1.91E-07	NO DATA	7.83E-07	4.52E-05	1.50E-05
CE143	2.33E-08	1.72E-08	1.91E-09	NO DATA	7.60E-09	9.97E-06	2.83E-05
CE144	4.29E-04	1.79E-04	2.30E-05	NO DATA	1.06E-04	9.72E-04	1.02E-04
PR143	1.17E-06	4.69E-07	5.80E-08	NO DATA	2.70E-07	3.51E-05	2.50E-05
PR144	3.76E-12	1.56E-12	1.91E-13	NO DATA	8.81E-13	1.27E-07	2.69E-18
ND147	6.59E-07	7.62E-07	4.56E-08	NO DATA	4.45E-07	2.76E-05	2.16E-05
W 187	1.06E-09	8.85E-10	3.10E-10	NO DATA	NO DATA	3.63E-06	1.94E-05
NP239	2.87E-08	2.82E-09	1.55E-09	NO DATA	8.75E-09	4.70E-06	1.49E-05

TABLE 7-13 - ACCIDENT ANALYSIS SUMMARY

Expected Injuries(1)

CLASSIFICATION	FATAL	NFDL	NDL
Fall of rib	0	1.17	0.26
Fall of roof	0.094	3.58	2.63
Drilling	0	0.059	0.036
Tramming	0.016	0.165	0.032
Rail haulage	0.049	0.927	0.510
Electrical	0.010	0.048	0.019
Fire/Ignition	0	0.179	0.36
TOTALS	0.169	6.128	3.856

(1) Injuries in nine years based on 200,000 work-hours per year

TABLE 7-14
NUMBER OF OPERATOR INJURIES, INJURY-INCIDENCE RATES PER 200,000 EMPLOYEE-HOURS, AND EMPLOYEE-HOURS, BY MINERAL INDUSTRY AND WORK LOCATION, JANUARY - SEPTEMBER, 1983. FROM MSHA (1983)

Mineral Industry And Work Location	Fatal	Fatal Incidence Rate	NFDL	NDL Incidence Rate	NFDL	NDL Incidence Rate	ALL Incidence Rate	Employee-Hours Reported
Underground Mines								
Metal-------------------------------	5	0.06	514	6.34	378	4.66	11.06	16,225,255
Nonmetal----------------------------	3	0.09	180	5.20	88	2.54	7.84	6,917,053
Stone-------------------------------	-	--	55	4.47	12	0.98	5.45	2,459,887
Total, Underground Mines	8	0.06	749	5.85	478	3.73	9.65	25,602,195

Table 7-12

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INHALATION DOSE FACTORS FOR ADULTS
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	1.18E-05	NO DATA	3.26E-10	NO DATA	NO DATA	6.06E-06	5.27E-05
ZR 95	1.34E-05	4.30E-06	2.91E-06	NO DATA	6.77E-06	2.21E-04	1.38E-05
ZR 97	1.21E-08	2.45E-09	1.13E-09	NO DATA	3.71E-09	9.84E-05	6.54E-05
NB 95	1.76E-06	9.77E-07	5.26E-07	NO DATA	9.67E-07	6.31E-05	1.30E-05
MO 99	NO DATA	1.51E-08	2.87E-09	NO DATA	3.64E-08	1.14E-05	3.10E-05
TC 99M	1.29E-13	3.64E-13	4.63E-12	NO DATA	5.52E-12	9.55E-08	5.20E-07
TC101	5.22E-15	7.52E-15	7.38E-14	NO DATA	1.35E-13	4.99E-08	1.36E-21
RU103	1.91E-07	NO DATA	8.23E-08	NO DATA	7.29E-07	6.31E-05	1.38E-05
RU105	9.88E-11	NO DATA	3.89E-11	NO DATA	1.27E-10	1.37E-06	6.02E-06
RU106	8.64E-06	NO DATA	1.09E-06	NO DATA	1.67E-05	1.17E-03	1.14E-04
AG110M	1.35E-06	1.25E-06	7.43E-07	NO DATA	2.46E-06	5.79E-04	3.78E-05
TE125M	4.27E-07	1.98E-07	5.84E-08	1.31E-07	1.55E-06	3.92E-05	8.83E-06
TE127M	1.58E-06	7.21E-07	1.96E-07	4.11E-07	5.72E-06	1.20E-04	1.87E-05
TE127	1.75E-10	8.03E-11	3.87E-11	1.32E-10	6.37E-10	8.14E-07	7.17E-06
TE129M	1.22E-06	5.84E-07	1.98E-07	4.30E-07	4.57E-06	1.45E-04	4.79E-05
TE129	6.22E-12	2.99E-12	1.55E-12	4.87E-12	2.34E-11	2.42E-07	1.96E-08
TE131M	8.74E-09	5.45E-09	3.63E-09	6.88E-09	3.86E-08	1.82E-05	6.95E-05
TE131	1.39E-12	7.44E-13	4.49E-13	1.17E-12	5.46E-12	1.74E-07	2.30E-09
TE132	3.25E-08	2.69E-08	2.02E-08	2.37E-08	1.82E-07	3.60E-05	6.37E-05
I 130	5.72E-07	1.68E-06	6.60E-07	1.42E-04	2.61E-06	NO DATA	9.61E-07
I 131	3.15E-06	4.47E-06	2.56E-06	1.49E-03	7.66E-06	NO DATA	7.85E-07
I 132	1.45E-07	4.07E-07	1.45E-07	1.43E-05	6.48E-07	NO DATA	5.08E-08
I 133	1.08E-06	1.85E-06	5.65E-07	2.69E-04	3.23E-06	NO DATA	1.11E-06
I 134	8.05E-08	2.16E-07	7.69E-08	3.73E-06	3.44E-07	NO DATA	1.26E-10
I 135	3.35E-07	8.73E-07	3.21E-07	5.60E-05	1.39E-06	NO DATA	6.56E-07
CS134	4.66E-05	1.06E-04	9.10E-05	NO DATA	3.59E-05	1.22E-05	1.30E-06
CS136	4.88E-06	1.83E-05	1.38E-05	NO DATA	1.07E-05	1.50E-06	1.46E-06
CS137	5.98E-05	7.76E-05	5.35E-05	NO DATA	2.78E-05	9.40E-06	1.05E-06
CS138	4.14E-08	7.76E-08	4.05E-08	NO DATA	6.00E-08	6.07E-09	2.33E-13
BA139	1.17E-10	8.32E-14	3.42E-12	NO DATA	7.78E-14	4.70E-07	1.12E-07

TABLE 7-16 - IGNITION SOURCES OF UNDERGROUND METAL AND NONMETAL FIRES (a)

SOURCE	% OF TOTAL
Electrical	42.1%
Welding Sparks/Hot Slag	15.8
Miscellaneous	13.0
Spontaneous Combustion	11.4
Engine Heat	7.0
Friction	6.1
Explosives	3.5

(a) From Mine Safety and Health, Vol. 6, n. 2, 1981.

TABLE 7-17 - ACCIDENT AVERAGE SEVERITY, 1981(1)

ACCIDENT CLASSIFICATION	PARTIAL PERMANENT DISABILITY	TEMPORARY NFDL
Electrical	-	14
Explosives and breaking agents	2,027	35
Fall of face, rib	-	45
Fall of roof	872	26
Fire	-	374
Powered Haulage	214	37
Hoisting	-	36
All accidents	283	26

(1) Data shown in terms of number of days lost for a given accident classification

8.0 CONCLUSIONS

The first phase of the HLW-PSSA project was primarily devoted to the gathering, sorting and assembling information as appropriate for a safety assessment of a nuclear waste repository. This section highlights the results obtained from the development of the logic models, the evaluation of consequences, and the data base required for quantitative evaluation of the accident scenarios. The application of these results in support of the tasks to be performed in the second project phase is also discussed.

8.1 ACCIDENT SCENARIOS

The accident scenarios developed for the basalt repository concept were identified using material flow diagrams for each specific waste handling process. This is similar to other conceptual studies of repository concepts (Bechtel, 1981, SRENCO 1978). Scenarios addressing emplacement and retrieval operations were kept separate to treat retrieval as an additional incremental risk contributor that could be examined separately.

During scenario development, initiating events identified from the material flow diagrams were screened (Pepping, 1981) to remove events of low frequency and consequence. Screening was performed separately for each consequence type initially considered, allowing the eventual development of accident scenario families for each consequence type (see Tables 2-18 through 2-22). Although the accident scenarios contributing to personnel radiological risk and repository availability were not used in the selected sample problem, the logic model development is complete and available for future use. In particular, personnel radiological and nonradiological risks may require further examination in the future as repository construction and operation are initiated.

