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PRELIMINARY ENVIRONMENTAL ANALYSIS OF A GENERIC FUEL REPROCESSING FACILITY

U.S. ENVIRONMENTAL PROTECTION AGENCY

Office of Radiation Programs

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OFFICE OF RADIATION PROGRAMS

ENVIRONMENTAL PROTECTION AGENCY

TASK ORDER NO. 68-01-1121

PRELIMINARY

ENVIRONMENTAL ANALYSIS OF A GENERIC FUEL

REPROCESSING FACILITY

R. Cooperstein, R. C. Erdmann, R. R. Fullwood

SAI Services

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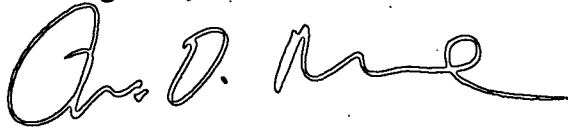
FOREWORD

The Office of Radiation Programs carries out a national program designed to evaluate the exposure of man to ionizing and nonionizing radiation, and to promote the development of controls necessary to protect the public health and safety and assure environmental quality.

Office of Radiation Programs technical reports allow comprehensive and rapid publishing of the results of intramural and contract projects. The reports are distributed to groups who have known interests in this type of information such as the Nuclear Regulatory Commission, the Energy Research & Development Administration, and State radiation control agencies. These reports are also provided to the National Technical Information Service in order that they may be readily available to the scientific community and to the public.

Comments on this analysis as well as any new information would be welcomed; they may be sent to the Director, Technology Assessment Division (AW-559), Office of Radiation Programs, U.S. Environmental Protection Agency, Washington, D.C. 20460.

W. D. Rowe, Ph.D.
Deputy Assistant Administrator
for Radiation Programs

PREFACE

Nuclear power has become a principal option to satisfy the
national need for a clean, safe and reliable energy supply.
As a result, the generation of light-water-cooled nuclear
power reactors, using enriched uranium fuel, is experiencing
rapid growth. This increase in nuclear power reactors will
require similar growth in the associated aspects of the fuel
cycle such as mining and milling of uranium ore, production
of nuclear fuel material, manufacture of fuel elements,
shipping, reprocessing of spent fuel elements and waste
disposal activities. To date, the controlled and accidental
releases of relatively small amounts of radioactivity from
operating nuclear power and reprocessing plants have been
maintained well below specified limits. However, these
operations may impinge to a greater extent on the environ-
ment as a result of their anticipated growth.

Projections of the Civilian Nuclear Power Program indicate
that the nuclear economy will expand to about 153 gigawatts
by 1980 and to about 735 gigawatts by the year 2000. Economic
analyses by the AEC and by commercial investors have con-
cluded that generation of electric power by nuclear plants
requires reprocessing of spent fuel to recover residual
fissile materials for re-use in new fuel elements. Approx-
imate total fuel reprocessing rates in metric tons per year
could reach 3500 in 1980 and increase to 15,000 in the year
2000. The total radioactivity due to beta emitters in the
accumulated wastes will increase from 210 megacuries (1970)
to 18,800 megacuries in 1980 and to 209,000 megacuries in
2000.

The number of reprocessing plants that will be required is a function of the individual plant designs assumed and the amount of spent fuel to be processed. It is assumed, for this study, that one aqueous separations plant will be required for each 1500 metric tons of LWR fuel. A plant of this size should be capable of processing fuel from about 50 power reactors since each typical 1000 MWe LWR will discharge approximately 30 metric tons of fuel each year.

Reprocessing involves destroying the integrity of the spent fuel elements to permit dissolution and separation of the fuel from its metal cladding prior to chemical separation of the useful fissile materials from waste products by some adaptation of the Purex solvent extraction process. Destruction of the integrity of the fuel elements which had been maintained through the cycle in the reactor represents the main source of radioactivity from the nuclear power industry which could potentially enter the environment.

The Environmental Protection Agency, whose charter is to assure protection of the environment by systematic abatement and control of pollution, sponsored a program through its Office of Radiation Program to perform an environmental analysis study of a generic fuel reprocessing facility in order to project what effects accidents, in such a facility, of potential environmental risk significance, may have on the public health and safety.

This report discusses the probabilities and consequences of hypothetical but credible accidents that could occur in the operation of a generic LWR fuel reprocessing plant which could have potential environmental impacts.

In preparing this report technical data was obtained from numerous sources, nevertheless as might be expected for an analysis of this type, "hard data" were not available in most cases and the authors were required to assume "best judgment values." The limitations which this type of approach places on these data and analysis should be recognized. However, it is our feeling that the methodology and approach used in this report are applicable to environmental analyses at fuel reprocessing plants, and that the information presented despite its limitations is the best available at this time.

SUMMARY

A generic reprocessing plant for light-water-reactor spent
fuel utilizing the Purex process has been synthesized from
a review of existing plants and those under construction.
This model is used to develop a quantitative description
of the probability of occurrence of a spectrum of poten-
tially credible accidents and resultant radioactive releases
during its operation. The results can be used to determine
the potential impact on the environment.

It is concluded from evaluation of the generic design, sys-
tems and components that the most probable off-site release
pathway that could endanger the public health and safety
would be via airborne releases through the ventilation
system in the event of accidents.

The postulated credible accidents considered include explo-
sions during various unit operations involving different
sources of radioactivity, fires, criticality, loss of cool-
ing to the high level liquid waste facility and accidents
which could derive from the occurrence of natural events
such as earthquakes, tornadoes and floods.

Accidents which might occur during normal operation were
emphasized over those that might occur during shutdown or
neptunium batch processing conditions. In addition, no
releases during decommissioning or start-up operations were
analyzed.

If, during an accident large releases of noble gases or
tritium were noted, these were not assessed because they

would normally be released in reprocessing spent fuel. The
installation of noble gas collectors, to minimize krypton
releases for example, were examined briefly. While such
systems exist in the laboratory, production-scale systems
are presently not available. Therefore the risk tradeoff
between continual release of such gases and possible acci-
dental release of stored quantities of such gases, after
some period of radioactive decay, was not conducted.

Hypothetical accident probabilities are estimated by fault
tree analysis of the model plant's safety and confinement
systems during operation. The expected responses to hypo-
thesized operating transients and postulated accidents,
including natural events, are evaluated. Realistic assump-
tions based upon existing process technology and experience
are used in the evaluations to determine the consequent
radioactive releases.

If a processing modification is incorporated into the
reprocessing cycle it will be a relatively simple matter to
estimate the likelihood of releases from such a change for
comparison with those documented herein. Thus, a measure
of the level of acceptability of a processing change from a
release or safety viewpoint is evident from this work.

In such a generic analysis, it is not possible to make a de-
tailed account of the operational and design data which
would be pertinent to an in-depth safety analysis. We have
attempted, where possible, to utilize existing failure data
supplementing this with estimates of how such an incident
during processing might take place. Our considerations
included types of instrumentation used, human behavior and

designs. Thus, our results may be suspect in that errors in design, etc., may have been overlooked. Barring these anomalies, the accuracy of our results may vary within a factor of 10 or less from the true situation.

The ventilation system (scrubbers, HEPA filters) played a very prominent role in decreasing the consequences of many of the accidents releasing nonvolatile species. If such releases are determined to be excessive, one should consider installation of additional release mitigating equipment as a positive means of removing hazardous airborne substances.

Since the spent fuel to be reprocessed has substantially less decay heat compared to when it is housed in a reactor, the concerns of decay heat removal at a reprocessing plant, while real are not severe. Interruptions of power for effecting the latter, can be more readily tolerated. Even so, auxiliary power generation equipment is available to provide plant power during emergencies.

Based upon fault tree analysis and consequence calculations, consequence/likelihood plots are drawn for selected isotopes including ruthenium, iodine, plutonium, and other representative actinides and fission products.

By selecting the dominant accidents from these plots and by applying a simplified meteorological case, the dose in rems as a function of distance for a number of pertinent isotopes are plotted utilizing likelihood as a parameter. These data permit quantification of accident impact on the environment for a generic reprocessing plant.

It is recognized by EPA that this report presents only an initial analysis of the potential environmental impact of accidents at a generic LWR fuel reprocessing facility. There remain further analyses which could be completed, using this technique, for both a generic facility and for any specific facility design sited at a particular location. Extensions of the current study to derive population dose estimates and predictions of the health effects which could result from these exposures are examples of two such possible additional studies.

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I.   INTRODUCTION

As of June 1, 1973, there were 33 operating reactors, 56 being built and 80 additional reactors on order. Based on nuclear power projections of 300,000 MWe by 1985, additional fuel reprocessing capability is needed to provide for the recovery of fissile material remaining in spent fuel elements[1]. The two existing fuel reprocessing plants, Nuclear Fuel Services and the Midwest Fuel Recovery Plant, have combined capacities for reprocessing the spent fuel discharges from the equivalent of about 35 1000 MWe light water cooled power reactors (1050 MTU/year). The Barnwell Nuclear Fuel Plant, under construction, will have a capacity for reprocessing an additional 1500 MTU/year of spent LWR fuel bringing the total annual reprocessing capacity to only 2550 MTU/year. This will be insufficient capacity for the industry within the balance of this decade thereby requiring the construction of added capability to ensure smooth, economical and timely operation of the nuclear fuel cycle for the anticipated energy requirements.

In the operation of a nuclear power reactor, the buildup of fission products and the depletion of fissile material (U-235 and Pu) requires that for maximum utilization of the fuel, the reactor operator must periodically replace about one-third of the fuel elements and redistribute the remaining partially spent fuel elements throughout the reactor core. The discharged spent fuel elements still contain between one-third and one-fourth of the U-235 in the fuel prior to irradiation and part of the fissile Pu that was produced from U-238. The fuel reprocessing plant recovers the unused fissile material so that it can be recycled in reactor reload fuel. Reprocessing also permits separation and removal

of the fission products from the fissile material for conversion into an acceptable form for long term isolation from the biosphere.

The sequential process of transforming uranium ore into usable fuel for nuclear power reactors and the operations to recover unused values of uranium as well as the plutonium and other desirable isotopes produced during irradiation in the reactors constitute the "nuclear fuel cycle"[2]. These operations are generally performed at separate installations in various parts of the country, depending for the most part upon the economics of transportation. The specific components comprising the LWR supporting fuel cycle are shown in Figure I-1, page 3, and include the following:

a) Mining uranium ore

b) Milling and refining ore to produce uranium concentrates (U_3O_8)

c) Production of uranium hexafluoride (UF_6) from uranium concentrates to provide feed for isotopic enrichment

d) Isotopic enrichment of uranium hexafluoride to attain reactor enrichment requirements using the gaseous diffusion process

e) Fabrication of nuclear reactor fuel including: converting UF_6 to uranium dioxide (UO_2), pelletizing, encapsulating in rods and assembling fuel elements

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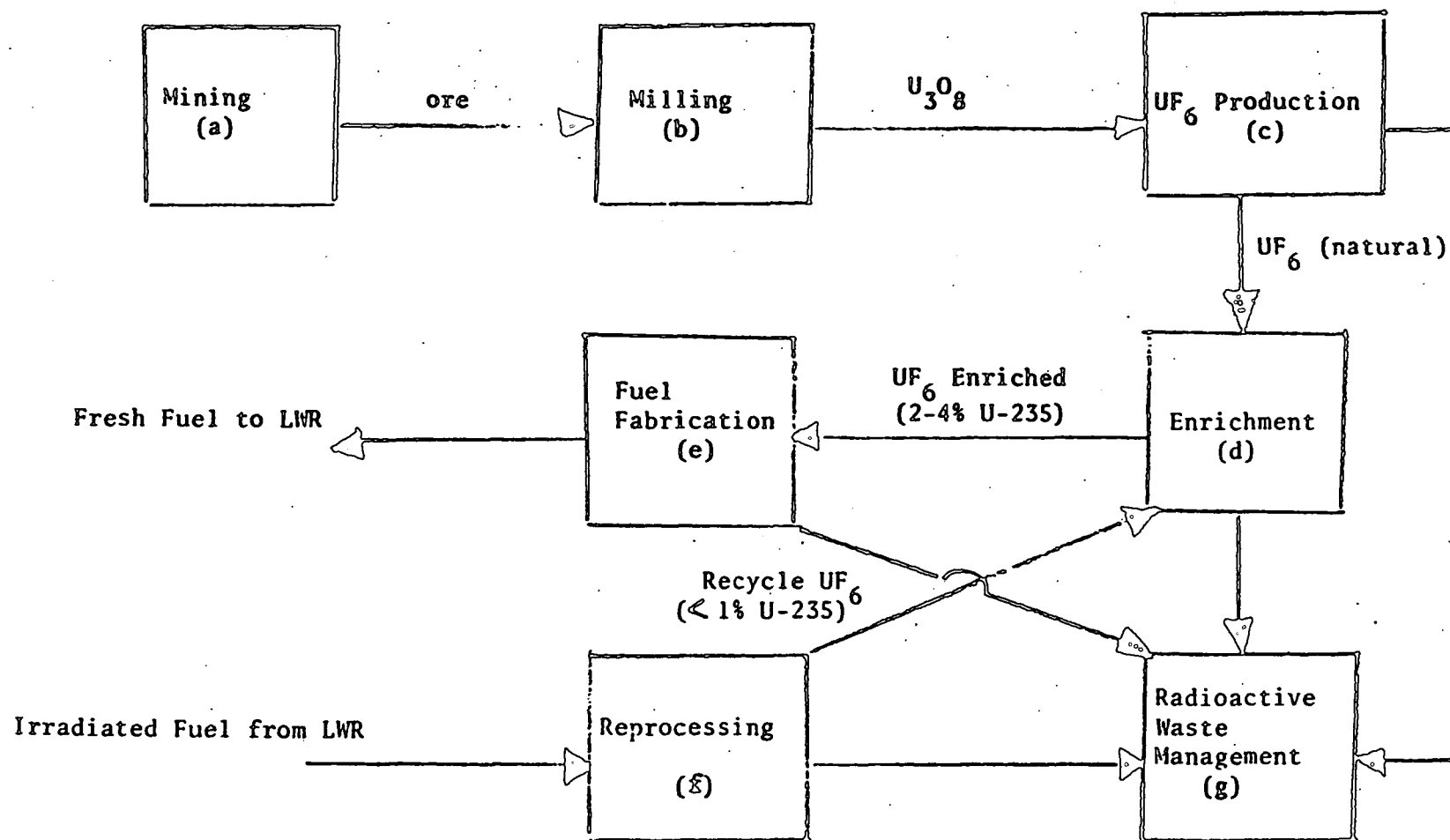
FIGURE I-1

Nuclear Fuel Cycle - Light Water Reactors

Uranium Dioxide Fueled - No Plutonium Recycle

f) Reprocessing irradiated fuel and converting uranium
to UF_6 for recycle through the gaseous diffusion
plant for re-enrichment

g) Radioactive waste management of high level and
other than high level wastes, including long-term
storage of wastes

h) Transportation activities associated with moving
materials to and from each of the above operations.

Uranium milling and refining (beneficiation of raw uranium
ore into U_3O_8) is usually done near the mines to avoid the
cost penalty involved in shipping the comparatively low
value ore over long distances. Uranium ore is refined into
U_3O_8 for shipment in drums to a "conversion" plant where it
is converted into uranium hexafluoride (UF_6). Uranium in the
UF_6 gaseous state is required as feed material for subsequent
"enrichment" in the gaseous diffusion process.

Uranium hexafluoride ·· , although refined as to total
uranium content, still contains less than 1% of the fis-
sionable isotope U-235 after "conversion". To be suitable
for fabrication into fuel elements for modern power reactors,
it must be "enriched" to approximately 2% to 5% U-235
(depending upon the specific design requirements of individual
reactors). Thus, UF_6 is fed into enrichment plants where
the U-235 isotope is upgraded to the required content.

After enrichment, the gaseous UF_6 is converted into a
metallic oxide (UO_2) for fabrication into fuel elements

which, together with control rods, structure and moderating components, form the nuclear reactor core.

In the reactor, fuel elements initiate and sustain the controlled fissioning "chain reaction" which produces vast quantities of heat necessary to generate electricity via the steam-powered turbine generators. On the average, one pound of slightly enriched uranium produces approximately 3.2 million kilowatt hours (KWH) of electricity. This compares with about 2.2 million pounds of coal required to generate the equivalent electricity.

During irradiation in the reactor, various fission products are created which tend to lower reactivity over time through absorption of neutrons. Excessive accumulation of fission products and burn-up of U-235 impair the chain reaction and eventually shut down the reactor. Irradiated fuel, therefore, must be replaced periodically after being only partially consumed. As the fuel is irradiated in the reactor core, part of the fertile uranium isotope U-238 is converted to plutonium, a portion of which undergoes fission thereby contributing to the reactor's heat output. The remaining plutonium and other fission products stay intact and become potential byproducts and waste residues.

Economic considerations favor the recovery of the fissile material remaining in the spent fuel elements. The net value of this residual fissile material, after allowing for the costs associated with reprocessing, waste disposal and related transportation services, amounts to about $50,000 per metric ton of irradiated fuel. Thus, during each year a 1500 MTU capacity reprocessing plant is operated at full capacity, it will reclaim fissile material having a net

worth of about $75,000,000. Moreover, by recovery of fissile
material, such a plant will conserve natural resources equiv-
alent to about a million and one-half tons of uranium ore
each year[3].

Reprocessing accomplishes the objectives of:
  a. reclaiming the unused uranium and plutonium for
     subsequent recycling into replacement fuel
  b. extracting valuable isotopes such as neptunium
     and separating the waste fission products from the
     reusable fissile material
  c. concentrating the fission products and associated
     irradiated wastes to permit safer, less complicated
     handling for permanent storage and more economical
     storage that will result in a minimal impact on
     the environment.

This study is concerned with determining the probability
of accidents associated with a generic light-water reactor
irradiated fuel reprocessing plant and the consequences of
these accidents to the environment. This analysis will
permit a quantitative risk comparison with other parts of
the nuclear fuel cycle and to other risks accepted by
society. The study is limited to developing the accident
risk envelope for a generic reprocessing plant that could
impact on the environment. Fault tree analyses are used
for the accident probability predictions. Only those trees
needed in the development of the risk envelope are evaluated
although a complete set of trees is presented to serve as
an aid for future work in risk assessments of fuel repro-
cessing facilities.

II. DESCRIPTION OF THE GENERIC FUEL REPROCESSING PLANT AND CONTROLS

Public health and safety is a principal concern in reprocessing spent irradiated fuel to a similar extent as for nuclear power reactors. In fact, the reprocessing plant may be a greater source of radioactivity in effluents than a reactor[4]. Thus, adequate safety margins are included in the design of reprocessing plants to prevent accidents and to assure that acceptable protection systems will function reliably to mitigate the consequences of accidents, if they should occur, because of multiple system failures or noncompliance with procedures provided to prevent accidents[5-7].

The function of a fuel reprocessing plant is to recover the residual fuel materials, uranium and plutonium, in a form suitable for reuse and to isolate radioactive wastes for storage and ultimate permanent disposal. The basic elements of reprocessing are illustrated in Figure II-1, page 8. This simplified flow diagram is generally applicable to any of the process techniques which have been applied for spent fuel reprocessing. The generic reprocessing technique for this study is the Purex process, a solvent extraction process, which has been in use for two decades in this country and is in use in other countries where spent fuels are processed[8].

1. Plant Description

The following assumptions are made regarding the generic plant:

- o the facility would be sited in conformance with AEC siting criteria as expressed in Part 100, Title 10 of the U.S. Code of Federal Regulations (10 CFR 100)[9].
- o sufficient water supply would be available for plant operation

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Spent Fuels
From Reactors

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┌─────────────────┐         ┌─────────────────┐         ┌─────────────────┐
│                 │         │                 │         │   Uranium       │
│ Fuel Receiving -│────────▶│   Chemical      │────────▶│   Plutonium   ─│──▶ Shipment
│                 │         │   Separation    │         │   Products      │
│ Feed Preparation│         │                 │         │                 │
└─────────────────┘         └─────────────────┘         └─────────────────┘
         │                           │
         │                           ▼
         │                  ┌─────────────────┐
         │                  │                 │
         └─────────────────▶│   Waste         │
                            │   Treatment     │
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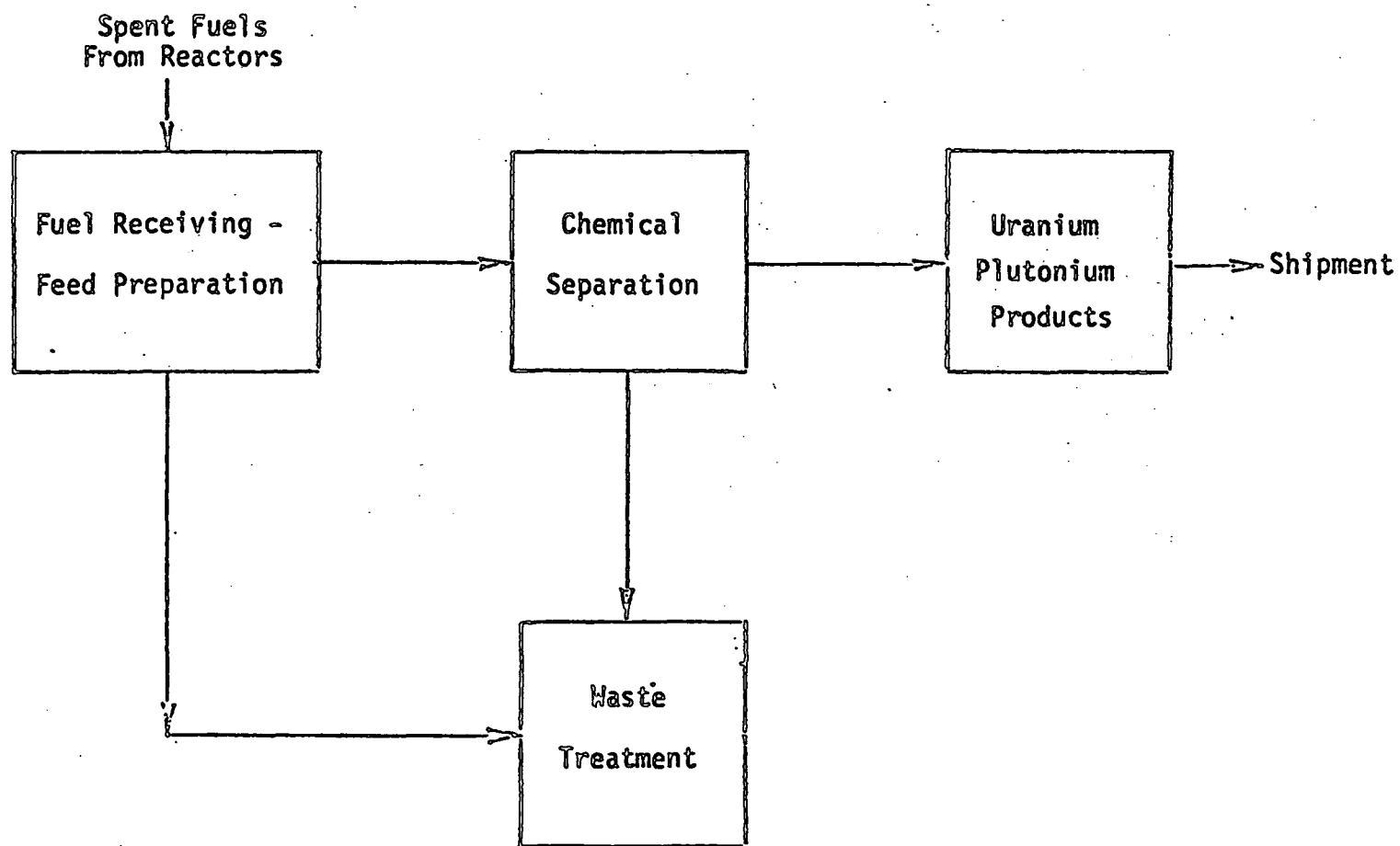
FIGURE II-1

Fuel Reprocessing - Generic Flow Diagram

o   it would have a processing rate of 5 metric tons
    of heavy metal (uranium and plutonium) per day of
    spent fuel from light-water power reactors.

o   the major facilities on the site would be:

    1. fuel receiving and storage facility
    2. main processing building housing the repro-
        cessing, product storage and waste solidifica-
        tion equipment
    3. radioactive-area-ventilation-air filtration
        and discharge system
    4. high level radioactive liquid waste storage
    5. offices
    6. warehouse and shops
    7. steam-generating plant
    8. cooling towers
    9. a retention basin
    10. product conversion facilities

All processing equipment and systems for processing irradiated
fuel elements, except for waste tank facilities, will be
housed in the process, fuel receiving, and storage station
building by the nature of the radiochemical operation. Because
of the radioactivity that must be handled, many of the process
systems must be heavily shielded; the equipment must be
operated, and some of it must be maintained from stations that
are remote from the equipment itself. Thus, as illustrated in
Figure II-2, page 10, the facility is likely to be a massive
concrete structure. This particular building was the proposed
Atlantic Richfield Reprocessing Center which has the basic
features of the Hanford and Barnwell Nuclear Fuel Plants[10].

The model plant for this study is synthesized from the features
of the three plants scheduled to be operational by the mid-
1970's. These are the Barnwell Nuclear Fuel Plant (BNFP), the
Nuclear Fuel Services Plant (NFS) and the Midwest Fuel Recovery
Plant (MFRP). Their respective reprocessing capacities are
1500, 750 and 300 metric tons per year of low-enrichment
uranium fuels irradiated in light-water power reactors[5-7].
The assumed model most closely resembles the BNFP because

Kold Chemical Preparation

Contact Cell Crane

Service Galleries

Contact Cell Process Equipment

Central Control Room

Observation Gallery

Exhaust Ventilation Duct

High Level Waste Storage

Analytical Caves

Remote Process Equipment

Hull Disposal Tunnel

Fuel Receiving

Cask Unloading Pool

Remote Mechanical Cell

Fuel Storage Pool

Fuel Assembly Transfer Area

FIGURE II-2

ARC PROCESS BUILDING

of its larger capacity, which is used for the purpose of
estimating the fractional releases of radioactive materials
for the accidents considered. This choice is based on
the belief that future plants will be designed for such a
capacity to effect low unit processing costs. High capacity,
small volume equipment will be used to minimize plant inventory
of both reactor fuel and process reagents thereby ensuring a
greater degree of overall safety and economy. Future plants
may have different characteristics from those used in this
study; however, it is expected that the derived quantities
of radionuclides that could be released in potential credible
accidents will remain unchanged or decrease as a result of
advancing technology[11].

2.   Comparison of Potential Hazards in a Reprocessing
          Plant and a Nuclear Power Plant

Performance criteria for engineered safety in reprocessing
plants are based upon those proposed for nuclear reactors
(Appendix A, 10 CFR 50; Code of Federal Regulations[9])
although the function and design of reprocessing plants
are significantly different. Potential hazards in a
separations facility differ considerably from those antici-
pated in power reactors due to the specific conditions
found in the reprocessing plant. Examples of these dif-
ferences follow:

o   The potential for a nuclear criticality is very low in a
    reprocessing plant albeit fissionable isotopes are present
    in quantity and are separated and purified in the course
    of operations. The use of soluble and fixed poisons,
    favorable geometry, concentration control and mass control
    mitigates the possibility of nuclear criticality where a
    nuclear chain reaction could take place.

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o Fuel reprocessing operations are generally of such a
  nature that the rate of approach of critical parameters
  (concentration, temperature, acidity, etc.) to prescribed
  upper limits will be relatively slow compared to power
  reactors. Although reliable instrumentation and control
  provisions are required, very rapid response is not
  necessary. The effect of exceeding prescribed operating
  limits does not usually present an immediate hazard.

o The cladding, which serves as the primary barrier to
  fission product escape from fuel in a reactor, must be
  breached in the reprocessing operation to recover the
  fissile materials. The potential hazard from having
  mobile radioactive materials in plant process systems,
  however, is relatively low since the systems do not
  contain large amounts of stored energy (like power
  reactor primary coolant systems) which could provide
  the driving force of radioactivity dispersal.

o In some instances, plant process streams will be corrosive.
  System failure due to corrosion with subsequent radio-
  nuclide releases may not be severe for the streams are
  doubly contained for leak tightness and operate at low
  pressures.

o Flammable and/or chemically reactive materials are used
  in fuel reprocessing operations. Well developed tech-
  nologies, however, are used for assuring that potential
  hazards from use of such materials are appropriately
  controlled.

o The potential hazards from loss of plant cooling capa-
  bility are low due to the lower stored energy levels of
  the limited fissile inventory in process which has

already undergone extended decay times. Emergency power
supply availability will enhance safety assurance by
providing positive off-gas release control and continuity
of process condition surveillance and safe shutdown
conditions. Continuous cooling is not critical although
necessary for high level wastes which are stored on an
interim basis. Cooling is used to avoid boiling in the
high level liquid wastes and/or overheating of the con-
tainment vessels. Overheating could compromise the
integrity of the storage tanks resulting in uncontrolled
releases of waste fission products to the environment.

o Fuel reprocessing operations are performed at low
   temperatures - limited to normal boiling points at
   atmospheric pressure - and at low pressures - limited
   by subatmospheric pressure maintained in the process
   system.

o Fission product wastes produced during irradiation of
   power reactor fuel are separated in the reprocessing
   plant and large quantities of these materials must be
   stored and controlled. Process operations are performed
   to reduce the volumes of solutions of such wastes to more
   manageable volumes which will not be released to local
   water courses. Alternatively, the wastes will be con-
   verted to low mobility forms (solids) for safe on-site
   retention until transferred to authorized permanent
   disposal facilities.

o Off-gas treatment processes and controlled effluent
   releases are required to assure that gaseous products
   which are not amenable to immobilization are not released
   from the plant at rates exceeding prescribed limits.

The Purex process basically involves solutions and as such makes the recovery of tritium especially difficult.

3.   Process Description

The overall process function is to recover the contained uranium and plutonium from spent fuel assemblies. The Purex solvent extraction process will be used to separate the latter from fission products. To accomplish the overall function, several processes must be used. They are described in the following and are illustrated by the simplified block diagram in Figure II-3, page 15, which is the particular flow diagram for the BNFP facility. For comparative purposes, the block diagrams for the MFRP and NFS processes are illus- trated in Figures II-4 and II-5, pages 16 and 17. Principal similarities and differences among the commercial plants and the assumed model are listed in Tables II-1 and II-2, pages 18 and 19.

o   **Fuel Receipt and Storage.** Irradiated fuel assemblies
    arrive at the reprocessing plant in shielded casks.
    These are monitored for outside contamination, cleaned,
    removed from the carriers and submerged in a pool of
    water for unloading the fuel assemblies. The cask is
    opened, the fuel assemblies removed and placed in storage
    canisters. The canisters are moved to a fuel storage
    pool where they are held until the fuel is scheduled for
    reprocessing.

o   **Mechanical Disassembly.** When scheduled for reprocessing,
    the fuel assembly is remotely transferred to a mechanical
    facility where it is sheared, as rods or as fuel bundles,
    into short lengths to expose the fuel to the dissolvent.
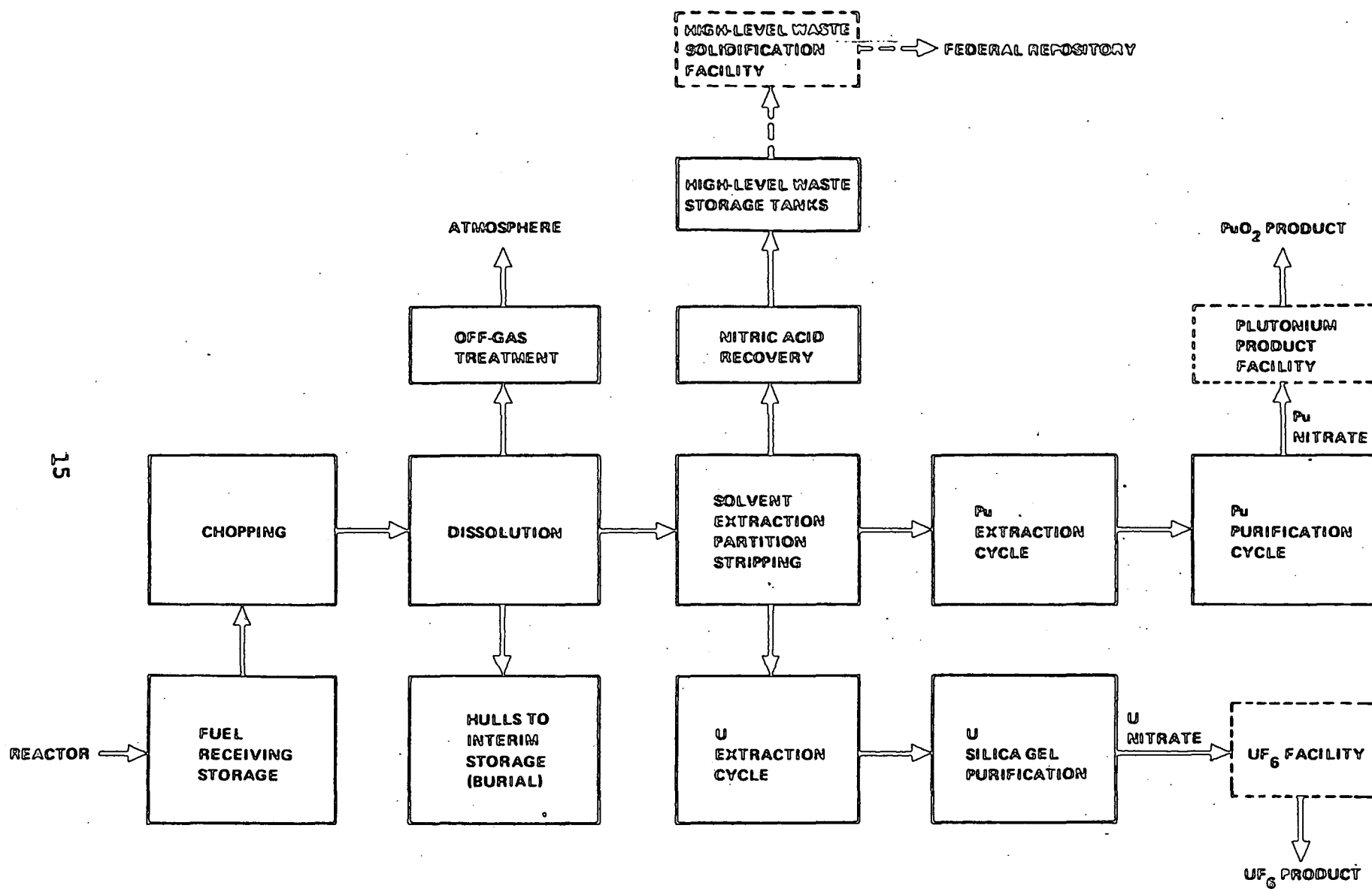
FIGURE II-3

Simplified Flow Diagram of BNFP Process[7]
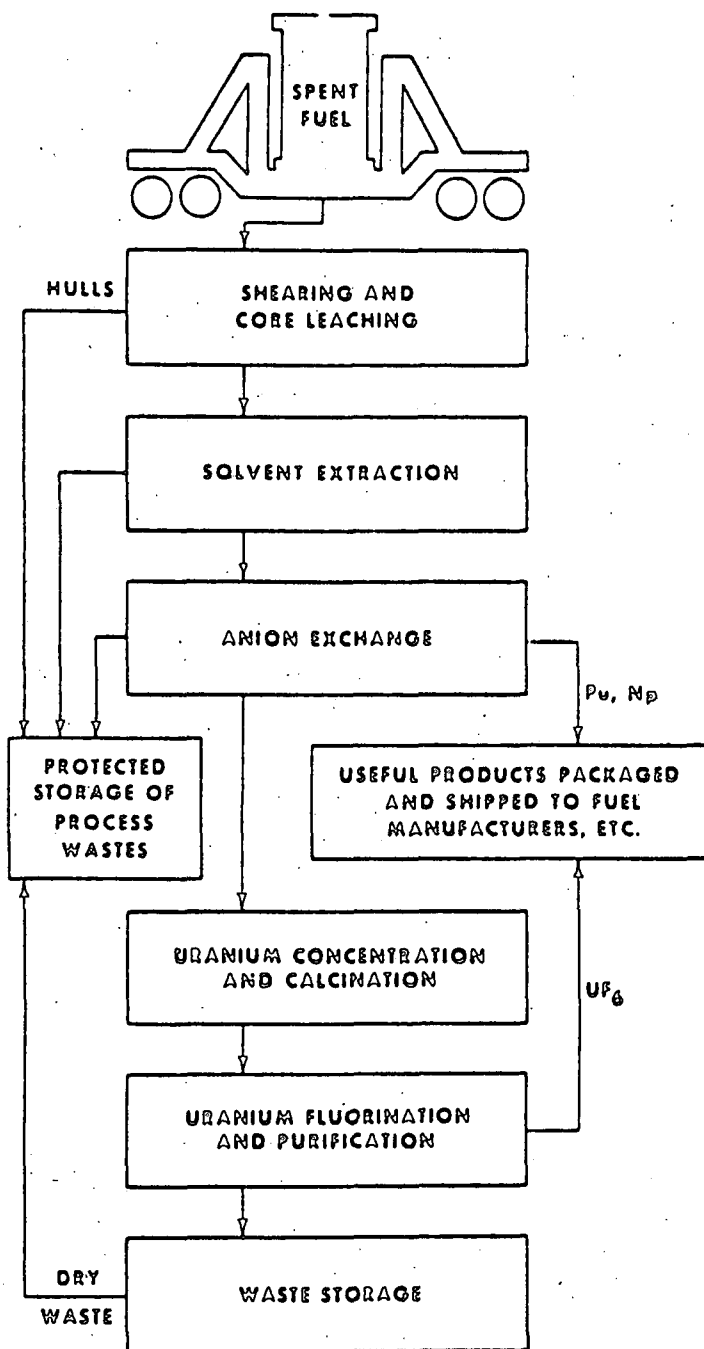
FIGURE II-4

MFRP   GE PROCESS [6]

General Electric's proposed Aqua-
fluor process makes use of aqueous
and fluoride volatility fuel recovery
technologies. The process uses
well-demonstrated fuel recovery
techniques including fuel shearing
and leaching, solvent extraction,
anion exchange, calcination, fluor-
ination, and uranium hexafluoride
distillation. Anion exchange is
used in the recovery and purifica-
tion of plutonium and neptunium.
Uranium will be converted to
the volatile uranium hexafluoride
(UF_6) and purified to make it
suitable for toll enrichment. The
high level liquid wastes will be
converted to solid form and stored
temporarily on-site in a waterfilled
concrete storage basin.