The concept of compromise of long-term repository viability was treated in this study only to identify potential contributing events. Further evaluation is beyond the scope of a preclosure analysis. Table 2-22 lists the initiating events/accident scenarios capable of compromising the long-term viability of the repository concept. These events are predominantly related to the movement of the canister after final inspection. A transfer cask is used for transport to underground placement and a hydraulic ram is used to eject the canister from the cask into the borehole. These operations have the potential to crease or puncture the canister and no concurrent visual observation/inspection mechanisms are identified in the conceptual design information. After the canister is loaded into the transfer cask at the hot cell outlet port, visual contact is lost. Transportation accidents such as crane drop, transporter collision, and waste cage restraint failure and emplacement accidents related to the cask positioning or hydraulic ram activities can all cause undetected canister damage. These activities may require additional administrative controls and/or design additions for inspection following borehole insertions.

Accident scenarios grouped into the remaining consequence categories will have different uses in future project phases. Scenarios contributing to both public and personnel radiological exposure are immediate concerns related to the licensing and operation of a nuclear fuel waste repository. These two consequence types will therefore receive the majority of attention in future tasks in order to support the repository development and licensing effort in a timeframe responsive to current repository goals.

8.2 CONSEQUENCE EVALUATION

Radiological and nonradiological consequences were evaluated from the standpoint of selecting the most appropriate approach for quantitative evaluation of each consequence type and recommending the specific consequence types for further analysis in future project tasks. The following consequence types were evaluated initially: (1) radiological consequence to the public, (2) radiological consequence to the worker, (3) nonradiological consequence to the worker, (4) impact on repository availability, (5) compromise of a repository's ability for long-term geologic isolation of high-level waste, and (6) financial impact. The compromise of long-term repository viability is not a risk contributor in the preclosure phase and, as mentioned earlier, has been included in the evaluation to identify event sequences and/or operations that are potentially capable of generating this type of consequence in order to facilitate consideration of preventive procedures and operations at an early stage in the repository design/licensing process. The other consequence types have been developed to the point where a representative body of accident scenarios exists for each, including the models necessary for quantitative evaluation of the consequence.

Of the different consequence types considered, the radiological (public and worker) and nonradiological occupational consequences were recommended for further evaluation. However, in the interest of providing useful information to the NRC for licensing purposes in a time frame consistent with the DOE repository design schedule, emphasis should be on radiological public and worker consequences.

In the quantitative evaluation of radiological risk to the public and worker, the transport and behavior of radionuclides play a key role and sophisticated transport models have been developed in previous studies. Reliable estimates of release fractions are difficult to obtain largely because of the accident-specific nature of the release and the lack of adequate experimental data to support postulated release assumptions. This large uncertainty in the release fraction has to be recognized and accounted for in future work.

For the next phase of the project which will involve quantitative analysis of a sample problem, release fraction for gases and volatiles given in previous studies will be used. Empirical models have been proposed in Section 3.1.5 to determine release fractions of particulates. For transport of these particulates in confined areas, the Indoor Air Quality model will be used, while the Gaussian plume dispersion model (CRAC2) will be used to determine aerosol release to the environment.

Accident scenarios capable of creating occupational injury hazards have been identified in previous sections; however, no scenario quantification effort is required. Occupational injury data for analogous activities already exist and can be used directly to estimate occupational hazards to repository workers. Some of these data have already been gathered and are given in Section 7 of this report. The second phase of this project will contain a subtask for the acquisition of further occupational hazard data in related activities, particularly, mining operations. These data will then be used to estimate expected occupational injury rates from repository development and operation. This is not a major task, however, The emphasis of future phases will be in estimating radiological incident frequencies and consequences.

The two other consequences mentioned in this report are repository availability and financial impact. Financial impact represents a monetary composite of the expense of all other consequences (see Section 3, Figure 3-1). At this time no effort is planned to combine the different types of risk contributors into a single financial risk curve. The information will be available to perform this summation, if interest is expressed in the future.

Accident scenarios leading to loss of repository availability have been identified in Section 2. Both radiological and nonradiological scenarios contribute to this consequence in addition to many unique scenarios. The need for emphasis on issues directly relevant to the repository licensing process dictates the exclusion of this consequence from further development at this time. Again, the information is still available should further definition of risk to repository availability be required in the future.

8.3 FAULT TREE DEVELOPMENT

Fault trees were developed for systems identified as intermediate events during event tree development and covered in sufficient detail in the basalt repository conceptual design description. The majority of the systems meeting these criteria are air circulation/ventilation systems for both surface and subsurface facilities. The fault trees have been constructed for the first phase of this project, and will be reduced and quantified in the second project phase. Preliminary conclusions can now be drawn concerning the systems described and the performance criteria specified. Due to the passive nature of the spent fuel/storage canister assemblies, the primary barriers to environmental radionuclide release are the cladding, the canister, and the closed confinement of the repository enclosure. Should a handling accident lead to the break of the canister (and presumably the cladding of the contained fuel) the only barrier left is the integrity of the repository atmosphere containment. There are two separate air tight containments in the waste handling building and another system for the subterranean environment. These systems have redundant backup fans/filter assemblies to ensure the continuation of air filtration at a net pressure slightly less than atmospheric. Fault tree models developed for these systems in Section 4 took credit for backup assemblies wherever possible.

Based on the conceptual design description, several specific system function modifications should be considered as system design matures to ensure that the desired redundancy and independence are obtained. They are discussed below.

The waste handling building has both a primary and a secondary confinement exhaust ventilation/filtration system. These systems serve different areas of the building. The intent of the design is to operate the primary system (the one serving the most contaminated areas) at a slightly lower pressure than the secondary to force any leakage to flow in the direction of higher contamination. There are several accident scenarios where either primary or secondary operation can provide sufficient ventilation to prevent radionuclide release to the environment. The two systems are not completely independent and neither are their respective backup units. All primary and secondary fan units exhaust to a common plenum with a single tornado damper/bird screen and a single stack. These commonalities could cause all confinement exhaust systems to fail in the event of damage to the plenum or stack (e.g. earthquake). The subterranean confinement exhaust system has a similar commonality. The exhaust of five fan/filter assemblies is routed to a common plenum, tornado damper/bird screen assembly, and exhaust stack.

Both the waste handling building and the subterranean confinement ventilation/filtration systems require intake supply fans to provide sufficient circulation. The power distribution system for the facility is not detailed enough to provide specific 'fan to load bus' description. However, the standby exhaust fans for all systems (primary, secondary, and subterranean) appear to be supplied from a standby power load center. In the event of a loss of normal power, the 13.8KV diesel generators would start and vital standby power loads would be powered directly from a standby load center.

Power to the intake supply fans appears to be supplied entirely from normal power load centers. Standby power can only be supplied to these loads by closing a bus cross-tie breaker between standby and normal power buses. This is not only an additional necessary equipment operation (probably manually initiated) but it also adds other loads to the standby bus, possibly exceeding generator capacity. This condition is not certain from the design information available; however, to preserve the redundancy and independence of backup systems, both the exhaust and the supply intake fans must be powered by redundant bus arrangements. This detail should be followed closely as repository design matures due to the importance of confinement ventilation/filtration systems to the prevention of radiological releases from reaching the open environment.

The waste transport shaft exhaust system is an additional confinement exhaust system that can play an important role in mitigating certain accident scenarios involving the loading and transportation from surface to subterranean locations. Two fan/filtration units are used continuously (no installed backup) to draw air up the waste transport shaft and through the filter assemblies. The power distribution description indicates these fans are supplied from a normal power bus, requiring the same cross-tie

operation described earlier to restore fan operation given a loss of off-site power. Final design should consider 1) a backup unit for this function and 2) the use of a standby power bus to supply fan loads.

The waste handling building confinement exhaust ventilation/filtration systems (both primary and secondary) can be cross-tied between the filter train outlets and the blower fan inlets. The schematic provided for these systems does not show a check valve on the inlet to the fan. In the case of the primary confinement system, a loss of the normal fan followed by the startup of the standby fan could allow bypass of the entire airflow path through the repository in favor of the lower resistance of pulling air back through the disabled fan and the cross-tie line. A check valve in the inlet to each fan (both primary and secondary systems) would eliminate this possibility.

Other systems identified in the accident scenarios did not have sufficient description from the conceptual design to adequately permit development of fault tree models. Instead, a data base (see Section 7) that contains industry-wide failure rate statistics for these types of systems in similar applications was assembled. For example, a subterranean radiation monitoring system is used to monitor the tunnels and airflow passageways. In the event that detectable levels of airborne contamination are present, this system sends a signal to the confinement exhaust system. The signal closes the dampers on the filter bypass ducts, opens the dampers on the inlet and outlet of the filter train, and starts the second stage of the exhaust fans. This monitoring system with associated alarm/trip functions is not described explicitly in the conceptual design description. However, data are available on the performance of radiation monitoring systems with alarm/trip functions in operating reactors and fuel handling facilities. The only assumption inherent in using these data is that the system installed in the repository will be at least as reliable as those in present day use. This is not considered to be a limiting assumption.