FIGURE II-5

NUCLEAR FUEL SERVICES, INC., SPENT FUEL PROCESS[5]
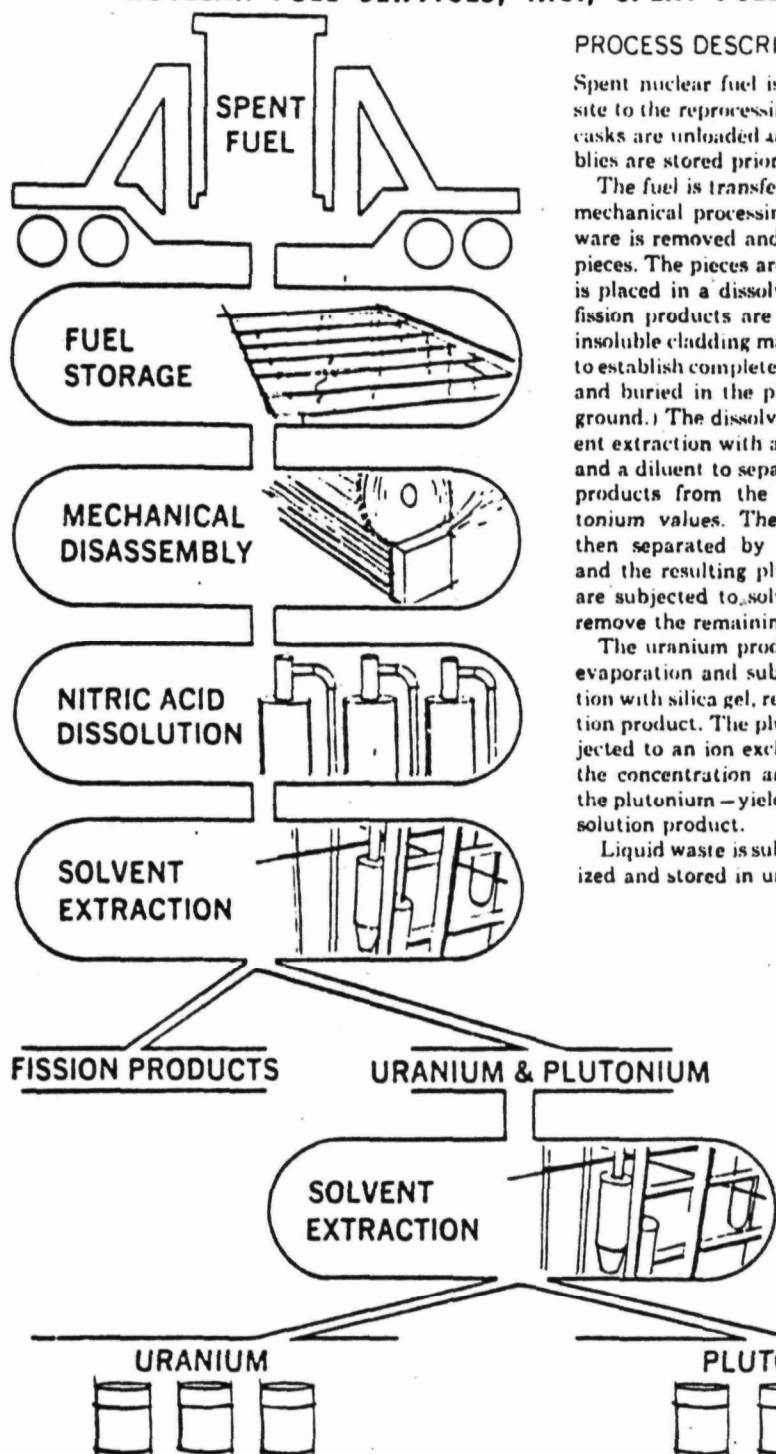
PROCESS DESCRIPTION

Spent nuclear fuel is transported from the reactor site to the reprocessing plant in shielded casks. The casks are unloaded underwater and the fuel assemblies are stored prior to reprocessing.

The fuel is transferred from the storage pool to a mechanical processing cell where end-fitting hardware is removed and the fuel is sheared into small pieces. The pieces are collected in a canister, which is placed in a dissolver, where the fuel values and fission products are dissolved in nitric acid. (The insoluble cladding materials—hulls--are monitored to establish completeness of dissolution of fuel values and buried in the plant's radioactive waste burial ground.) The dissolver solution is subjected to solvent extraction with a mixture of tributyl phosphate and a diluent to separate the majority of the fission products from the contained uranium and plutonium values. The plutonium and uranium are then separated by solvent extraction techniques and the resulting plutonium and uranium streams are subjected to solvent extraction purification to remove the remaining fission products.

The uranium product stream is concentrated by evaporation and subjected to a final decontamination with silica gel, resulting in a uranyl nitrate solution product. The plutonium product stream is subjected to an ion exchange treatment to effect both the concentration and further decontamination of the plutonium—yielding the final plutonium nitrate solution product.

Liquid waste is subjected to evaporation, neutralized and stored in underground liquid waste tanks.

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TABLE II-1
PRINCIPAL SIMILARITIES OF THE MODEL SEPARATIONS FACILITY
COMPARED TO OTHER COMMERCIAL REPROCESSING FACILITIES

Comparison Item	Model	BNFP	NFS	MFRP
Fuel Unloading and Storage	Underwater	Underwater	Underwater	Underwater
Headend Process	Chop-Leach	Chop-Leach	Chop-Leach	Chop-Leach
Stored Fuel Criticality Safety	Spacing	Spacing	Spacing	Spacing
Fuel Chopping	Mechanical Shear	Mechanical Shear	Mechanical Shear	Mechanical Shear
Fuel Dissolution Material	Nitric Acid	Nitric Acid	Nitric Acid	Nitric Acid
Fuel Dissolution Technique	Semicontinuous	Semicontinuous	Batch	Semicontinuous
Fuel Dissolution Equipment	Baskets in Dissolvers	Baskets in Dissolvers	Baskets in Dissolvers	(No Similarity) Leaching trough
Solvent Cleanup	Alternate Contact with Sodium Carbonate and Nitric Acid Solutions	Alternate Contact with Sodium Carbonate and Nitric Acid Solutions	Alternate Contact with Sodium Carbonate and Nitric Acid Solutions	Alternate Contact with Sodium Carbonate and Nitric Acid Solutions
Final Exhaust Filters	Roughing and HEPA	Roughing and HEPA	Deep Fiberglass and HEPA	(No Similarity) Sand Filter

TABLE II-2

PRINCIPAL DIFFERENCES OF THE MODEL SEPARATIONS FACILITY

COMPARED TO OTHER COMMERCIAL REPROCESSING PLANTS

Comparison Item	Model	BNFP Separations Facility	NFS	MFRP
Location	Compliance with 10 CFR 100	South Carolina	New York	Illinois
Design Capacity	5 MTU/day	5 MTU/day	3 MTU/day	1 MTU/day
Shearing	Entire fuel elements including end fittings	Entire fuel elements including end fittings	End fittings may be removed before shearing	Pins removed from fuel elements before shearing pins only
Criticality Control During Dissolution	Soluble poison	Soluble poison	Geometric limitations	Geometric limitations
Fuel Dissolution Technique	Semicontinuous	Semicontinuous	Batch	Semicontinuous
Fuel Dissolution Equipment	Baskets in Dissolvers	Baskets in Dissolvers	Baskets in Dissolvers	Vibratory Leacher Tray
HA Contactor	Centrifugal Contactor	Centrifugal Contactor	Pulse Column	Pulse Column
Partitioning	Ion Exchange	Electropulse column	Solvent extraction with chemical valence adjustment	Ion exchange
Interim High-Level Waste Storage Form	Acidic solution (1-5 molar)	Acidic solution (1-5 molar)	Acidic solution	Calcined solid
Iodine Removal from Process Off-gas	Mercuric Nitrate Iodine Scrubbers plus Iodine Silver Zeolite Adsorption Beds	Mercuric Nitrate Iodine Scrubbers plus Iodine Silver Zeolite Adsorption Beds	Mercuric nitrate scrubbers plus silver zeolite adsorption beds.	Sodium Hydroxide Scrubbers. Heated Silver Zeolite
Process Vent Filters	Roughing and HEPA	Roughing and HEPA	Multiple HEPA	Packed Fiberglass Filter
Final Exhaust Filters	Roughing and HEPA	Roughing and HEPA	Deep Bed Fiberglass plus HEPA	Sand Filter
Feed Clarification	Centrifuge	Centrifuge	None	None
Tritium Disposal	As vapor up stack	As vapor up stack	As water to creek	As vapor up stack
Uranium Product Form	Hexafluoride	Nitrate solution	Hexafluoride	Hexafluoride
Liquid Effluent	Noncontaminated	Noncontaminated	Contaminated	Noncontaminated

The fuel segments fall into or are fed into a dissolver
vessel.

Fuel Dissolution. The segmented fuel containing the
plutonium, neptunium and fission products formed during
irradiation, as well as unspent uranium is dissolved out
of the cladding hulls with nitric acid to form the feed
for subsequent remote liquid-liquid solvent extraction
steps. After dissolution, the undissolved cladding
hulls, made of zirconium, zirconium alloys or stainless
steel, are separated from the solution, rinsed, monitored
for residual fissile material and transferred to a pro-
tected interim vault storage area. Gases generated during
the dissolutions are channeled to an off-gas treatment
system. This system contains decontaminating units such
as condensers, scrubbers, chemical traps for iodine
removal and particulate filters which remove
radioactive gases and particulates other than inert gas,
e.g., Kr-85 and tritium, to levels below allowable
release limits before being exhausted to the atmosphere.
Nitrogen oxides formed during the dissolutions are also
retained by the ventilation treatment system to minimize
their release to the atmosphere.

o   Solvent Extraction. The chemically adjusted aqueous
    feed solution is then subjected to a Purex-type extrac-
    tion. It is contacted countercurrently in a centrifugal
    contactor with an organic solution of tributyl phosphate
    (TBP) dissolved in normal paraffin hydrocarbon diluent
    (dodecane). The organic solution preferentially extracts
    the tetravalent plutonium and hexavalent uranium, leaving
    about 95% of the fission products in the aqueous solution.
    The organic solution from the extraction passes through

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a scrubbing column where it is washed with additional
nitric acid solution. This step removes about 96% of
the residual extracted fission products from the organic
product solution. The wash solution is recycled back
to the centrifugal contactor. The aqueous solution
leaving the codecontamination cycle contains about 99.8%
of the fission products from the initially dissolved
solution. It is routed to a high level waste treatment
system where it is concentrated for protected interim
liquid waste storage and/or calcined to an immobile
solid form, loaded into specially designed containers
and transferred to protected storage facilities for
ultimate authorized off-site disposal. A simplified
block diagram illustrating the solvent extraction cycle
is shown in Figure II-6, page 22.

o **Product Separation.** Anion exchange or electrochemical
   reduction is used to partition plutonium and uranium
   into separate streams following the codecontamination
   step. In the former operation, solution from the solvent
   extraction system concentrator is fed through a cooler
   to a series of semicontinuous ion exchange contactors,
   in an ion-exchange cell, where the plutonium is sorbed
   on the resin as anionic $Pu(NO_3)^{--}$ and subsequently
   removed as a nitrate solution for concentration and
   loadout. Alternatively, the organic solution from the
   codecontamination step can be passed through a partitioning
   column where tetravalent plutonium is electrochemically
   reduced to the less extractable trivalent state. The
   reduced plutonium is then stripped into an aqueous nitric
   acid solution containing hydrazine as a holding reductant.
   The organic uranium containing solution is then stripped
   into acidified water. Electrochemical reduction and
   solvent extractions eliminate the need for chemical

Salting Agent
Scrub Solution

U, Pu Products
in Organic Phase

Back Extractant Solution

Organic To Treatment

Scrub
Stages

Back-
Extraction
Stages

Feed Solution
U, Pu, Fission
Products

Extraction
Stages

Organic
Extractant

Waste Solution
Salting Agents and
Fission Products

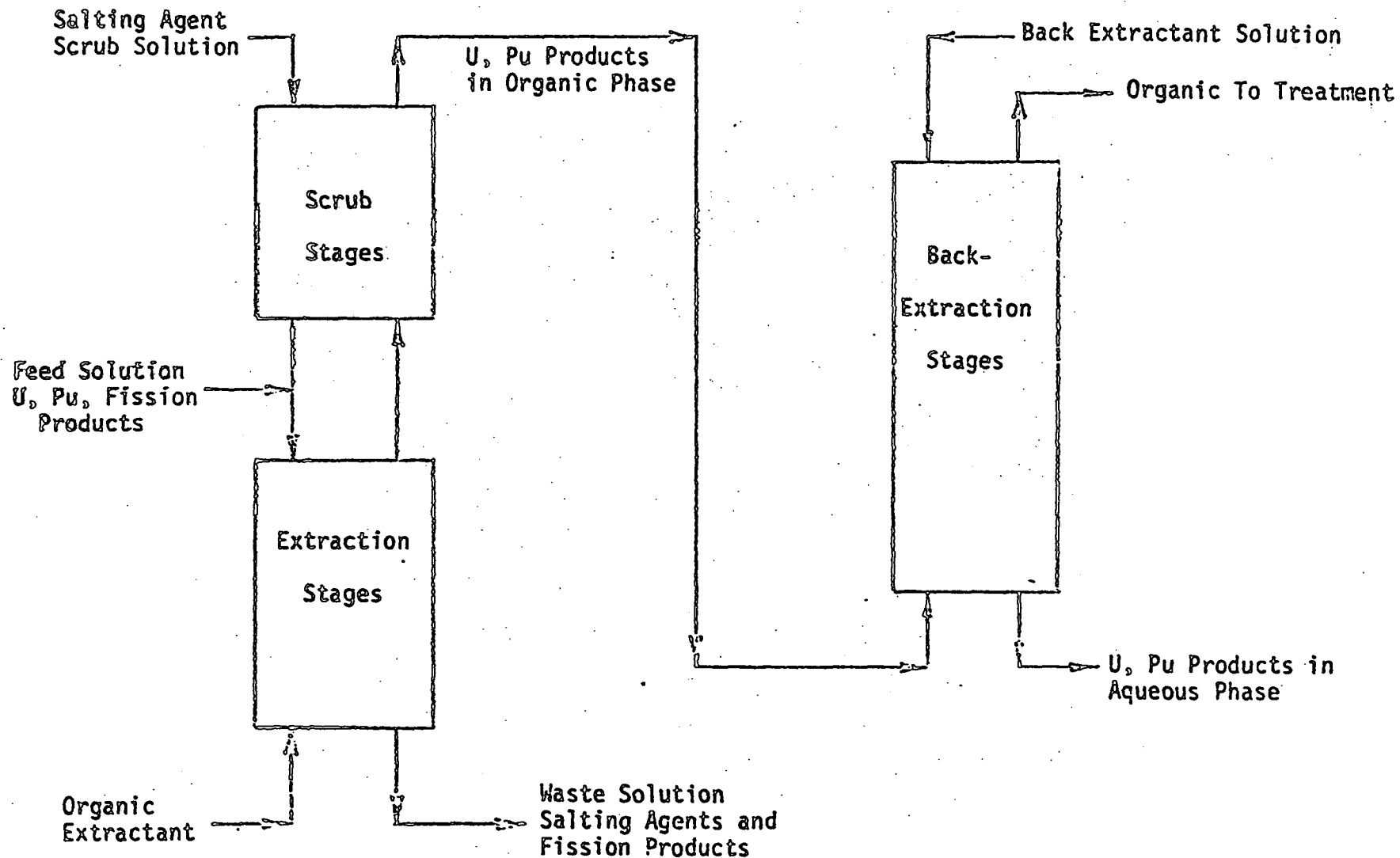U, Pu Products in
Aqueous Phase

22

FIGURE II-6

SOLVENT EXTRACTION CYCLE

additions for valence adjustment and the use of ion-exchange resin columns. The quantities of waste to be subsequently handled and disposed of, are also reduced. In this analysis, the former unit operation is reviewed for its accident potential.

o **Uranium Purification and Recovery.** The aqueous uranium strip solution is concentrated and its acidity is adjusted prior to resubjecting it to another partitioning cycle and filtration through a silica gel bed for final concentration loadout as uranyl nitrate solution. The operations remove additional residual fission products and plutonium further ensuring that the uranium solution meets product specifications for reuse in the fuel cycle. The nitrate solution is ultimately shipped as such or converted to UF_6 in an associated facility prior to shipment for reuse in the fuel cycle.

o **Plutonium Purification and Recovery.** Plutonium in the aqueous stream may be cyclically converted to anionic $Pu(NO_3)_6^{--}$ and sorbed on strong base anion exchange resin while the associated uranium and fission products are washed out in the raffinate. The sorbed plutonium from the last purification cycle is eluted as nitrate solution, concentrated and stored pending conversion to plutonium dioxide for use in plutonium recycle or breeder reaction fuel elements. Alternatively, the aqueous plutonium stream leaving the partitioning column may be reoxidized to the extractable tetravalent state and subjected to additional extraction-scrubbing sequences as described above to further decontaminate the product solution from fission products. The final plutonium nitrate solution, after concentration, is analyzed and stored in geometrically favorable tanks until it is converted in the Plutonium

Product Facility to the solid oxide form for storage and/or off-site shipment.

o   <u>Organic Solvent Systems</u>. The organic solvent waste streams from the decontamination and partition cycles are washed successively with dilute aqueous solutions of sodium carbonate, followed by nitric acid and neutralized by sodium carbonate to remove organic degradation products by extraction or precipitation. Precipitated solids are removed by filtration. As required, fresh TBP or diluent is added to maintain the TBP concentration and the total solvent inventory.

<u>Liquid Waste Treating and Storage</u>. The aqueous raffinate streams from the plutonium and uranium cycles are reprocessed for residual fissile material content by extraction into TBP organic solution which is recycled back to the decontamination cycle for recovery. The aqueous raffinate, essentially depleted of radioactive materials, is then concentrated in the low level waste concentrator. The radioactive waste streams from all the solvent extraction cycles are concentrated in the high-level or low-level waste concentrators to recover nitric acid and water for reuse in the process while reducing the waste volumes to be stored in appropriate waste facilities; miscellaneous waste streams containing salts, minimal fission products and no appreciable fissile material are acidified and concentrated. The concentrator bottoms are stored in appropriate waste tanks for ultimate disposal at a federal repository and the condensed over-heads are vaporized to the stack.

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o Process Off-Gas Treatment. In order to maintain near-
atmospheric internal working pressures, with very few
exceptions, all of the process equipment pieces -
vessels, extractors, condensers, etc. - are vented to
one or more process vent systems. The vent gases are
treated by scrubbing with circulating mercuric nitrate
solution to remove radioactive iodine, then treated in
an absorber to convert nitrogen oxides to nitric acid
suitable for recycling. The dissolver off-gas and
vessel off-gas streams are combined and passed through
a second decontamination train which includes an iodine
scrubber unit, iodine adsorber beds and a series of high
efficiency filter banks before being released to the
stack. Thus, the relatively small amounts of radio-
activity including most of the remaining radioiodine are
removed from the vent gases prior to discharge, via a
stack, to the atmosphere. The vent gases will probably
contain most of the tritium as a result of operations
in the acid recovery system. Virtually all of the
tritium from the original fuel follows the water phase,
as tritiated water, to the waste system and thence to
the process condensates derived during evaporation-acid
recovery operations. In early process designs, this
material was discharged to the environment following
evaporative and chemical treatments, for removal of
entrained non-volatile activity. Current design
approaches recycle process condensates in the process,
as process water, with evaporation of a small-volume
remainder of the condensate to discharge tritium to the
atmosphere via the off-gas system and the stack.

The gaseous wastes from a reprocessing plant include
a ventilation air, which is indirectly exposed to the
process, besides the process off-gases which are
directly exposed to the process and which contain some
volatile radionuclides. Both systems are shown by
Figure II-7, page 27. In this block diagram, the
process building is shown schematically as the large
block on the left. The building operating spaces are
divided into four zones of ventilation control, with
corresponding increasing levels of potential contamina-
tion by radioactivity, and increasing degrees of access
control requirements. As plotted on the diagram, the
building ventilation system is engineered to maintain
pressure differentials between zones such that the air
flow is always toward the zone of greater potential
contamination. The Zone 1 spaces, shown at a slightly
positive air pressure, include offices, lunchrooms, etc.,
and no special radiological control measures are required.
Examples of Zone 2 spaces include parts of the analytical
laboratories. The potential for radioactive contamination
is low, but controlled access is required. Zone 3 spaces
include plutonium product loadout spaces, other spaces
of the analytical laboratories, etc. Access is super-
vised. Zone 4 spaces are those which are expected to
be routinely contaminated, such as process cells. The
negative pressure in Zone 4 is typically an inch of
water.

A relatively large volume of air continuously flows
through the process building zones, through an efficient
filtration system, and to the atmosphere via a stack.
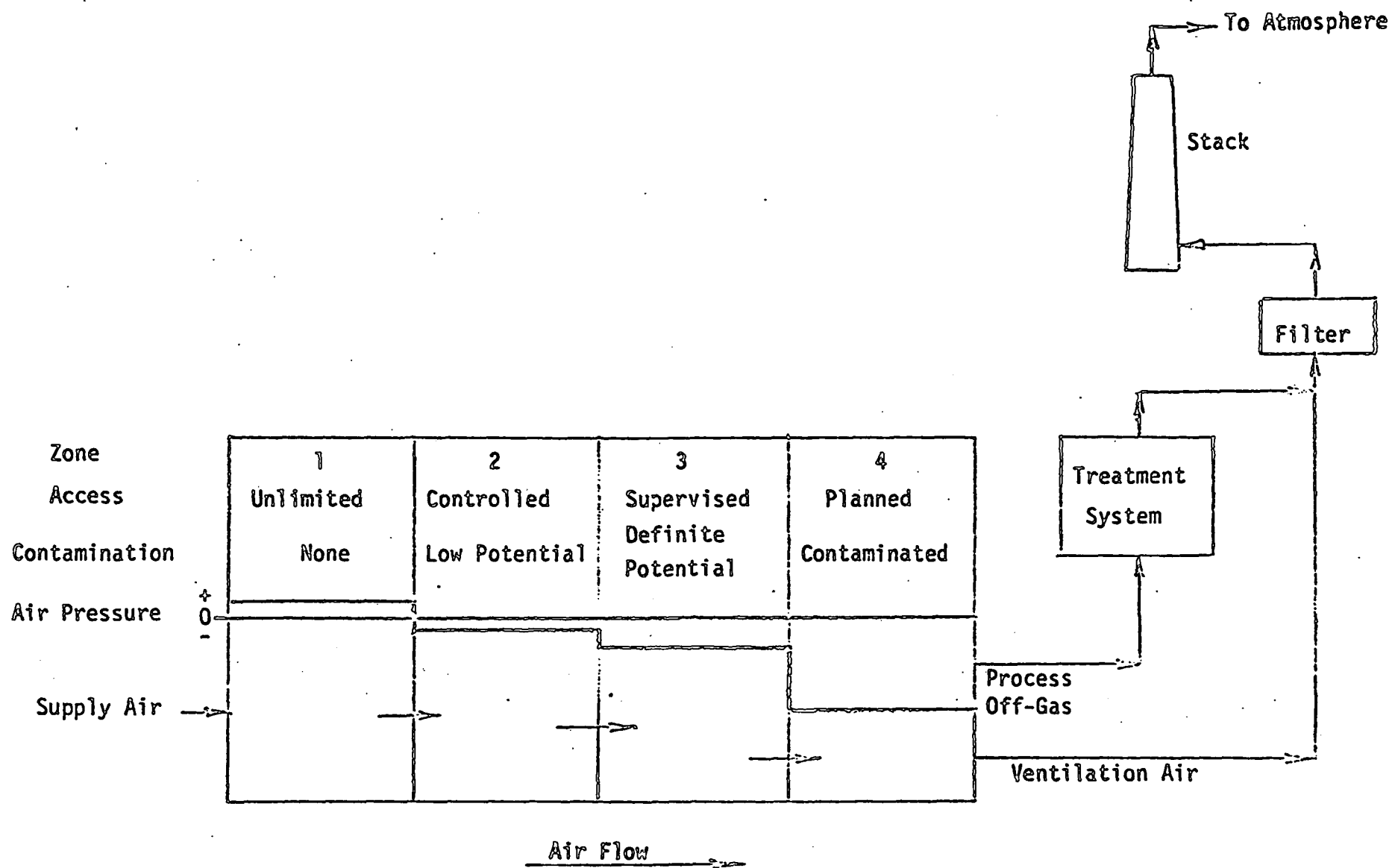
FIGURE II-7

OFF-GAS AND VENTILATION WASTES

With the exception of incoming fuel shipments and the transfer of solid scrap to burial, all radioactive material handled in the operation of the plant is located in an exclusion area. All processing equipment and systems for processing irradiated fuel elements, except for waste tank facilities, may be housed in the process and fuel receiving and storage station building as exemplified in the isometric drawing, Figure II-2, page 10. A summary of the process functions and chemical reactions involved in reprocessing is given in Appendix A.

4.   Radioactivity Confinement

The general concept of radioactivity confinement used in the generic plant analysis has been used for many years and has performed reliably in plants processing LWR fuels decayed for half a year or more.

The reprocessing facility design and process flow operations include multiple methods and systems to insure control of radioactive material so that in-plant and off-site exposures exceeding allowable limits, are not experienced throughout the life of the plant under normal operating conditions and routine maintenance. Confinement is based upon the use of one or more of the following:

   o Barriers to confine radioactive materials in the plant which include process piping and equipment, building structures, casks and storage tanks.

   o Channeling to confine the flow of fluid within fixed barriers or direct it through fixed barriers by pressure differentials or by gravity. Channeling is used to direct potentially contaminated fluids away from plant personnel and/or points of uncontrolled release toward a treatment system or point of controlled release.

o Treatment to remove radioactive material from fluids or gases discharged to the environment so that established limits are not exceeded.

o Burial to confine certain non-mobile radioactive solid wastes within the site boundaries.

The methods used, depending upon the mobility, quantity, type and intensity of the associated radioactivity for the unit operation involved, are shown in Table II-3, page 30.

Accidental radioactive releases, as measured by an increase in off-site radiation level, could follow three pathways: release to liquid effluents, release to ground water and release to the atmosphere. The confinement and ventilation systems in fuel reprocessing plants remove particulates of non-volatiles dispersed under accidental conditions and liquid releases to such an extent that off-site environmental risks are dominated by airborne releases of volatile and semi-volatile materials only. Therefore, the airborne release pathway was the major one considered for evaluations of the off-site environmental risks from potential accidents in the plant operations. Secondary emphasis was placed on liquid releases to the ground. A radioactive material flow diagram for a reprocessing plant, as exemplified by BNFP, is shown schematically in Figure II-8, page 31.

5.   Administrative Controls

The operation of a radiochemical separations plant is in most respects like the operation of any large chemical plant except for the complications introduced by the radioactive nature of the process materials. Working with radioactive materials necessitates adherence to extensive government regulations for their control. These are found

TABLE II-3

CONFINEMENT

| | Confinement Method Used | | | |
| | Barriers | | Channeling | Treatment |
	Single	Multiple		
a. Pool Water	X		X	X
b. Fuel Elements-Undamaged		X		
c. Fuel Elements-Damaged	X		X	X
Process				
d. Fluids		X	X	X
e. Solids	X		X	X
Product				
f. Uranium (Storage & Shipping)	X			
g. Plutonium & Neptunium		X	X	X
° h. Plutonium & Neptunium (Storage & Shipping)		X		
Waste				
i. High Level Liquid		X	X	X
° j. Intermediate Liquid		X	X	X
° k. Low Level Liquid with Tritium	X		X	X
l. Other Low Level Liquid	X		X	X
m. Solids (Hulls & Equipment) in cell	X		X	X
° n. Hulls during Transport		X		
° o. Contaminated Equipment & Solid ° Radioactive Wastes during Transport	X(1)	X(1)		

° (1) Number of barriers will be dependent upon the radiation level of the material. If the
°   radiation level is high, shielding will be provided during transport.

BNFP RADIOACTIVE MATERIAL FLOW
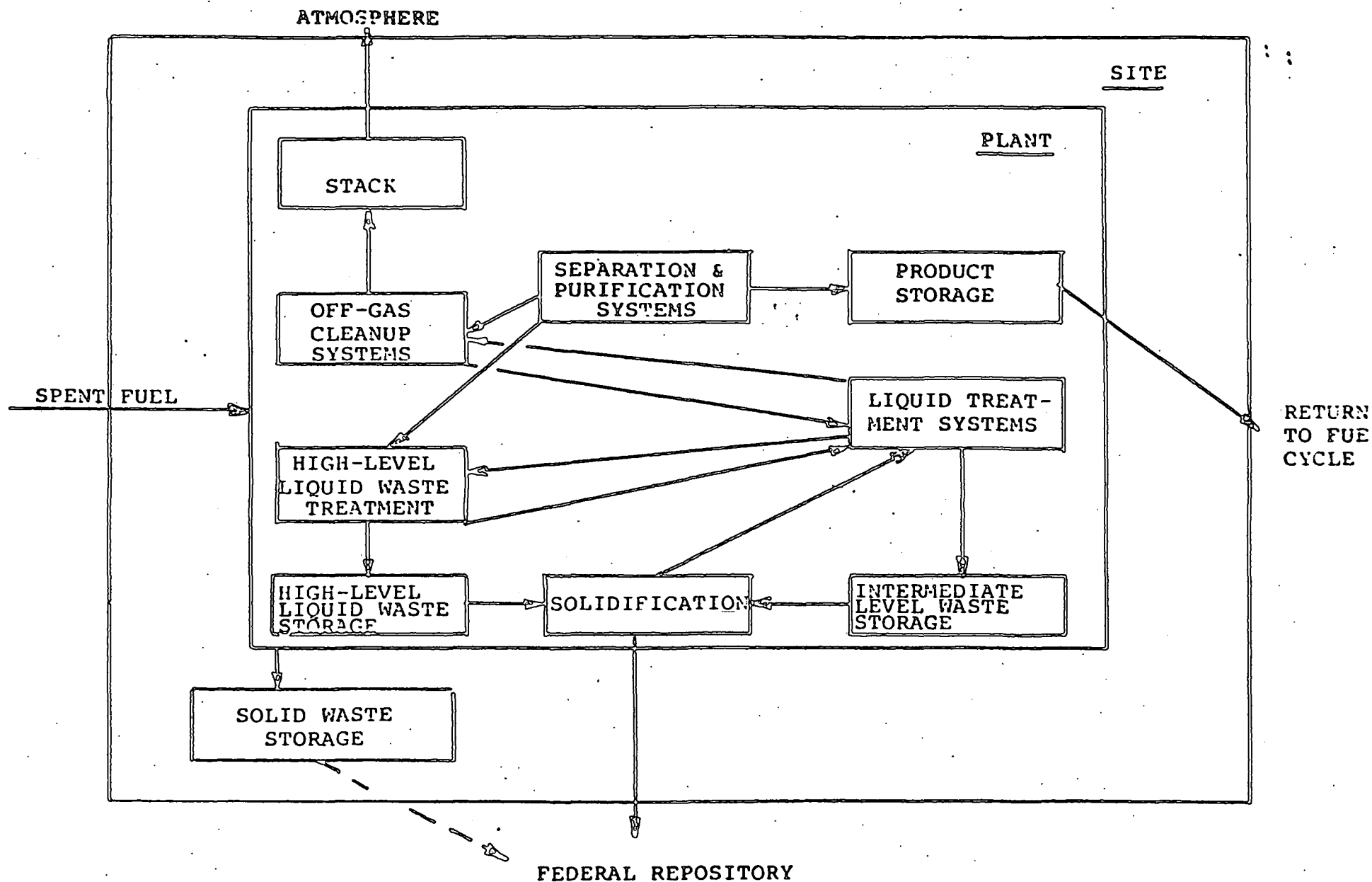
FIGURE II-8

under Code of Federal Regulations, Title 10, Chapter I[9].
These regulations serve as minimum limits for operational
safety.

The principles upon which criticality safety and radio-
active contamination controls are based, are implemented
in plant design and plant operation. In addition, however,
administrative responsibilities are assigned to specific
individuals or groups, at the plant, for plant functions
to assure that the reprocessing plant is operated and
maintained under the full range of normal and potential
accident conditions without risk to public health and
safety.

The daily functioning of the fuel reprocessing plant is
governed by an on-site organization which is self-sufficient
in regard to assuring public, plant personnel and facility
safety on a day-to-day basis. Functional components in
this organization include engineering, production, safety
and analytical services, employee and community relations
and financial groups. These licensed personnel have
specific qualifications for overseeing criticality and
radiation safety, accountability for special nuclear materials,
plant operation, plant maintenance, plant assistance and plant
services which include decontamination and waste disposal
operations. The on-site staff assure that all safety-
related activities are performed in accordance with established
procedures. Reviews and audits of plant safety are performed
at appropriate intervals both on an internal and independent
basis for compliance with prescribed requirements.

A criticality-safety control committee appointed by manage-
ment establishes the limits on the operating variables that
have a bearing on criticality safety. This committee,

composed of representatives from all the functioning groups,
reviews proposed changes in equipment or in operating
procedure. The committee's approval is required before
any change is implemented.

Radiation survey inside the plant as well as in the site
environs within a radius of up to 50 miles verifies the
effectiveness of contamination control.

Through a system of checks and balances among the functional
components in carrying out their daily operations of the
plant, the maintenance of performance records, the institu-
tion of training programs and testing of personnel to assure
their ability to discharge safety-related responsibilities
and the performance of audits at regular intervals, plant
safety under all normal and abnormal operation conditions
as well as full compliance with license and regulatory
requirements are further assured.