Other systems will also be quantified directly without breaking them into component contributions using fault tree logic. Surface radiation monitoring systems, air lock seal doors and compressed air systems are assigned industry failure rates. As the repository system design matures, many of the systems should be explicitly modeled using fault tree logic to obtain more facility specific failure estimates.

8.4 SAMPLE PROBLEM SELECTION

A sample problem was selected to demonstrate the overall methodology and to verify analytical methods at the event tree level. The problem was chosen as a subset of the accident scenarios leading to public radiological exposure risk during emplacement operations (see Table 2-18). Contributing scenarios were selected from all major facility processing areas identified in Table 2-1. Intermediate events required for the sample problem scenarios include all the systems fault trees.

Simplification at the event tree rather than at the fault tree level was chosen because of available computer programs. At the fault tree level, the SETS computer program provides fault tree reduction capability and the VALUE computer program quantifies the SETS output expression, including the generation of the modified Fussell-Vesely component importance measure. As both these programs were extensively used previously and verified for other applications, it is cost-effective to only solve the fault tree expressions once.

Computer programs required to quantify event tree expressions, including dependency and importance measure, are not currently available. The sample problem was chosen to be small enough to allow a check calculation to be performed by hand. It should be emphasized that the simplicity of this family of scenarios (due to omission of some 90 scenarios) requires that the final estimates of risk and importance rankings be used only for demonstration purposes. The results do <u>not</u> represent a preliminary evaluation or order of magnitude estimate of the final results. Inclusion of all the identified accident sequences, completion of the human error analysis, and estimation of common cause contribution must be performed in order to draw final conclusions.

8.5 IMPORTANCE RANKING

The Fussell-Vesely importance measure (normalized) was chosen as the best measure for the purposes of this project (see Section 6 and the appendix). The other importance measures either did not adequately meet project objectives, did not consider every occurence of the component, or were not readily usable in already existing computer programs.

8.6 DATA BASE DEVELOPMENT

A list of relevant data was derived from the literature and government/industry data banks with emphasis on events identified for the sample problem. The data gathering task will continue in the next study phase with emphasis on human error and accidents related to mining activities.

Initiating event frequencies, intermediate event probabilities, and component failure rates and repair times were derived from various sources such as the Government-Industry Data Exchange Program (GIDEP), Nuclear Plant Reliability Data System (NPRDS), System Reliability Service Data Bank, IEEE Standards, Federal Aviation Administration Statistical Handbook, Department of Transportation publications, EPRI studies, and government-sponsored studies on nuclear reactors and waste management.

Data for radiological consequence evaluation were derived primarily from government-sponsored studies and consisted of data on radionuclide inventories, release fractions, and dose factors. While meteorological information on the Hanford site is known to be available and can be obtained from the National Climatic Center in North Carolina, no specific information was compiled at this time. The more detailed consequence

evaluation in the next study phase will help define the specific meteorological information required to calculate popu'ation dose.

The data on personnel injuries in mine-related activities resulted from the work performed by Engineers International (See Section 7). This data base will be augmented in future work. The data collected, so far, have provided useful insights into areas of likely concern when evaluating the impact of repository operations to the workers.

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APPENDIX

AN EVALUATION OF IMPORTANCE MEASURES FOR PROBABILISTIC RISK ANALYSIS

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ABSTRACT

Several importance ranking measures that have been used in the nuclear industry to study systems performance were evaluated for the purpose of identifying one or two measures that would be used in risk evaluations of preclosure activities at proposed high-level-waste repositories. They are: the Birnbaum measure, structural measure, criticality measure of basic event importance, upgrading function, Fussell-Vesely measure of basic event and minimal cut set importance, Barlow-Proschan measure of basic event and minimal cut set importance, and the sequential contributory measure of basic event importance.

Using a set of criteria, a preliminary screening of these measures was performed to identify those that are applicable to the safety and risk evaluation of a nuclear waste repository. The importance measures selected from the preliminary screening were then applied to a specially selected fault-tree problem to test each measure's performance under various conditions.

On the basis of the results of this study, the Fussell-Vesely importance measure is recommended for future work. This measure provides the most appropriate means of ranking component importance both as a function of system unavailability and unreliability. Furthermore, the inherent straightforwardness of the Fussell-Vesely equation allows for ease in the calculation of probability distributions.

1.0 INTRODUCTION

Safety and reliability analyses have become increasingly important in the conceptual design, developmental and operational phases of complex facilities and systems. However, it is not always sufficient to know just how safe or reliable a system is (or can be). We often need to identify the components or events that contribute most significantly to system failure in order that changes or modifications to improve safety or reliability can be implemented in a logical and cost-effective manner. Prioritization or importance ranking of components is very useful to a decision maker with limited resources who is faced with the problem of implementing many modifications.

In the nuclear industry, where safety has been of high concern, several ways of prioritizing components, subsystems or systems with regard to their contribution to overall plant risk have been proposed. The increased activity in recent years towards the closing of the back-end of the nuclear fuel cycle have focused attention to the risks involved in the permanent storage of spent fuel and of high-level waste in geologic repositories. At this important time where conceptual repository designs are being developed and potential sites selected, it is necessary to have available a suitable methodology to rank components and basic events according to their contributions to the safety of operating a nuclear waste repository facility. The selected importance measure(s) need to be applicable to fault tree analysis which is a basic analytic tool for the quantitative safety assessment of preclosure activities in a repository. These importance measures could be used to help streamline the regulatory and licensing process and, at the same time, to provide useful information to ease public concern regarding the safety of this operation.

In this report, we present the results of an evaluation of several importance ranking measures that have been used in the nuclear industry for fault tree analysis applications. Using a set of criteria that are discussed in Section 3, a preliminary screening of measures for applicability to the preclosure phase of the repository is performed. The selected importance measures are then applied to a specially selected fault-tree problem to determine how each measure performs under various conditions. On the basis of the results generated, an importance measure is recommended for use in future risk evaluations of preclosure operations at proposed high-level waste repositories.

2. IMPORTANCE MEASURES

This section reviews the basic attributes of several importance measures that have been proposed or used in nuclear safety or in risk-related studies for importance ranking of components or system and the available computer programs for calculating these measures.

2.1 IMPORTANCE MEASURES FOR FAULT TREE APPLICATIONS

A number of importance measures have been proposed or used in connection with fault tree analysis. A fault tree is a deductive method of analyzing a system's degree of safety or reliability by postulating a failure state of the overall system and by identifying the components or events that are necessary and sufficient to contribute to the occurrence of that failure state.

Some importance measures are applicable only to non-repairable components and are dependent on the system unavailability. Others work only for repairable components and are dependent on the expected number of system failures for a given time interval. However, under the assumptions that failures are completely independent and have low probabilities (i.e., the component failure probability does not exceed 1%) it can be shown that these measures are closely related to each other. Thus, the choice of which measure to use depends essentially on the kind of information that one seeks regarding the system.

A comprehensive assessment of importance measures as applied to fault-tree analysis was performed by H. Lambert (Ref. 1). Engelbrecht-Wiggans and Strip have also evaluated various importance measures and have shown them to be closely related to the concept of probabilistic values in game theory (Ref. 2).

2.1.1 Birnbaum Measure

The Birnbaum measure of importance is defined as the probability that the system is in a critical state for component i, i.e., the difference between the probability that the system fails with component i failed and the probability the system fails with component i functioning, or:

$$I_i^B = g\,(1_i, \underline{Q}(t)) - g(0_i, \underline{Q}(t)) \qquad (1)$$

where: $\underline{Q}(t)$ is the failure probability of the components other than i which contribute to system failure,

$g\,(1_i, \underline{Q}(t))$ is a probability function of the state vector having 1 in the i^{th} position and $\underline{Q}(t)$ in all other positions;

and $g(0_i, \underline{Q}(t))$ is a probability function of the state vector having 0 in the i^{th} position and $\underline{Q}(t)$ in all other positions.

The Birnbaum measure is a conditional probability since the state of component i is fixed. As such, it is not a function of the failure probability of component i. Regardless of the failure probability of the top event, a component present as a single order minimal cut set will always be ranked higher in importance than a component in a higher order minimal cut set provided the failure probability of any component in the system under consideration is less than 1. Therefore, the measure has limited applications. However, many of the importance measures which will be discussed below can be defined in terms of Birnbaum's measure.

2.1.2 Structural Measure of Importance

The structural measure of importance is a deterministic measure which evaluates the importance of a basic event or component to system performance based on the structural position of a given component in a fault tree without regard for the actual failure probability of this component. It only considers two states for that component, either failed or not failed. Generally, a component present in a single order cut set will always have the highest structural importance ranking.

The structural measure of importance for component i is defined as the number of critical states for component i divided by the total number of states for the n-1 remaining components (2^{n-1}). It is, therefore, the fractional number of system states (for the n-1 remaining components) which are critical for component i.