6.   Auxiliary Plant Systems and Controls

a.   Ventilation and Off-Gas System

In the generic plant, only gaseous and solid radioactive
wastes will eventually leave the site. No radioactive
liquid effluents will be exposed to the environment.

The primary effluent that could have an impact on the
environment during operation of the facility is the gaseous
effluent from the stack. This was discussed briefly in an
earlier section and in more detail in the following.

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The gaseous effluent from the main stack consists of
building ventilation air, vaporized process condensate
and the off-gases from the dissolver and vessel off-gas
treatment system. The building ventilation exhaust air
is routed through at least two stages of high efficiency
(HEPA) filters prior to discharge to the atmosphere.
Excess process condensate is decontaminated by evaporation
and condensation. The decontaminated water is recycled
to the process or alternatively may be revaporized and
discharged to the atmosphere via the 100-meter tall stack.
The dissolver off-gases (DOG) are routed through a treat-
ment train consisting of a condenser, knock-out pot, iodine
scrubber and an NO_2 absorber. The NO_2 absorber is designed
to recover 70 percent of the NO_x as 45 percent nitric acid.
The treated dissolver off-gas, which still contains residual
amounts of NO_x and iodine, is further treated through the
vessel off-gas treatment system (VOG) which also collects
gases vented from various other process or storage vessels.
The VOG treatment system consists of a condenser, knock-
out pot, iodine scrubber, pre-filter, iodine absorber and
a two-stage high efficiency (HEPA) filter. The stack is
equipped with samplers, monitors and alarms which identify
the amounts of radioactivity in the effluent. The system
is illustrated in Figure II-9. The expected radioactive
release from use of the treatment systems shown in Figure
II-9 in processing high-exposure spent fuel (average annual
fuel exposure of 32,000 MWD/MTU at 40 MW/MTU) cooled 180
days from reactor discharge, and processed at the rate of
1500 MTU/year is listed in Table II-4, page 36.

b.   Water Supply

Water must be provided for process streams, for makeup
water for closed loop cooling systems, for cooling towers

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MAIN PROCESS SYSTEM

TREATMENT SYSTEM FOR NON-CONDENSABLE EFFLUENTS

TO ATMOSPHERE

NOTE 1

FUEL — SHEAR & DISSOLVER

① DISSOLVER DUST SCREEN

DISSOLVER OFF-GAS CONDENSER AND KNOCK-OUT POT

NO. I IODINE SCRUBBER

NO₂ ABSORBER

FILTER NOTE 2

LAB. VENTILATION

MAIN PROCESS

VESSEL OFF-GAS

② VESSEL OFF-GAS CONDENSER AND KNOCK-OUT POT

NO. 2 IODINE SCRUBBER

FILTERS

IODINE ADSORBERS

FILTERS

NOTE 2

⑥

MAIN STACK

FILTER NOTE 2

BLDG. VENTILATION

③ CONDENSER AND KNOCK-OUT POT

(ALTERNATE)

ALTERNATE IODINE ABSORBER

ALTERNATE FILTERS

NOTE 2

TREATMENT SYSTEM FOR CONDENSABLE EFFLUENTS

⑦

④ HAW CONCENTRATOR

CONDENSER

LAW CONCENTRATOR

ACID FRACTIONATOR

CONDENSER

VAPORIZER

NOTE 3

⑤

LIQUID WASTE STORAGE

GENERAL PURPOSE CONCENTRATOR

CONDENSER

NOTE 3

LEGEND
——— EFFLUENT STREAM
– – – INTERNAL PROCESS STREAM

NOTES:

1. THIS STREAM IS CONTINUOUSLY SAMPLED FOR RADIOACTIVE CONTENT.

2. THESE STREAMS ARE SAMPLED FOR RADIOACTIVE CONTENT AS REQUIRED.

3. THESE STREAMS ARE CONTINUOUSLY SAMPLED & MONITORED WITH PROVISION FOR DIVERSION & RECYCLE IF REQ'D.
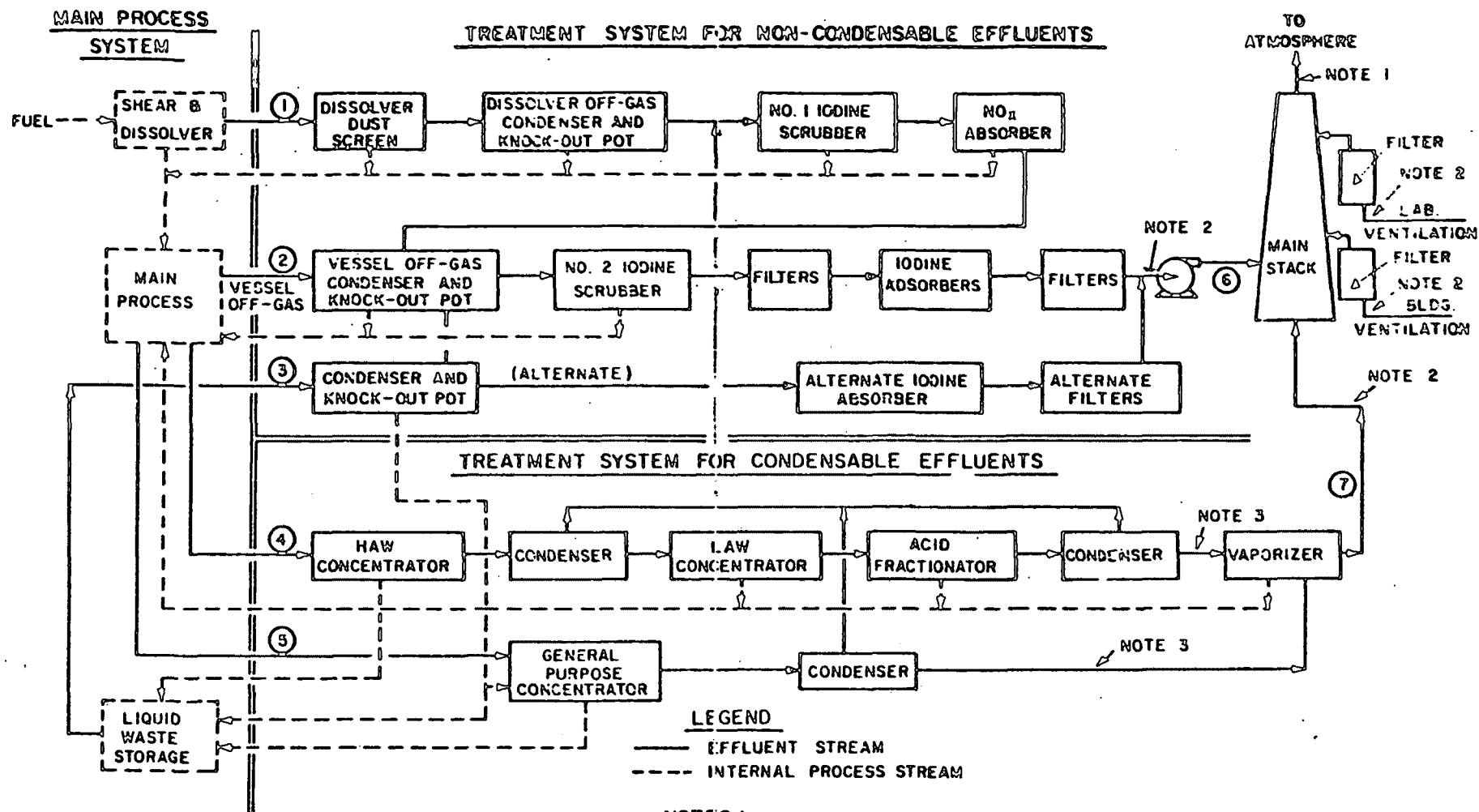
FIGURE II-9

SEPARATIONS FACILITY EFFLUENT TREATMENT SYSTEM[7]

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TABLE II -4

ESTIMATED AVERAGE RADIOACTIVE EFFLUENTS
(curies/sec) - Separations Plant Main Stack

Stream No. →	1	2	3	4	5	6	7	Total To Stack
°H-3	3.6E-3	--	--	1.4E-2	--	3.6E-3	1.4E-2	1.8E-2
°Kr-85	4.3E-1	--	--	--	--	4.3E-1	--	4.3E-1
Sr-89	2.0E-6	1.0E-5	2.0E-5	4.1	1.1E-3	6.5E-9	1.1E-9	7.6E-9
°Sr-90	2.3E-6	1.2E-5	2.3E-5	4.8	1.3E-3	7.6E-9	1.3E-9	8.8E-9
Y-90	2.3E-6	1.2E-5	2.3E-5	4.8	1.3E-3	7.6E-9	1.3E-9	8.8E-9
Y-91	4.2E-6	2.1E-5	4.2E-5	8.7	2.4E-3	1.4E-8	2.3E-9	1.6E-8
Zr-95	7.4E-6	3.8E-5	7.4E-5	15.4	4.2E-3	2.4E-8	4.0E-9	2.8E-8
Nb-95	1.4E-5	7.1E-5	1.4E-4	29	7.9E-3	4.6E-8	7.6E-9	9.3E-8
Ru-103	2.2E-6	1.1E-5	2.2E-5	4.6	1.2E-3	7.2E-9	1.2E-9	8.4E-9
Ru-106	1.3E-5	6.7E-5	1.3E-4	28	7.6E-3	4.3E-8	7.2E-9	9.0E-8
°I-129	1.4E-6	--	--	0.1E-0	8.1E-0	1.4E-10	1.1E-9	1.5E-9
°I-131	1.1E-9	--	--	9.3E-7	9.3E-7	1.1E-9	1.1E-8	1.2E-8
°Cs-134	5.0E-6	2.6E-5	9.0E-5	11	2.9E-3	1.7E-8	2.8E-9	1.9E-8
°Cs-137	3.2E-6	1.6E-5	3.2E-5	6.7	1.0E-3	1.0E-8	1.7E-9	1.2E-8
Ce-141	1.4E-6	7.3E-5	1.4E-5	3.0	0.1E-4	4.7E-9	7.8E-10	5.5E-9
Ce-144	2.2E-5	1.1E-4	2.2E-4	46	1.2E-2	7.2E-8	1.2E-8	8.4E-8
Pm-147	4.0E-6	2.0E-5	4.0E-5	8.3	2.2E-3	1.3E-8	2.2E-9	1.5E-8
U-234	4.3E-11	2.4E-10	2.2E-12	4.4E-11	3.3E-12	4.4E-14	2.2E-14	7.0E-14
U-235	4.2E-13	2.4E-12	2.2E-14	4.4E-9	2.2E-10	4.4E-16	2.2E-16	7.0E-16
U-238	9.9E-12	5.8E-11	5.4E-13	1.1E-7	5.4E-9	1.4E-14	5.4E-15	2.0E-14
°Pu-238	1.0E-7	3.8E-7	1.1E-8	2.2E-3	5.6E-9	7.9E-11	5.3E-11	1.3E-10
°Pu-239	9.0E-9	2.5E-8	9.9E-10	2.0E-4	5.0E-6	7.1E-12	4.8E-12	1.2E-11
°Pu-240	1.6E-8	4.6E-8	1.8E-9	3.6E-4	9.1E-6	1.3E-11	8.7E-12	2.1E-11
°Pu-241	4.2E-6	1.2E-5	4.5E-7	9.1E-2	2.3E-3	3.3E-9	2.2E-9	5.5E-9
Pu-242	8.6E-11	2.4E-10	9.5E-12	1.9E-6	4.8E-8	6.8E-14	4.6E-14	1.1E-13
°Am-241	0.7E-9	4.3E-8	0.7E-8	1.4E-2	4.3E-6	2.8E-11	4.3E-12	3.2E-11
Am-242	1.7E-10	8.4E-10	1.7E-9	2.7E-4	8.4E-8	9.5E-13	8.4E-14	6.3E-13
°Cm-242	1.0E-6	5.1E-6	1.0E-5	1.7	5.1E-4	3.2E-9	5.1E-10	3.7E-9
Cm-243	6.7E-10	3.4E-9	6.7E-9	1.1E-3	3.4E-7	2.1E-12	3.4E-13	2.5E-12
°Cm-244	1.1E-7	5.4E-7	1.1E-6	1.8E-1	5.4E-9	3.4E-10	5.4E-11	3.9E-10

°Quantitatively and Biologically Significant.

Short lived decay products are also present in equilibrium quantities.

For Peak Release, multiply Krypton and Iodine values in Stream 07 by 2.2, and remaining values in Streams 07 and 08 by 1.2.

[1] ACTIVITY BASIS:
SPECIFIC POWER	= 40 kw/kg
COOLING TIME	= 180 days
FUEL ACTIVITY	= 32,000 MWD/MTU
PRODUCTION RATE	= 1,500 MTU/Yr.

RELEASE BASIS:
1000 Kr-85 & H-3 Released
Iodine Decontamination Factor = 1,000
DOG/VOG Decontamination Factor = 10 and Filter Efficiency = 99.80
HAW/Vaporizer Decontamination Factor = 10[12]
GPP/Vaporizer Decontamination Factor = 10[6]
Pu Product Recovery = 99%
Pu Loss to Waste = 1%

[2] Average source during reprocessing is assumed to be less than peak. Values are based on instantaneous rate averaged over a calendar year.

NOTE: 1E-10 = 1 x 10^{-10}

and for the fuel storage pools. In addition, water must be provided as backup for the plant fire water system, emergency cooling for the HLLW storage facilities, for the emergency utility area, and for the 76°F cooling loop.

In the generic plant, it is assumed that water will be available from deep wells on the site. (At BNFP, water is obtained from three deep wells. Normal cooling is by circulation through the cooling tower but the wells can supply straight-through cooling emptying into a pond which serves as an emergency reservoir. The BNFP pond is a 15-acre pond having a capacity of 60 million gallons of water.)[7] Recirculation, for several months, with such a pond is possible by means of an emergency diesel-powered pump. Redundant pumps are connected to separate emergency busses with automatic start features as a back-up safety measure.

c.   On-Site Electrical Power

The loss of on-site electrical power, if sustained, could lead to an unsafe plant condition. To minimize this possibility, consideration is given to a high reliability source of off-site power. Should off-site power fail, standby diesel powered generators will come on and assume priority loads. These and day-tank fuel supplies are located in structures designed to meet earthquake and tornado criteria[5,6,7]. An uninterruptable source of ac power is used to supply power to the process control equipment and to provide control for starting the diesel generators. A 24V dc battery supply provides a highly reliable supply of power for monitors and some control functions.

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d.   Compressed Air Supply

Oil free compressed air is required for process control,
instrumentation and starting the emergency diesel powered
generators. It is used for air lifts, air pulsers for the
extraction column operations, for air circulation as in the
high-level liquid waste storage tanks, and for air purging
of radiolytic hydrogen concentrations, generated during
plant operation, to prevent formation of potentially
explosive concentrations.

The generic plant is provided with two air compressors to
provide a doubly redundant supply of air. These compressors
are provided with pressure relief valves as a precaution
in the event of failure of the pressure cut-off switch.
Accumulator tanks located at critical locations about the
plant are equipped with reverse flow check valves in their
feed lines so compressor failure or pipe failure will not
necessarily result in immediate pressure failure. .As an
example, the emergency diesels have accumulator tanks
capable of 5 starts without resupply.

e.   Steam Supply

Steam is used for process transfer jets, process heating,
space heating, steam turbines, decontamination, deaerator
heating and stripping, and yard steam tracing.

In general, it is not a critical service except for two
aspects: the high-level liquid waste transfer and the
control of steam pressure (100 psig) and hence the temperature
(270°F). To inhibit the possibility of a "red-oil" explosion
which requires approaching a threshold temperature as one of

the conditions, the steam pressure is limited to 40 psig in the solvent extraction operations and subsequent processing of the effluent streams containing organic solvents. Redundant pressure regulators are used to maintain the maximum allowable process temperatures during the various unit operations.

The generic plant, having a large quantity of high-level liquid waste, has an emergency steam generator for emergency liquid waste transfer which is designed to earthquake acceleration criteria and is housed in an enclosure built to tornado criteria.

III.   FAULT TREE CONSTRUCTION OF ACCIDENT SEQUENCES

1.   Background

Quantitative safety analysis has been developing rapidly
due to the requirements of the space program and to the
growth in size and numbers of nuclear power reactors. The
techniques which have developed in these disciplines are
used in this study for the analysis of hypothetical nuclear
fuel reprocessing accidents and therefore a brief review
of the development of reactor safety analysis will be pre-
sented.

The accident potential in nuclear power reactors has been
recognized for some time. Safety analysis of nuclear power
reactors has been approached by designing them for protec-
tion against the maximum credible accident (MCA). Later,
in a desire for more realism, the design basis accident(s)
(DBA) was defined and mitigated against. Currently there
is a desire to present accidents on a numerical risk basis
so that comparisons can be made with risks already accepted
by society. Early work in probabilistic reactor accident
assessment was done by Mulvihill[12], Garrick et al[13], Farm-
er[14] and Otway and Erdmann[15]. All of these authors except
Farmer used fault tree analysis for the calculation of
accident probabilities. This work is actively continuing
under AEC sponsorship. To date, the most complete descrip-
tions of reactor risk are provided by Otway and Erdmann[15]
and Starr[16] who place these risks in perspective by compar-
isons with other social risks. The present work provides
a similar safety analysis for a generic nuclear fuel repro-
cessing plant.

The central problem of probabilistic safety analysis is the representation of a complex system such as a fuel reprocessing plant in a form suitable for safety analysis. A plant is represented by construction, plumbing, electrical drawings, process flow charts, etc.; these must be synthesized into a unified description of the plant accident spectrum which is suitable for probabilistic analysis. This synthesis can be done through the use of fault trees.

Engineering for safety is not a new concept. Common examples are pressure relief valves on a boiler, or the safety on a loaded firearm. In both of these examples two simultaneous events must occur for an accident to result. Early safety engineering was usually done after the fact of an accident to prevent its recurrence. And in fact, the beginning of probabilistic safety was concerned with analyzing after-the-fact missile misfirings. Nuclear safety engineering has introduced a new aspect; namely, the calculation of the occurrence probability of accidents that in many cases have never occurred, or have never occurred with the safety system being analyzed.

Normally, reasoning proceeds from cause to effect; this process, when applied to safety, is called failure mode and effects analysis. This method proceeds by a series of "what if" statements to the final undesired event but generates many final results that are not of particular interest.

Fault tree analysis begins with a final result that is significant and proceeds back through the system identifying causes. In a certain sense, time is going backwards

in the logic of developing the tree. In this manner, many extraneous paths that would be generated by a failure-mode and effects analysis are eliminated and only those paths that lead to the top event are generated.

The logical structure dictated by the form of the fault tree prescribes the manner in which the probabilities must be combined to yield both the probability of the top node (final hazard event), and the probability of individual event sequences.

The overall accuracy of event sequence prediction, therefore, depends directly upon both the availability and quality of the basic probability data .

2.   Fault Tree Symbolic Language

Fault tree analysis is basically a two-state Boolean logic and as such uses the operation of logical addition often referred to as logical "or" ("union").
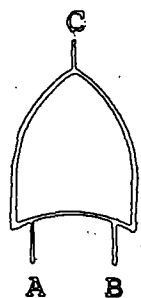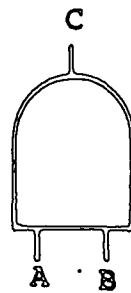

A + B = C

Truth Table

A	B	C
1	0	1
0	1	1
0	0	0
1	1	1

Thus if A or B have a true input, the output is true. In the case of fault trees, true corresponds to failure. If either of the inputs A or B are in the failed state the gate output is in the failed state.

Another logical operation is logical multiplication and
is referred to as logical "and" ("intersection").

	Truth Table	
A	B	C
1	0	0
0	1	0
0	0	0
1	1	1

$A \cdot B = C$

Thus if A _and_ B are true, the output is true and two
safety systems must simultaneously fail to get system
failure. In the earlier example of a loaded firearm, the
pulling of the trigger, and the safety being off must
both occur for the weapon to fire.

Other logical operations have been defined and are used
by safety analysts. In this report, care has been used
to avoid other operations because: they usually can be
represented by combinations of "and" and "or" gates, for
reasons of clarity and because these operations are the
only ones that can be directly treated in existing fault
tree computer programs.

The accident events are represented by a square which de-
scribes the event logically represented by a gate.

A circle is used to represent an event for which probabil-
ity-of-occurrence data is available. Hence it represents
a terminal event requiring no further development.

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A diamond is used to represent an event which is not fur-
ther developed because it is not believed to be of signifi-
cance or adequate information is unavailable.
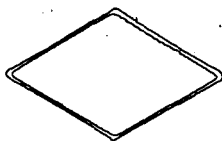
The triangle is used as a linking symbol.

LINK IN   LINK OUT

A triangle, always to the right of an event rectangle, rep-
resents a linking out and indicates that the whole tree
should be transferred over to the linking-in connection
which is always at bottom center of the link-in point. The
linking operation is used to avoid repetitious drawing and
space limitations. In this report, the link-in rectangle
may give information by further specifying the tree being
linked-into. In this way a general repeated tree is partic-
ularized for a specific application. This procedure is
similar to a subroutine calling sequence which transfers
parameters to the subroutine calculation.

3.   Fault Tree Construction for a Nuclear
          Fuel Reprocessing Plant

A chemical plant is a very complex entity in both its arch-
itecture and processing operations. Correspondingly a fault
tree modeling the safety analysis of the plant is also highly
complex. To systematize and maintain control over the comple-
tion of the safety modeling a procedure for fault tree con-
struction known as Leak Path Analysis was used.[17] Briefly,

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the method enumerates the intersection of barrier failures
encountered in tracing all paths from radioactive sources
to the environment. The union of these Leak Paths forms the
top event - the uncontrolled release of radiation (URR) to
the environment. This very large equation is both unwieldy
and contains redundancies but when factored into most com-
pact form (terms appearing only once) or as close thereto
as can be achieved, it is suitable for conversion to a fault
tree of the plant. This is performed by the replacement of
intersection operations by the fault tree symbol AND and
similarly union by OR.

The top of a fault tree constructed in this manner is shown
in figure III-1. The modes of release are air and liquid
pathways. Solid pathways were not considered.

The release by liquid pathways was developed in a series of
fault trees for qualitative evaluation of critical paths
but based on historical data from high level liquid waste
storage[18] was not further analysed and the event is now
"diamond-off" on the fault tree.

Similarly, an investigation of risks under neptunium proces-
sing failed to identify consequences as severe as those which
could occur during normal processing. Since neptunium pro-
cessing is performed about once a year, the probability is
correspondingly reduced and the risks are not further con-
sidered.

An investigation of URR during shutdown did not reveal risks
comparable to normal operation and was therefore excluded
from detailed analysis.

Figure III-1

Generic Reprocessing Plant: Top of Tree
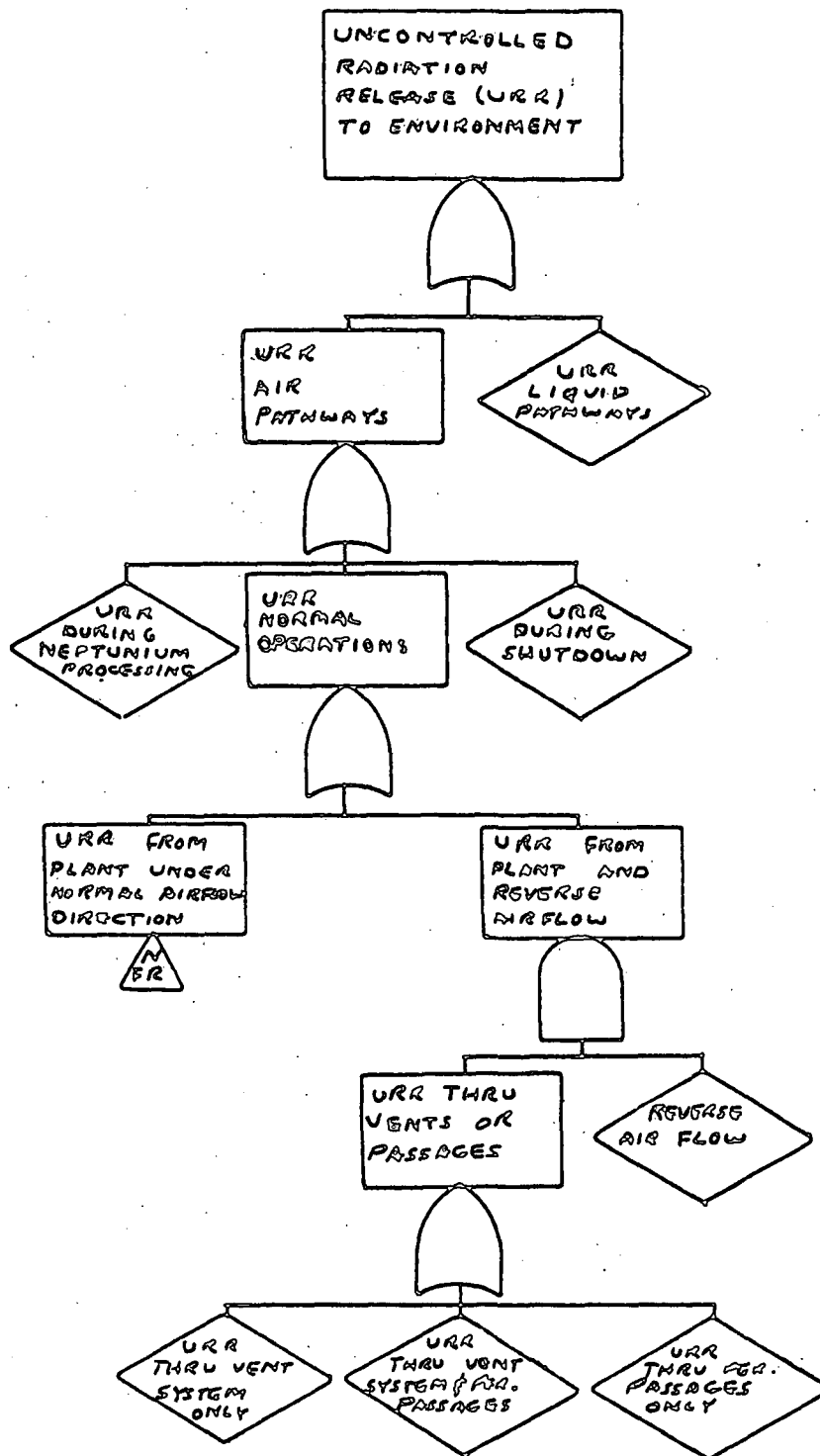
Release by air pathways under normal plant operation is divided into normal and reverse air flow because reverse air flow allows particulate from cells such as the PPC to impact the environment without any high efficiency filtering. In the investigation it was found that reverse air flow cannot occur by equipment failure alone. There is one exception, however, the air pulser can in principle result in a higher pressure in the process cells than outside the building. Considering the reverse flow ventilation damper, the redundant protection for maintaining normal air flow, the interlock on the pulser and the volume of building and site of air flow, this mechanism was determined to be ineffective and was not pursued. Fault trees were constructed for the reverse air flow diamonded events shown in Figure III-1 but are not included because they were not numerically evaluated and void in the risk assessment.

A tornado is a natural event that can result in reverse air flow from its depressurization and this is included in the tornado analysis. Because of the uncertainty of failure modes under tornado conditions, little value is derived from the reverse flow fault trees.

The fault tree development continues with the linking tree NFR (figure III-2). The failure probability of 10^{-2}/yr or 10^{-3}/yr for single and double HEPA filters respectively are from references 19 and 20. This is the probability of the filter having a particulate transmission greater than design specification. Clogging of the filters was not considered to be a failure mode for present purposes.

The event URR in RPC is developed in figure III-3. The releases from this are further divided into those due to
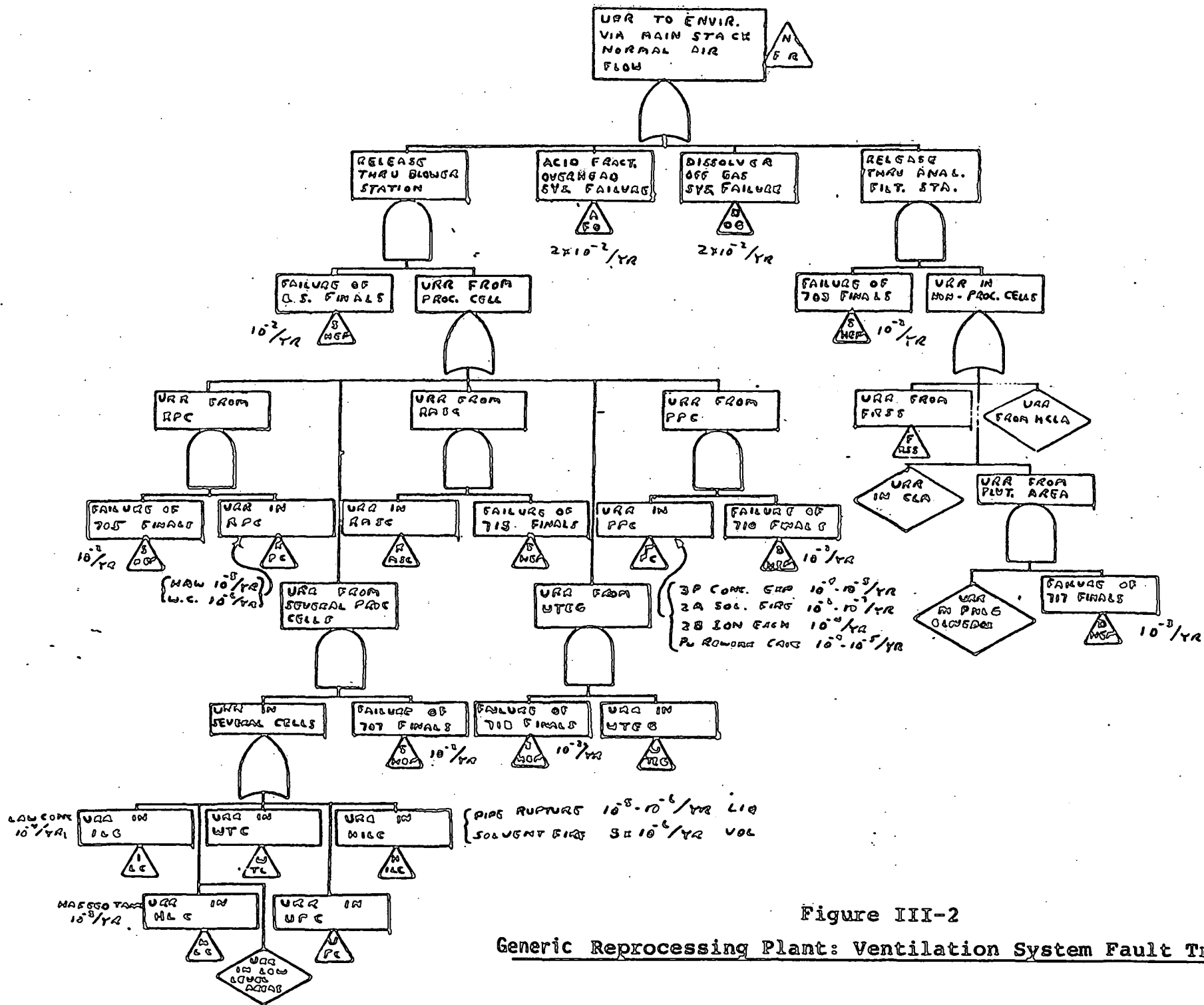
Figure III-2

Generic Reprocessing Plant: Ventilation System Fault Tree

LIST OF EQUIPMENT

| Code	Description
10-C-101	C1 DISSOLVER
10-C-102	C2 DISSOLVER
10-C-103	C3 DISSOLVER
C1-D-044	HOT SLAB POT
10-G-102	C1 DISSOLVER HEATER/COOLER
10-G-103	C2 DISSOLVER HEATER/COOLER
10-G-104	C3 DISSOLVER HEATER/COOLER
C1-G-C01	HAW CONCENTRATOR RECOILER
C1-G-008	HAW CONCENTRATOR
10-H-104	C1 DISSOLVER SCREEN POT
10-H-109	C2 DISSOLVER SCREEN POT
10-H-190	C3 DISSOLVER SCREEN POT
C3-H-001A/B	MAINTENANCE CEILING PLUG
06-H-C01C/D	MAINTENANCE CEILING PLUG
06-H-C01	SHIELDING DOOR
C3-T-C01	FUEL TRANSFER CONVEYOR
C3-T-002	FUEL TRANSFER CONVEYOR DRIVE
10-T-101	TRANSVERSE FUEL PUSHER
18-T-102	MAINTENANCE CRANE 50-TON
18-T-103	CASKET HANDLING CRANE
18-T-104	POWER MANIPULATOR
18-T-106A/B	DISSOLVER BASKET STORAGE RACK
10-T-107	FUEL TRANSFER TABLE
40-T-401	DISSOLVER BASKET DUMPER
C3-T-003	HULL MONITOR
40-T-411	SUMP JET OASE
40-T-412	SUMP JET OASE
C3-T-043	PASS THROUGH PORT
07-V-C04	SHIELDING WINDOW
07-V-C06	SHIELDING WINDOW
07-V-C06	SHIELDING WINDOW
07-V-C06	SHIELDING WINDOW
07-V-010A/B	PERISCOPE SLEEVE O PLUG
07-V-010C/F	DITTO
07-V-010C/G	DITTO
07-V-010D/H	DITTO
07-V-C01	SHIELDING WINDOW
14-Y-101	SHEAR
14-Y-104	SHEAR SUPPORT FRAME
14-Y-106	SHEAR FEED MAGAZINE
14-Y-106	SHEAR MAG. TRANSITION DUCT.
14-Y-107	SHEAR FEED CHAIN O MAGAZINE
14-Y-108	SHEAR FEED RAM DRIVE UNIT
10-Y-102	DIVERTOR
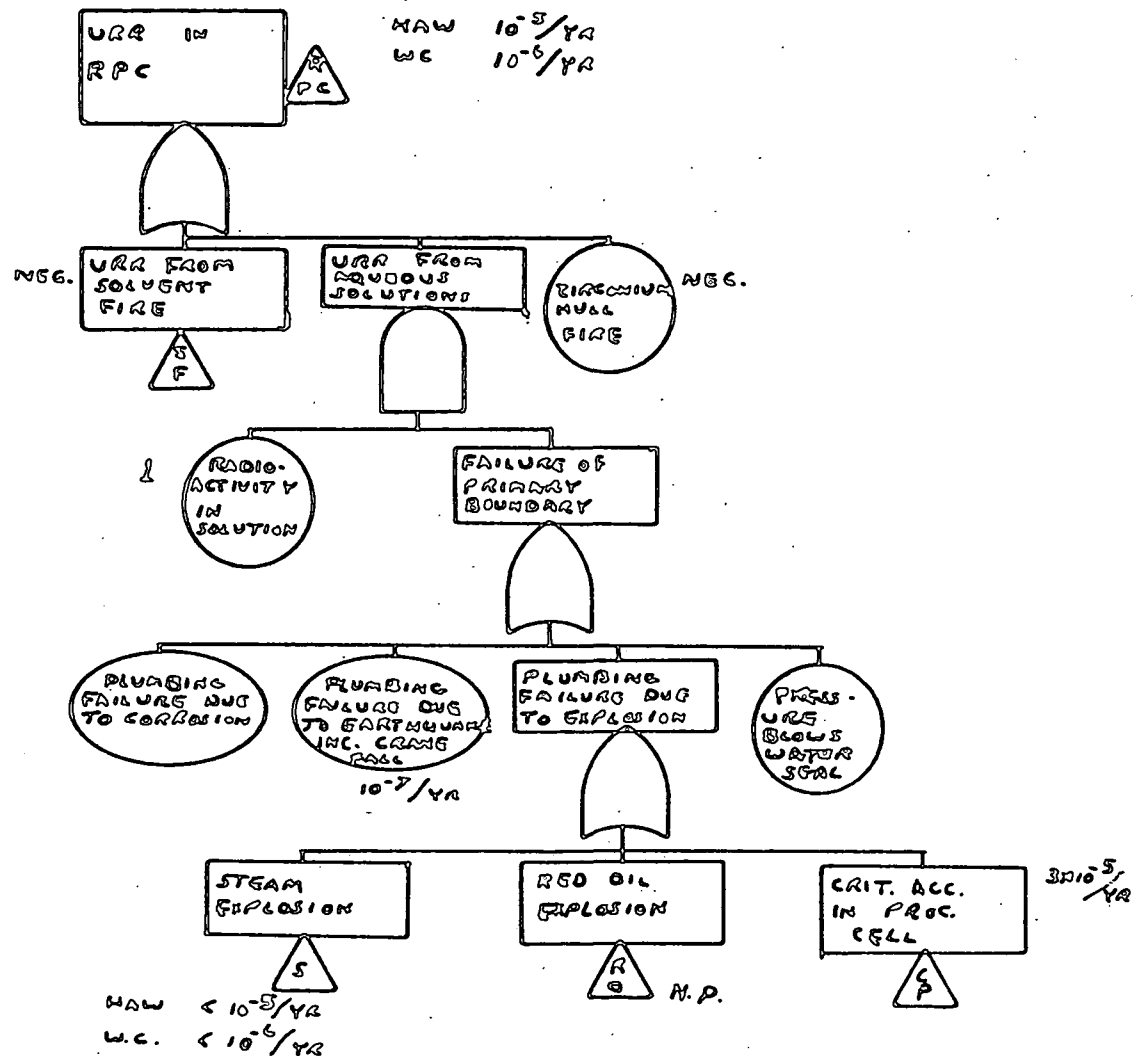
Figure III-3

Generic Reprocessing Plant:

Fault Tree For URR in the Remote Process Cell

solvent fire, aqueous solutions and zirconium hull fires.
The aqueous solution spill is developed to material failure
from chemical, physical and natural phenomena. Statement
of the use of water (aqueous) seals for confining the gase-
ous products within the plumbing boundary is expressed
explicitly even though the probability of occurrence is
one for about 1 psig overpressure.

In the design of the fault trees for the reprocessing plant,
it was decided to present the generic tree for the occur-
rences in that cell. This is further supplemented by the
list of equipment present in the cell. In order to evalu-
ate the accident probability in that cell, repeated appli-
cations of the tree to encompass all the equipment given in
the list must be performed. This reduces the repetitive
complexity of the trees and helps to maintain a better
perspective over the details.

The data base used in evaluating figure III-3 and the other
trees included in this report comes from many sources. It
should first be stated that there are no evaluated data
bases for nuclear fuel reprocessing plants, therefore fail-
ure rate data from similar equipment in similar environments
must be used.

Probably the best source of component data would be from
chemical plants using solutions of similar acidity. Acci-
dents are reported to the Manufacturing Chemists Associa-
tion[21] but there is no reporting of component failure and
there are no compilations of this data.

Some component failures and all accidents are reported to the USAEC but there is no systematic compilation of this data. The body of this data is for nuclear reactors, an example of which is a recent analysis of pipe rupture occurrences. Using data reported in 1972, it is found that the log mean pipe failure rate for PWR and BWRs is

$$1.6 \times 10^{-5}/yr\text{-}ft.$$

Similar data is available in the General Electric Pipe Rupture Study series.[22] The report GEAP-10207-25 contains pipe break data for both nuclear and fossil power plants.

Anyakora et al[23] have published instrument failure rate data on chemical plants in Great Britain in three environmental categories. Their work is summarized in a compilation by Powers and Tompkins.[24] Additional British data is contained in the useful text by Bourne and Green[25] and the article by Bourne.[26]

Data on U.S. instruments can be found in MIL-HDBK-217A[27] and in the recent IEEE survey of industrial plants.[28] Data on the reliability of fire prevention systems may be found in the paper by Miller.[29]

General collections embracing electrical, instrumentation and plumbing failure rates are found in the publications by Garrick et al[30] and Atomics International memo.[31] A very useful recent nuclear plant reliability evaluation is that due to Erdmann et al.[32]

The probabilities for natural phenomena are quite uncertain and subject to the siting of the generic plant. A probability of $10^{-7}/yr$ is estimated for exceeding the design

basis earthquake by a factor of 2 in acceleration. This
is based upon an unpublished study prepared in connection
with an environmental impact analysis for a plant in
Richland, Washington. The probability of 10^{-5}/yr for a
design basis tornado is from an unpublished study of a
plant in Oklahoma.

Using these data, the fault tree of III-3 can be evaluated.
The linking tree steam explosion needed for the bottom
left hand event is presented in figure III-4. This steam
explosion is presented as the intersection of pressure
buildup, the failure of pressure relief devices and operator
failure. In some portions of the plant where aqueous seals
are used, a damaging explosion is not possible. The seal
will blow and some of the entrapped radioactivity will
constitute a small release within that cell.

Figure III-5, a red oil explosion shows the elements that
must be present to result in such a reaction: the presence
of heavy ions, excess acidity, organic solvent, excessive
temperature and the failure of the operator to correct the
upset condition. These events are developed into subsidiary
trees as required.

Figure III-6 shows the fault tree for a criticality acci-
dent. This is developed in a general way and some care must
be exercised in applying it to assure that all the failure
modes are indeed possible. The valve failures shown were
originally intended to be developed into subsidiary trees
to include possible common mode problems. Because these
were not found, chemical plant failure rate values were
used. The failure rate for instrument power used in figure
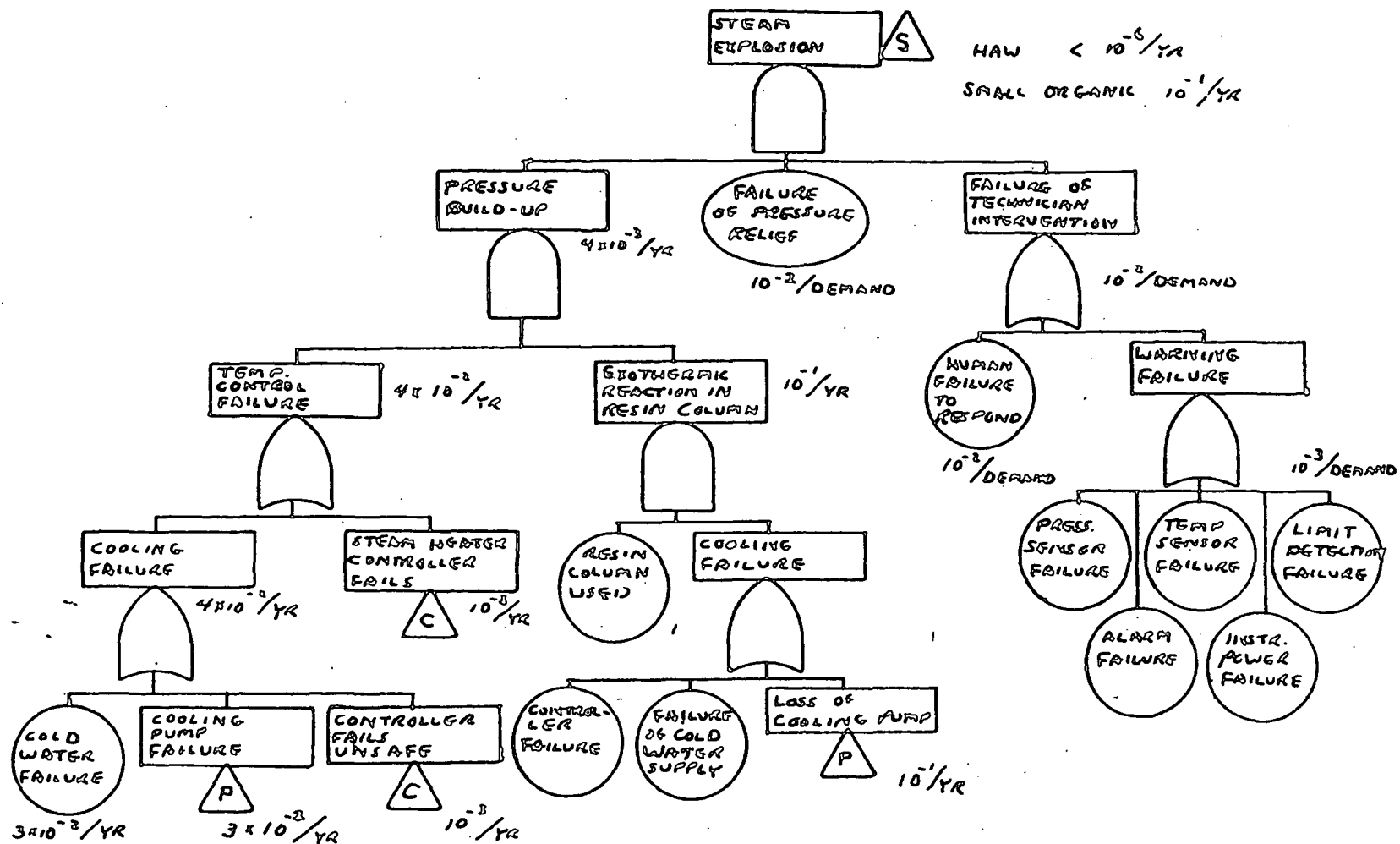
Figure III-4

Generic Reprocessing Plant: Fault Tree for Steam Explosion Accident

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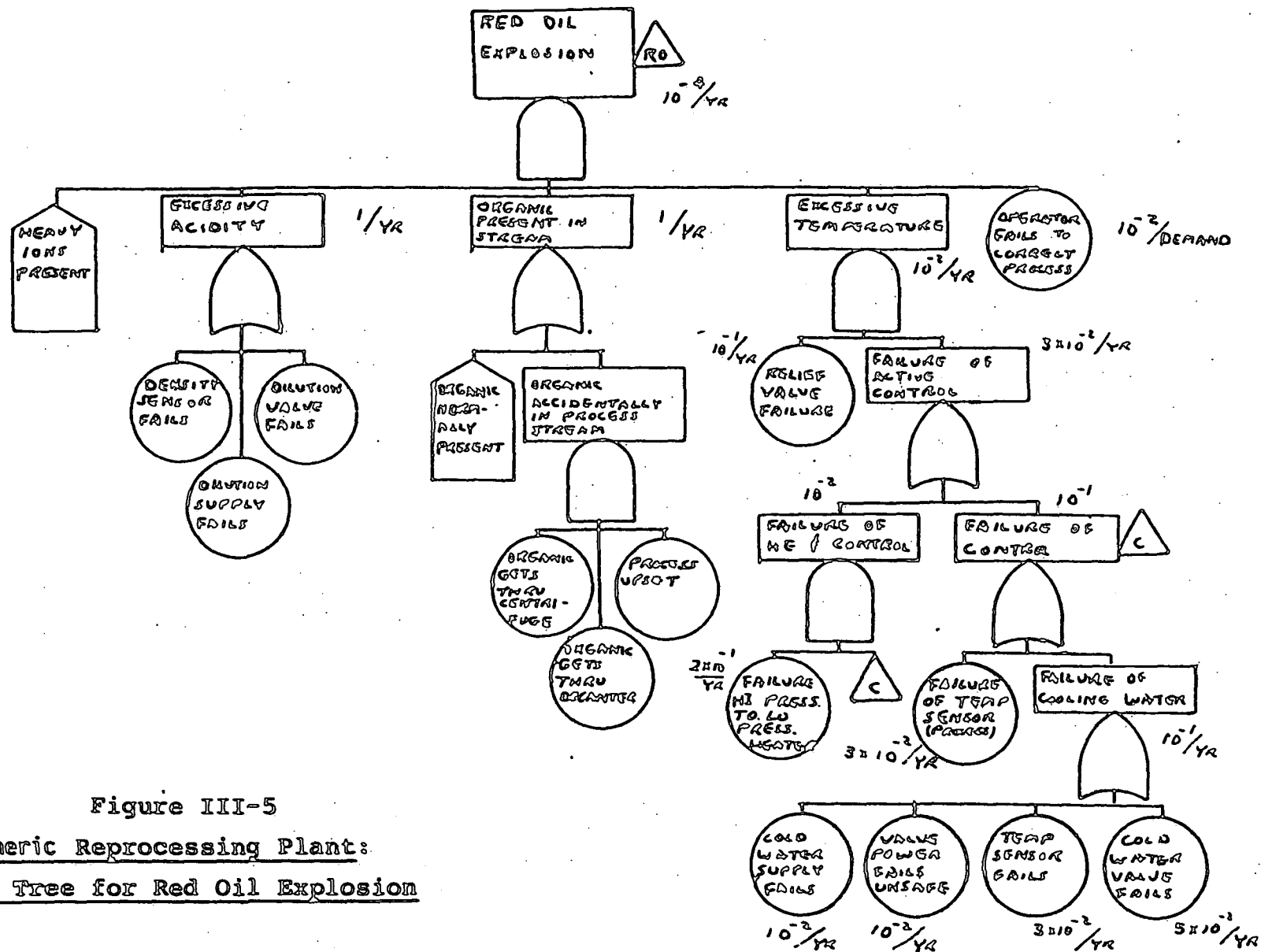
Figure III-5

Generic Reprocessing Plant:

Fault Tree for Red Oil Explosion

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Figure III-6

Generic Reprocessing Plant:

Fault Tree for Criticality Accident

in Process Cell
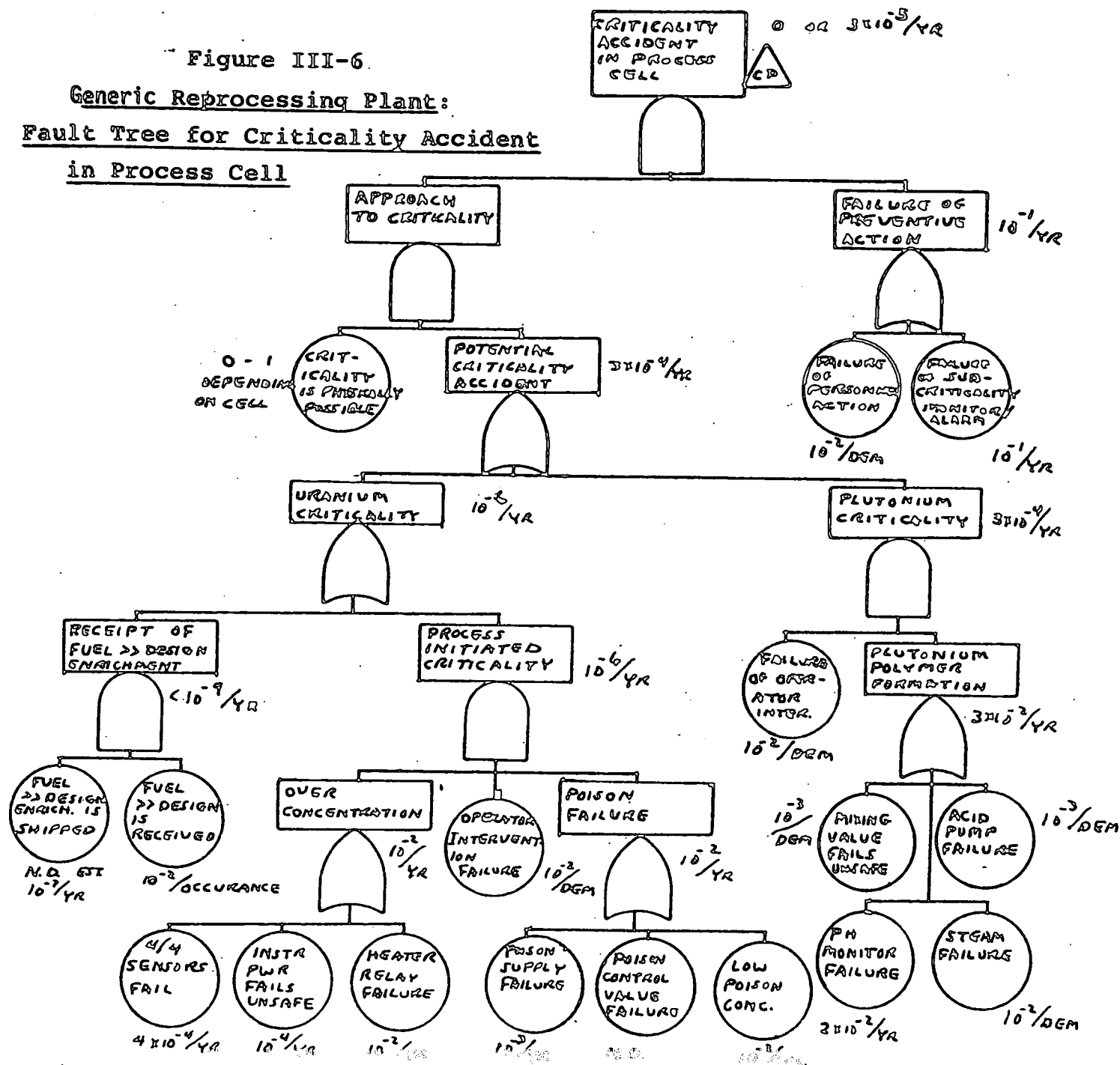
III-6 was taken from the Final Safety Analysis Report for the uninterruptable instrument power system in the Monticello power plant.[33] This also agrees with other data on d.c. system power.

This completes the failure probability analysis that began with figure III-3. The analysis continues by reference to figure III-2 and picking up another process cell. The complete collection of fault trees used in this report are presented in Appendix B.

IV.   ACCIDENT ANALYSIS

1.   General

A fuel reprocessing plant represents a substantial poten-
tial release source for radioactivity because of the
presence of fissile material and fission products, the use
of organic solvents and the handling of quantities of
radioactivity in aqueous solutions. Because of the redun-
dant safety systems designed into the plant and the inherent
nature of the process, however, it is believed that most of
the potential accidents have a low probability of occurrence.

Anticipated operational accidents are considered to be
situations in which only one independent failure, human or
equipment, is involved. Environmental radioactive releases
from such events would be expected to be less than those
allowable by 10 CFR 20 although the technical specifications
on releases may be exceeded. An upper limit accident is
considered to be a situation resulting from multiple opera-
tor errors, from multiple equipment malfunctions or from
stresses imposed by natural phenomena which may have notice-
able potential environmental consequences.

2.   History of Accidents in Reprocessing

Most of the accidents that have occurred during irradiated
fuel reprocessing operations took place at AEC installations
when the industry was in its infancy. These experiences, in
general, might not correspond with anticipated commercial
plant experience. The Nuclear Fuel Services plant at West
Valley, New York is the only commercial light water irradiated

fuel reprocessing plant which has operated in this country.
It had 6 years of operation before it suspended operation for
modification to allow higher throughput[5]. Reprocessing
experience that has been applied and operated on a produc-
tion basis in the U.S. employing essentially the same
process technologies, however, is greater than 100 plant-
years[10].

A literature survey was made to accumulate a data base of
accidents which were categorized into types of accidents
for the nuclear energy and related chemical processing in-
dustries. The survey showed that they do happen in spite
of safety precautions that have been taken. The data base
provided guidance in the evaluation of the safety features
assumed for the generic reprocessing plant too. Specific
descriptions of the accident circumstances from the survey
are presented in Appendix C.

The types of accidents which have occurred or are probable
during reprocessing operations, according to the litera-
ture, are categorized as follows:

    a) Criticality accidents
    b) Fires
    c) Explosions
    d) Fuel receiving and storage accidents
    e) Waste storage accidents
    f) Natural phenomena events.

These are discussed in the following section.

3.   Postulated Accidents; Bases for Selection
      in the Hazards Analysis

Accident situations selected for analysis were hypothesized
from a review of reprocessing experiences, the design of
the generic reprocessing plant and the unit operations
involved in the process. The accidents selected are be-
lieved to have the most severe consequences in terms of
potential exposure to the environment. These events, of
low probability, are credible only if one assumes simultan-
eous failure of engineered safety features and where per-
tinent, compromise of administrative procedures established
as safety barriers. The types of incidents considered are
those most likely to result in the dispersion of radio-
activity beyond the primary confinement. These include:

   o Nuclear criticality - a solution criticality in a
      process vessel in the Remote Process Cell

   o Fires involving solvent and process equipment; a
      fire in leached zirconium alloy cladding was also
      considered. The consequences of such an incident,
      however, were found to have a negligible poten-
      tial environmental impact

   o Explosions involving different types of radioactiv-
      ity sources

   o Fuel receiving and storage maloperation

   o Loss of cooling to the high level waste facility

   o Earthquake and tornado consequences on the model
      reprocessing plant's structures and systems con-
      taining the normal inventory of design basis fuel
      were also reviewed and evaluated.

The potential accidents are discussed in the following
in terms of malfunctions or errors required for the acci-
dent to occur, the probabilities of their occurring, the
protective measures available and the off-site consequen-
ces that could result from such accidental releases.

a.   Criticality Accidents

In the 30 years of the nuclear age, representing 432 plant-
years, there have been 30 criticality accidents in the
U.S.[34,35]. Most of these occurred in experimental facili-
ties; some of the early accidents are attributable to the
limited understanding of nuclear energy at the time.
There have been no criticality accidents since 1968. Seven
of the accidents which occurred were in solution systems
which could exist in a fuel reprocessing plant. The
balance of the incidents involved conditions that would
not be encountered in a reprocessing facility. A summary
of the relevant accidents is presented in Table IV-1,
page 61. The tabulated incidents occurred during reproces-
sing operations but not at a production facility. However,
they provide anticipated general characteristics for crit-
icality accidents which could occur in a processing plant's
solution systems. These are summarized as follows:

1. The number of fissions in such an event would be
   less than 10^{20}

2. The accident would have to be sustained over many
   minutes to produce a fission yield approaching 10^{20}
   fissions

TABLE IV-1

Solution Criticality Accidents[34,35]

DATE	LOCATION	ACTIVE MATERIAL	GEOMETRY	TOTAL FISSIONS	CAUSE	PHYSICAL DAMAGE
June 16, 1953	Y-12 Processing Plant, Oak Ridge, Tenn.	2.5 kg ^{235}U $UO_2(NO_3)_2$ in 56 liters H_2O	Cylinder concrete reflected below	1.3×10^{18}	Wash water added to $UO_2(NO_3)_2$ solution	None
December 30, 1958	LASL, New Mexico Pu Processing Plant	3.27 kg Pu $PuO_2(NO_3)_2$ in $\sim$ 168 liters H_2O	Cylinder water reflected below	1.5×10^{17}	Agitator created critical geometry	None
October 16, 1959	Chemical Processing Plant Idaho Reactor	34.5 kg ^{235}U $\sim$800 liters H_2O-$UO_2(NO_3)_2$	Cylinder concrete reflected below	$\sim$4 $\times 10^{19}$	Solution siphoned from safe to unsafe geometry	None
January 25, 1961	Chemical Processing Plant Idaho Reactor Testing Area	8 kg ^{235}U $UO_2(NO_3)_2$ in 40 liters H_2O	Cylinder	6×10^{17}	Solution pumped from safe to unsafe geometry	None
April 7, 1962	Hanford Works, Richland, Wash.	1.55 kg Pu	Cylinder unreflected	8×10^{17}	Concentrated solution incorrectly siphoned	None
July 24, 1964	The Wood River Junction, R.I. scrap recovery facility	2.64 kg ^{235}U in $UO_2(NO_3)_2$	Cylinder unreflected	1.3×10^{17}	Concentrated solution poured into unsafe geometry. Additional moderation in tank	None
Jan. 30, 1968	Y-12 Processing Plant - Oak Ridge, Tenn.	3.3 Kg U-233 $UO_2(NO_3)_2$ in 20 ltr. water	Sphere water reflected	1.1×10^{16}	Solution surged from safe to unsafe geometry	None

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3. The rate of energy release would be too low to be explosive, i.e., no shock front generation would be anticipated

4. The event would be associated with a change from normal procedures

5. The environmental impact would be very small; total property loss would be less than $70,000 as indicated from the incidents which occurred.

Criticality incidents have typically resulted in initial bursts of less than 10^{18} fissions followed in some instances by subsequent bursts of less than 3×10^{16} fissions per second. Little or no damage resulted to the confinement equipment from the criticality events.

In this analysis, a criticality is assumed to occur in the Remote Process Cell. The assumptions concerning the radioactive release are discussed in section V on consequences. Accidental criticality in fuel receiving and storage operations is unlikely because the areas where these operations are performed are designed to be subcritical with unirradiated fuel of 5% enrichment. Light water reactor fuel is normally enriched to less than 4%. After a burnup of 30,000 MWD/tonne the enrichment is reduced to less than 3%. Fission products generated in the irradiation also contribute neutron poisons in the elements thereby further reducing their fuel worth.

Criticality in the Remote Process Cell could accidentally occur by overfilling a dissolver through a failure to

switch the diverter chute. Depending on geometry, this would have to occur in conjunction with a reduction in neutron poison and failure of monitoring instruments.

In the multi-batch dissolution process if the filter (40 mesh) fails or is eliminated it is possible to transfer plutonium fines to the dissolver transfer tank and to the accountability tank. Accidental transfer of fines to the HAF feed tank is minimized or eliminated by centrifugal clarification of the feed after it leaves the accountability tank.

Some possibility of criticality exists in the centrifuge. This is made unlikely by an interlock on excessive current to prevent further operation if it is loaded with more than 1.5 kg of insoluble fines.

Following centrifugal contacting and concentration of uranium and plutonium, criticality is inhibited by controlling the concentrations of fissile material in the solutions. Criticality could result from the formation of an excessive fissile material concentration due to out-of-specification process control coincident with the failure of the concentration monitors. As an example, the dilute aqueous plutonium stream (1BP) is continuously monitored for plutonium concentration. The density is also checked by the analysis of grab samples. In addition to these monitors, neutron monitors are mounted on the HS column (scrubber) and the 1B electropulse column (product separation) to warn of excessive plutonium concentration.

A criticality accident could occur in this part of the process from a failure of process control that results in

higher-than-normal fissile material concentrations in
solution concurrently with multiple monitoring failures or
it could result from administrative error by processing
higher enrichment fuel under specifications normally used
for lower uranium enrichment.

Criticality in product loadout is also possible. It would
be of higher consequence in the plutonium loadout area than
in the uranium loadout area. Criticality is avoided by
density control, neutron monitoring, use of equipment hav-
ing favorable geometry, and fixed neutron poisons. The
formation of plutonium polymers is controlled by keeping
the solution greater than 1 M acid. The plutonium product
is stored in product tanks of favorable geometry designed
to earthquake and tornado criteria both as to integrity
and position.

Criticality in product loadout could result from failure
of concentration control coincident with monitor failure.
It could result from plutonium polymer formation coincident
with monitor and acid control failure. It could result from
flooding of the plutonium product cell or greater than
design basis earthquake or tornado.

b.   Fires

Three postulated incidents involving dispersal of radioac-
tive material through the agency of fire have been analyzed.
One involves a contaminated solvent, assumed to contain a
substantial loading of iodine along with plutonium and
fission products. Another involves a solvent fire in the
plutonium extraction cycle. The third fire evaluated is

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assumed to occur with the ion-exchange resin during product purification.

A fire involving leached hulls (chopped cladding containing residual fuel material) was initially considered too. Experience in six years of shearing and handling the leached zirconium hulls indicated a very low probability of a major fire in this material[5]. More than 95% of the radioactivity associated with the leached hulls is induced radioactivity and is an integral part of the metal itself. In order for this radioactivity to escape, the metal itself must burn, potentially producing volatile particulates. The remainder of the radioactivity associated with the hulls is fission product and transuranic species in the form of refractory oxides. All of this radioactivity which remains with the hulls failed to be removed from the hulls during several hours of vigorous boiling in nitric acid and subsequent washing. In order for a hull fire to occur, there would have to be source of ignition of sufficient energy to initiate an active fire in the zirconium hulls which would either go undiscovered, or not respond to firefighting efforts using dry chemical extinguishing agents. There have been instances under which leached zirconium hulls have glowed but at no time has this ever affected more than a small fraction of the hulls nor has it ever resulted in anything more active than a transient glowing of the few hulls involved. The probability of such an event going unnoticed is small also, as there is an operator present at the viewing window of the hull inspection and canning station whenever they are being handled. In comparison to the hypothetical solvent fire, a zirconium fire results in lower potential off-site releases and the heat release in

such an incident is estimated to be approximately 5% that for the postulated solvent fire[36].

1.   Solvent Fires

Codecontamination is the operation which removes most of the fission products and other undesirable impurities from solutions of uranium and plutonium without separating the uranium and plutonium components. Partitioning is the operation in which the uranium is separated from the plutonium (neptunium). Solvent extraction cycles, employing 30 volume percent tributyl phosphate (TBP) in a normal paraffinic hydrocarbon (dodecane) as the water immiscible solvent, is used for these operations in conjunction with various chemical adjustments. Because of the moderate flash point of the organic solvent, 70°C, there is a potential for the occurrence of solvent fires during these processing steps due to upsets or system leaks. Operating temperatures are held below the 70°C flash point by temperature controls and flow rates are monitored to avoid spills and to maintain the desired compositions in all feed and discharge streams of the equipment used in these processing steps. This applies to the use of anion exchange resin columns or electrochemical reductions for the partitioning steps.

A solvent fire could result from a failure of temperature control which would allow the flash point to be reached. Loss of temperature control could be caused by failure of the temperature sensor, temperature control servo or mixing valve failure. Leaks or spills due to process control upsets or pipe rupture under earthquake conditions could result in a solvent fire if failure of sump level sensors to

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solvent accumulations on the cell floors occurred.

2.   Ion Exchange Resin Fires

Ion exchange resin columns are used to partition plutonium, uranium and/or neptunium into separate streams while providing for additional fission product decontamination following the initial codecontamination step. Potential autothermal resin-nitrate reaction in these ion exchange columns can be prevented in this processing step by limiting the temperature to less than 135°C and by avoiding contact of the material with oxidizing agents.

A resin fire could result from failure of process control to limit the acidity as monitored by specific gravity measurement of the feed streams. Monitoring is done continuously and alarmed on high density indication. Failure of such alarms and monitors to indicate a potentially hazardous condition could lead to a resin fire.

Failure of temperature control to maintain the resin temperature to less than 135°C could also result in a resin-nitrate reaction. Temperature regulation is performed by sensors with servo controls. Thus, anything that can upset this control, such as failure of the sensor or the servo-mechanism, could lead to excessive heating and an incident.

Low pressure steam is commonly used for heating purposes in reprocessing plants. Under upset conditions such that the process came into temperature equilibrium with the steam, a temperature of 135°C could be attained which could lead to the onset of a resin-nitrate reaction.

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If the resin beds were highly loaded with plutonium, radio-
active heating could augment the temperature of the bed.
This could occur in the plutonium purification and recovery
operation carried out in the plutonium product cell. Fail-
ure to remove the product from loaded resins in the event
of a plant shutdown would also be conducive to initiating
such an incident.

Mitigating safeguards, i.e., the use of heavy bar grids to
inhibit the expulsion of resin from the columns and pressure
relief instrumentation, are incorporated in the equipment
design to minimize the consequences of such an event.

Advancing reprocessing technology may preclude the need for
utilizing ion exchange resins in separations operations. The
occurrence of such an accident was therefore excluded from
consideration in this analysis.

c.   Explosions

Postulated incidents involving the dispersion of radioactiv-
ity by explosions in process equipment are analyzed. Ex-
plosions considered included the following:

   o   an explosion in the high aqueous feed tank (HAF)

   o   an explosion in the high aqueous waste concentra-
       tor (HAW concentrator)

   o   an explosion in the low aqueous waste concentra-
       tor (LAW concentrator)

   o   an explosion in the silver zeolite iodine adsorber

   o   an explosion in the waste calciner

   o   an explosion in the plutonium product calciner.

An explosion in the high aqueous feed tank could conceiv-
ably be caused by ignition of an explosive mixture of radio-
lytically generated hydrogen in the air above the liquid.

An explosion in the HAW or LAW concentrator could conceivably be caused by ignition of an explosive mixture of radiolytically generated hydrogen in the air above the liquid in the evaporator or a "red-oil" explosion. An explosion in the silver zeolite iodine adsorber could result from the formation of silver azide due to the presence of excess ammoniacal material in the off-gases. A waste calciner explosion could also conceivably occur from the excessive presence of hydrogen and/or "red-oil" in the equipment used in this unit operation.

An explosion in the plutonium product calciner could conceivably result from ignition of an explosive mixture of hydrogen which may be used in the process of decomposing the intermediate plutonium oxalate product or from an accelerated decomposition of moist oxalate crystalline material.

These circumstances are normally precluded from developing by means of both design and operational safety features; multiple failures of these protective systems could conceivably lead to the incidents noted.

"Red-oil" is a material that can be formed from a heavy metal nitrate, e.g., uranium, and/or nitric acid solutions mixed with tributyl phosphate solvent at temperatures exceeding 135°C[37,38]. The exact nature of the reaction of tributyl phosphate (TBP) with hot concentrated solutions of heavy metal nitrates and nitric acid has not been determined. However, under optimum conditions, the reaction becomes explosive and oxides of nitrogen are evolved. Although Purex-process conditions do not approach those

giving rise to such a reaction, safeguards are provided in the concentration of uranium, plutonium and nitric acid solutions to prevent their accidental occurrence. One method of avoiding an explosion is to keep the evaporation temperature below 135°C, the minimum temperature at which the reaction occurs. Another method is to remove the TBP from the aqueous stream prior to evaporation by steam stripping the aqueous stream.

In order for a "red-oil" explosion to occur, several independent instrument control failures and administrative failures would have to precede the occurrence. The code-contamination column would have to be out of control dumping solvent with the high aqueous waste (HAW) stream for a number of hours without being noticed or corrected; the controller for the reduction of 150 psig steam (181°C) to less than 40 psig steam (131°C) would have to malfunction; the relief valve which restricts the low pressure steam would have to fail to operate; the pressure controller on the steam to the evaporator heating coils would have to malfunction, causing steam pressure to rise above 40 psig, and not be noticed for an extended period of time, and the evaporator bottoms product would have to be substantially overconcentrated while all the normal indicators of this condition either malfunction or are ignored. Given these conditions, an explosion involving complexes of a heavy metal, TBP and nitric acid is possible.

Radiolysis of aqueous solutions results in the production of hydrogen and oxygen. Thus, all process streams would be expected to evolve hydrogen. It is estimated that 3 ft^3 of hydrogen (STP) per 10^6 BTU of fission product heat in

acid waste and 10 $ft^3/10^6$ BTU in alkaline waste is formed by radiolysis[11]. The combustion threshold of hydrogen in air occurs at 4% concentration. To avoid reaching this hydrogen concentration, dilution of the off-gases with continuously flowing air is used in the processing operations. The HAF storage tank, the HAW and LAW concentrators and the high and intermediate level liquid waste systems have the highest potential for such an explosion due to hydrogen gas accumulation as a result of failure of the air purge system. Such a failure would constitute a common mode failure to all the systems.

To reduce the likelihood of air flow failure, the plant is designed with redundant air flow features which have been discussed under special safety systems. These systems include redundant ventilation blowers with a spare blower in reserve, emergency electrical power, air reservoirs and spare air compressors.

A hydrogen explosion might occur as a result of failures in the process ventilation system such as two DOG/VOG blowers failing together or their power failing, filter blockages, ventilation control failure such that air flow balancing dampers are closed and failure of emergency compressed air which could result in a hydrogen explosion in the HLLW or ILLW storage tanks if the waste is not stored under self-boiling conditions.

The process off-gas system, composed of the Dissolver Off-Gas (DOG) and Vessel Off-Gas (VOG) systems removes iodine, particulate radioactive contaminants and nitrogen oxides which are volatilized during the fuel segment dissolution and subsequent processing steps. These off-gases are then combined with the ventilation air, filtered through two stages of HEPA off-gas filters and discharged through the stack.

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Silver zeolite sorbents are incorporated in the subsystem scrubbing trains to further restrict volatile radioiodine releases to the environment. [39]

An explosion in the waste calciner facility can be postulated on essentially the same basis as the high level waste concentrator explosion for the calciner would be fed from the HAW concentrator. The calciner, however, operates at a much higher temperature than the concentrator (>450°C). This would tend to increase the quantity of ruthenium that could be volatilized during an accident by about a factor of 10 over the amount that could be volatilized from a HAW concentrator explosion [40-43]. Aside from this variation and the fact that the equipment employed in this operation, having a small holdup which would limit the probability of such an incident, the results of a waste calciner explosion should be essentially the same as for the HAW concentrator explosion analysis.