Lambert (Ref. 1) defines structural importance as:

$$I_i^{ST} = g(1_i, \underline{1/2}) - g(0_i, \underline{1/2}) \quad (2)$$

where: $g(1_i, \underline{1/2})$ is the function of the state vector having 1 in the i^{th} position and 1/2 in all other positions; and

$g(0_i, \underline{1/2})$ is the function of the state vector having 0 in the i^{th} position and 1/2 in all other positions.

For the series-parallel system shown in Fig. 1, the three critical cut sets for component i are {1}, {1,2} and {1,3}. This is verified by the following equation:

$$C_i = 2^{n-1} \{g(1_i, \underline{1/2}) - g(0_i, \underline{1/2})\} \quad (3)$$

where C_i is the number of critical cut sets for component i.

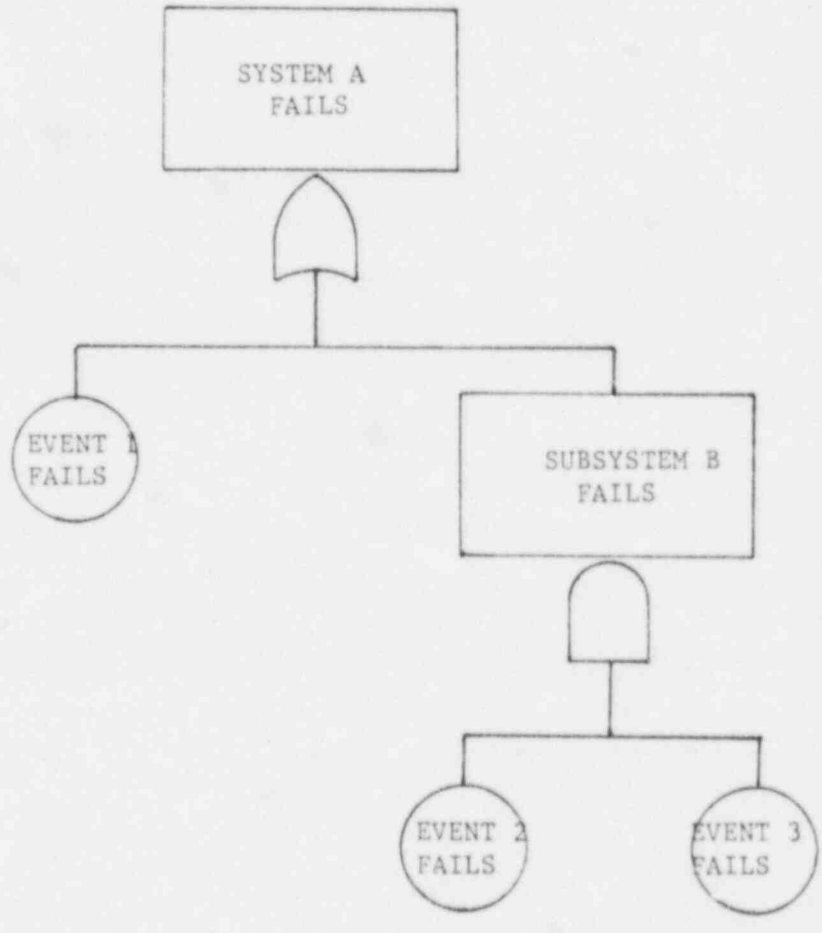

Fig. 1. Series-parallel system.

2.1.3 Criticality Measure

The criticality measure of basic event importance is the Birnbaum measure weighted by the contribution of component i to the system unavailability and can be expressed mathematically as:

$$I_i^{CR} = \frac{I_i^B P_i}{P_T} \tag{4}$$

where

I_i^B = Birnbaum measure or criticality

P_i = failure probability of component i in time interval t

P_T = System failure (or unavailability) in time interval t.

The system failure probability is calculated using the minimal cut set upper-bound equation:[1]

$$P_T = 1 - \sum_{i=1}^{k} (1 - P(MCS)_i) \tag{5}$$

where: $P(MCS)_i$ is the failure of minimal cut set i.

Unlike Birnbaum's measure which is neither a function of component nor of system failure probability, the criticality measure provides a more meaningful importance ranking. For example, a component with high failure probability but present in a high order cut set may be ranked higher in importance if system failure probability is high.

2.1.4 Upgrading Function of Basic Event Importance

The upgrading function measures the fractional reduction in the probability of the top event of a fault tree when the failure rate of component i is reduced fractionally. The upgrading function is applicable

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[1]A minimal cut set is a smallest combination of component failure which, if they all occur, will cause the top event to occur. See attachment for more detailed discussion of the minimal cut set upper bound equation and how it compares to the rare event approximation for top event failure probability.

only to non-repairable systems and only when relative failure rates (proportional hazards) are known. The proportional hazard, a , is defined by

$$F(t) = 1 - e^{-aR(t)} \tag{6}$$

where $F(t)$ is the probability that a non-repairable component fails in time interval t and $R(t)$ is the common hazard assumed to be shared by all components; a can be regarded simply as a failure rate.

This measure is useful during the design stages of a system where data on repair times may not be available and only the relative failure rates are known.

2.1.5 Fussell-Vesely Measure of Importance

The Fussell-Vesely (F-V) measures include the basic event importance and the minimal cut set importance measures. The importance measure for basic event i, I_i^{FV}, is defined as the probability that basic event (or component) i contributes to system failure, given that the system has failed within time interval t. Mathematically, this is the probability of the union of all minimal cut sets containing basic event i given that the system has failed (i.e., the Top Event has occurred):[2]

$$I_i^{FV} = \frac{P\left(\bigcup_{k=1}^{J} (MCS)_k\right)}{P_T} \tag{7}$$

where J = number of minimal cut set terms containing basic event i,

where $P\left(\bigcup_{k=1}^{J} (MCS)_k\right)$ = probability of the union of minimal cut set k containing basic event i,

P_T = probability of the top event

The F-V basic event importance measure provides a numerical algorithm for ranking a basic event not only according to how many minimal cut set terms the basic event is part of but also according to how important these terms are to the occurrence of the top event.

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[2]See Attachment for presentation of a mathematical expression that is an approximation of Equation 7.

The F-V measure of cut set importance ranks cut sets according to their fractional contribution to the top event and is given by

$$I_{CS}^{FV} = \frac{P(MCS)}{P_T} \tag{8}$$

where P(MCS) = minimal cut set failure probability
P_T = top event failure probability.

2.1.6 Barlow-Proschan Measure of Importance

The Barlow-Proschan (B-P) measures are based on the concept of interval reliability which assess the contribution of a basic event or a minimal cut set over an interval of time and is useful for the analysis of catastrophic failures.

The B-P basic event importance is the probability that an initiating event causes the top event to occur. It is computed from the Birnbaum measure, weighted by the ratio of the failure frequency of the basic event to the expected number of system failures (the integral of the top event rate over mission time) for a given time interval, i.e.:

$$I_i^{BP} = \frac{\int_o^t \left[g\,(1_i\,,\,\underline{Q}(t')) \;-\; g(0_i\,,\,\underline{Q}(t')) \right] \; W_{f,i}\,(t')\,dt}{E\left[N_s(t)\right]} \tag{9}$$

where $g(1_i, \underline{Q}(t')) - g(0_i, \underline{Q}(t'))$ = Birnbaum measure for basic event i

$W_{f,i}$ = failure frequency of basic event i

$E\left[N_S(t)\right]$ = expected number of system failures for a given mission time

The B-P measure is used to rank only initiating events identified in the fault tree. The sum of all initiating event importances in the B-P measure of importance equals unity. Essentially, this measure is useful when information on the way in which system failure has occurred and on the most probable cause of failure are required.

The B-P measure of cut set importance is the probability that a minimal cut set causes system failure. For a given minimal cut set to cause system failure an initiating event must have occurred and all other basic events in that cut set must have failed at the time the initiating event occured. If each minimal cut set contains an initiating event, the B-P measure of cut set importance is the same as Fussel Vessely's cut set importance measure.

Lambert has pointed out that the Fussell-Vesely definition of cut set importance always assigns more importance to a cut set of a lower order than a cut set of a higher order when basic event probabilities are equal (Ref. 1). This is true for the B-P measure only when no replication (i.e., event is present in many cut sets) of events occurs. A case where the B-P measure ranks a higher order cut as more important than a lower order cut set is discussed in Reference 1. The system contains ten components and the top event is represented by 21 minimal cut sets. One cut set contains four components which do not appear in other cut sets. The other 20 cut sets were obtained by taking all combinations of three components from the remaining six components. It was shown that for a top event probability of approximately 0.64, the four-component cut set has a greater probability of causing the system to fail than a three-component cut set. If the events contained in the lower order cut set were not replicated in other cut sets, this lower order cut would be ranked more important. Thus, for the B-P measure, when no replication of events occur, lower order cut sets will generally be more important than a higher order cut set when basic event probabilities are equal.

The B-P measures of importance are limited in that only initiating events and cut sets containing initiating events are ranked.

2.1.7 Sequential Contributory Measure of Importance

The sequential contributory (SC) measure like the B-P measure of importance is based on the concept of interval reliability. The SC measure of importance is the probability that component i (enabling event) is contributing to system failure when another component j (initiating event) causes the system to fail. In some cases, an event can be either an initiating event or an enabling event. For a redundant system of two components whose failure will cause system failure, the component that

fails first is an enabling event and the component that fails last is the initiating event. Thus, the SC importance measure is expressed as:

$$\sum_{\substack{j \\ i=j}} \frac{\int_{o}^{t} I^{B}_{i,j}\, Q_i\,(t')\; W_{f,j}\,(t')\; dt}{E\left[N_s(t)\right]} \tag{10}$$

where $I^{B}_{i,j}$ = criticality function for (Birnbaum) enabling event i and initiating event j = $g_i(1_i, 1_j, Q(t')) - g_i(1_i, 0_j, Q(t'))$

Q_i = failure probability of enabling event i

$W_{f,j}$ = failure frequency function for initiating event j

$E\left[N_s(t)\right]$ = expected number of system failures.

2.1.8 Significance Indices

A. Bhattacharya and S. Ahmed have defined a set of three significant indices which quantify the importance of each component, with respect to the mean and variance of the top event probability (Ref. 3). These indices are described as follows:

a. $$S_m^{(1)} = \frac{\bar{P}_m}{\bar{P}} \frac{\delta \bar{P}}{\delta \bar{P}_m} \tag{11}$$

where $\bar{P}_m$ is the mean failure probability of component m and $\bar{P}$ is the mean top event probability, and δ symbolizes the change in a particular parameter. $S_m^{(1)}$ measures the percentage change in the mean top event probability with respect to unit percentage change in component mean. This significance index is essentially the same as the upgrading function defined by Lambert (Ref. 1).

b. $$S_m^{(2)} = \frac{\bar{P}_m}{\sigma^2} \frac{\delta \sigma^2}{\bar{P}_m} \tag{12}$$

where σ^2 is the variance of the top event probability distribution.

$S_m^{(2)}$ is the percent change in the top event variance with respect to unit percentage change in component mean.

c. $$S_m^{(3)} = \frac{\sigma_m^2}{\sigma^2} \frac{\delta \sigma^2}{\delta \sigma_m^2} \tag{13}$$

where σ_m^2 is the corresponding variance of the m^{th} component

$S_m^{(3)}$ is the percent change in the top event variance with respect to unit percentage change in component variance.

Significance Indices $S_m^{(2)}$ and $S_m^{(3)}$ provide information on the scatter of component failure data, i.e., they identify the components which significantly contribute to the uncertainty of the top event.

2.1.9 Risk Importance Measures

W. Vesely, et al (Ref. 4) have introduced two measures of risk importance, risk reduction worth and risk achievement worth.

The risk reduction worth is the reduction in risk if a system feature were assumed to be optimized or if it were assumed to be made perfectly reliable. Depending on how the decrease in risk is measured, it can be defined as a ratio or an interval.

On a ratio scale, the risk reduction worth D_i of feature i is defined as:

$$D_i = R_o/R_{\bar{i}} \tag{14}$$

where R_o = present risk level

$R_{\bar{i}}$ = decreased risk level with the feature optimized or assumed to be perfectly reliable.

On an interval basis, the risk reduction worth δ_i is given by:

$$\delta_i = R_o - R_{\bar{i}} \tag{15}$$

The risk achievement worth is the increase in risk if a system feature were assumed not to be present or to be failed. Again, depending on how the increase in risk is measured, the risk achievement worth can either be

defined as a ratio or an interval. On a ratio basis, the risk achievement worth A_i of feature is:

$$A_i = R_i^+/R_o \qquad (16)$$

where R_i^+ = increased risk level without feature i or with feature i assumed failed.

On an interval basis, the risk achievement worth a_i is:

$$a_i = R_i^+ - R_o \qquad (17)$$

Vesely has shown that these risk measures can be equated in terms of Birnbaum's measure of importance or the F-V measure of importance.

The Birnbaum measure is shown to be the sum of the risk achievent worth (a_i) and the risk reduction worth (d_i) of component i. The F-V measure is expressed in terms of the decrease in risk level, R_i^-, and is related to the risk reduction worth on a ratio scale, D_i. The bases for the derivation of the expressions relating the risk achievement worth and risk reduction worth to the Birnbaum and F-V importance measures are not presented explicitly in Reference 4. In view of the objectives of the HLW-PSSA project, we do not believe there is any advantage in using these risk measures instead of the more basic importance measures that have been discussed in previous sections.

2.2 COMPUTER PROGRAMS

Of the computer programs that are currently available for ranking basic events and cut sets in their order of importance to system failure (i.e., top event occurs) we have considered the following ones because of their ability to accept as input the minimal cut sets nerated from the SETS (Ref. 5) fault-tree reduction code:

For systems modeled with fault trees, IMPORTANCE (Ref. 6) allows calculation of various importance measures such as the structural measure, the Birnbaum measure, the criticality measure, the Fussell-Vesely (F-V) measure, the Barlow-Proschan (B-P) measure, and the sequential contributory (S-C) measure.

In addition to minimal cut sets (obtained from SETS), this code requires as input the failure rates and repair times of all basic events contained in the minimal cut sets. The failure and repair distributions are assumed to be exponential. All measures are computed assuming statistical independence of basic events.

The B-P and S-C measures are time integrated quantities which depend upon the sequences of events leading to system failure and are dependent on assuming a pseudo repair time for all initiating events, which is the sum of the repair time of all initiating events. The current version of the code cannot handle differing repair times for initiating events.

The SEP (Set Evaluation Program) computer code (Ref. 7) provides a means of measuring the contribution of each basic event to system failure by taking the product of the Birnbaum measure and the event probability. When normalized by the system failure (top event) probability, this expression essentially becomes the Fussell-Vesely measure (Ref. 7). SEP calculates these measures separately for the noncomplemented and complemented occurrence of each event. These measures are calculated assuming that the event and its complement are independent events. The sum of these two measures yields the true importance measure for the event.

VALUE (Ref. 8) ranks the basic event and minimal cut set contribution to top event probability using the F-V equation, modified to yield importance rankings which sum to unity.

The computer code STADIC-2 (Ref. 9), which is a general purpose Monte Carlo simulation code, can also be used to calculate importance measures such as the F-V measure and to propagate data uncertainties, thus, providing importance ranking factors in the form of distributions.

3. EVALUATION

A major objective of the High-Level Waste Preclosure Systems Safety Analysis (HLW-PSSA) project, of which this evaluation is a part, is the development of a systematic methodology to identify and quantitatively prioritize the structures, components, systems, and operations which are important to safety during the preclosure phase of the HLW repository. Although the repository is still at the conceptual stage, risk assessment at this time is directed mainly at identifying the components or systems that are dominant risk-contributors in order that design modifications can be made (if needed) before construction and operation.

The selection of importance measure(s) for use in this project has to consider several factors relevant to the repository such as the extent of the availability of repository-specific data, compatibility with analytical tools that will be used in the preclosure risk assessment, the ease of applicability to repository situations, etc.

The importance measures discussed in Section 2 require information on basic event (component) failure rate, repair time, mission time, initiating and enabling event identification, etc. For a geologic repository, failure rate and repair data of relevant equipment and operations can be obtained (although no geologic repository is yet in operation) since many are standard industry equipment.

A set of criteria has been established to screen out the importance measures which are inadequate in meeting our task objective and to select one or two importance measures that can best prioritize systems and components of importance to preclosure operations in a geologic respository. These criteria are:

1. The importance measure should use readily available repository data (i.e., component failure rates and repair times) and not require a prohibitive level of data detail.

2. Useful insights on a repository system performance should be provided by the importance measure.

3. The measure should be easily applicable to repository situations and it should yield scrutable results.

4. Numerical importance ranking should be provided by the measure.

5. Uniform ranking of all components (or events) represented in the fault tree model should be possible.

6. The measure should be applicable to repairable and non-repairable components.

7. A computer program should be available to calculate the rankings.

3.1 PRELIMINARY SCREENING

The significance index is directly related to the upgrading function. The risk reduction worth and risk appreciation worth measures do not seem to offer any advantage over the other basic importance measures that were discussed in Section 2. Hence, these measures are not addressed explicitly in the remainder of this report.

The different features of the importance measures were evaluated using the above criteria. The results are summarized in Table 1. Since the IMPORTANCE code is capable of calculating all seven importance measures, it was used to determine the importance rankings of basic events for the simple fault tree model shown in Fig. 2. This fault tree was first reduced to its minimal cut sets using the SETS code.