An explosion in the plutonium product concentrator can be postulated on essentially the same basis as the high level waste concentrator explosion. The hydrogen generation rate from a plutonium product solution, however, is much lower than from a high level waste solution. Hoover and Ingalls [44] quote a hydrogen generation rate for plutonium as 0.05-0.06 ml/day/gram Pu-239 at the probable nitric acid concentration range anticipated in the evaporator concentrate. Increasing this generation rate by an order of magnitude, to reflect the higher specific activity of design basis plutonium, and assuming an equilibrium quantity of process solution of 10 liters at 200 grams per liter of plutonium, the rate of hydrogen generation in the evaporator would be about 50 ml per hour. At a volume free space in the evaporator of 38 liters, at least 30 hours would be required to reach the minimum lower explosive limit of hydrogen content, disregarding the flow of

instrument air to the probes. (This in itself would be
sufficient to keep all hydrogen generated at a concentra-
tion below the lower explosive limit for the equipment
has a small holdup capacity.)

Although the plutonium product concentration explosion
is highly improbable, it has been evaluated as an upper
limit accident for the inventory of the plutonium in the
concentrator is large - 2,000 grams or 23,000 total curies
at the time of the postulated explosion[36]. Using the same
assumptions as were employed to analyze the high activity
waste evaporator explosion, the radioactivity release
is calculated to be 1.5×10^{-2} curies.

A criticality and/or explosion accident during conversion
of plutonium nitrate to oxide powder is also considered
to be highly improbable. In this operation, plutonium
nitrate in solution from anion exchange or evaporation is
precipitated as the oxalate. This product is filtered
and washed. The wet oxalate crystals are dried at 400°C
for a fixed period and then calcined at 750°C also for a
fixed period, in a slow stream of air or hydrogen. To
ensure criticality control, fissile concentration controls
and favorable geometry with fixed poisons are employed in
the operations. Batch sizes of solids processed are always
maintained subcritical. The product powder is screened,
sampled, weighed and sealed in metal containers for subse-
quent shipment or storage.

The governing radioactive material that could be released
to the environment due to such an accident would be in a

particulate form. Decomposition of plutonium oxalate to oxide results in particles having an indicated Mass Median Diameter of 8-12 microns[45]. Oxide particles were found to be 26 to 68% of the size of their precursor. These measurements were made under laboratory conditions; plant oxide produced under accident conditions will probably be coarser. Mishima et al, report fractional releases up to 1% of the source when heating either the oxalate or partially oxidized oxalate in an upsweep of air[46]. Their finely divided, free flowing powder was composed of spheres with a Mass Median Diameter of 32 microns. As much as 0.9% of the plutonium used in the source was made airborne during a 1-hour period at temperatures up to 1000°C and air velocities up to 100 cm/sec (~2.5mph).

For the postulated accident, it is assumed that in the startup of the calcining furnace, hydrogen is introduced before the air is displaced with nitrogen, violating the procedure and resulting in a hydrogen-air explosion. The oxalate batch size being calcined is assumed to contain the equivalent of 3 kg of Pu. Overpressurization of the furnace will not be enough to destroy the furnace. Consequently, the explosion would be directed toward the ends of the furnace. The powder is assumed to be entrained at a windspeed of 20 mph. At this condition, up to 15% of the solids would be expected to be airborne into the ventilation exhaust system. The airborne powder particles' size is so large that the filtering efficiency of the three stages of HEPA filters in series would be almost 100%. A filter release factor of 8×10^{-9} is assumed[20].

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If the furnace does explode, at most the surrounding glove
box might rupture and powder would be ejected into the
room. This would create a temporary internal contamina-
tion problem in the building. However, the size of the
particles would be too large to cause a sizeable external
release of radioactivity. It is assumed a 1% release to
the exhaust ventilation system could occur. The anticipa-
ted stack releases for the cases discussed are as follows
for a Pu mixture having a specific activity of 14.1
Ci/g^{19}.

1. Furnace intact, direct release to vent system:

$$(3000 \text{ g Pu}) (0.15) (8 \times 10^{-9}) = 3.6 \times 10^{-6} \text{g Pu} \approx 5.1 \times 10^{-5} \text{ Ci}$$

2. Furnace explodes with release to room:

$$(3000 \text{g Pu}) (0.01) (8 \times 10^{-9}) = 2.4 \times 10^{-7} \text{g Pu} \approx 3.4 \times 10^{-6} \text{ Ci}$$

The off-site release from such incidents will have insig-
nificant environmental impact even under the worst disper-
sion conditions for the HEPA filters will reduce these values
by a factor of $\gg 10^{+3}$.

d.   Fuel Receiving and Storage Area Accident

An accident in the fuel receiving and storage area result-
ing in the release of radioactivity that could have an
environmental impact is a credible event. The consequences
of an uncontrolled release in this plant area could be
serious although the probability of such an occurrence
would be small. Consideration of such an event was made in
this analysis.

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Based on regulatory standards and requirements for package
design, quality assurance and handling and monitoring
procedures, the probability of a cask being breached is
low[47].

A hypothetical incident which may bound a variety of events
as to the nature and the magnitude of a release in the
fuel receiving and storage area is assumed for this eval-
uation and is described as follows:

> In shipment, it is assumed that the spent fuel
> cask loses its heat removal capability. The
> spent fuel rods self-heat due to fission pro-
> duct decay heating to a temperature approaching
> 1225°C. This causes cladding failure and re-
> lease of a large fraction of the more volatile
> fission products to the hermetically sealed, dry
> cask cavity. After receipt of the shipping cask
> at the reprocessing plant, it is transferred to
> the cask unloading pool in the fuel receiving
> and storage area where it is opened. On open-
> ing the cask, mobile radioactive species are
> expelled from the cask cavity as a stream of
> bubbles which rise to the pool surface. Some of
> the fission products released in the cask cavity
> will plate out on the cask's internal surface;
> some will remain in the pool water. The balance
> of the fission products - primarily volatile
> species - will be airborne within the building
> area and are assumed to pass through the vessel
> off-gas system. Of the fission products in the
> spent fuel cask inventory, it is assumed that

all of the noble gases, tritium, halogens,
cesium, strontium and ruthenium in the breached
elements are released to the unloading pool.[48,49]
For this evaluation, the airborne release of
the noble gases and tritium is neglected since
they are accepted as normal releases to the
atmosphere. Ninety percent of the halogens
(essentially iodine), ruthenium and cesium and
all of the strontium expelled from the cask
are conservatively assumed to remain in the pool
water. The balance are airborne and pass through
the vessel off-gas system.

Table IV-3 lists the anticipated release data for the
radioactive species involved.

e. Leakage of Fission Products From High Level
    Waste Tanks

High level wastes will be stored on an interim basis in
high integrity tanks as solutions, with the radiolytic
heat removed by heat transfer to a cooling water system.
As a safety device for the eventuality of failure of the
cooling system water supply, provision is included for
venting the storage tanks to the atmosphere via an off-gas
system designed for a total decontamination factor, boil-
ing waste to atmosphere, of at least 1×10^{10}.[36,40,41]
Because of the defenses in depth which would be operative -
high integrity design of the cooling water supply system,
tank-saucer-vault containment design, a failure and/or

TABLE IV-3[50,51]

Anticipated Releases - Fuel Receiving

And Storage Area Incident

Group	Isotope:	Ci/Tonne	Source/4.5 tonne Shipment	Fraction Released
Ru:				
	Ru-103	1.2×10^5	$5.4 (10^5)$	10%
	Ru-106	6.1×10^5	$27.4 (10^5)$	
I:				
	I-129	3.6×10^{-2}	$16.2 (10^{-2})$	10%
	I-131	1.6	7.2	
	Cs-134	2.4×10^5	$10.8 (10^5)$	10%
OFP☆	Cs-137	1.2×10^5	$5.4 (10^5)$	

☆ Other Fission Products

accident in this part of the facility which could have an environmental impact is not expected. Tank storage of high level wastes has been accomplished safely over the 25 years since they began to be generated, despite the fact that some leaks have occurred (see Appendix C). Extensive measurements at the leak locations show that the material released has remained in the vicinity of the tanks. It is anticipated that developing technology will incorporate additional capability to detect and contain any leakage that might occur.

f.   Natural Phenomena Incidents

The reprocessing plant is designed, evaluated and constructed to criteria and guidelines accepted as adequate to provide reasonable assurance that the plant could be operated without undue risk to the health and safety of the public[53]. These criteria include requirements that the structures, systems and components important to safety, be designed to withstand the effects of natural phenomena (earthquakes and tornadoes). The design bases that satisfy the natural phenomena criteria reflect:

- o Appropriate consideration of the most severe of the natural phenomena that has historically been reported for the site and the surrounding area

- o Appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena

- o The importance of the safety functions performed.

Because (1) operating and accident stresses are lower, (2) process parameters cannot change as rapidly, and (3) consequences of any failure are generally less severe for fuel

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reprocessing operations, no plant components have perform-
ance requirements that are critical to safety assurance in
the same sense as power reactor safety system response or
cooling continuity requirements. Protection against uncon-
trolled release of radioactive materials is assured by
maintaining the mechanical and structural integrity of
relatively passive confinement and off-gas treatment facil-
ities and of certain monitoring and control system compon-
ents. No violent expulsion of process materials would be
anticipated in the event of a natural phenomenon incident
as system components are not highly stressed during opera-
tion. Operating loads are accounted for in design and
construction because system components and interconnections
are generally of small size (over 90% of the process piping
is less than 2 inches in diameter) and relatively thick -
walled as a result of corrosion allowances provided. The
availability of multiple confinement barriers further
assures that radioactive materials escaping from the pri-
mary process systems or from effluent collection, treatment
or disposal facilities are monitored and controlled so as
not to result in uncontrolled releases to the environment[5-7].

An earthquake may negate some or all of the confinement of
individual systems but will have little, if any, direct
dispersive mechanisms.

The potential off-site exposures from such an event would
be much less than one percent of the accident exposure
guideline[36].

The plant structure will also protect the radioactive inven-
tory from the effect of a design basis tornado. The Class
1 Ventilation System which ventilates the cells and the

emergency cooling system for potentially self-boiling solutions will maintain the process in a safe shutdown condition. The inventories of the plant areas which have a potential for release during or subsequent to a design basis tornado would be negligible. Although the tornado has great potential for releasing radioactivity, it has great potential for dispersing the release thereby mitigating the consequences. The maximum potential off-site releases and exposures would be well within the guidelines[36].

V.   Consequences of Accidents

For the purpose of demonstrating the degree of inherent
safety of the generic reprocessing plant and its confinement
systems, the credible consequences of operational upsets
and of stresses which might be imposed by the design basis
natural phenomena were selected in evaluating the postulated
accidents. The accidents examined are those believed to
have the most severe consequences in terms of potential
exposure to the environment. These accidents are events of
low probability which are credible only if one assumes simul-
taneous failure of engineered safety features and where
pertinent, administrative procedures established as safety
barriers are bypassed.

For each accident probability sequence, there is, at that
point in the reprocessing operations, a corresponding
consequence of radioactivity dispersal beyond the primary
confinement. These source term release values have been
calculated for generic reprocessing plant accidents using
realistic values for many of the fault tree branches discussed
in Section III and Appendix B.

The types of accidents considered were discussed in Section
IV. The cases examined are listed in Table V-1. Some of
these accidents could appear in more than one location on a
fault tree or on several fault trees. The physical assump-
tions used in the analyses are generally based upon the
design of the Barnwell Nuclear Fuel Plant.[7]

TABLE V-1

Postulated Reprocessing Plant Accidents Examined

HAW Concentrator Explosion

Solvent Fire in the Plutonium Extraction Cycle

Solvent Fire in the Co-Decontamination Cycle[*]

Explosion in the LAW Concentrator

Ion Exchange Resin Fire[*]

Nuclear Criticality Incident

Explosion in the HAF Tank

Waste Calciner Explosion

Fuel Receiving and Storage Accident

[*]Accident Examined Was In The Neptunium Recovery Cycle

Accident releases might be initiated in any of the number of process cells in the reprocessing plant. Airborne releases would obviously have to pass through the plant's ventilation system prior to escaping to the environment. In the realistic calculations the following measured filter efficiencies were used in developing the source terms:[19,20]

First HEPA filter removes   99.9%

Second HEPA filter removes   99.0%

Third HEPA filter removes   94.0%

Fourth HEPA filter removes   83.0%

The accidents listed in Table V-1 are considered in detail in the following:

1.   HAW Concentrator Explosion

The assumptions and results for this accident as calculated in the Barnwell SAR[7] are given in Table V-2.

TABLE V-2

SAR Source Calculation HAW Concentrator Explosion (Barnwell)

Group*	Isotope	Ci/Tonne	X SF	X RF	X EF =	Ci Released
Ru:			1.0	1.0×10^{-3}	1.05	
	Ru-103	1.2×10^5				126
	Ru-106	6.1×10^5				640
Zr-Nb:			1.0	4.6×10^{-7}	1.05	
	Zr-95	3.5×10^5				.17
	Nb-95	6.5×10^5				.31
I:			8.3×10^{-2}	1.0	1.05	
	I-129	3.6×10^{-2}				3.2×10^{-3}
	I-131	1.6				0.14
OFPT:			1.0	4.6×10^{-7}	1.05	
	Sr-89	9.0×10^4				.043
	Sr-90	8.4×10^4				.042
	Y-90	8.4×10^4				.042
	Y-91	1.9×10^5				.094
	Cs-134	2.4×10^5				.119
	Cs-137	1.2×10^5				.060
	Ce-141	7.9×10^4				.039
	Ce-144	8.8×10^5				.434
	Pm-147	1.4×10^5				.070
	Am-241	250.				1.2×10^{-4}
	Am-242	4.0				2.1×10^{-6}
	Cm-242	4.4×10^4				.021
	Cm-243	34.0				1.7×10^{-5}
	Cm-244	5.7×10^3				.0028
Pu:			0.01	4.6×10^{-7}	1.05	
	Pu-238	4.3×10^3				2.0×10^{-5}
	Pu-239	3.2×10^2				1.5×10^{-6}
	Pu-240	6.3×10^2				2.9×10^{-6}
	Pu-241	1.7×10^5				8.3×10^{-4}
	Pu-242	3.6				1.7×10^{-8}

Release Factor:

Non-volatile fraction dispersed:

Concentrator volume = 600 liters of solution.
Cell volume = 2850 m^3; dispersion = 100 mg/m^3
Total solution dispersed = 0.285 kg of concentrate
Density of concentrate = 1.4 kg/ℓ
Fraction of concentrate dispersed = 3.3×10^{-4}
Dispersion passing through filter = 0.14 mg/m^3
Filter factor = 1.4×10^{-3}

Element	Non-Volatile Fraction Dispersed	X Filter Factor	+ Volatile Fraction	= RF
Ru	3.3×10^{-4}	1.4×10^{-3}	1×10^{-3}	1.0×10^{-3}
Zr-Nb	3.3×10^{-4}	1.4×10^{-3}	0	4.6×10^{-7}
I	0		1.0	1.0
OFPT	3.3×10^{-4}	1.4×10^{-3}	0.	4.6×10^{-7}
Pu	3.3×10^{-4}	1.4×10^{-3}	0	4.6×10^{-7}

Equivalence Factor:

EF = 600 liters of concentrate/567ℓ/MTU = 1.05 MTU

Since the HAW Concentrator is in the Remote Process Cell (RPC), there are 2 HEPA filters in series before stack release. To carry out a realistic calculation one needs the non-volatile release fraction which is (2850 m^3 cell volume)/(100mg/m^3), the denominator being a representative density.[3] This yields a release factor RF = 3.33 x 10^{-4} for non-volatiles. In addition 100% of the iodine and 0.1% of the ruthenium are volatilized. With these data the following releases are calculated.

TABLE V-3

Volatile Source From HAW Concentrator Explosion

Group	Ci/tonne	X SF	X EF	+ Volatile Fraction	Volatile Sources Ci
Ru-103	1.2E5	1	1.05	1.0E-3	1.26E2
Ru-106	6.1E5	1	1.05	1.0E-3	6.41E2
I-129	3.6E-2	8.3E-2	1.05	1.0	3.14E-3
I-131	1.6E0	8.3E-2	1.05	1.0	1.39E-1

❋ Note: 1E5 = 1 x 10^5

Table V-4

Non-Volatile Source From HAW Concentrator Explosion

Group	Isotope	Ci/Tonne	X	SF	X	EF	X	RF	No Filter Working	One Filter Working	Two Filters Working
Ru:				1.0		1.05		3.3E-4			
	Ru-103	1.2E5							4.16E1	4.16E-2	4.16E-4
	Ru-106	6.1E5							2.12E2	2.12E-1	2.12E-3
Zr-Nb				1.0		1.05		3.3E-4			
	Zr-95	3.5E5							1.21E2	1.21E-1	1.21E-3
	Nb-95	6.5E5							2.25E2	2.25E-1	2.25E-3
I:				8.3E-2		1.05		0			
	I-129	3.6E-2							0	0	0
	I-131	1.6E0							0	0	0
OFPT:				1.0		1.05		3.3E-4			
	Sr-89	9.0E4							3.12E1	3.12E-2	3.12E-4
	Sr-90	8.4E4							2.91E1	2.91E-2	2.91E-4
	Y-90	8.4E4							2.91E1	2.91E-2	2.91E-4
	Y-91	1.9E5							6.63E1	6.63E-2	6.63E-4
	Cs-134	2.4E5							8.33E1	8.33E-2	8.33E-4
	Cs-137	1.2E5							4.17E1	4.17E-2	4.17E-4
	Ce-141	7.9E4							2.74E1	2.74E-2	2.74E-4
	Ce-144	8.8E5							3.05E2	3.05E-1	3.05E-3
	Pm-147	1.4E5							4.81E1	4.81E-2	4.81E-4
	Am-241	250.0E0							8.67E-2	8.67E-5	8.67E-7
	Am-242	4.0E0							1.39E-3	1.39E-6	1.39E-8
	Cm-242	4.4E4							1.52E1	1.52E-2	1.52E-4
	Cm-243	34.0E0							1.18E-2	1.18E-5	1.18E-7
	Cm-244	5.7E3							1.98E-0	1.98E-3	1.98E-5
Pu:				0.01		1.05		3.3E-4			
	Pu-238	4.3E3							1.49E0	1.49E-3	1.49E-5
	Pu-239	3.2E2							1.11E-1	1.11E-4	1.11E-6
	Pu-240	6.3E2							2.18E-1	2.18E-4	2.18E-6
	Pu-241	1.7E5							5.9E1	5.9 E-2	5.9 E-4
	Pu-242	3.6E0							1.25E-3	1.25E-6	1.25E-8

✩ Note: $1.2E5 = 1 \times 10^5$

2.   Solvent Fire in Pu Extraction Cycle

The Barnwell SAR[7] calculation is given below.

TABLE V-5

SOLVENT FIRE IN THE Pu SOLVENT EXTRACTION CYCLE

Group	Isotope:*	Ci/Tonne	X	SF	X	RF	X	EF	= Ci Release
Pu:									
	Pu-238	4.3×10^3		1.0		1×10^{-4}		0.034	0.015
	Pu-239	3.2×10^2		1.0		1×10^{-4}		0.034	0.001
	Pu-240	6.3×10^2		1.0		1×10^{-4}		0.034'	0.002
	Pu-241	1.7×10^5		1.0		1×10^{-4}		0.034	0.58
	Pu-242	3.6		1.0		1×10^{-4}		0.034	1.3×10

* Other radioisotopes are considered to be negligible relative to plutonium.

Equivalence Factor:

   · Organic consumed = 14 liters
   Processing rate = 0.208 MTU/hr
   3AP normal flow = 85 ℓ/hr = 409 ℓ/MTU

$$EF = \frac{14\ \ell}{409\ \ell/MTU} = 0.034\ MTU$$

Release Factor:

   Pu dispersed = 1%
   Filter release = 1%
   $RF = (.01)(.01) = 1 \times 10^{-4}$

This incident would occur in the Plutonium Product Cell (PPC). There are 3 HEPA filters in series before stack release. The calculation yielded the following data:

TABLE V-6

Solvent Fire in the Pu Extraction Cycle

							Stack Release Curies		
Group	Isotope:*	Ci/Tonne	X SF	X EF	Fraction Dispersed	No Filter Working	One Filter Working	Two Filters Working	Three Filters Working
Pu:									
	Pu-238	4.3×10^3	1.0	0.034	.01	1.46E0☆	1.46E-3	1.46E-5	8.76E-7
	Pu-239	3.2×10^2	1.0	0.034	.01	1.09E-1	1.09E-4	1.09E-6	6.54E-8
∞∞	Pu-240	6.3×10^2	1.0	0.034	.01	2.14E-1	2.14E-4	2.14E-6	1.28E-7
	Pu-241	1.7×10^5	1.0	0.034	.01	5.77E1	5.77E-2	5.77E-4	3.46E-5
	Pu-242	3.6	1.0	0.034	.01	1.22E-3	1.22E-6	1.22E-8	7.32E-10

☆ $1.46E0 = 1.46 \times 10^0$

The Barnwell[7] Calculation is given in Table V-7.

TABLE V-7 - SAR Results

Group°	Isotope	Ci/Tonne	x	DF°°	x	SF	x	RP	x	EP	=	Ci Released
Ru:						0.01		0.10		0.085		
	^{106}Ru	1.2×10^9		0.16									1.69
	^{103}Ru	6.1×10^8		0.73									38
Zr-Nb:						0.01		1×10^{-6}		0.085		
	^{95}Zr	3.5×10^8		0.14									4.1×10^{-3}
	^{95}Nb	6.5×10^8		0.14									7.7×10^{-3}
I:						0.04		1.0		0.085		
	^{129}I	3.6×10^{-3}		1.0									1.2×10^{-5}
	^{131}I	1.6		0.032									1.8×10^{-4}
OFPT:						0.001		1×10^{-6}		0.085		
	^{90}Sr	9.0×10^6		0.20									1.5×10^{-6}
	^{90}Sr	8.4×10^6		1.0									7.2×10^{-6}
	^{90}Y	8.4×10^6		1.0									7.2×10^{-6}
	^{91}Y	1.9×10^8		0.23									3.6×10^{-4}
	^{144}Ce	2.4×10^8		0.79									1.7×10^{-3}
	^{143}Ce	1.2×10^8		0.96									1.0×10^{-3}
	^{141}Ce	7.9×10^6		0.13									8.4×10^{-5}
	^{144}Pm	8.8×10^8		0.66									5.0×10^{-3}
	^{147}Pm	1.4×10^8		0.90									1.1×10^{-3}
	^{241}Am	290.		1.0									2.1×10^{-6}
	^{242}Am	4.0		1.0									3.4×10^{-8}
	^{242}Cm	4.4×10^6		0.43									1.6×10^{-6}
	^{244}Cm	34.0		0.99									2.9×10^{-7}
	^{246}Cm	5.7×10^8		1.0									4.8×10^{-3}
Pu:						0.01		1×10^{-6}		0.085		
	^{238}Pu	4.3×10^8		1.0									3.6×10^{-4}
	^{239}Pu	3.2×10^8		1.0									2.7×10^{-3}
	^{240}Pu	6.3×10^8		1.0									5.4×10^{-8}
	^{241}Pu	1.7×10^8		1.0									1.5×10^{-2}
	^{242}Pu	3.6		1.0									2.7×10^{-7}

°OFPT = Other fission products and transuranium elements.
°°DF = Decay Factor; corrects for simultaneous accumulation and decay of stored waste
for one year before Np recovery campaign.

Release Factor:

99% filter efficiency.
1% of contaminants in burned organic dispersed.
10% of Ru and 100% of I volatilizes and passes out the stack.

Element	Fraction Dispersed	x	Fraction Passing Filter	+	Volatile Fraction	=	RP
Ru	0.01		0.01		0.10		0.10
Zr-Nb	0.01		.01		-		1×10^{-6}
I	-		-		1.0		1.0
OFPT	0.01		0.01		-		1×10^{-6}
Pu	0.01		0.01		-		1×10^{-6}

Equivalence Factor:

Area of the HILC = 1080 ft^2.
Total burned = 100 gallons = 378 liters.
HAF = 2260 liters/MTU during Np campaign.
HAF/HAX = 0.51 during Np campaign.
HAX = HAF = 4431 liters/MTU.

$$EF = \frac{378 \text{ liters}}{4431 \text{ liters/MTU}} = 0.085 \text{ MTU}$$

This incident takes place in the High Intermediate Level Cell (HILC). There are 2 HEPA filters before stack release. The volatile releases are given in Table V-8 and the non-volatile releases are given in Table V-9.

TABLE V-8

VOLATILE RELEASES

Group	Isotope	Ci/Tonne	X	DF	X	SF	X	EF	Volatile Fraction	Stack Release Ci
Ru:						0.01				
	Ru-103	1.2×10^5		0.16				0.085	0.1	1.63E0*
	Ru-106	6.1×10^5		0.73						3.78E1
I:						0.04		0.085	1.0	
	I-129	3.6×10^{-2}		1.0						1.22E-4
	I-131	1.6		0.032						1.74E-4

* 1.63E0 = 1.63 x 10^0

TABLE V-9

NON-VOLATILE RELEASES

										Stack Release Ci		
Group	Isotope	Ci/Tonne	X	DF**	X	SF	X	EF	Fraction Dispersed	No Filter Works	One Filter Works	Two Filters Works
Ru:						0.01		0.085		1.63E-1	1.63E-4	1.63E-6
	Ru-103	1.2×10^5		0.16					.01	1.63E-1	1.63E-4	1.63E-6
	Ru-106	6.1×10^5		0.73						3.78E0	3.78E-3	3.78E-5
Zr-Nb:						0.01		0.085				
	Zr-95	3.5×10^5		0.14					.01	4.17E-1	4.17E-4	4.17E-6
	Nb-95	6.5×10^5		0.14						7.74E-1	7.74E-4	7.74E-6
I:						0.04		0.085				
	I-129	3.6×10^{-2}		1.0					0	0	0	0
	I-131	1.6		0.032						0	0	0
OFPT:						0.001		0.085				
	Sr-89	9.0×10^4		0.20					.01	1.53E-2	1.53E-5	1.53E-7
	Sr-90	8.4×10^4		1.0						7.13E-2	7.13E-5	7.13E-7
	Y-90	8.4×10^4		1.0						7.13E-2	7.13E-5	7.13E-7
	Y-91	1.9×10^5		0.23						3.71E-2	3.71E-5	3.71E-7
	Cs-134	2.4×10^5		0.79						1.61E-1	1.61E-4	1.61E-6
	Cs-137	1.2×10^5		0.96						9.78E-2	9.78E-5	9.78E-7
	Ce-141	7.9×10^4		0.13						8.74E-3	8.74E-6	8.74E-8
	Ce-144	8.8×10^5		0.66						4.94E-1	4.94E-4	4.94E-6
	Pm-147	1.4×10^5		0.90						1.07E-1	1.07E-4	1.07E-6
	Am-241	250.		1.0						2.12E-4	2.12E-7	2.12E-9
	Am-242	4.0		1.0						3.4 E-6	3.4 E-9	3.4 E-11
	Cm-242	4.4×10^4		0.43						1.6 E-2	1.6 E-5	1.6 E-7
	Cm-243	34.0		0.99						2.86E-5	2.86E-8	2.86E-10
	Cm-244	5.7×10^3		1.0						4.84E-3	4.84E-6	4.84E-8
Pu:						0.01		0.085		3.65E-2	3.65E-5	3.65E-7
	Pu-238	4.3×10^3		1.0					.01	2.72E-3	2.72E-6	2.72E-8
	Pu-239	3.2×10^2		1.0						5.35E-3	5.35E-6	5.35E-8
	Pu-240	6.3×10^2		1.0						1.44E0	1.44E-3	1.44E-5
	Pu-241	1.7×10^5		1.0						3.06E-5	3.06E-8	3.06E-10
	Pu-242	3.6		1.0								

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4. Explosion in the LAW Concentrator

The Barnwell[7] calculation is given in Table V-10.

TABLE V-10 - SAR Results

Group*	Isotope:	Ci/Tonne	×	SF	×	dF	×	LF	=	Ci Released
Ru:				0.02		0.001		3.5		
	Ru-103	1.2×10^5								8.6
	Ru-106	6.1×10^5								43.0
Zr-Nb:				0.02		1.1×10^{-7}		3.5		
	Zr-95	3.5×10^5								.0028
	Nb-95	6.5×10^5								.0052
I:				0.032		1.0		3.5		
	I-129	3.6×10^{-2}								4.0×10^{-3}
	I-131	1.6								0.18
OFPT:				0.002		1.1×10^{-7}		3.5		
	Sr-89	9.0×10^4								7.0×10^{-5}
	Sr-90	8.4×10^4								6.6×10^{-5}
	Y-90	8.4×10^4								6.6×10^{-5}
	Y-91	1.9×10^5								1.5×10^{-4}
	Cs-134	2.4×10^5								1.8×10^{-4}
	Cs-137	1.2×10^5								9.4×10^{-5}
	Ce-141	7.9×10^4								6.2×10^{-5}
	Ce-144	8.8×10^5								6.9×10^{-4}
	Pm-147	1.4×10^5								1.1×10^{-4}
	Am-241	250.								Nil
	Am-242	4.0								Nil
	Cm-242	4.4×10^4								3.5×10^{-5}
	Cm-243	34.0								2.7×10^{-7}
	Cm-244	5.7×10^3								4.5×10^{-6}
Pu:				9×10^{-4}		1.1×10^{-7}		3.5		
	Pu-238	4.3×10^3								1.5×10^{-6}
	Pu-239	3.2×10^2								1.1×10^{-7}
	Pu-240	6.3×10^2								2.2×10^{-7}
	Pu-241	1.7×10^5								6.0×10^{-5}
	Pu-242	3.6								Nil

*OFPT = Other fission products and transuranium elements.

Release factor:

Non-volatile fraction dispersed:

Concentrator volume = 1500 liters of concentrate
Cell volume = 1510 m^3
Aerosol content = 100 mg/m^3
Concentrate dispersed = 1.5×10^5 mg = 0.15 kg
Concentrate density = 1.3 kg/ℓ
Non-volatile fraction dispersed = 8×10^{-5}
Dispersion passing through filter = 0.14 mg/m^3
Filter factor = 1.4×10^{-3}

Element	Non-Vol. Fraction Dispersed	×	Filter Factor	+	Volatile Fraction	=	RF
Ru	8×10^{-5}		1.4×10^{-3}		.001		0.001
Zr-Nb	8×10^{-5}		1.4×10^{-3}		0		1.1×10^{-7}
I					1.0		1.0
OFPT	8×10^{-5}		1.4×10^{-3}		0		1.1×10^{-7}
Pu	9×10^{-5}		1.4×10^{-3}		0		1.3×10^{-7}

Equivalence factor:

Concentrate = 90 ℓ/hr = 4.17 ℓ/MTF (does not recycle Pu fuel)
EF = 1.4 MTU

The LAW concentrator is in the High Intermediate Level Cell
(HILC) and there are 2 HEPA filters prior to stack release.
Table V-11 lists the volatile releases and Table V-12
lists the non-volatile releases.

TABLE V-11

VOLATILE RELEASES

Group	Isotope	Ci/Tonne	X	SF	X	EF	Volatile Fraction	Stack Release Ci
Ru:				0.02		3.5	.001	
	Ru-103	1.2×10^5						8.4E0
	Ru-106	6.1×10^5						4.27E1
I:				0.032		3.5	1.0	
	I-129	3.6×10^{-2}						4.03E-3
	I-131	1.6						1.79E-1

$8.4E0 = 8.4 \times 10^0$

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TABLE V-12

NON-VOLATILE RELEASES

Group*	Isotope:	Ci/Tonne	X	SF	X	EF	Fraction Dispersed	Stack Release Ci No Filter Works	One Filter Works	Two Filters Works
Ru:				0.02		3.5	8.0E-5☆			
	Ru-103	1.2×10^5						6.72E-1	6.72E-4	6.72E-6
	Ru-106	6.1×10^5						3.42E0	3.42E-3	3.42E-5
Zr-Nb:				0.02		3.5	8.0E-5			
	Zr-95	3.5×10^5						1.96E0	1.96E-3	1.96E-5
	Nb-95	6.5×10^5						3.64E0	3.64E-3	3.64E-5
I:				0.032		3.5	0			
	I-129	3.6×10^{-2}						0	0	0
	I-131	1.6						0	0	0
OFPT:				0.002		3.5	8.0E-5			
	Sr-89	9.0×10^4						5.02E-2	5.02E-5	5.02E-7
	Sr-90	8.4×10^4						4.7 E-2	4.7 E-5	4.7 E-7
	Y-90	8.4×10^4						4.7 E-2	4.7 E-5	4.7 E-7
	Y-91	1.9×10^5						1.06E-1	1.06E-4	1.06E-6
	Cs-134	2.4×10^5						1.34E-1	1.34E-4	1.34E-6
	Cs-137	1.2×10^5						6.72E-2	6.72E-5	6.72E-7
	Ce-141	7.9×10^4						4.42E-2	4.42E-5	4.42E-7
	Ce-144	8.8×10^5						4.83E-1	4.83E-4	4.83E-6
	Pm-147	1.4×10^5						7.84E-2	7.84E-5	7.84E-7
	Am-241	250.						1.40E-4	1.40E-7	1.40E-9
	Am-242	4.0						2.24E-6	2.24E-9	2.24E-11
	Cm-242	4.4×10^4						2.46E-2	2.46E-5	2.46E-7
	Cm-243	34.0						1.90E-5	1.90E-8	1.90E-10
	Cm-244	5.7×10^3						3.91E-3	3.19E-6	3.19E-8
Pu:				9×10^{-4}		3.5	8.0E-5			
	Pu-238	4.3×10^3						1.08E-3	1.08E-6	1.08E-8
	Pu-239	3.2×10^2						8.06E-5	8.06E-8	8.06E-10
	Pu-240	6.3×10^2						1.59E-4	8.06E-8	8.06E-10
	Pu-241	1.7×10^5						4.28E-2	4.28E-5	4.28E-7
	Pu-242	3.6						9.07E-7	9.07E-10	9.07E-12

☆ 8.0E-5 = 8.0×10^{-5}

5. Ion Exchange Resin Fire (Neptunium Recovery Campaign)

The Barnwell[7] Calculation is given in Table V-13.

TABLE V-13

SAR Results

Isotope	Source Activity, Ci/tonne	x	SF	x	RF	x	EF =	Activity Released, Ci
Ru-103	1.24×10^5		3.0×10^{-6}		5.1×10^{-2}		5	9.49×10^{-2}
Ru-106	7.22×10^5		3.0×10^{-6}		5.1×10^{-2}		5	0.552
Zr-95	3.58×10^5		6.6×10^{-6}		1.0×10^{-3}		5	1.18×10^{-2}
Nb-95	6.82×10^5		6.6×10^{-6}		1.0×10^{-3}		5	2.25×10^{-2}
I-129	3.55×10^{-2}		3.1×10^{-7}		.50		5	2.75×10^{-8}
I-131	1.6		3.1×10^{-7}		.50		5	1.24×10^{-6}
Sr-89	8.2×10^4		1.0×10^{-8}		1.0×10^{-3}		5	4.10×10^{-6}
Sr-90	8.2×10^4		1.0×10^{-8}		1.0×10^{-3}		5	4.10×10^{-6}
Cs-134	1.79×10^5		1.0×10^{-8}		1.0×10^{-3}		5	8.95×10^{-6}
Cs-137	1.25×10^5		1.0×10^{-8}		1.0×10^3		5	6.25×10^{-6}
Ba-137m	1.10×10^5		1.0×10^{-8}		1.0×10^{-3}		5	5.50×10^{-6}
Ce-144	5.6×10^5		1.0×10^{-8}		1.0×10^{-3}		5	2.80×10^{-5}
Cm-242	4.0×10^4		1.0×10^{-8}		1.0×10^{-3}		5	2.00×10^{-6}
Cm-244	4.9×10^3		1.0×10^{-8}		1.0×10^{-3}		5	2.45×10^{-7}
Np-238	6.38		0.95		1.0×10^{-3}		5	3.03×10^{-2}
Pu-238	4.37×10^3		5.0×10^{-4}		1.0×10^{-3}		5	1.09×10^{-2}
Pu-239	3.2×10^2		5.0×10^{-4}		1.0×10^{-3}		5	8.00×10^{-4}
Pu-240	6.3×10^2		5.0×10^{-4}		1.0×10^{-3}		5	1.58×10^{-3}
Pu-241	1.6×10^5		5.0×10^{-4}		1.0×10^{-3}		5	0.40

Element	Non-Volatile Fraction Passing Filter	+	Volatile Fraction	=	RF
Ru	9.0×10^{-4}		.05		5.1×10^{-2}
Zr-Nb	1.0×10^{-2}		0		1.0×10^{-3}
I	0		.50		.50
Sr, Ce, Cs, Ba, Cm	1.0×10^{-3}		0		1.0×10^{-3}
Pu	1.0×10^{-3}		0		1.0×10^{-3}
Np	1.0×10^{-3}		0		1.0×10^{-3}

Equivalence Factor (EF)

$EF = (5 MTU/day)(1\ day) = 5 MTU$

The accident occurs in the Plutonium Product Cell (PPC) where there are three HEPA's in series before stack release. 10% of the Ru and 100% of the I are volatilized during the burning. The results are given below for the volatile and non-volatile releases. These results are included for completeness. They are not included in risk assessment considerations since it is anticipated that this unit operation will not be used in the reprocessing plants under consideration.