In running IMPORTANCE, one can input failure data either in terms of failure rate and restore time (for repairable components only) or in terms of proportional hazards (i.e., relative failure rates). The upgrading function can be calculated only if proportional hazards are used as input. The fault tree shown in Fig. 2 was therefore evaluated using two sets of input to allow calculation of all the studied importance measures. The fault tree and the data used for this exercise are the same as those in the example of Reference 6. The results are shown in Tables 2 and 3. The Birnbaum measure and structural measure yield the same order of ranking, i.e., single event cut sets are ranked higher in importance regardless of their failure probability. The remaining measures of importance shown in Table 2 yield approximately the same ranking.

The inherent nature of the different importance measures (as explained in Section 2) and the limitations in the IMPORTANCE code itself, prevented us from comparing all the measures on the same scale. The IMPORTANCE code classifies the different measures into two general categories: (1) those that are weighted by the system unavailability, and (2) those that are weighted by the expected number of system failures for a given time interval and are useful for ranking continuously operating systems for which failure cannot be tolerated. Birnbaum, Structural, Criticality, Upgrading Function, and Fussell-Vesely (F-V) measures of importance belong to the first category and are generally applied to non-repairable systems. The Barlow-Proschan (B-P) and the Sequential Contributory (SC) measures fall into the second category. In the example shown, if we combine the calculated B-P and SC importance values, the resulting event ranking is the same as the F-V ranking (see Tables 2 and 3). This is true if there is only one initiating event in a minimal cut set.

It is possible to apply the importance measures in the first category to repairable systems. However, the importances will be calculated as a function of the limiting system unavailability which is expressed as

$$\bar{U} = R\tau_S , \tag{23}$$

where R is the top event failure rate and
τ_s is the mean time to repair of the initiating events.

The top event rate is a function of the criticality and failure frequency functions of the initiating events present in the fault tree:

$$R = \sum_{i=1}^{n} \Delta g_i W_{f,i} , \tag{24}$$

where Δg_i = the criticality function (or Birnbaum's measure) for initiating event i and

$W_{f,i}$ = failure frequency of initiating event i.

From Equations 23 and 24, it can be seen that the limiting system unavailability can be orders of magnitude higher than the unavailability of a non-repairable system with equivalent component failure rates, depending on the value of the repair time, τ_s. Thus, in general, rankings based on system unavailability differ from those based upon interval reliability.

The present version of IMPORTANCE cannot handle different repair times for initiating events. We consider this a serious limitation, particularly for the purpose of our investigation.

Based on the results presented in Tables 1 to 3, the following measures have been screened out as inadequate in meeting our objectives:

- Birnbaum's measure - as discussed in Sec. 2.1.1, this measure by itself is not very useful since it is not a function of failure probability of the top event. Hence, component in a single order cut set will always be ranked higher in importance over those components in higher order cut sets, provided the failure probability of any component is less than 1.

- Structural measure - this is a deterministic (not probabilistic) measure and it depends on the structural position of a component in the fault tree. Thus, a component present as a single event minimal cut set will always be considered to be more important regardless of its failure probability.

- o Upgrading function - this measure is applicable to non-repairable components and only when relative failure rates are known. Although its application can be useful during the design phase, other importance measures that are more versatile can provide the same information, and perhaps more.

- o Barlow-Proschan measure - this measure is limited in that only the contribution of initiating events to overall system failure is considered. Only the IMPORTANCE code is currently capable of calculating this measure as applied to fault tree problems. However, the IMPORTANCE code assumes only one repair time for all initiating events. Thus, the results will not be meaningful, particularly, for cut sets containing two or more events. Whereas the concept of interval reliability is quite interesting and useful, the application of the Barlow-Proschan method to the safety analysis of repository operations may not be adequate since we need to determine the sensitivity of a system failure to all components or events present and not only to initiating events.

3.2 FINAL SCREENING

The remaining importance measures, namely, criticality, F-V and SC were subjected to a more rigorous analysis to determine whether they rank in a consistent manner and to determine how the ranking behaves as a function of various factors such as repair time, top event probability, minimal cut set order, etc.

In many respects, the criticality measure is equivalent to the F-V measure and one purpose of this exercise has been to determine the conditions (if any) under which the criticality measure differs from F-V. If such conditions do exist, it is important to find out which measure produces more reasonable or meaningful results.

The SC measure like the B-P measure, is relevant for interval reliability analysis. However, it is better handled in the IMPORTANCE code, since enabling events can have different repair times. Furthermore, it is easy to manipulate the fault tree by creating dummy events to serve as initiating events, so that one can get a ranking of all events in the fault tree, by assuming these events to be enabling events.

Fig. 3 shows a fault tree specially developed to study the three importance measures (i.e., F-V, criticality and SC) at the appropriate level of detail. The fault tree is relatively simple, allowing easy review of the results. The fault tree represents various combinations of Boolean logic, with minimal cut sets containing one to five components. Event, E1, is present in six minimal cut sets. The variations in the combinations of component/event failure allow adequate testing of the features offered by the importance ranking methods. Table 4 lists the minimal cut sets derived upon reduction of the fault tree in Figure 3 using the SETS code. Table 5 gives the assumed failure rate and repair data for each event. Event, E1, appears in many cut sets of different orders but it has a low failure rate

and small repair time. There is also a high-order minimal cut set with events that are not present in any other cut sets.

The F-V measures (basic event and minimal cut sets) were evaluated using both the IMPORTANCE and the VALUE codes. The criticality and SC measures were calculated using IMPORTANCE. The results of the ranking are shown in Table 6.

All three measures practically give the same ranking when the probability of the top event is less than 0.1. However, as the probability of the top event increases, the criticality importance value for E18 decreases by comparison with the corresponding F-V value (see Table 7). This condition exists for components or events in series. From Equation 4, the criticality measure can be shown to be directly proportional to the Birnbaum measure. In our example there are two events in series which are the main contributors to system failure. Event 18 has a failure probability which is 100 times smaller than that of Event 17. The Birnbaum measure for component E18 decreases as the actual failure probability of component E17 increases. Hence, the criticality measure for component E18 decreases exponentially as the probability of the top event approaches unity. By comparison, the F-V measure is the ratio of the sum of the probabilities of the minimal cut sets containing component E18 to the probability of the top event. Hence, as top event probability approaches 1, the F-V ranking for component E18 tends to increase and approaches .01. The F-V measure is considered to be a better importance measure because it ranks both critical and non-critical components in the system and it is possible that a component can contribute to system failure without being critical.

As mentioned earlier, we have included dummy events to serve as initiating events in order to rank all events using the SC measure method. The order of ranking for the basic events is the same as that obtained with the F-V measure. The concept of interval reliability may be useful when a system is in the operational stage and one wishes to understand the sequence of events contributing to system failure. However, during the design stage, when insufficient data are usually available and enabling events are either difficult or impossible to identify, the SC measure of importance is not useful.