TABLE V-14

VOLATILE SOURCE

Isotope	Source Activity Ci/tonne	X	SF	X	EF	Volatile Fraction	Stack Release Ci
Ru-103	1.24×10^5		3.0×10^{-6}		5	.05	9.3E-2*
Ru-106	7.22×10^5		3.0×10^{-6}		5	.05	5.42E-1
I-129	3.55×10^{-2}		3.1×10^{-7}		5	.5+	2.75E-8
I-131	1.6		3.1×10^{-7}		5	.5	1.24E-6

* $9.3E-2 = 9.3 \times 10^{-2}$
+ Reference 5

TABLE V-15   NON-VOLATILE SOURCE

Isotope	Source Activity Ci/tonne	SF	EF	Fraction Dispersed	No Filter Works	One Filter Works	Stack Release Ci Two Filters Work	Three Filters Work
Ru-103	1.24×10^5	3.0×10^{-6}	5	.45	8.37E-1☆	8.37E-4	8.37E-6	5.02E-7
Ru-106	7.22×10^5	3.0×10^{-6}	5	.45	4.87EO	4.87E-3	4.87E-5	2.92E-6
Zr-95	3.58×10^5	6.6×10^{-6}	5	.5	5.91EO	5.91E-3	5.91E-5	3.54E-6
Nb-95	6.82×10^5	6.6×10^{-6}	5	.5	1.13EO	1.13E-2	1.13E-4	6.78E-6
I-129	3.55×10^{-2}	3.1×10^{-7}	5	0	0	0	0	0
I-131	1.6	3.1×10^{-7}	5	0	0	0	0	0
Sr-89	8.2×10^4	1.0×10^{-8}	5	.5	2.05E-3	2.05E-6	2.05E-8	1.23E-9
Sr-90	8.2×10^4	1.0×10^{-8}	5	.5	2.05E-3	2.05E-6	2.05E-8	1.23E-9
Cs-134	1.79×10^5	1.0×10^{-8}	5	.5	4.48E-3	4.48E-6	4.48E-8	2.69E-9
Cs-137	1.25×10^5	1.0×10^{-8}	5	.5	3.13E-3	3.13E-6	3.13E-8	1.88E-9
Ba-137m	1.10×10^5	1.0×10^{-8}	5	.5	2.75E-3	2.75E-6	2.75E-8	1.65E-9
Ce-144	5.6×10^5	1.0×10^{-8}	5	.5	1.40E-2	1.40E-5	1.40E-7	8.40E-9
Cm-242	4.0×10^4	1.0×10^{-8}	5	.5	1.0 E-3	1.0 E-6	1.0 E-8	6.0 E-10
Cm-244	4.9×10^3	1.0×10^{-8}	5	.5	1.18E-4	1.18E-7	1.18E-9	7.08E-11
Np-238	6.38	0.95	5	.5	1.52E1	1.52E-2	1.52E-4	9.12E-6
Pu-238	4.37×10^3	5.0×10^{-4}	5	.5	5.47EO	5.47E-3	5.47E-5	3.28E-6
Pu-239	3.2×10^2	5.0×10^{-4}	5	.5	4.0E-1	4.0E-4	4.0E-6	2.40E-7
Pu-240	6.3×10^2	5.0×10^{-4}	5	.5	7.88E-1	7.88E-4	7.88E-6	4.37E-7
Pu-241	1.6×10^5	5.0×10^{-4}	5	.5	2.0E2	2.0E-1	2.0E-3	1.20E-4

☆ 8.37E-1 = 8.37×10^{-1}

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6. Nuclear Criticality Incident

The Barnwell[7] results for a criticality incident in the Remote Process Cell (RPC) are given in Table V-16.

TABLE V-16

SAR Results
NUCLEAR CRITICALITY INCIDENT (10^{10} Fissions)

Isotope	Activity Released,* Ci@t=0
I-131(8.05d)	0.75
I-132(2.4h)	3.30
I-133(20.5h)	18.0
I-134(52.5m)	450.
I-135(6.68hr)	48.0
Xe-135m(15m)	395.
Xe-138(17m)	1050.
Kr-87(1.3h)	112.
Kr-83m(1.86h)	13.5
Kr-88(2.8h)	69.5
Kr-85m(4.4h)	18.5
Xe-135(9.2h)	36.4
Xe-133m(2.3d)	0.20
Xe-133(5.27d)	2.70
Xe-131m(12.0d)	6.0×10^{-2}
Kr-85(10.4y)	2.0×10^{-3}

*The iodine is assumed to be volatile. Of the amount generated, 1% is assumed to be released to the vent system, the rest remaining in solution.

7.   Explosion in the HAF Tank

This accident is assumed to occur in the High Level Cell
(HLC). The consequences of this accident have been evaluated
assuming that there is only one HEPA filter before stack release.
The release data are given in Tables V-17 and V-18.

    Note: It is more likely that for present and future plant
           designs that at least two filters will be in line
           between the HLC and the stack.

TABLE V-17

VOLATILE SOURCE

Group	Isotope	Ci/Tonne	Split Factor	Equivalence Factor	Volatile Fraction	Stack Release Ci
Ru:			1.0	4	.001	
	Ru-103	1.2×10^5				4.8E2*
	Ru-106	6.1×10^5				2.44E3
I:			0.1	4	1.0	
	I-129	3.6×10^{-2}				1.4E-2
	I-131	1.6				6.4E-1

*4.8E2 = 4.8×10^2

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TABLE V-18 NON-VOLATILE SOURCE

Group	Isotope	Ci/Tonne	Split Factor	Equivalence Factor	Fraction Dispersed	Stack Release Ci No Filter Works	One Filter Works
Ru:			1.0	4	5.9E-6☆		
	Ru-103	1.2×10^5				2.83E0	2.83E-3
	Ru-106	6.1×10^5				1.44E1	1.44E-2
ZR-Nb:			1.0	4	5.9E-6		
	Zr-95	3.5×10^5				8.26E0	8.26E-3
	Nb-95	6.5×10^5				1.53E1	1.53E-2
I:			0.5	4	0		
	I-129	3.6×10^{-2}				0	
	I-131	1.6				0	
OFPT:			1.0	4	5.9E-6		
	Sr-89	9.0×10^4				2.12E0	2.12E-3
	Sr-90	8.4×10^4				1.98E0	1.98E-3
	Y-90	8.4×10^4				1.98E0	1.98E-3
	Y-91	1.9×10^5				4.48E0	4.48E-3
	Cs-134	2.4×10^5				5.66E0	5.66E-3
	Cs-137	1.2×10^5				2.83E0	2.83E-3
	Ce-141	7.9×10^4				1.86E0	1.86E-3
	Ce-144	8.8×10^5				2.08E1	2.08E-2
	Pm-147	1.4×10^5				3.31E0	3.31E-3
	Am-241	250.				5.90E-3	5.90E-6
	Am-242	4.0				9.45E-5	9.45E-8
	Cm-242	4.4×10^4				1.04E0	1.04E-3
	Cm-243	34.0				8.03E-4	8.04E-7
	Cm-244	5.7×10^3				1.35E-1	1.35E-4
Pu:			1.0	4	5.9E-6		
	Pu-238	4.3×10^3				1.02E-1	1.02E-4
	Pu-239	3.2×10^2				7.55E-3	7.55E-6
	Pu-240	6.3×10^2				1.49E-2	1.49E-5
	Pu-241	1.7×10^5				4.01E0	4.01E-3
	Pu-242	3.6				8.50E-5	8.50E-8

☆ $5.9E-6 = 5.9 \times 10^{-6}$

8.   Waste Calciner Explosion

The calciner would be fed from the High Aqueous Waste (HAW)
concentrator. It would probably be located in the same
cell area, which in the case of Barnwell, is the Remote
Process Cell (RPC). The calciner operates at several
hundred degrees Celsius and this would raise the amount
of Ru volatilized during an accident by about a factor of
10 over the HAW concentrator explosion results. Aside
from this variation, the results of a waste calciner ex-
plosion would be essentially what they were in the HAW
concentrator explosion analysis.

9.   Fuel Receiving and Storage Accident

Incidents that release radioactivity in the receiving and
storage area are varied. A scenario which may bound many
similar incidents in the nature and magnitude of the release
was chosen for evaluation.

While being shipped the spent fuel cask has lost its heat
removal capability. The fuel rods fail the clad and release
a large fraction of the more volatile fission products.
When the cask is opened in the receiving and storage area
it will be submerged in water. The cask interior may be
dry and thus, much of the fission activity leaves the cask
in a large stream of bubbles that rise to the surface. Some
of the fission products will have plated out on the cask
interior walls and some will remain in the pool water. The
released products will either enter the Fuel Receiving and
Storage area (FRS) atmosphere or they may enter the DOG/VOG
system.

Table V-19 lists some expected release data for Ru and I.
The tritium and noble gas releases are neglected as being
part of the normally accepted release. The material quan-
tities used were taken from the data for rail shipments
given in Table V-20.

TABLE V-19

Expected Release Data[50,51]

Group	Isotope:	Ci/Tonne	Source/4.5 tonne Shipment train	Building Fraction Released	Ci Released
Ru:					
	Ru-103	1.2×10^5	5.4×10^5	10%	5.4E4
	Ru-106	6.1×10^5	27.4×10^5		2.74E5
I:					
	I-129	3.6×10^{-2}	16.2×10^{-2}	10%	1.62E-2
	I-131	1.6	7.2		7.2 E-1
OFP*	Cs-134	2.4×10^5	10.8×10^5	1%	1.08E4
	Cs-137	1.2×10^5	5.4×10^5		5.4E3

*OFP = Other Fission Products

TABLE V-20

Volatile Isotope Activity in Spent Fuel Shipments[50]

Isotope	Type of Shipment	MTU per Shipment	Total Activity per Shipment	% Released To Plenum	Activity In Plenum
Kr-85	Truck	0.45	4350	17	740
H-3	Truck		187	1	1.87
I-131	Truck		0.64	2.3	0.015
Kr-85	Rail	4.5	43,500	17	7400
H-3	Rail		1,870	1	18.7
I-131	Rail		6.4	2.3	0.15
Kr-85	Water	18	174,000	17	29,600
H-3	Water		7,480	1	75
I-131	Water		25.6	2.3	0.59

The probability of cask heat transfer failure during shipment was estimated at 10^{-3} to 10^{-4} per shipment. With approximately 250 shipments per year, this became 2.5×10^{-1} to 2.5×10^{-2} per year likelihood. We chose 10^{-1} per year for a 4.5 tonne delivery of failed fuel.

Prior to the cask being opened it was assumed that 100% of the I, 10% of the Ru and 1% of the Cs was volatile. The amount that rises through the pool after the cask is opened under water was assumed as 10% of what was initially volatile for the Ru and I and 0.1% for the Cs.

The released gases subsequently pass through an iodine scrubber (VOG) and then through 2 HEPA filters. It was assumed that 7% of the Ru, 0.01% of the I and 0.1% of the Cs would pass through the iodine scrubber. It was also assumed that the passage through the iodine scrubber would cause the volatiles to become non-volatile. Hence the HEPA filter failure probabilities are taken as 10^{-2} and 10^{-3} per demand, as previously noted. Table V-21 presents the results. Noble gases were considered as normal releases.

TABLE V-21

FUEL RECEIVING AND STORAGE ACCIDENT

Isotope	Ci in a 4.5 tonne Shipment		Amt. in Cask Atmosphere Before Release		Amt. in FRSS Atmos.after Cask is Opened		Amt.After Iodine Scrubber	Amt.After 1 Filter	Amt. After 2 Filter
Ru 103	5.4×10^5	(.10)☆	5.4E4	(.1)☆	5.4E3	(.07)☆	3.78E2	3.78E-1	3.78E-3
Ru 106	2.74×10^6	(.10)	2.74E5	(.1)	2.74E4	(.07)	1.92E3	1.92E0	1.92E-2
I 129	1.62×10^{-1}	(1.0)	1.62E-1	(.1)	1.62E-2	(10^{-4})	1.62E-6	1.62E-6	1.62E-6
I 131	7.2	(1.0)	7.2E0	(.1)	7.2E-1	(10^{-4})	7.2E-5	7.2E-5	7.2E-5
Cs 134	1.08×10^6	(.01)	1.08E4	(.001)	1.08E1	(10^{-3})	1.08E-2	1.08E-5	1.08E-7
Cs 137	5.4×10^5	(.01)	5.4E3	(.001)	5.4E0	(10^{-3})	5.4E-3	5.4E-6	5.4E-8

☆ Fraction available for release

VI.   Risk Assessment

1.   Release Likelihood Spectra

To quantify the risk from a generic reprocessing plant re-
quires the synthesis of accident likelihoods and their
consequences. In Section 3, a number of fault trees keyed
to each process cell were constructed. While these fault
trees are generic in nature they do indicate the complica-
tions that are required in order to have an accident and
also indicate the probable likelihood of such an event. In
Section 5, the consequences from each of these accident
sequences identified in the fault trees are evaluated. In
this section, the results of Sections 3 and 5 are combined
to produce a spectrum of release likelihood curves for a
variety of isotopes.

Since there are 10 to 15 accidents that were considered,
an alphabetical code is employed in plotting up the data for
accident identification. This alphabetical code is given
in Table VI-1. Also shown in Table VI-1 are the number of
HEPA filters normally found between the process cell of
interest and the final exhaust stack. Both the process
cell and the fault trees that were utilized to evaluate
the likelihood of the incident occurring in the cell are
given in the table as well as the probability of occurrence
of the incident.

Figures VI-1 through VI-5 are plots of the data for the
classes of isotopes considered. Iodine and ruthenium are

TABLE VI-1

ACCIDENTS CONSIDERED IN PRESENT ANALYSIS

ACCIDENT	LOCATION	NO. OF HEPA FILTERS	TREES	LIKELIHOOD/YEAR
A. HAW Conc Explosion	RPC	2	RPC, S, RO	10^{-5}
B. Solvent Fire Pu Extraction Cycle	PPC	3	PPC, SF	10^{-6}
C. Solvent Fire Codecontamination Cycle	HILC	2	HILC, SF	10^{-6}
D. LAW Conc Explosion	ILC	2	ILC, S, RO	10^{-4}
E. Ion Exchange Resin Fire (Not Plotted)	PPC	3	PPC, IER	10^{-4}
F. Criticality Incident	Various (RPC Typical)	–2	RPC, CP	10^{-5}
G. HAF Tank Explosion	HLC	1*	HLC, H	10^{-5}
H. Waste Calciner Explosion	(RPC)	2	RPC, S, RO, H	10^{-6}
I. Fuel Receiving and Storage Accident	FRSS		FRS	10^{-1}

* The analysis presented in this report assumes that there is only one HEPA filter between HLC and stack. However, it is more likely that for present and future plant designs that at least two filters will be in line between HLC and the stack.
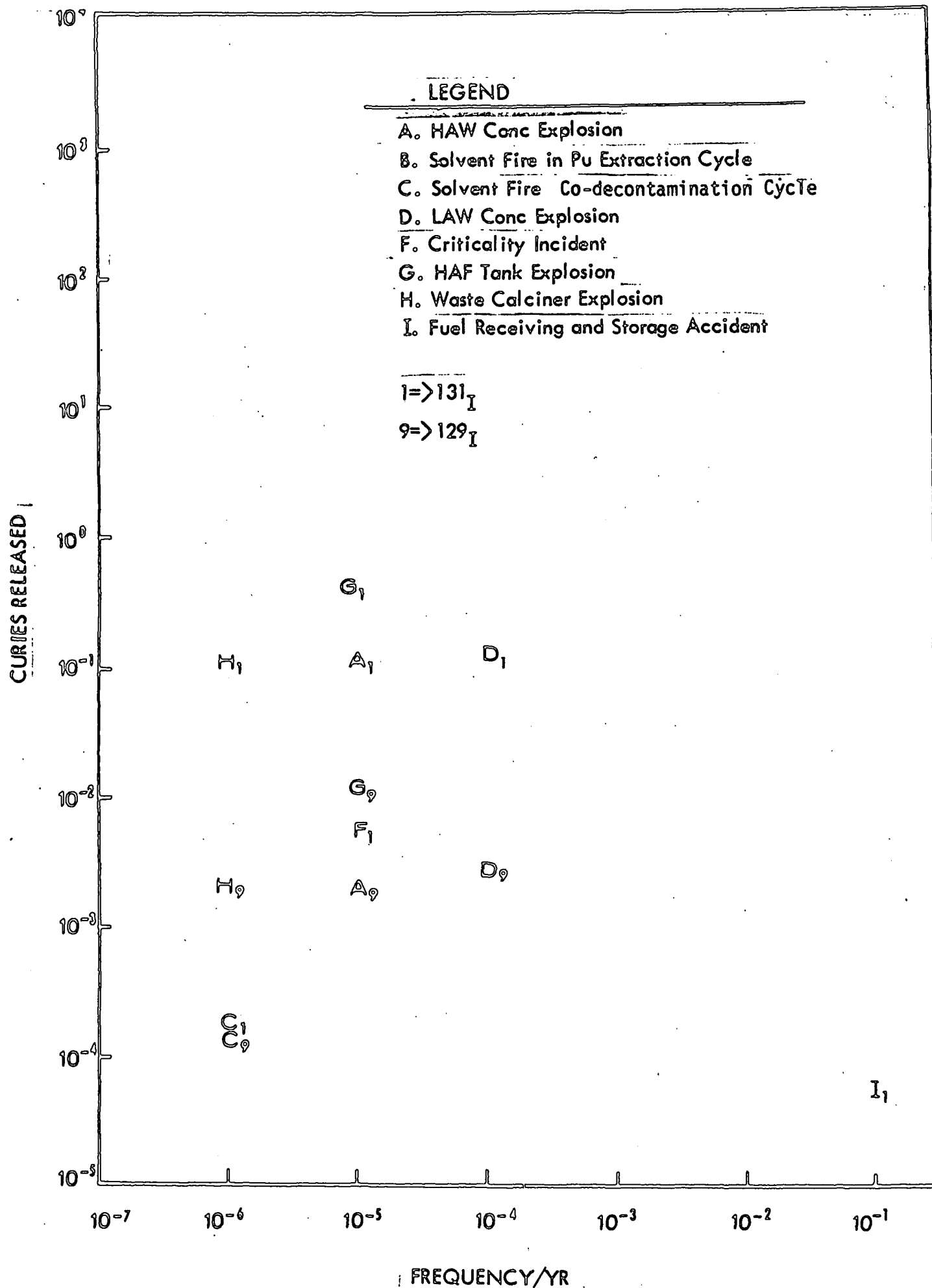
Figure VI -1   IODINE RELEASES ANTICIPATED FOR THE   107
HYPOTHETICAL ACCIDENTS ANALYZED
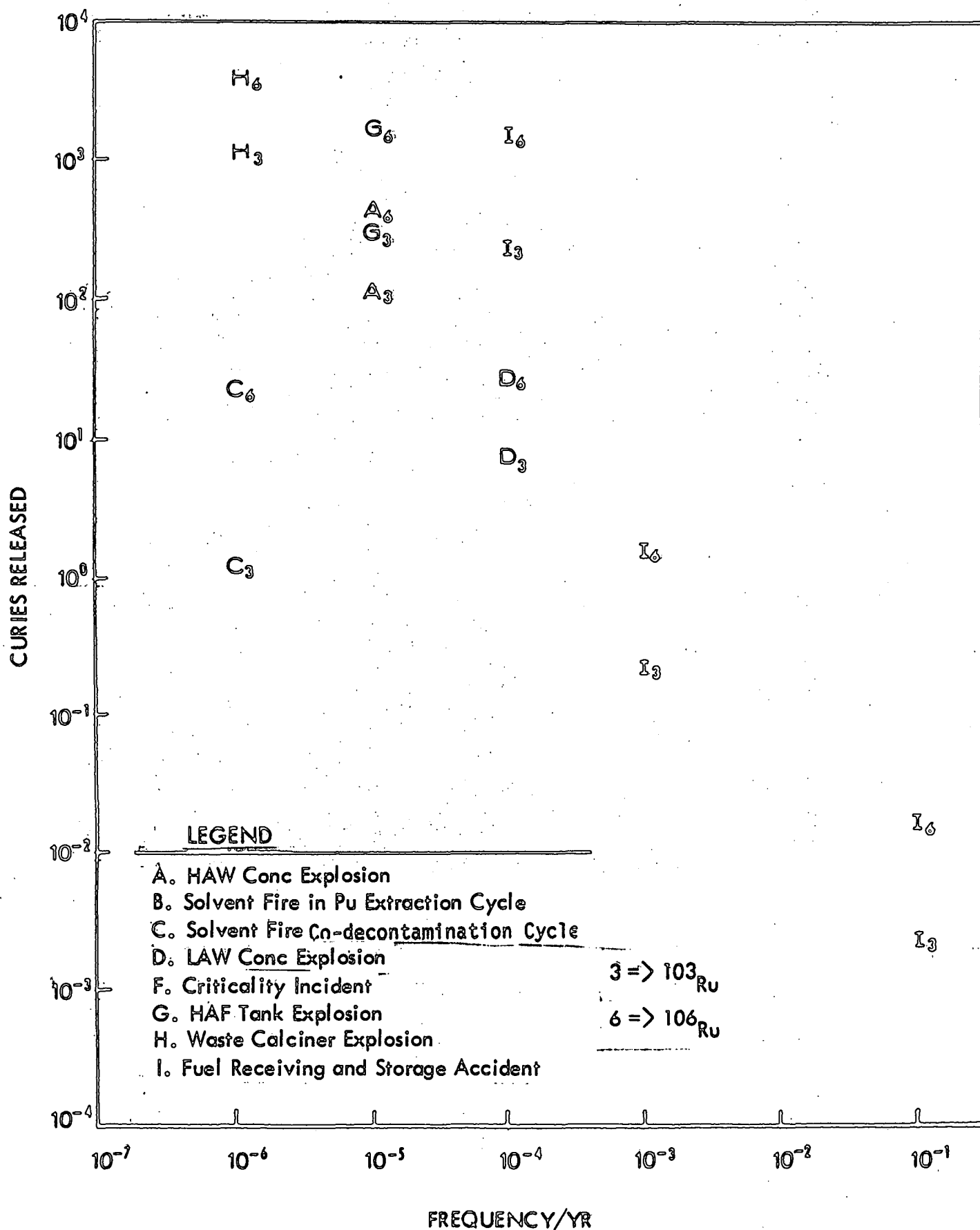
Figure VI-2 RUTHENIUM RELEASES ANTICIPATED FOR
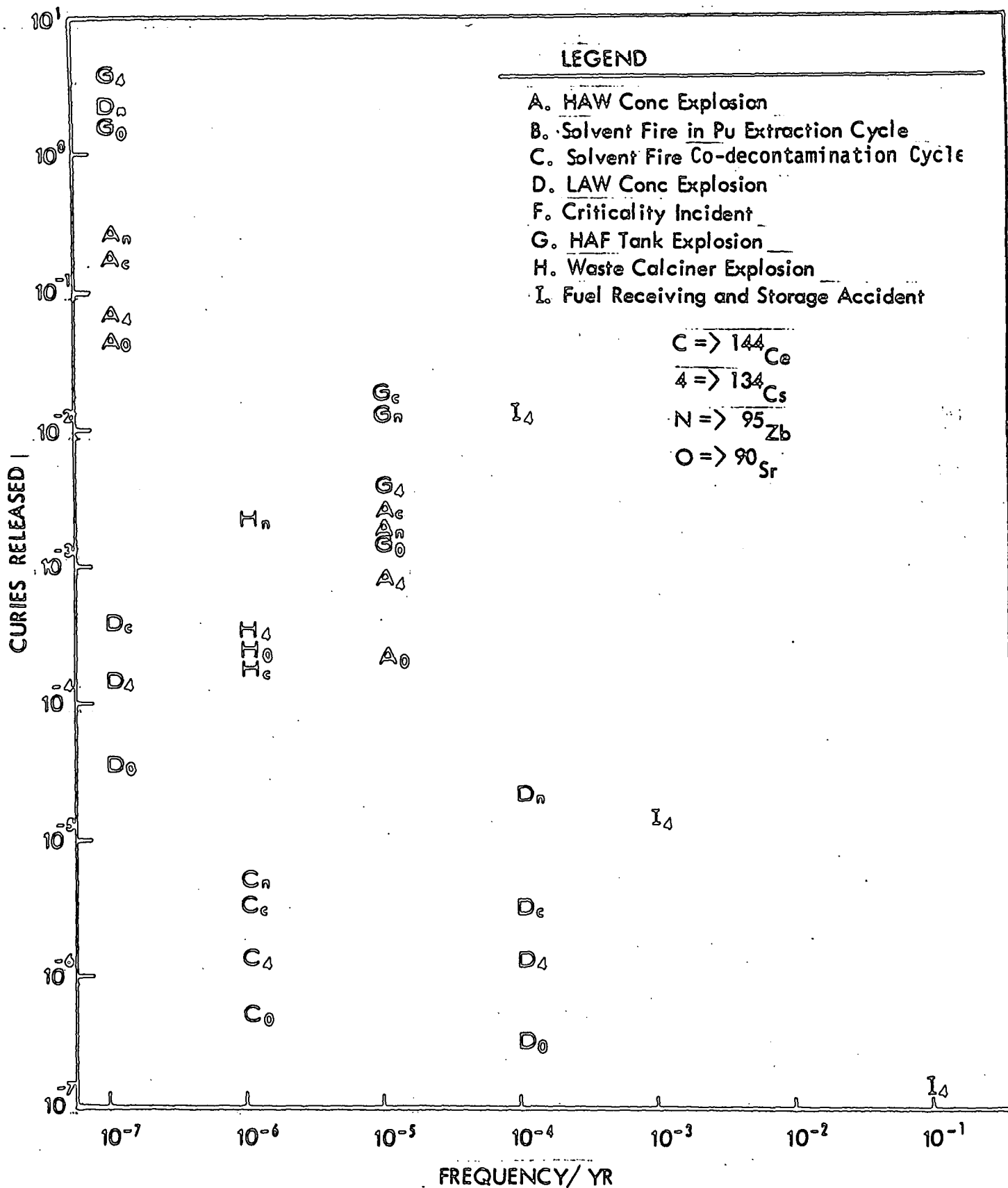THE HYPOTHETICAL ACCIDENTS ANALYZED
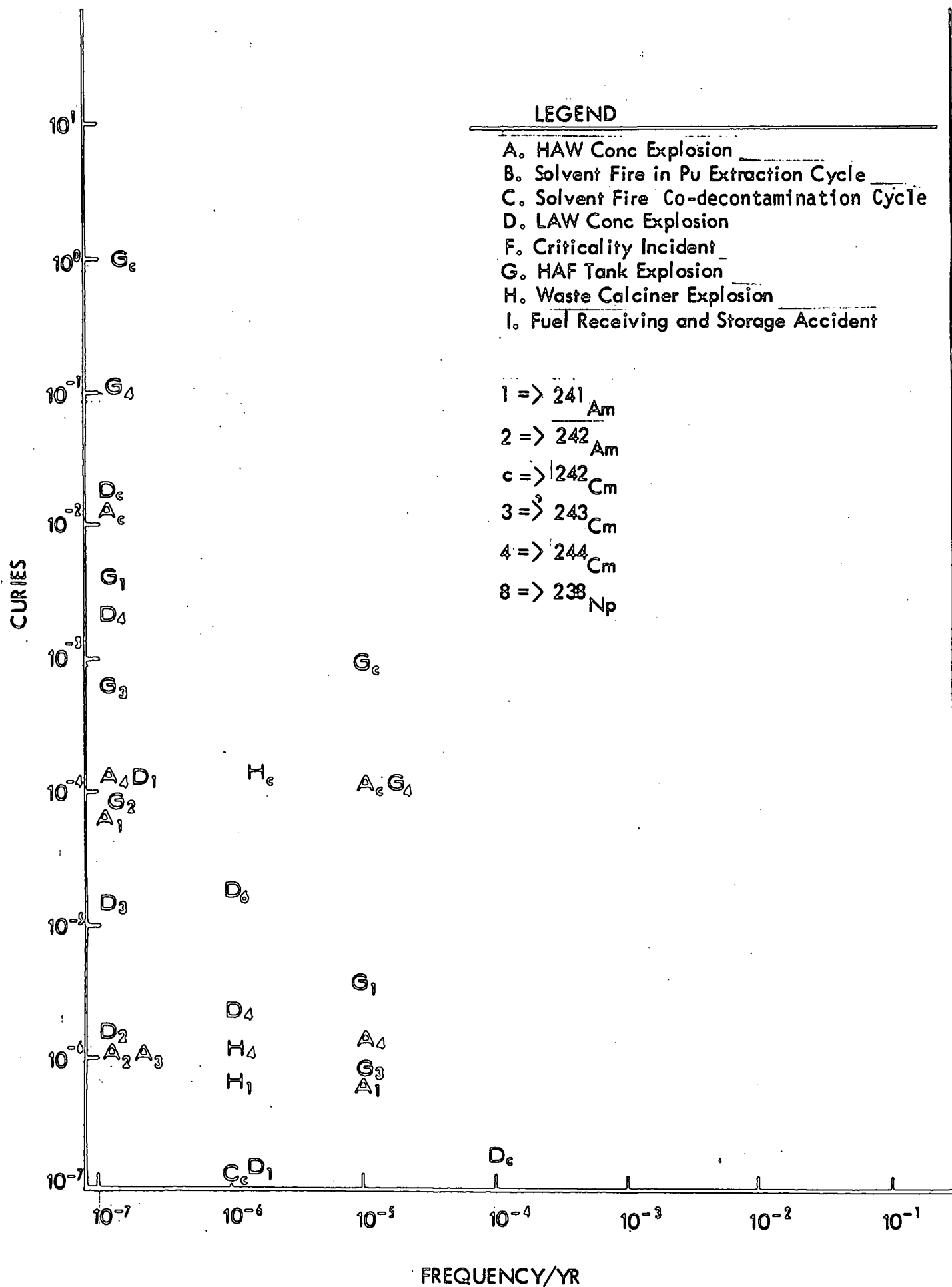
Figure VI-3 NON-VOLATILE ACTINIDE RELEASES ANTICIPATED
FOR THE HYPOTHETICAL ACCIDENTS ANALYZED

10^2

10^1

10^0

10^{-1}

10^{-2}

10^{-3}

10^{-4}

10^{-5}

10^{-6}

10^{-7}

CURIES (Pu)

LEGEND

A. HAW Conc Explosion
B. Solvent Fire in Pu Extraction Cycle
C. Solvent Fire Co-decontamination Cycle
D. LAW Conc Explosion
E. Ion Exchange Resin Fire (Not Plotted)
F. Criticality Incident
G. HAF Tank Explosion
H. Waste Calciner Explosion
I. Fuel Receiving and Storage Accident

$8 \Rightarrow 238_{Pu}$

$9 \Rightarrow 239_{Pu}$

$0 \Rightarrow 240_{Pu}$

$1 \Rightarrow 241_{Pu}$

$2 \Rightarrow 242_{Pu}$

10^{-2} 1 Filter Fails on Demand

10^{-3} All Filters Fail on Demand

G_1

G_0

A_1
D_1

G_0

G_9

G_1

A_0
D_9

H_1

A_1

$A_0 D_0$
$A_9 G_2$
D_9

G_0

$B_1 D_1$

$C_1 H_0$

G_0
A_0

G_9

A_0

A_2

D_8

A_0
A_9

D_2

B_0
H_9
C_0

D_1

B_0 B_9

10^{-7} 10^{-6} 10^{-5} 10^{-4} 10^{-3} 10^{-2} 10^{-1}

FREQUENCY/VR

Figure VI-4 PLUTONIMUM RELEASES ANTICIPATED FOR
THE HYPOTHETICAL ACCIDENTS ANALYZED

Figure VI-5 OTHER NONVOLATILE FISSION PRODUCT RELEASES ANTICIPATED
FOR THE HYPOTHETICAL ACCIDENTS ANALYZED

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plotted separately because of their expected importance in the analysis. The other three curves identify the actinide releases, the plutonium releases and the remaining fission products which are all nonvolatile. Only I^{131} is plotted for the criticality incident in the RPC.

To obtain the abscissa values shown in the figures it was necessary to calculate or obtain from the literature a probability of HEPA filter failure. We have found that a probability of failure of 10^{-2} per demand for the first filter is realistic. If more than one filter occurs in the series then we have assumed that an additional factor of ten is needed for the likelihood of complete filter bank failure. That is for two or three HEPA filters in series the probability of failure would be 10^{-3} per demand. In the case of volatile releases the filters are not assumed to work at all. Hence, in the case of iodine and partially in the case of ruthenium the filters do not work because of the volatile nature of the release.

The likelihood values in the figures are then the products of the numbers in Table VI-1 with the probability of filter failure. For example in Figure VI-3, the probability of a HAF tank explosion occurring is 10^{-5}/year. When the filter is working this appears as G_1 for Am^{241} and as Gc for Cm^{242} at the 10^{-5} vertical line. When the filter is failed these two points are shifted to the 10^{-7}/year line, as seen in the figure, in which the probability of filter failure of 10^{-3}/demand has been factored. Since these curves are plotted on log/log paper the points that appear furthest from the origin are those of most significance. For example in the iodine curve, Figure VI-1, the points G_1 and D_1 dominate the spectrum. These figures therefore graphically indicate those accidents of most significance.

2.   Dose Quantification

We have selected some of the data given in Figures VI-1
through VI-5 and have calculated the likelihood of receiving
a certain dose at a given distance from the generic reproces-
sing plant site. To do this we utilized the following sim-
plified meteorological model.[55] For a puff release following
an accident we have

$$D = Q \circ K \circ (X/Q) \tag{6.1}$$

where

Q is the source strength in curies

K is a dose conversion factor in rem meters^3/curie second

D is the received dose in rem

and where

$$(X/Q) = [\pi \bar{u} \; \sigma_y \; \sigma_z]^{-1} \; exp- \; [(y^2/2\sigma_y^2)+(h^2/2\sigma_z^2] \tag{6.2}$$

with

h = release height or stack height in meters

y = cross wind or off centerline distance in meters

$\bar{u}$ = average wind speed in meters/second

σ_y= cloud horizontal dispersion with distance, in meters

σ_z= cloud vertical dispersion with distance, in meters.

The following data were used in this analysis

h = 100 meters

$\bar{u}$ = 2 meters/second

y = 0.

Based upon data given in reference 1 for Category D turbu-
lence, Table VI-2 was developed for (X/Q) as a function of
distance from the plant site. K values, obtained from ref-
erence 56 are listed in Table VI-3.

TABLE VI-2

Category D Meteorology Parameters

Distance From Source [Meters]	σ_y	σ_z	(X/Q) [sec/meter^3] h=0, y=0	(X/Q) [sec/meter^3] h=100 meters, y=0
100	6.	5.	5.32E-3 *	0
500	30.	19.	2.79E-4	2.76E-10
1,000	55.	32.	9.0 E-5	6.74E-7
5,000	220.	90.	8.06E-6	4.29E-6
10,000	400.	114.	3.50E-6	2.38E-6
50,000	1500.	310.	3.43E-7	3.25E-7
100,000	2700.	420.	1.41E-7	1.37E-7

* $5.32\text{E-}3 = 5.32 \times 10^{-3}$

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TABLE VI-3

Isotopic Dose Conversion Factors [1]

Group	Isotope	K factors $\left[\dfrac{\text{rem } m^3}{\text{Ci sec}}\right]$	Group	Isotope	K factors $\left[\dfrac{\text{rem } m^3}{\text{Ci sec}}\right]$
Ru:			OFPT:		
	Ru-103	15.9		Sr-89	47.6
	Ru-106	238		Sr-90	238
Zr-Nb:				Y-90	
	Zr-95	47.6		Y-91	47.6
	Nb-95	15.9		Cs-134	119
I: [2]				Cs-137	95
	I-129	$4.73(10^5)$		Ce-141	9.5
	I-131	$8.5\ (10^4)$		Ce-144	238
				Pm-147	15.9
I: [3]	I-131	$1.6\ (10^3)$		Am-241	$9.52(10^4)$
		$3.8\ (10^2)$		Am-242	
				Cm-242	$6.47(10^4)$
				Cm-243	
				Cm-244	$1.26(10^5)$
			Pu:		
				Pu-238	$3.81(10^5)$
				Pu-239	$3.81(10^5)$
				Pu-240	$3.81(10^5)$
				Pu-241	$3.81(10^2)$
				Pu-242	$3.81(10^5)$

(1) Conversion factors for dose to lung from inhalation of insoluble particles except for iodine for which the organ of reference is the thyroid.

(2) Ingestion of milk by infant.

(3) Inhalation by adult.

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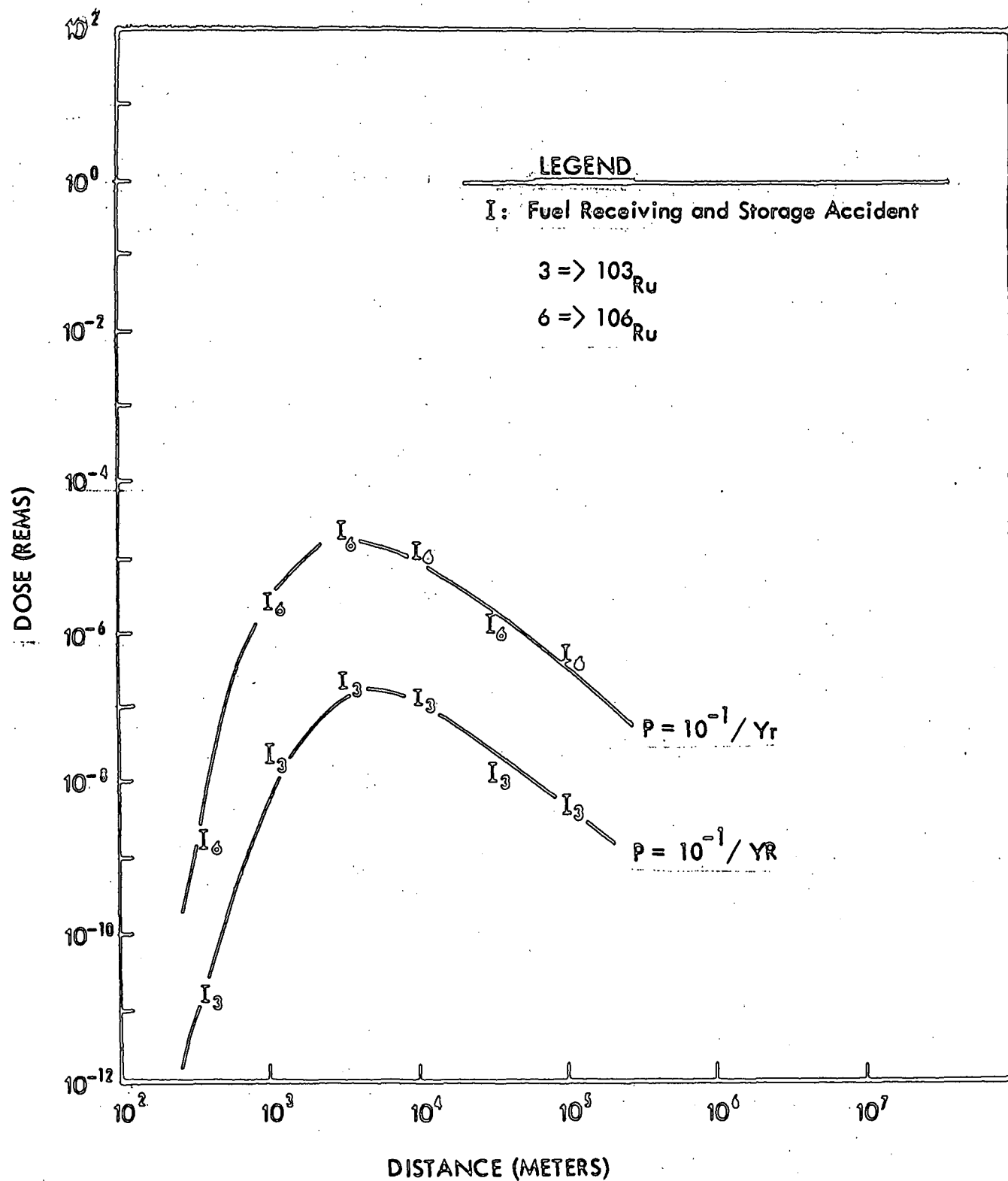
Figure VI-6a RUTHENIUM DOSE AT DISTANCE FOR A CERTAIN LIKELIHOOD

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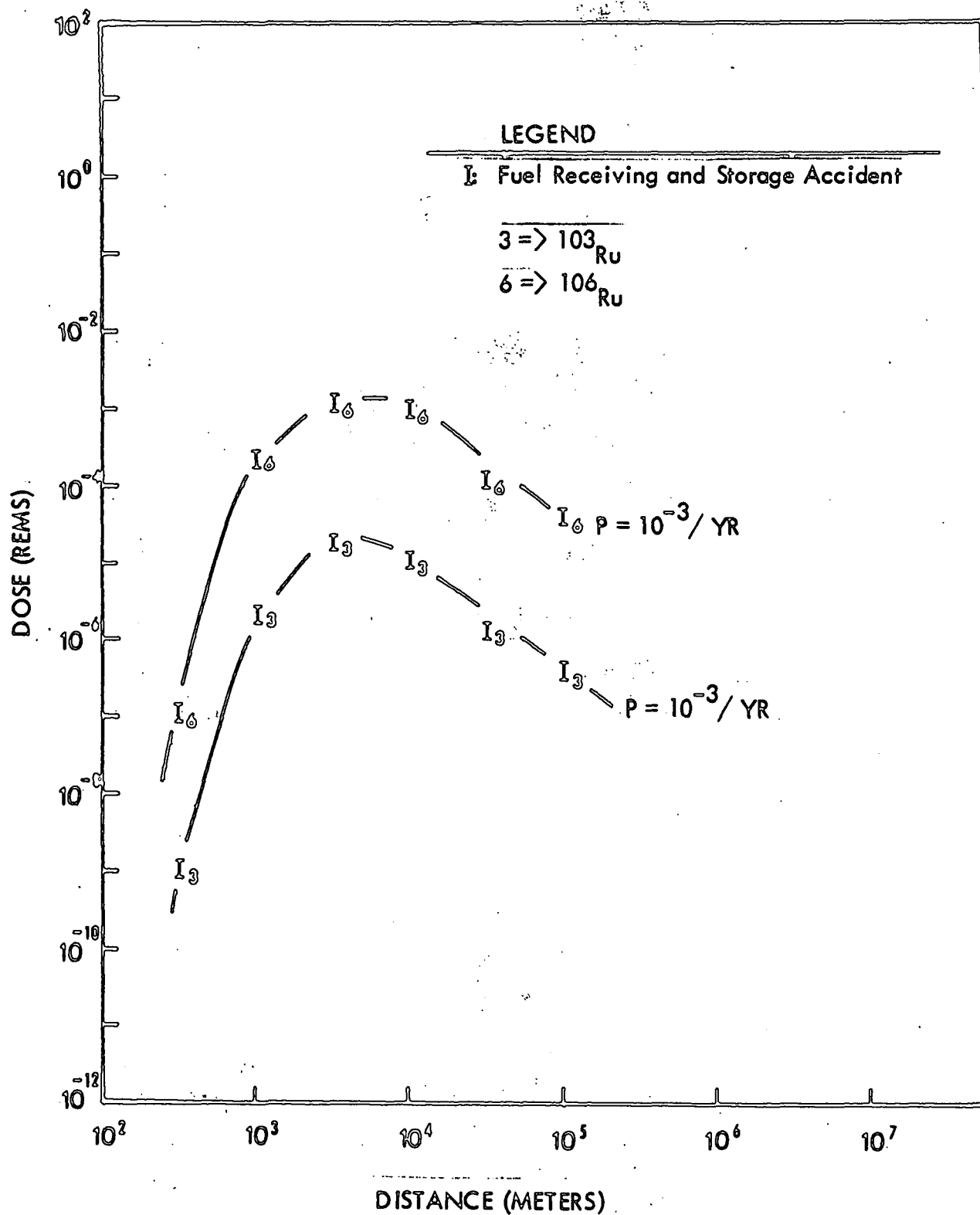
Figure VI-6b RUTHENIUM DOSE AT DISTANCE FOR A LIKELIHOOD OF 10^{-3}/YR

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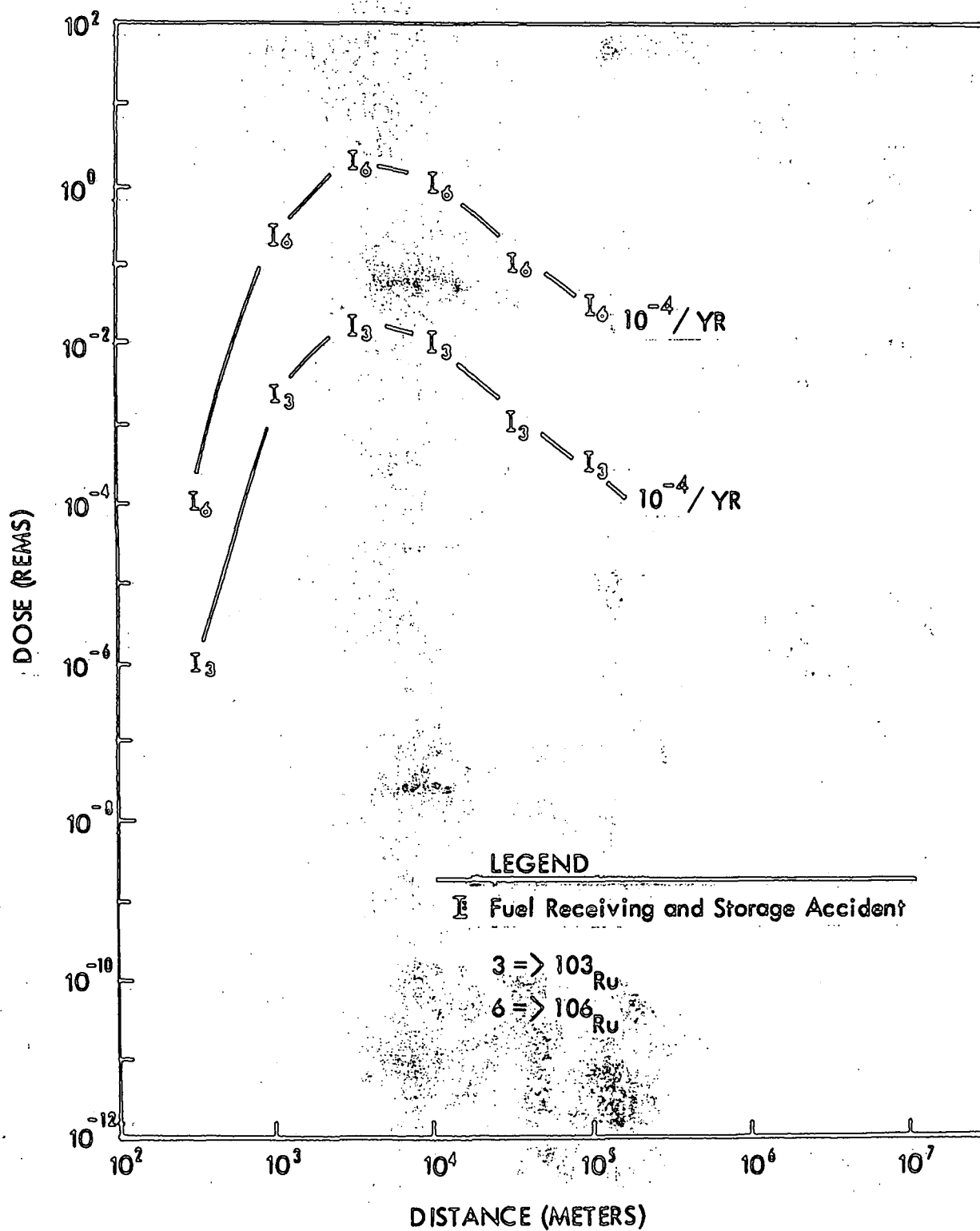
Figure VI-6c RUTHENIUM DOSE AT DISTANCE FOR A LIKELIHOOD OF 10^{-4}/YR
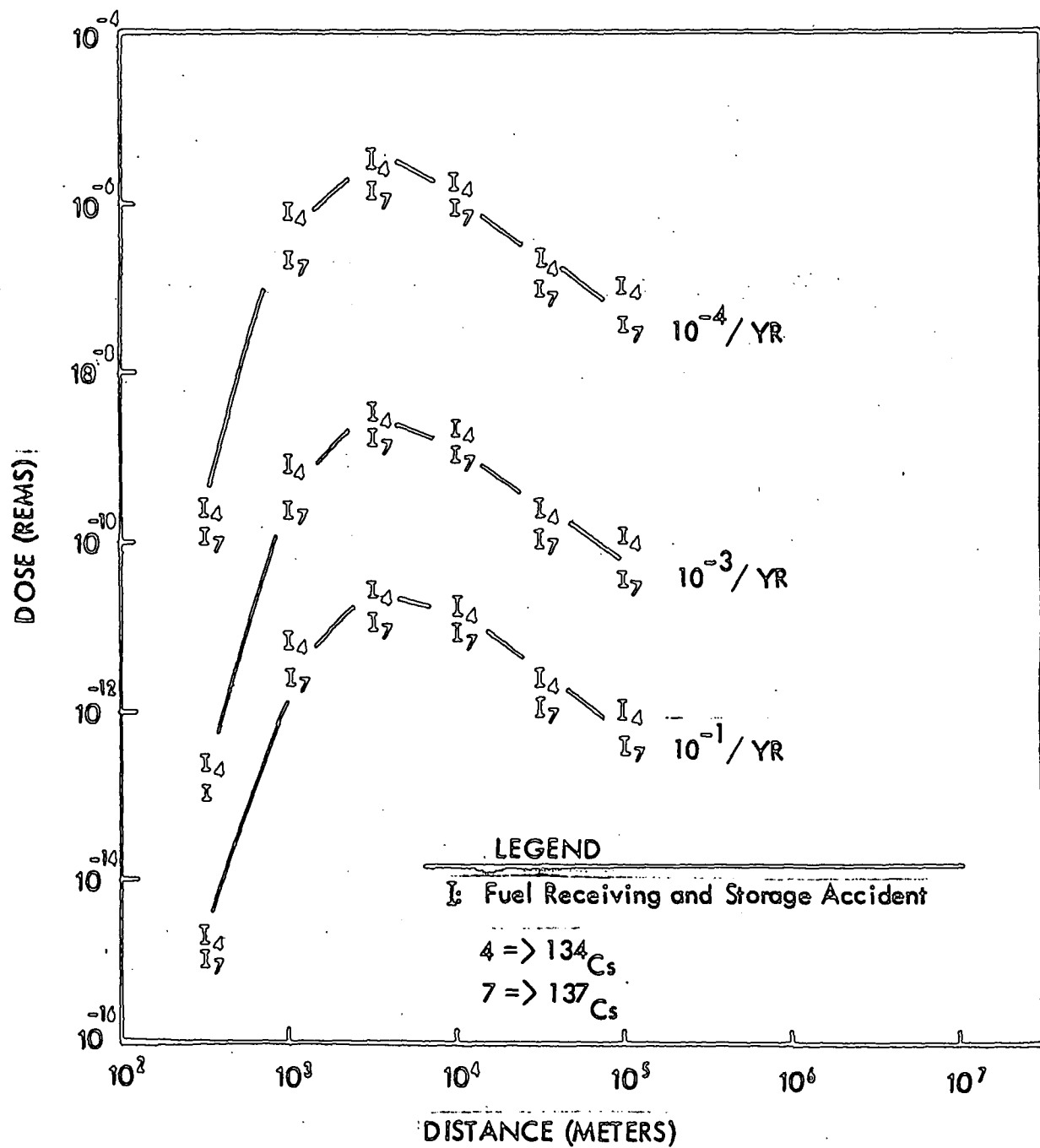
Figure VI-7 CESIUM DOSE AT DISTANCE FOR SEVERAL LIKELIHOODS

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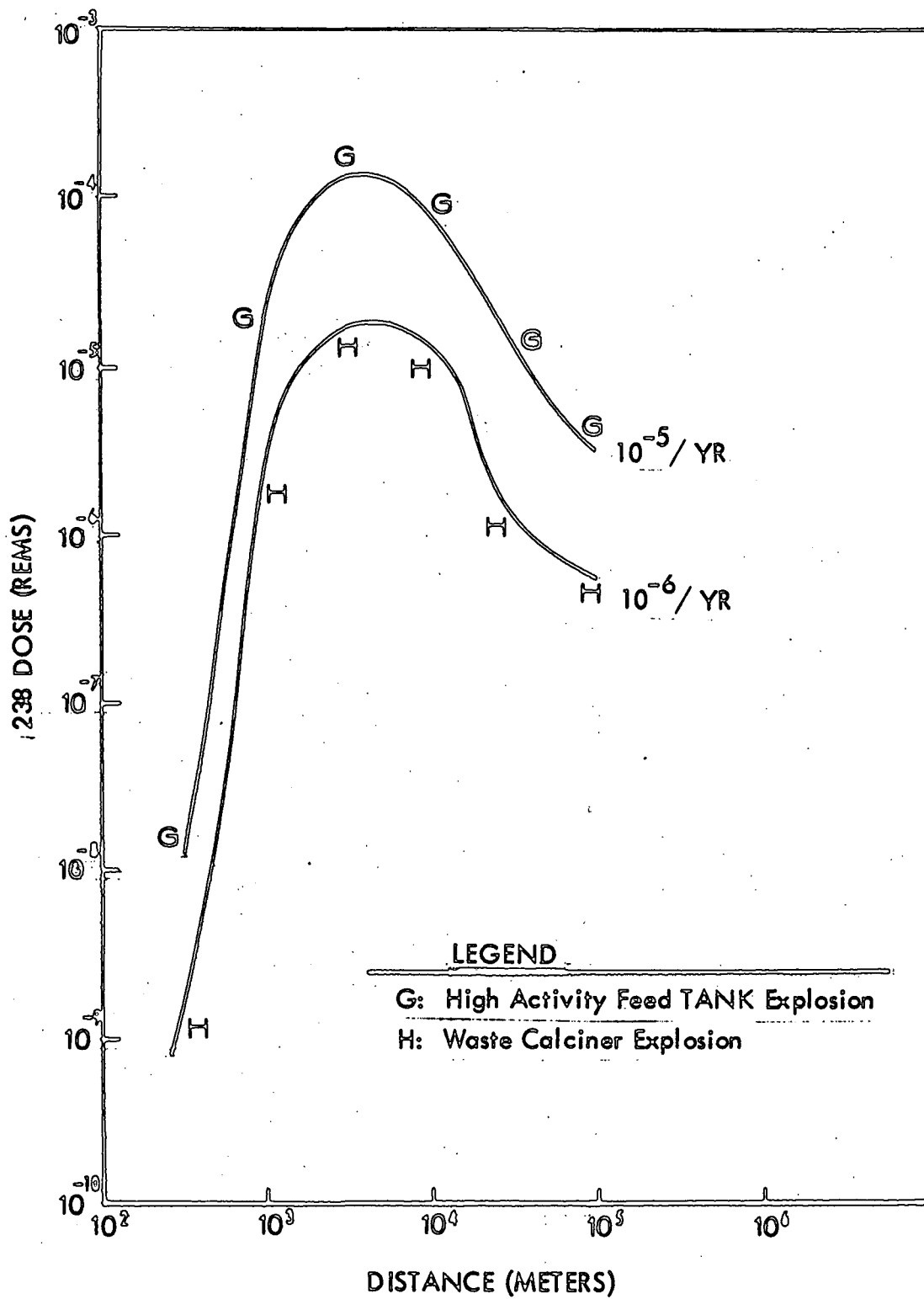
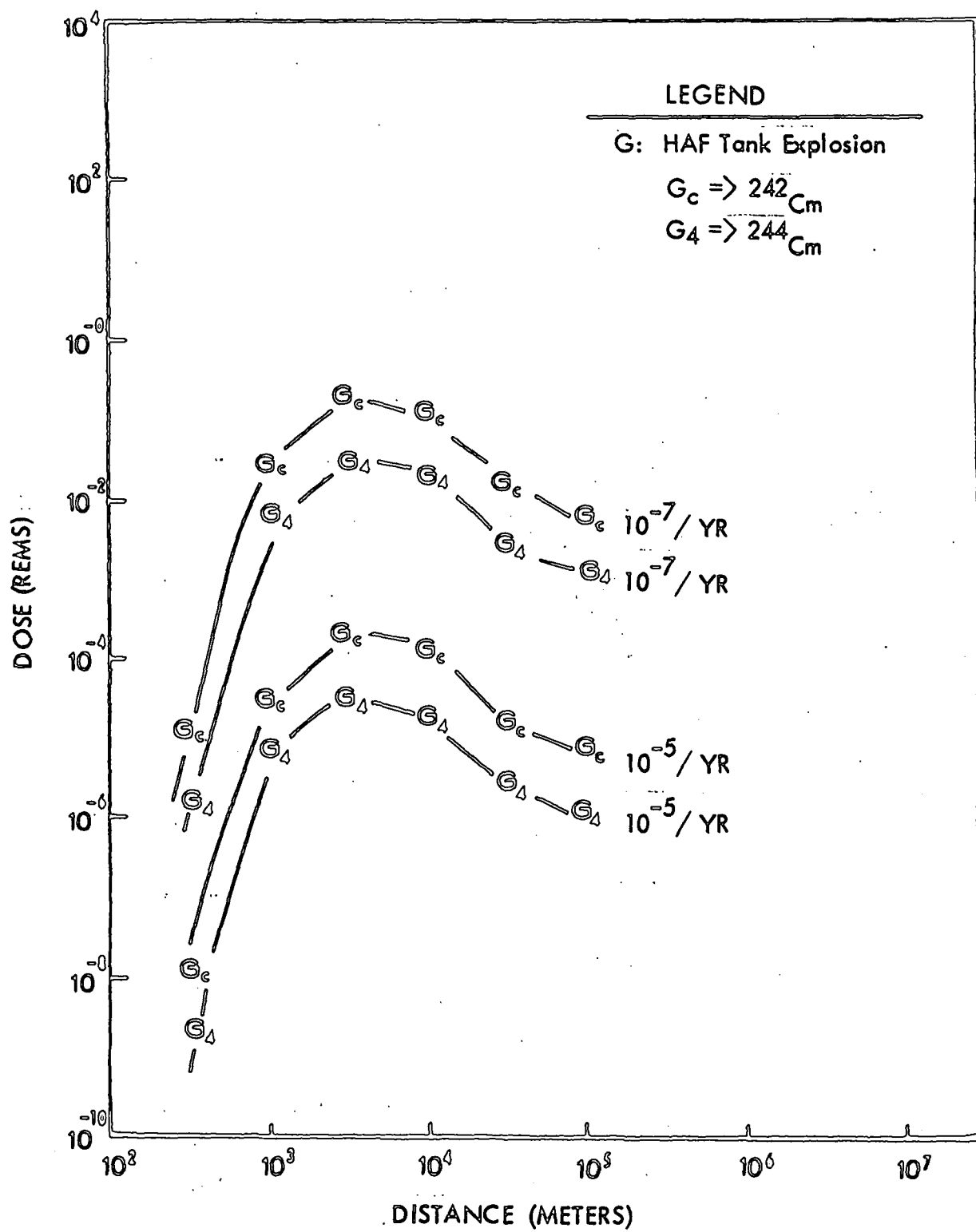Figure VI-8 LUNG DOSE FROM PLUTONIUM VS DISTANCE
FOR SEVERAL LIKELIHOODS

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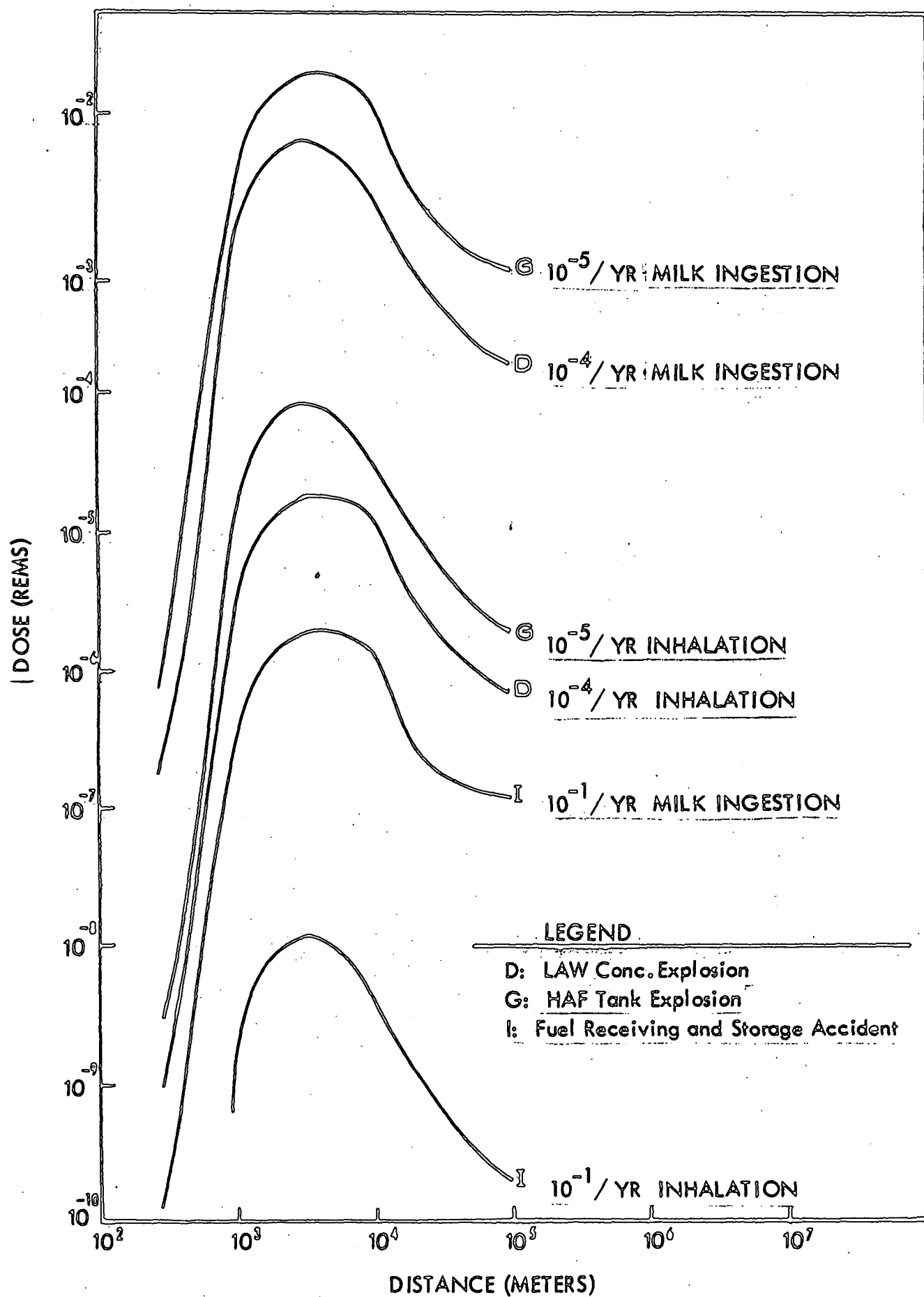Figure VI-9 CURIUM DOSE AT DISTANCE FOR SEVERAL LIKELIHOODS

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Figure VI-10a THYROID DOSE FROM IODINE-129 VS DISTANCE

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LEGEND

D: LAW Conc. Explosion
G: HAF Tank Explosion
I: Fuel Receiving and Storage Accident

G 10^{-5}/YR MILK INGESTION

D 10^{-4}/YR MILK INGESTION

G 10^{-5}/YR INHALATION

D 10^{-4}/YR INHALATION

I 0.1/YR MILK INGESTION

I 0.1/YR INHALATION

DISTANCE (METERS)

DOSE (REMS)
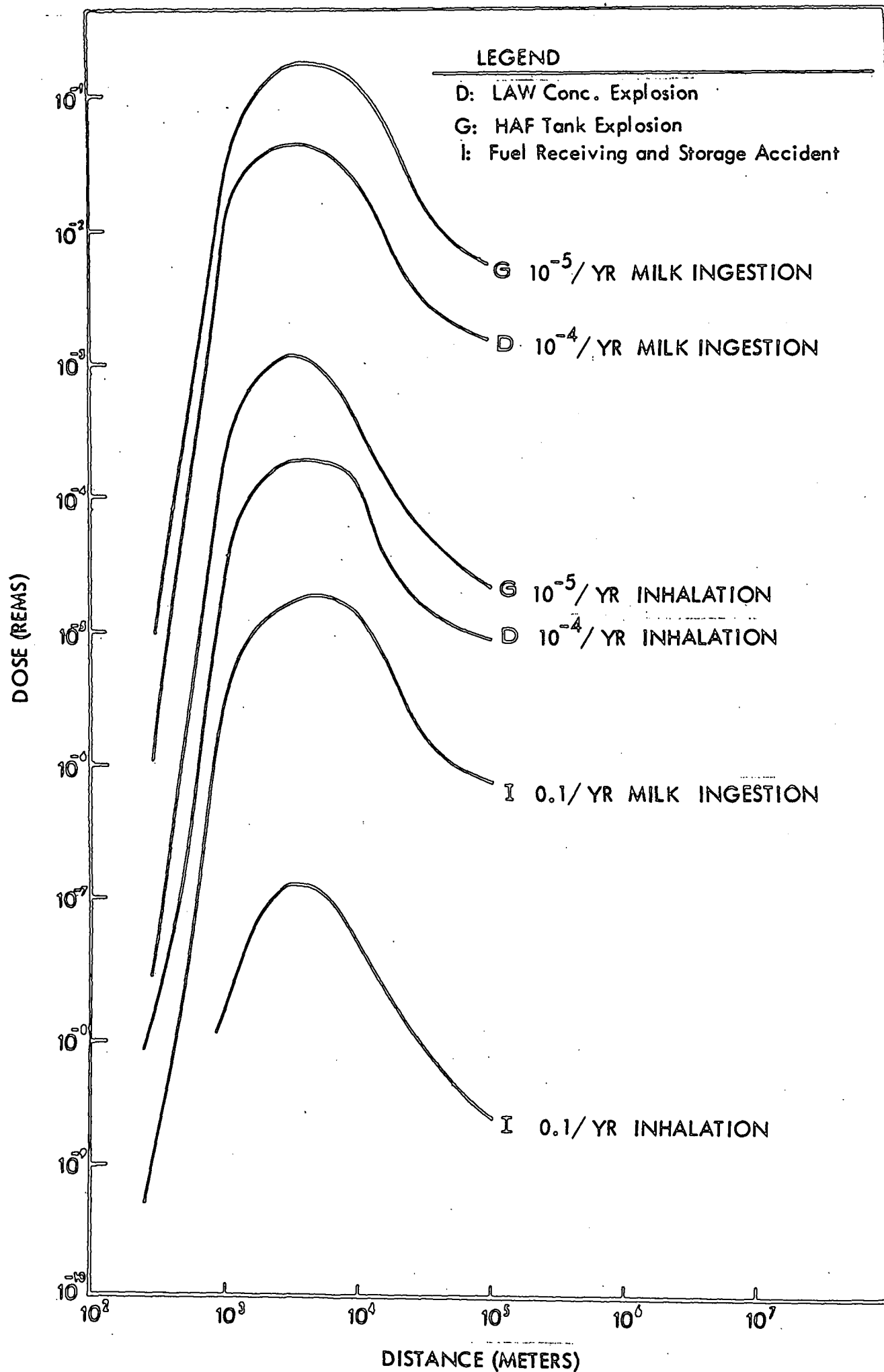
Figure VI-10b THYROID DOSE FROM IODINE-131 vs DISTANCE

By inspecting Figures VI-1 through VI-5 those points that
provided the larger releases with the higher probabilities
were used as source terms to generate the doses at distance
given in Figures VI-6 through VI-10. The isotopes plotted
were ruthenium 103 & 106, cesium 134 & 137, plutonium 238 &
241, curium 242 & 244, and iodine 129 & 131. Any of the
other isotopes could be plotted in a similar manner.

The distance where the largest dose occurs is approximately
5,000 meters from the plant and this appears to be common
for each of the isotopes examined. Category D meteorology
is perhaps that which occurs with the highest frequency.
A more pessimistic, less likely meteorology could, of course,
lead to a series of different dose/distance curves.

Inspection of the results presented in this chapter indicate
that incidents involving the high activity feed tank and the
fuel receiving and storage cask, dominate much of the re-
lease data for non-plutonium releases. Plutonium releases
are most evident in ion exchange resin fires and in the
high activity waste concentrator and waste calciner explos-
ions. Other incidents, such as low activity waste concentra-
tion explosions, contribute to a lesser extent.

3.   Site Related Events

It is difficult in a generic study to utilize actual site
related data in the accident analysis. We have assumed that
the generic plant would be built to withstand forseeable

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site related events. Listed in Table VI-4 are several severe phenomena whose occurrence might damage a portion of the plant. While we cannot do a specific failure analysis for these initiating events, the following statements appear appropriate.

Were a severe tornado to strike a reprocessing plant it might initiate the following failures:

 a. loss of offsite electric power
 b. filter failure
 c. missile penetration of a portion of the building
 d. stack structural failure, either partial or complete
 e. possible loss of storage pool water.

There appears to be practically no way in which a tornado can cause the process cell walls to fail. The loss of electric power or storage pool water are expected to be temporary in nature. The possible release of excess radioactivity due to filter failure or stack collapse will be counteracted by the extensive turbulence and dispersion caused by the tornado itself. Hence little radiological risk is expected from tornado induced events.

An earthquake can cause the following malfunctions:

 a. possible structural failure
 b. loss of offsite electric power
 c. internal piping failures
 d. stack collapse
 e. loss of pool water.

TABLE VI-4

Selected Natural Event Data

Event	Frequency of Occurrence [19,57,58]
Tornado [19]	$6(10^{-4})$/yr.
Earthquake [19] Intensity IX	10^{-5}/yr.
Meteorites [57]	10^{-9}/yr.
Airplane Impact [58]	10^{-6}/yr.

The incident of importance here is the piping failure possibility. This event could cause releases to occur in many of the cells simultaneously. Building collapse is not expected to occur even during severe earth tremors. The releases would probably be liquid in nature and consequently would not contribute significantly to airborne releases. Moreover the frequency of 10^{-5}/year is about the same as the process cell incidents. Hence the earthquake induced releases will not impact the upper limit process cell accidents to any noticeable extent.

Airplane or meteorite impact with subsequent fire would cause local process cell failure. However these likelihoods are smaller than cell initiated events and are therefore not significant.

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Summary Table Describing the Basic Operations, Process Functions and Chemical Reactions in the Generic Spent Fuel Reprocessing Plant (Modeled after BNFP).