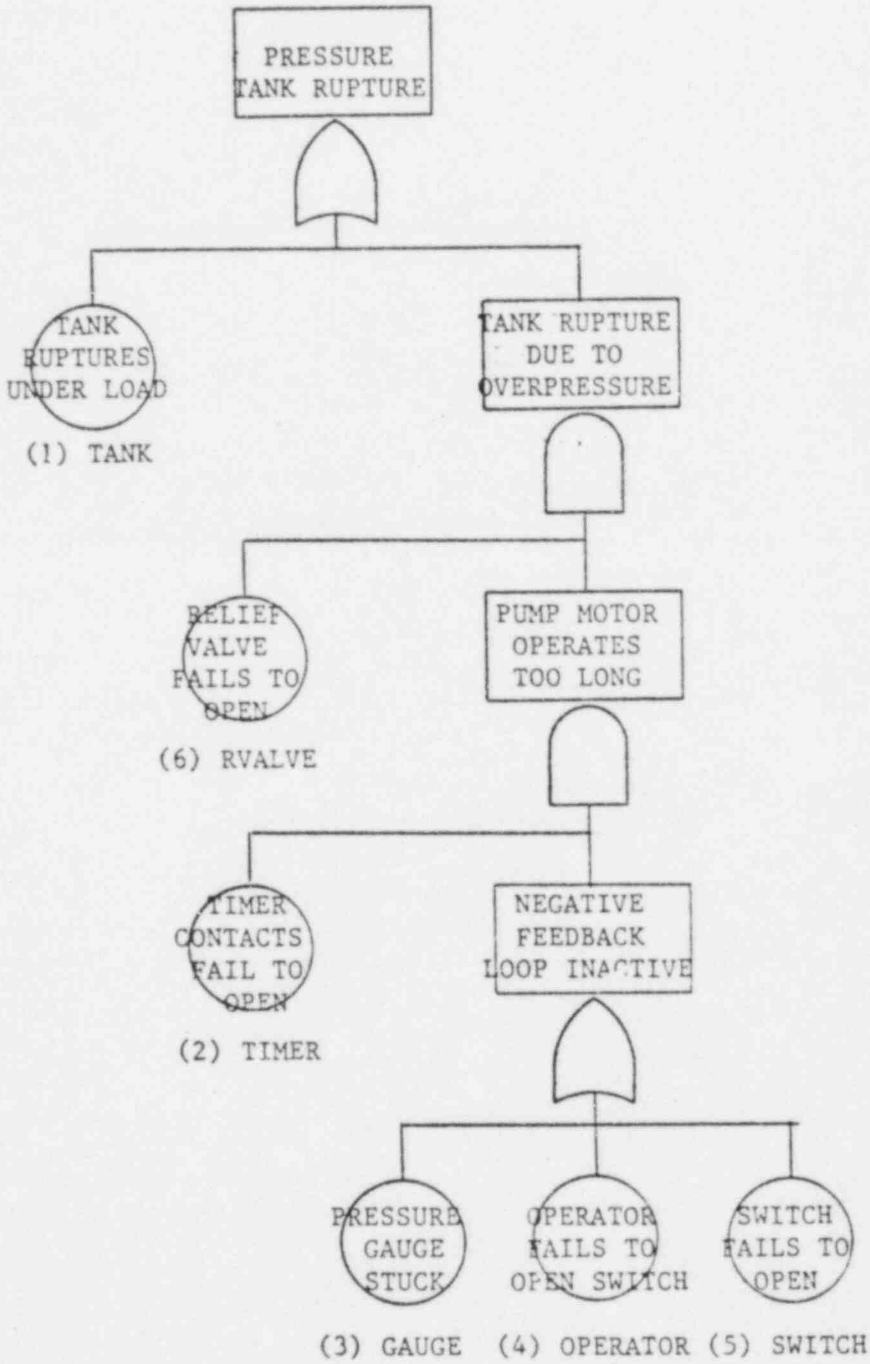

Fig. 2. Fault tree for pressure tank rupture.

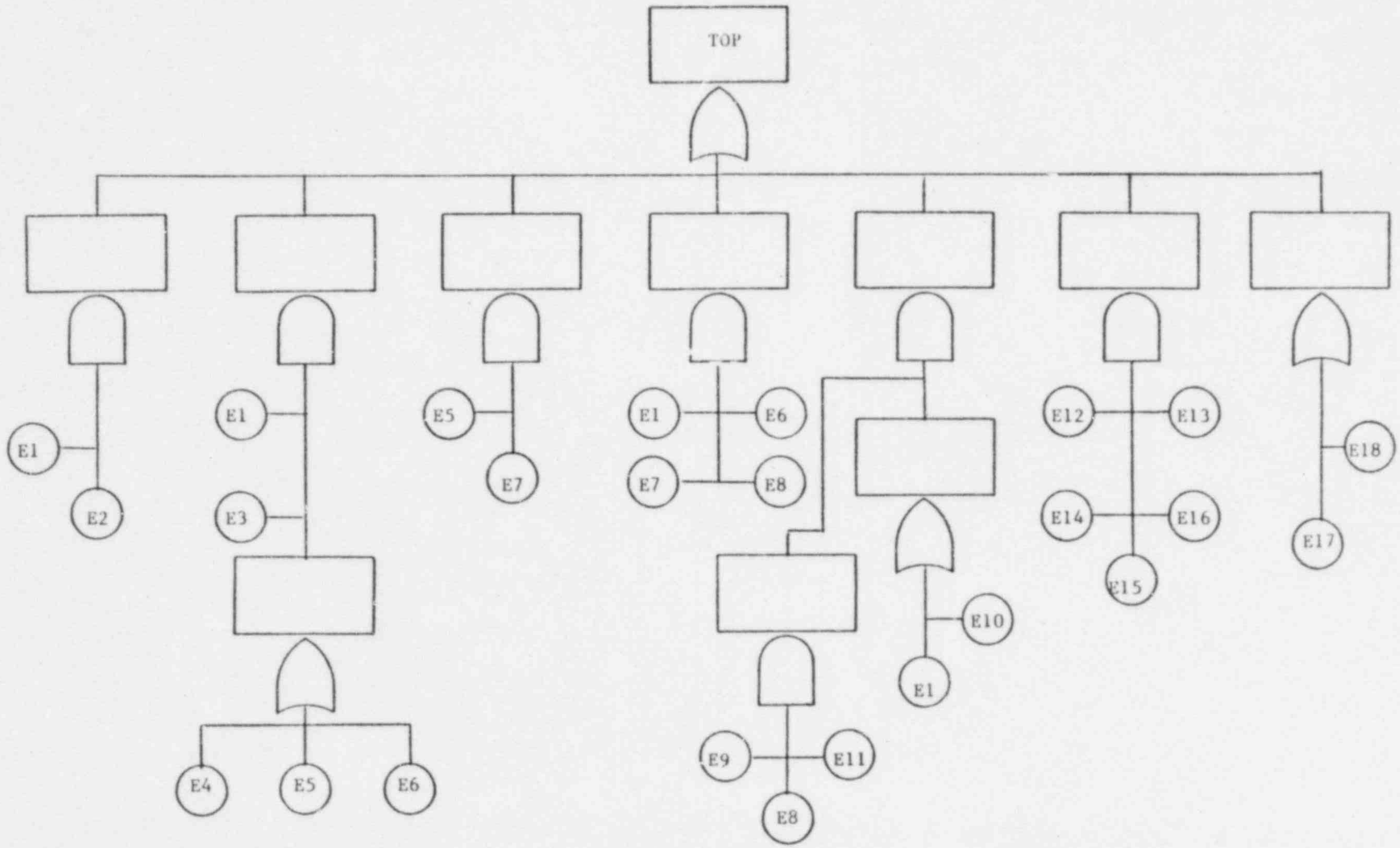

Fig. 3. Hypothetical fault tree for importance measure evaluations.

TABLE 1
IMPORTANCE MEASURE SELECTION CRITERIA

CRITERIA	BIRNBAUM'S	STRUCTURAL	CRITICALITY	UPGRADING FUNCTION	FUSSELL-VESELY[a]	BARLOW-PROSCHAN	SEQUENTIAL CONTRIBUTORY
1. Uses readily available repository data	yes	yes	yes	yes	yes	yes (to some extent)	yes (to some extent)
2. Provides useful insights on a repository system performance	yes	yes	yes	yes	yes	yes	yes
3. Easily applied to repository situations and provides scrutable results	yes	yes	yes	yes	yes	yes (to some extent)	yes (to some extent)
4. Provides numerical ranking	yes	yes	yes	yes	yes	yes	yes
5. Uniformly ranks all components (or events) represented in the fault tree model	no	no	yes	yes	yes	Initiating events only	Enabling events only
6. Applicable to repairable and non-repairable components	yes	yes	yes	Non-repairable only	yes	Repairable only	yes
7. Computer program available to calculate the rankings	yes	yes	yes	yes	yes	yes	yes

TABLE 2
IMPORTANCE MEASURES DEPENDENT ON SYSTEM UNAVAILABILITY(a)

BASIC EVENT	PROPORTIONAL HAZARDS	IMPORTANCE VALUE (RANK)(b)				
		BIRNBAUM'S	CRITICALITY(c)	STRUCTURAL	UPGRADING	FUSSELL-VESELY(c)
1. Tank	1.00-03	.99 (1)	.235-02 (5)	.670 (1)	.235-02 (5)	.790-03 (4)
2. Timer	1.00	.38 (2)	.88 (2)	.249 (2)	.868 (2)	.333 (1)
3. Gauge	10.0	.489-02 (5)	.103 (4)	.132 (3)	.916-01 (4)	.035 (3)
4. Operator	100.0	.492-02 (4)	.445 (3)	.132 (3)	.109 (3)	.149 (2)
5. Switch	100.0	.492-02 (4)	.445 (3)	.132 (3)	.109 (3)	.149 (2)
6. RValve	10.0	.468-01 (3)	.99 (1)	.249 (2)	.874 (1)	.333 (1)

(a) Based on Top Event Probability = 1.00-02
(b) Numbers in parentheses indicate numerical ranking of basic events.
(c) Normalized to 1.0.

TABLE 3
IMPORTANCE MEASURES DEPENDENT ON EXPECTED NUMBER OF SYSTEM FAILURES

BASIC EVENT	FAILURE RATE (PER HR)	REPAIR TIME (HRS)	IMPORTANCE VALUE (RANK)	
			BARLOW-PROSCHAN[c]	SEQUENTIAL CONTRIBUTORY
1. Tank [a]	.100-07	720.0	.156 (2)	
2. Timer [a]	.100-04	720.0	.844 (1)	
3. Gauge [b]	.100-03	22.0	-	.150 (3)
4. Operator [b]	.100-01	1.0	-	.681 (2)
5. Switch [b]	.100-04	22.0	-	.151-01 (4)
6. RValve [b]	.100-03	4320.0	-	.843 (1)

(a) Initiating events
(b) Enabling events
(c) Normalized to 1.0.

TABLE 4

MINIMAL CUT SETS FOR HYPOTHETICAL FAULT TREE MODEL

E17

E18

E1 * E2

E5 * E7

E1 * E3 * E4

E1 * E3 * E5

E1 * E3 * E6

E1 * E6 * E7 * E8

E1 * E8 * E9 * E11

E10 * E8 * E9 * E11

E12 * E13 * E14 * E15 * E16

TABLE 5
FAILURE RATE AND REPAIR TIME DATA

BASIC EVENT	NO REPAIR[a] FAILURE RATE (per hr)	WITH REPAIR[b] FAILURE RATE (per hr)	WITH REPAIR[b] REPAIR TIME (hrs)
E1	1.0-5	1.0-6	10
E2	2.4-2	1.0-3	24
E3	1.0-2	1.0-2	10
E4	1.0-3	1.0-4	1
E5	7.2-4	1.0-6	720
E6	7.2-5	1.0-7	720
E7	1.5-3	1.5-4	10
E8	1.8-2	1.8-3	10
E9	4.4-4	2.0-5	22
E10	1.5-3	1.5-4	10
E11	1.8-1	2.5-4	720
E12	7.2-3	1.0-5	720
E13	1.2-1	5.0-3	24
E14	1.6-3	2.0-4	8
E15	1.0-2	1.0-3	10
E16	1.0-4	1.0-5	10
E17	1.0-1	1.0-1	1.0
E18	1.0-3	1.0-3	1.0

(a) Input to calculate F-V and Criticality Measures as a function of unavailability.
(b) Input to calculate SC measure as a function of expected number of failures.

TABLE 6
IMPORTANCE MEASURE RESULTS

EVENT	FUSSELL-VESELY VALUE CODE[a]	FUSSELL-VESELY IMPORTANCE CODE[b]	CRITICALITY	SEQUENTIAL CONTRIBUTORY
E17	.990	.991	.990	.991
E18	.990-02	.991-02	.892-02	.991-02
E5	.107-04	.107-04	.962-05	.102-04
E7	.107-04	.107-04	.962-05	.102-04
E1	.238-05	.238-05	.214-05	.226-05
E2	.238-05	.238-05	.214-05	.226-05
E8	.213-07	.213-07	.192-07	.203-07
E11	.213-07	.213-07	.192-07	.203-07
E9	.213-07	.213-07	.192-07	.203-07
E10	.212-07	.212-07	.191-07	.201-07
E3	.177-08	.178-08	.160-08	.169-08
E4	.990-09	.991-09	.891-09	.942-09
E6	.715-10	.715-10	.643-10	.681-10
E12	.137-10	.137-10	.123-10	.130-10
E13	.137-10	.137-10	.123-10	.130-10
E14	.137-10	.137-10	.123-10	.130-10
E15	.137-10	.137-10	.123-10	.130-10
E16	.137-10	.137-10	.123-10	.130-10

(a) Normalized to one.
(b) Unnormalized.

TABLE 7

COMPARISON OF THE FUSSELL-VESELY

AND CRITICALITY MEASURES FOR VARYING TOP EVENT PROBABILITY

TOP EVENT PROBABILITY	BASIC EVENT	FUSSELL-VESELY	CRITICALITY
.0505			
	E17	.991	.990
	E18	.991-02	.941-02
.101			
	E17	.991	.990
	E18	.991-02	.892-02
.503			
	E17	.995	.990
	E18	.995-02	.498-02

4. RECOMMENDED MEASURE

From the results of our analyses, we conclude that the F-V measure is the most suitable method for ranking component importance under a variety of conditions (i.e., as a function of system unavailability, unreliability, or both). Furthermore, the F-V method provides ranking for both basic events and minimal cut sets. The case presented in Section 2.1.6, which showed that the B-P minimal cut set measure provided a better ranking than the F-V measure, involved components (or basic events) which were replicated in many cut sets and their failure probabilities were equal. Such condition is considered extreme and not likely to occur in real-life situations. The dominant accident sequences that will be evaluated in the preclosure risk assessment of a geologic repository will involve different systems with varying failure probabilities. Hence, the use of the F-V measure will, in general, provide importance rankings that would be both useful and meaningful in assessing the safety of repository systems and operations.

Although none of the computer codes for calculating importance measures included an analysis of the uncertainty in the failure rate and repair data, the inherent straightforwardness of the F-V equation easily permits the calculation of probability distributions for the importance measure. We also recommend the use of STADIC-2 to perform these calculations in the HLW-PSSA project (Ref. 9). This program is a validated tool for combining distributions according to an input algorithm. The F-V measures for both components and systems can then be evaluated in the form of distributions.

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ATTACHMENT

This appendix discusses the basis for using the minimum upper bound approximation for calculating top event failure probability as used in the IMPORTANCE and VALUE computer codes. The mathematical expressions presented here were derived primarily from Reference A-1.

If basic events are not replicated in cut sets and all basic events are statistically independent, then

$$P\text{ (top event)} = \coprod_{j=1}^{N_K} \prod_{i \epsilon K_j} q_i \qquad (1)$$

where $i\ K_j$ means "for all basic events contained in minimal cut set K_j"

N_K = total number of minimal cut sets representing the fault tree structure

$\coprod$ = the logical operator for an "OR" gate;

for a pair of events: $\coprod_{i=1}^{2} q_i = q_1 + q_2 - q_1 q_2$

Π = the logical operator for an "AND" gate;

for a pair of events: $\prod_{i=1}^{2} q_i = q_1 q_2$

q_i = failure probability of basic event i.

In general, basic events are replicated and Equation 1 is not valid. Esary and Proschan (Ref. A-2) proved that the following bounds always hold when the basic events are statistically independent:

$$\prod_{r=1}^{N_p} \coprod_{i \in P_r} q_i \leq P\text{ (top event)} \leq \coprod_{j=1}^{N_K} \prod_{i \in K_j} q_i \qquad (2)$$

where

$i \in P_r$ means "for all basic events contained in minimal path set P_r"

N_p = total number of minimal path sets representing the fault tree structure.

The term ($\prod_{r=1}^{N_p} \coprod_{i \in P_r} q_i$) is the minimal path set lower bound while the term ($\coprod_{j=1}^{N_K} \prod_{i \in K_j} q_i$) is the minimal cut set upper bound. In general, the upper bound is very close to the "exact" value when q_i's are small. The overprediction which occurs for $.1 \leq q \leq 1$ is acceptable for most engineering calculations.

The first order expansion of the minimal cut set upper bound is called the "rare event approximation". In this approximation we neglect the simultaneous occurrence of two cut sets. As a rule of thumb, the rare-event approximation is accurate when $q_i \lessapprox .01$. For example, for a two-out-of-three system, the minimal cut set upper bound is:

$$1 - (1-q_1q_2)\ (1 - q_1q_3)\ (1 - q_2q_3) \qquad (3)$$

The rare event approximation of Equation (3) is

$$q_1q_2 + q_1q_3 + q_2q_3 \qquad (4)$$

The IMPORTANCE and VALUE computer codes use the minimal cut set upper bound to approximate the top event probability. The same method of approximation can be used to calculate the probability of the union of

minimal cut sets containing basic event i. Thus, the Fussell-Vesely equation given as

$$I^{FV} = \frac{P\left(\bigcup_{K=1}^{J} (MCS)_k \right)}{P_T} \qquad (5)$$

can be approximated by

$$I_i^{FV} \cong \frac{\sum_{K=1}^{j} P(MCS)}{P_T} \qquad (6)$$

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A-2. J. D. Esary and F. Proschan, "Coherent Structures with Non-identical Components," Technometrics, 5, 191 (1963).

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The major effort for this project has been on the gathering, organizing, and assembling of information pertinent to the safety assessment of a nuclear waste repository during preclosure operations. Specific issues addressed in this report are:

1. Detailed analysis of a conceptual basalt repository design in order to identify potential initiating events/accident scenarios capable of causing radiological and/or nonradiological consequences.
2. Evaluation of radiological and nonradiological consequences relevant to a nuclear repository and recommendation of an approach for quantitative evaluation of these consequences.
3. Comparative evaluation of several importance ranking measures that had been used in the nuclear industry in order to select a measure to best meet the needs of the program.
4. Development of event and fault tree models for those initiating events which have passed the preliminary screening process.
5. Compilation of specific data such as initiating event frequencies, component/system failure rates and repair times, personnel injury, and basic information necessary for more detailed radiological consequence evaluations at a later time.
6. Selection of a set of accident scenarios to be quantified in the next study phase to demonstrate the applicability of the proposed methodology that will identify and quantitatively prioritize structures, components, systems, and operations which are important to safety during the preclosure phase of a HLW rep.

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