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Cask receiving and handling	Receipt and preparation of shipping cask for unloading	Cask and carrier will be monitored for outside contamination and washed to remove outside dirt. The cask will be removed and the condition of fuel and coolant determined by temperature, pressure, and coolant radioactivity measurements. The cask will be vented to the vessel off-gas system and the primary coolant replaced, if necessary. The cask will be placed in the cask unloading pool where the lid will be removed and fuel elements unloaded remotely under water shield. Empty casks will be decontaminated, monitored, and returned to customer.
Fuel storage and transfer	Storage of fuel elements until dissolution	Fuel element identity will be confirmed and the elements placed in storage canisters in the storage pool. Pool water will be circulated through heat exchangers, inorganic ion exchange beds, and filters to remove fuel decay heat and radioactive contaminants. Elements will be remotely transferred from the pool to the feed mechanism of shear.
Shear	Preparation of fuel for dissolution	Fuel elements will be mechanically chopped into small segments, exposing oxide fuel inside the elements to dissolution while outside cladding (stainless steel or zircaloy) will remain undissolved.

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Dissolution and feed preparation Dissolution	Conversion of the fuel to a liquid solution $3UO_2 + 8HNO_3 \xrightarrow{<8M} 3UO_2(NO_3)_2 + 4H_2O + 2NO$ $UO_2 + 4HNO_3 \xrightarrow{>8M} UO_2(NO_3)_2 + 2H_2O + 2NO_2$ FISSION PRODUCTS + x $HNO_3 \rightarrow$ F.P. $(NO_3)_x + yH_2O+z$ NO $PuO_2 + 4HNO_3 \longrightarrow Pu(NO_3)_4 + 2H_2O$ $NO + 2HNO_3 \xrightarrow{>8M} 3NO_2 + H_2O$	The chopped fuel elements will be contacted with hot, concentrated nitric acid which will convert uranium, plutonium, and most of the fission products to soluble nitrate salts. Undissolved cladding (hulls) will remain in dissolver basket. Gases generated during dissolution will be channeled to off-gas treatment system. Nitrate salt solution will be transferred to tanks for sampling measurement and final acid adjustment.
Solids handling and waste	Disposal of undissolved cladding hulls	The cladding hulls will be rinsed and transferred by shielded trailer to a burial ground. Intermittently, or in case of abnormalities during dissolution, batches of hulls will be checked for complete dissolution of plutonium and uranium.
Co-decontamination and partition cycle	Separation of the plutonium and uranium from the bulk of the fission products and partitioning of the plutonium from the uranium	Adjusted aqueous feed solution and tributyl phosphate (TBP) diluted in a normal paraffin hydrocarbon will be mixed counter-currently in a bank of centrifugal contactors. The organic solution, which preferentially extracts the

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Extraction	$UO_2^{++} + 2NO_3^- + 2TBP$ $\xrightarrow{n-C_{12}H_{26}} UO_2(NO_3)_2 \cdot 2TBP$ $Pu^{+4} + 4NO_3^- + 2TBP \xrightarrow{n-C_{12}H_{26}}$ $Pu(NO_3)_4 \cdot 2TBP$ $PuO_2^{++} + 2NO_3^- + 2TBP$ $\xrightarrow{n-C_{12}H_{26}} PuO_2(NO_3)_2 \cdot 2TBP$	nitrate complexes of tetravalent plutonium and hexavalent uranium, will exit from the centrifugal contactor and pass through a pulsed scrub column where an aqueous nitric acid solution will remove extracted fission products from the organic stream. The organic stream will pass through a partitioning column where plutonium will be re-
Reduction and Partitioning	$Pu^{+4}_{aq} + 1e^- \longrightarrow Pu^{+3}_{aq}$ $UO_2^{++}_{aq} + 2e^- + 4H^+ \longrightarrow$ $U^{+4}_{aq} + 2H_2O$ $U^{+4}_{aq} + 2Pu^{+4}_{aq} + 2H_2O \longrightarrow$ $UO_2^{++} + 2Pu^{+3} + 4H^+$ $N_2H_4 + 2HNO_2 \longrightarrow N_2 + N_2O +$ $3H_2O$	duced to the inextractable trivalent state and stripped into another aqueous nitric acid stream containing hy- drazine. The organic stream will pass through another column where the uranium will be stripped into acidified water. **(Alternatively, anion exchange would be used for partitioning plutonium and uranium into separate streams)**
Stripping	$UO_2(NO_3)_2 \cdot 2TBP + H_2O + 2H^+ \rightarrow$ $UO_2^{++}_{aq} + 2HNO_3 + H_2O + 2TBP$	
Second uranium cycle	Further decontamination of uranium from fission products	Nitric acid will be added to the aqueous strip stream containing the uranium, and the uranyl nitrate complex will again be preferentially extracted by another TBP solution in a pulsed column. Before leaving the column,

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PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
		the organic stream will be scrubbed successively with strong and dilute nitric acid solutions which remove extracted ruthenium and zirconium-niobium, respectively. Uranium will be stripped from the organic stream in another column, using acidified water, and this solution will be subsequently concentrated by evaporation.
Uranium silica gel, product storage	Final decontamination and disposition of uranium	Concentrated uranium solution will be passed through silica gel beds to remove traces of zirconium-niobium. Uranyl nitrate product solution will be analyzed and stored in tanks until shipment.
Second and third plutonium cycle, storage, and shipping	Final decontamination and disposition of plutonium	Plutonium in aqueous stream leaving partitioning column will be reoxidized to the extractable tetravalent state, which will be preferentially extracted into the TBP organic stream in a pulsed column. In the same column,
Oxidation	$Pu^{+3} + 2NO_2 + H^+ \longrightarrow$ $Pu^{+4} + HNO_3 + NO$	the organic stream will be scrubbed successively with strong and dilute nitric acid solutions, which will remove
Extraction	$Pu^{+4} + 4NO_3^- + 2TBP \xrightarrow{n-C_{12}H_{26}}$ $Pu(NO_3)_4 \cdot 2TBP$	extracted ruthenium and zirconium-niobium, respectively. The organic stream will pass through a strip column where
Reduction and Stripping	$Pu^{+4}_{aq} + 1e^- \longrightarrow Pu^{+3}_{aq}$ $UO_2^{++}{}_{aq} + 2e^- + 4H^+ \longrightarrow$ $U^{+4}_{aq} + 2H_2O$	plutonium will be reduced to inextractable trivalent state, which will transfer to the aqueous stream of dilute nitric acid and hydrazine. The extraction-stripping

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Scrubbing	$U^{+4}_{aq} + 2Pu^{+4}_{aq} + 2H_2O \longrightarrow$ $UO_2^{++} + 2Pu^{+3} + 4H^+$ $TBP + UO_2^{++} + 2NO_3^- \xrightarrow{n-C_{12}H_{26}}$ $UO_2(NO_3)_2 \cdot 2TBP$	sequence will be repeated in the third plutonium cycle for further decontamination. A TBP scrub stream will remove residual uranium from the plutonium aqueous stream as it leaves the last strip column. Plutonium concentration will be accomplished by maintaining a high ratio of organic to aqueous flow in the strip columns. Final plutonium nitrate solution will be washed with an organic stream of normal paraffin hydrocarbon (diluent for TBP) to remove traces of TBP and phosphate. Product solution will be analyzed and stored in tanks until shipment. Solvent streams leaving plutonium cycles will pass through a strip column to remove residual inextractable species of uranium and plutonium and will be recycled to the co-decontamination cycle.
No. 1 solvent system Carbonate wash	Removal of degradation products from solvent $Na_2CO_3 + 2(C_4H_9)_2 HPO_4 \longrightarrow$ $2(C_4H_9)_2 Na PO_4 + H_2CO_3$ $2RCH_2NO_2 \rightleftharpoons 2RCH = NOOH +$ $Na_2CO_3 \longrightarrow 2RCH = NOONa +$ H_2CO_3	Organic solvent stream from co-decontamination and partition cycle will be washed successively with dilute aqueous solutions of sodium carbonate, nitric acid, and sodium carbonate (or sodium hydroxide) to remove organic degradation products by extraction or precipitation; precipitated solids will be removed by a filter. Fresh TBP or diluent (normal paraffin hydrocarbon) will be added,

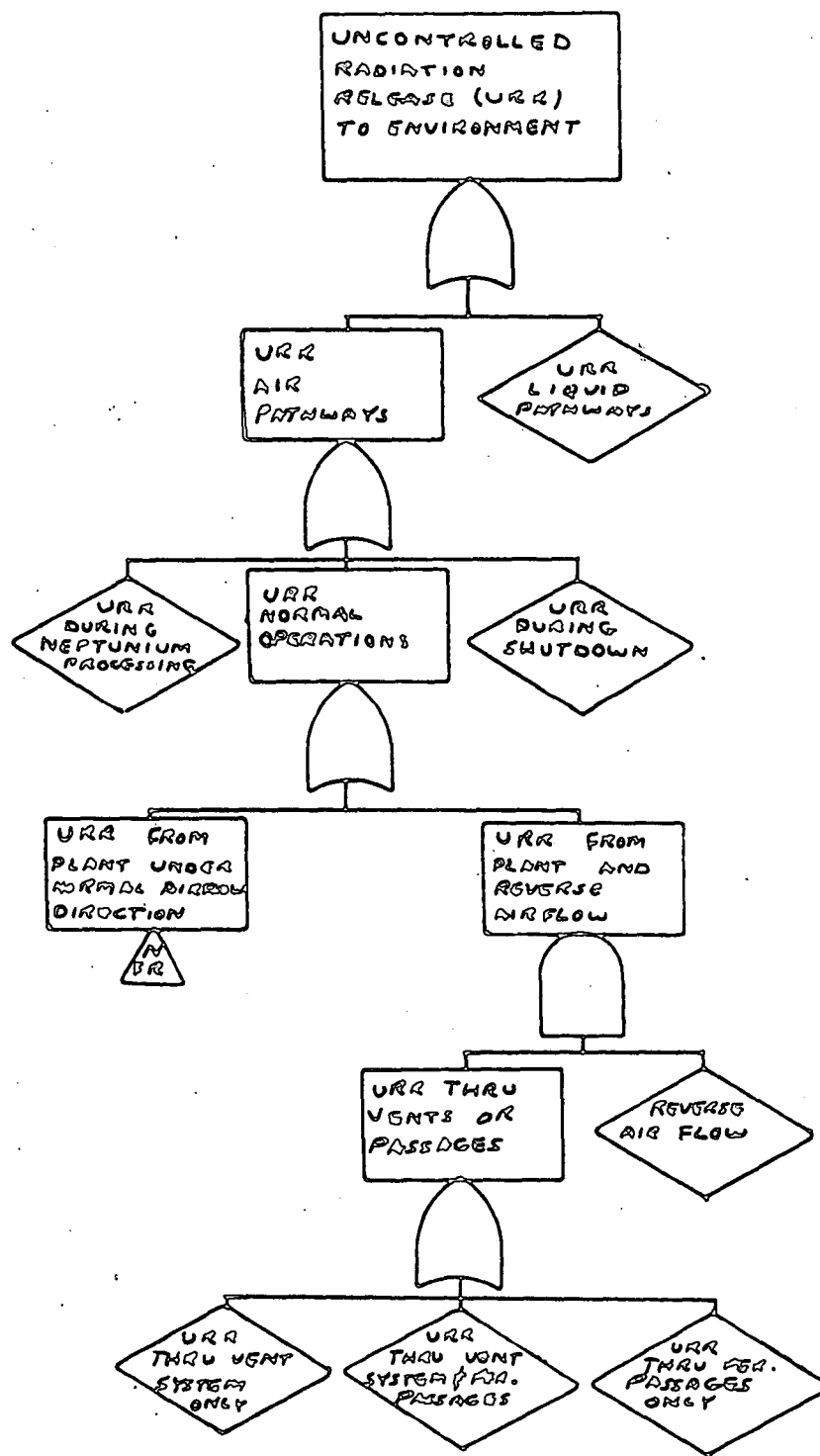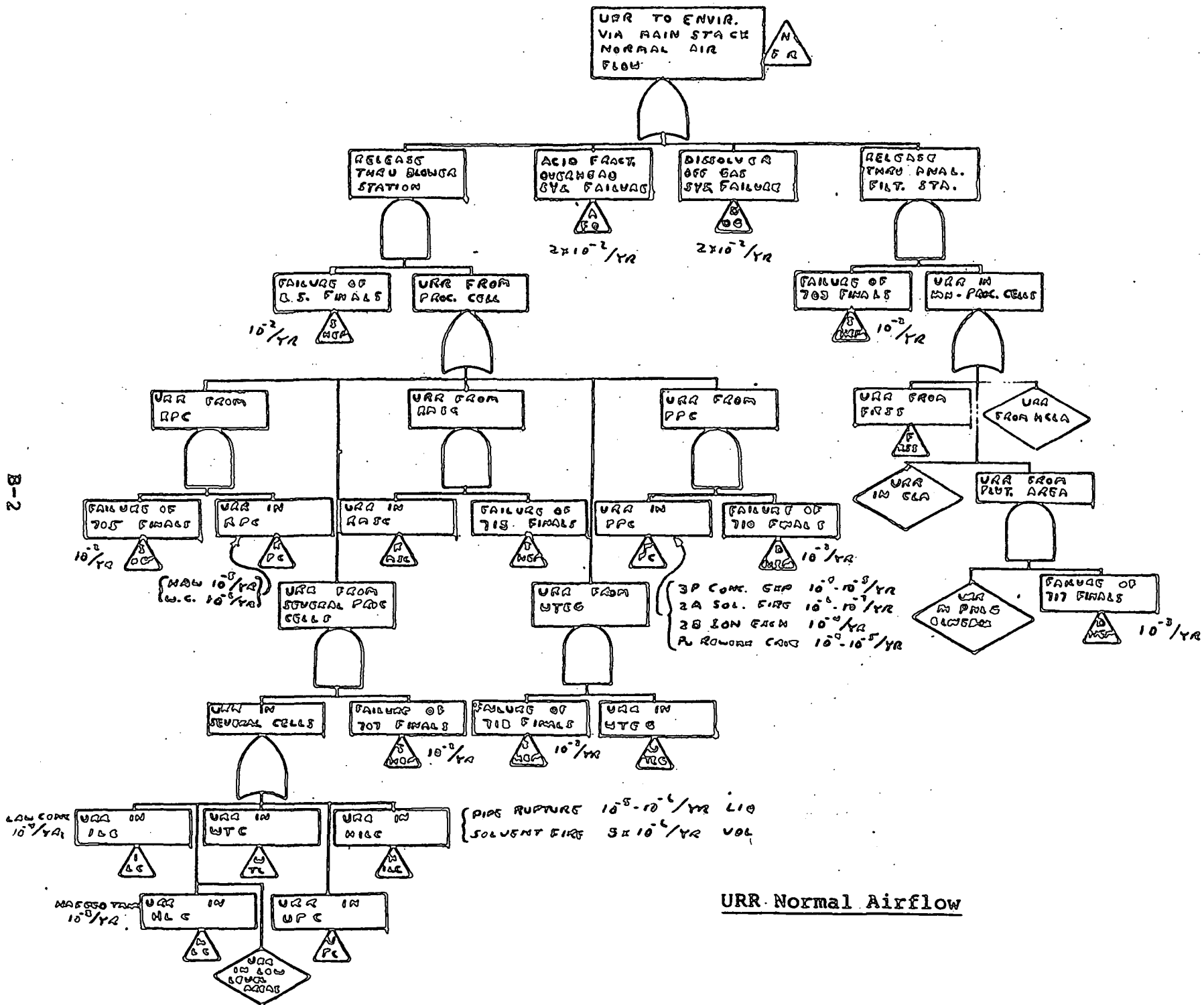| PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
		as required, to maintain proper TBP concentration or total solvent inventory.
No. 2 solvent system	Removal of degradation products from solvent	Organic solvent stream from second uranium cycle will be treated similarly to No. 1 system, except the second alkaline wash will be omitted.
Liquid waste treating and storage	Disposal of liquid waste streams with minimum residual waste volume for storage.	The highly radioactive waste stream from the co-decontamination cycle will be concentrated by evaporation; acidity of the concentrated bottoms will be reduced to permit long-term storage in stainless steel tanks by reacting with a
Acid reduction	$C_{12}H_{22}O_{11} + 18.2\ HNO_3 \longrightarrow$ $12CO_2 + 14.9\ NO + 3.3\ NO_2 +$ $20.1\ H_2O$	sugar solution; overheads will be fed to the low-activity evaporator for further decontamination. Most of the remaining nitric acid waste streams containing low levels of fission products, uranium, and plutonium will be concentrated in the low-activity waste evaporator; concentrated bottoms will be recycled to the co-decontamination cycle; overheads will be condensed and fed to the acid recovery system. Miscellaneous waste streams, containing salts, low levels of fission products and no appreciable uranium or plutonium, will be acidified and concentrated in the general-purpose evaporator; bottoms will be stored; overheads will be monitored for radioactivity content and then

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
		discharged.
Nitric acid recovery and storage	Recovery of nitric acid and reduction of nitrogen oxides release to the environs.	Overheads from LAW evaporator will contain most of the tritium (as tritiated water) and some undestroyed nitric acid from the process; they will be condensed and fed to the fractionator which concentrates nitric acid. Recovered acid will be stored and used in make-up of various acid streams; overheads containing tritiated water will be monitored for radioactivity (other than tritium) and released to the stack.
Off-gas treating ° Iodine Scrub	Removal of radioactive and other pollutants from gaseous effluents $^{a}Hg(NO_3)_2 + 4I^- \longrightarrow$ $HgI_4^= + 2NO_3^-$ $^{a}Hg(NO_3)_2 + 3I^- \dashrightarrow$ $HgI_3^- + 2NO_3^-$ $^{a}Hg/I_2$ mole ratio $\geqslant 4$	Off-gas from dissolver will pass through a scrubber where radioactive iodine will be removed by contact with dilute aqueous solution of nitric acid and mercurous/mercuric nitrate; it will subsequently pass through an acid absorber where nitrogen oxides will be removed. Dissolver off-gas and vessel off-gas streams will be combined, passed through another mercurous/mercuric nitrate scrubber, an iodine adsorber bed, and a high-efficiency filter before release to the stack.

Appendix B: Fault Trees Used in Risk Assessment

Top of the Reprocessing Plant Fault Tree

UNCONTROLLED RADIATION RELEASE (URR) TO ENVIRONMENT

URR AIR PATHWAYS

URR LIQUID PATHWAYS

URR DURING NEPTUNIUM PROCESSING

URR NORMAL OPERATIONS

URR DURING SHUTDOWN

URR FROM PLANT UNDER NORMAL AIRFLOW DIRECTION

URR FROM PLANT AND REVERSE AIRFLOW

URR THRU VENTS OR PASSAGES

REVERSE AIR FLOW

URR THRU VENT SYSTEM ONLY

URR THRU VENT SYSTEM & FUR. PASSAGES

URR THRU FUR. PASSAGES ONLY

URR Normal Airflow

URR in Fuel Receiving and Storage

LIST OF EQUIPMENT

19-C-101	01 DISSOLVER
10-C-102	02 DISSOLVER
19-C-103	03 DISSOLVER
41-D-000	HOT KILO POT
10-G-100	01 DISSOLVER HEATER/COOLER
10-G-100	02 DISSOLVER HEATER/COOLER
10-G-104	03 DISSOLVER HEATER/COOLER
41-G-001	HAW CONCENTRATOR REBOILER
41-G-000	HAW CONCENTRATOR
19-H-190	01 DISSOLVER SCREEN POT
19-H-199	02 DISSOLVER SCREEN POT
19-H-190	03 DISSOLVER SCREEN POT
00-H-001A/B	MAINTENANCE CEILING PLUG
00-H-001C/D	MAINTENANCE CEILING PLUG
00-H-001	SHIELDING DOOR
00-T-000	FUEL TRANSFER CONVEYOR
00-T-000	FUEL TRANSFER CONVEYOR DRIVE
10-T-101	TRANSVERSE FUEL PUSHER
10-T-100	MAINTENANCE CRANE 20-TON
10-T-100	BASKET HANDLING CRANE
10-T-104	POWER MANIPULATOR
10-T-100A/B	DISSOLVER BASKET STORAGE RACK
10-T-107	FUEL TRANSFER TABLE
00-T-001	DISSOLVER BASKET DUMPER
00-T-000	NULL MONITOR
00-T-011	DUMP JET CAGE
00-T-012	DUMP JET CAGE
00-T-000	PASS THROUGH PORT
07-V-000	SHIELDING WINDOW
07-V-000	SHIELDING WINDOW
07-V-000	SHIELDING WINDOW
07-V-000	SHIELDING WINDOW
07-V-010A/B	PERISCOPE SLEEVE & PLUG
07-V-010C/D	DITTO
07-V-010E/G	DITTO
07-V-010&/H	DITTO
07-V-011	SHIELDING WINDOW
10-Y-101	SHEAR
10-Y-104	SHEAR SUPPORT FRAME
10-Y-100	SHEAR FEED MAGAZINE
10-Y-100	SHEAR MAG. TRANSITION DUCT.
10-Y-107	SHEAR FEED CHAIN & MAGAZINE
10-Y-100	SHEAR FEED RAM DRIVE UNIT
10-V-102	DIVERTER
10-V-103	DIVERTER DRIVE UNIT
10-Z-101	FUEL TRANSFER HATCH

URR in Remote Process Cell

EQUIPMENT LIST

15-D-107	NO. 1 DISSOLVER TRANSFER TANK				
15-D-108	NO 2	"	"	"	
15 D 109	NO 3	"	"	"	
15 D 117	NO 4	"	"	"	
15 D 140	NO 1	"	HEAD POT	"	
15 D 141	NO 2	"	"	"	
15 D 142	NO 3	"	"	"	
15 D 143	NO. 4	"	"	"	

17 D 103	NO. 1 FEED ADJUSTMENT TANK
17 D 104	NO. 2 " " "
17 D 111	DISSOLVER FLUSH ACCUMULATOR
17 D 113	FEED SURGE TANK
17 D 144	DISSOLVER PRODUCT HEAD POT

10 D 104	ACCOUNTABILITY TANK

21 D 201	HA FEED TANK
21 D 240	HAF CONSTANT LEVEL HEAD POT
22 D 203	2DU HEAD POT

41 D 404	HCU CATCH TANK
41 D 405	HCU SAMPLE TANK
41 D 417	HAU SURGE TANK
41 D 431	HAU HEAD POT

43 D 400	GENERAL PURPOSE CONCENTRATOR TANK
43 D 412	SUMP COLLECTION TANK
43 D 447	G. P. FEED HEAD POT

61 D 602	DISSOLVER ACID SURGE TANK
61 D 617	HULL RINSE SURGE TANK
61 D 655	HULL RINSE CONSTANT LEVEL HEAD POT
61 D 656	HULL RINSE MIX FEED HEAD POT
61 D 659	NO 1 DISSOLVER ACID HEAD POT
61 D 660	NO 2 " " " "
61 D 661	NO 3 " " " "
61 D 662	NO 4 " " " "
61 D 663	DISSOLVER HULL RINSE SEAL HEAD POT

VALVE POT NO	SAMPLE POINT	DESCRIPTION
00 D 001	N-01	ACCOUNTABILITY TANK
00 D 002	N-02	NO.1 FEED ADJUSTMENT TANK
00 D 012	N-12	HAU SURGE TANK
00 D 013	N-13	LAU CONCENTRATOR FEED TANK
00 D 015	N-15	GEN PURPOSE CONCENTRATOR FEED TANK
00 D 010	N-10	HCU SAMPLE TANK
00 D 020	N-20	SUMP COLLECTION TANK
00 D 032	N 32	LAUD CHECK TANK
00 D 053	N-53	2DU LINE
00 D 061	N-61	DISSOLVER FLUSH ACCUMULATOR
00 D 062	N 62	NO.2 FEED ADJUSTMENT TANK
00 D 070	N-70	NO 1 DISSOLVER TRANSFER TANK
00 D 071	N 71	NO 2 " " "
00 D 072	N 72	NO. 3 " " "
00 D 073	N-73	NO. 4 " " "
00 D 074	N-74	FEED SURGE TANK

43 E 410	SUMP COLLECTION TANK REFLUX CONDENSER
43 E 421	GPC FEED TANK REFLUX CONDENSER

16 A 153	CENTRIFUGE

URR in High-Level Cell

HA FEED TANK 10^{-5}/YR

FAILURE OF PRIMARY BOUNDARY

RADIOACTIVE MATERIAL IN SOLUTION

PLUMBING FAILURE DUE TO CORROSION $10^{-5} - 10^{-6}$/YR

PLUMBING FAILURE DUE TO EARTHQUAKE $< 10^{-7}$/YR

PLUMBING FAILURE DUE TO EXPLOSION

PRESSUREBLOWS WATER SEAL

STEAM EXPLOSION — N.A.

HYDROGEN EXPLOSION 10^{-5}/YR

CRIT. ACC. IN PROCESS CELL 3×10^{-5}/YR

URR IN HLC

B-6

EQUIPMENT LIST

URR IN
HILC

PIPE RUPTURE $10^{-5}-10^{-6}$/YR LIQ.
SOLVENT FIRE $10^{-6}-10^{-7}$/YR VOL.

FAILURE OF
PRIMARY EQV.
CONTAINMENT
BOUNDARY

URR FROM
SOLVENT FIRE $10^{-6}-10^{-7}$/YR

PLUGGING
FAILURE DUE
TO CORROSION $10^{-5}-10^{-6}$/YR

PLUGGING
FAILURE DUE
TO CARTRIDGE

PLUGGING
FAILURE DUE
TO EXPLOSION

PRODUCT
BLOWS
MOTOR
SEAL

STEAM
EXPLOSION   NEG.

RSO OIL
EXPLOSION   NEG.

CRIT. ACC.
IN PROC. CELL 3×10^{-5}/YR

URR in High Intermediate-Level Cell

URR in Intermediate Level Cell

Text visible in the diagram:

URR IN ILC — I LC

LAW CONC FAILURE 10^{-9}/YR

FIRE IN OFFGAS FILTERS

FAILURE OF PRIMARY AQU BOUNDARY

URR FROM SOLVENT FIRE — OF — NA.

$10^{-9} - 10^{-6}$/YR

PLUMBING FAILURE DUE TO CORROSION

PLUMBING FAILURE DUE TO EARTHQUAKE — NEG. $< 10^{-3}$/YR

PLUMBING FAILURE DUE TO EXPLOSION

PRESENCE OF WATER SOME

SOLVENT BURNER EXPLOSION — NEG.

STEAM EXPLOSION — S — NEG.

RED OIL EXPLOSION — RO — 10^{-9}/YR

CRIT. ACC. IN PROCESS CELL — CP — NOT POSSIBLE

RELEASE OF WATER INTO BURNER

BURNER CONTROLS FAILURE

USE OF MORE VOLATILE SOLVENT

EQUIPMENT LIST

45-C-400	NO₃ ABSORBER
54-C-550	ACID FRACTIONATOR
42-C-40	SOLVENT BATCH STRIPPING TANK
42-C-402	SOLVENT BURNER FEED TANK
42-C-404	LAB CONCENTRATOR FEED TANK
42-C-434	ACID CHECK TANK
42-C-440	SOLVENT BURNER QUENCH POT
42-D-445	42-D-405 HEAD POT
42-C-446	ACID HEAD POT
42-D-46	42-C-440 HEAD POT
45-D-452	VACUUM BREAKER TANK
45-C-46	VESSEL OFF-GAS KO POT
45-C-472	NO₃ ABSORBER BOTTOMS HEAD POT
54-C-560	AF CHLORIDE REMOVAL HEAD POT
62-D-605	RECOVERED ACID STORAGE TANK
42-E-402	LAB CONCENTRATOR REBOILER
42-E-46	LAB CONCENTRATOR
45-E-4 5	VESSEL OFF-GAS CONDENSER
45-E-43	OFF-GAS HEATER (STEAM)
45-E-452	OFF-GAS HEATER (ELECT)
54-E-504	ACID FRACTIONATOR REBOILER
54-E-5 5	ACID FRACTIONATOR CONDENSER
54-E-5 C	RECOVERED ACID COOLER
42-F-40	WASTE SOLVENT BURNER
45-R-425	NO 1 IODINE SCRUBBER
45-R-426	NO 2 IODINE SCRUBBER
00-C-810	SAMPLER VALVE POT
00-D-81	SAMPLER VALVE POT
00-D-817	SAMPLER VALVE POT
00-D-827	SAMPLER VALVE POT
00-C-847	SAMPLER VALVE POT

EQUIPMENT LIST

URR IN PPC

SP CONC EXP. 10^{-4}-10^{-5}/YR
2A SOLVENT FIRE 10^{-6}-10^{-5}/YR
2B ION EXCHANGE FIRE 10^{-4}/YR
PU REWORK CRIT 3×10^{-5}/YR

2B 10^{-4}/YR — ION EXCHANGE RESIN FIRE (I GR)

FAILURE OF PRIMARY AQU. BOUNDARY

URR FROM SOLVENT FIRE 10^{-6}-10^{-7}/YR (S F)

PLUMBING FAILURE DUE TO CORROSION — NEG.

PLUMBING FAILURE DUE TO EARTHQUAKE — NEG.

PLUMBING FAILURE DUE TO EXPLOSION

PRESSURE BLOWS WATER SEAL

STEAM EXPLOSION (S)

RED OIL EXPLOSION (RO)   SP 10^{-4}-10^{-5}/YR

CRITICALITY ACCIDENT IN PROCESS CELL (C P)   PU REWORK TANK 3×10^{-5}/YR

URR in Plutonium Process Cell

```
70D 720          C W EXPANSION TANK
70E-737          C W EXCHANGER
70G-790A/B       C W CIRCULATION PUMPS

        PLUTONIUM NITRATE STORAGE

 36A-301          PUMP GLOVE BOX #1
□36A-302          PUMP GLOVE BOX #2
 36A-303          VALVE GLOVE BOX #1
□36A-304          VALVE GLOVE BOX #2
 36A-305          SAMPLE GLOVE BOX #1
□36A-306          SAMPLE GLOVE BOX #2
 36A-307          PN LOADOUT GLOVE BOX
 36A-308          MAINTENANCE GLOVE BOX

 36D-308A THRU F  PLUTONIUM NITRATE STORAGE TANKS
 36D-309A THRU F  PLUTONIUM NITRATE STORAGE TANKS
 36D-310A THRU F  PLUTONIUM NITRATE STORAGE TANKS
 36D-311A THRU F  PLUTONIUM NITRATE STORAGE TANKS
□36D-312A THRU F  PLUTONIUM NITRATE STORAGE TANKS
□36D-313A THRU F  PLUTONIUM NITRATE STORAGE TANKS
□36D-314A THRU F  PLUTONIUM NITRATE STORAGE TANKS
□36D-315A THRU F  PLUTONIUM NITRATE STORAGE TANKS
 36D-316          50 LITER NITRATE SAMPLE TANK
 36D-317          10 LTR NITRATE MEASURING TANK

 36G-308          P.H.S. TRANSFER PUMP
 36G-309          P.H.S. TRANSFER PUMP
 36G-310          PHS TRANSFER PUMP
 36G-311          PHS TRANSFER PUMP
□36G-312          PHS TRANSFER PUMP
□36G-313          PHS TRANSFER PUMP
□36G-314          P.H.S. TRANSFER PUMP
□36G-315          P.H.S. TRANSFER PUMP
 36G-316          PNLO TRANSFER PUMP
 36G-317          SUMP PUMP PN CELL #1
□36G-318          SUMP PUMP PN CELL #2

 75B-750A/B       TANK VAULT COOLING UNIT
□75B-751          TANK VAULT COOLING UNIT

 75K-750          PNLO GLOVE BOX INLET FILTER
 75K-751          PNLO GLOVE BOX EXH FILTER
 75K-752          PUMP GLOVE BOX INLET FILTER
 75K-753          PUMP GLOVE BOX EXH. FILTER
 75K-754          PUMP GLOVE BOX INLET FILTER
□75K-755          PUMP GLOVE BOX INLET FILTER
□75K-756          PUMP GLOVE BOX EXH. FILTER
□75K-757          PUMP GLOVE BOX INLET FILTER
 75K-758          VALVE GLOVE BOX INLET FILTER
 75K-759          VALVE GLOVE BOX EXH. FILTER
 75K-760          VALVE GLOVE BOX INLET FILTER
□75K-761          VALVE GLOVE BOX INLET FILTER
□75K-762          VALVE GLOVE BOX EXH FILTER
□75K-763          VALVE GLOVE BOX INLET FILTER
 75K-764          SAMPLE GLOVE BOX EXH. FILTER
 75K-765          SAMPLE GLOVE BOX INLET FILTER
□75K-766          SAMPLE GLOVE BOX EXH. FILTER
□75K-767          SAMPLE GLOVE BOX INLET FILTER
 75K-768A/B       TANK VAULT RECIRCUL. FILTER
 75K-769          EMERGENCY INLET FILTER

 80D-8105         SAMPLE VALVE POT
 80D-8106         SAMPLE VALVE POT
 80D-8107         SAMPLE VALVE POT
 80D-8108         SAMPLE VALVE POT
□80D-8109         SAMPLE VALVE POT
□80D-8110         SAMPLE VALVE POT
□80D-8111         SAMPLE VALVE POT
□80D-8112         SAMPLE VALVE POT
```


URR in Plutonium Loadout Cell

LIQUID & PART. RELEASE TO SOIL — LPS — 3×10^{-7}/YR

GASEOUS & VAPOROUS RELEASE TO UPS — UV — 3×10^{-3}/YR

STACK RELEASE FROM HIGH LEVEL LIQUID WASTE TANK — UTS — 4×10^{-7}/YR

FAILURE OF CELL LINER LEAK DETECT. — 10^{-4}/YR

FAILURE OF W.T. CELL LINER — 10^{-3}/YR

HLLW TANK FAILS INTO WASTE TANK VAULT — 3×10^{-3}/YR

EXCESSIVE RELEASE OF FISS. PROD. VAPORS DUE TO HEAT — 10^{-5}/YR

FAILURE OF WASTE OFF GAS SYSTEM — OG — 2×10^{-2}/YR

FAILURE OF VESSEL OFF GAS SYSTEM — OG — 2×10^{-2}/YR

TANK MAT. FAILS — 2×10^{-3}/YR

EARTHQUAKE IN EXCESS OF DBA FAILS TANK — 10^{-7}/YR

TANK FAILS FROM H2 EXPLOSION — H — 10^{-3}/YR

TANK FAILS FROM EXCESSIVE HEAT

MATERIAL WEAKNESS — 10^{-3}/YR

GTCM FAILURE — 10^{-3}/YR

FAILURE OF HEAT TRANS. FROM HLLW — HT — 3×10^{-5}/YR

GTCM RATE HIGHER THAN DESIGN — 10^{-1}/YR

FAILURE TO ACT ON GTCM PATCH INDICAT. — 10^{-2}/DEMAND

URR from High-Level Liquid Waste Tank

Failure of Heat Transfer from HLLW

FAILURE TO INSTITUTE EMERGENCY REFLOOD
— ER — 10^{-2}/DEMAND

$< 10^{-6}$/YR

FAILURE OF WARNING — $< 10^{-4}$/YR

FAILURE OF INSTR POWER — 10^{-5}/YR

FAILURE OF WIRING

CORRECT PERSONNEL ACTION FAILURE — 10^{-2}/DEMAND

DISSOLVER OFF GAS SYSTEM FAILURE — DOG — 2×10^{-2}/YR

FAILURE OF DOG BEFORE VD

FAILURE OF VD — VD — 10^{-2}/YR

REACTOR STEAM FAILURE — N.O. EST. $< 10^{-4}$/YR

FAILURE OF PRIMARY BOUNDARY — 10^{-5}/YR

COOLING WATER SUPPLY FAILURE — CW — 10^{-2}/YR

VESSEL OFF GAS SYSTEM FAILURE — VOG — 2×10^{-2}/YR

FAILURE OF VOG BEFORE VD

FAILURE OF VD — VD — 10^{-2}/YR

FAILURE OF PRIMARY BOUNDARY — 10^{-5}/YR

COOLING WATER SUPPLY FAILURE — CW — 10^{-2}/YR

Failure to Institute Emergency Reflood

Cooling Water Failure
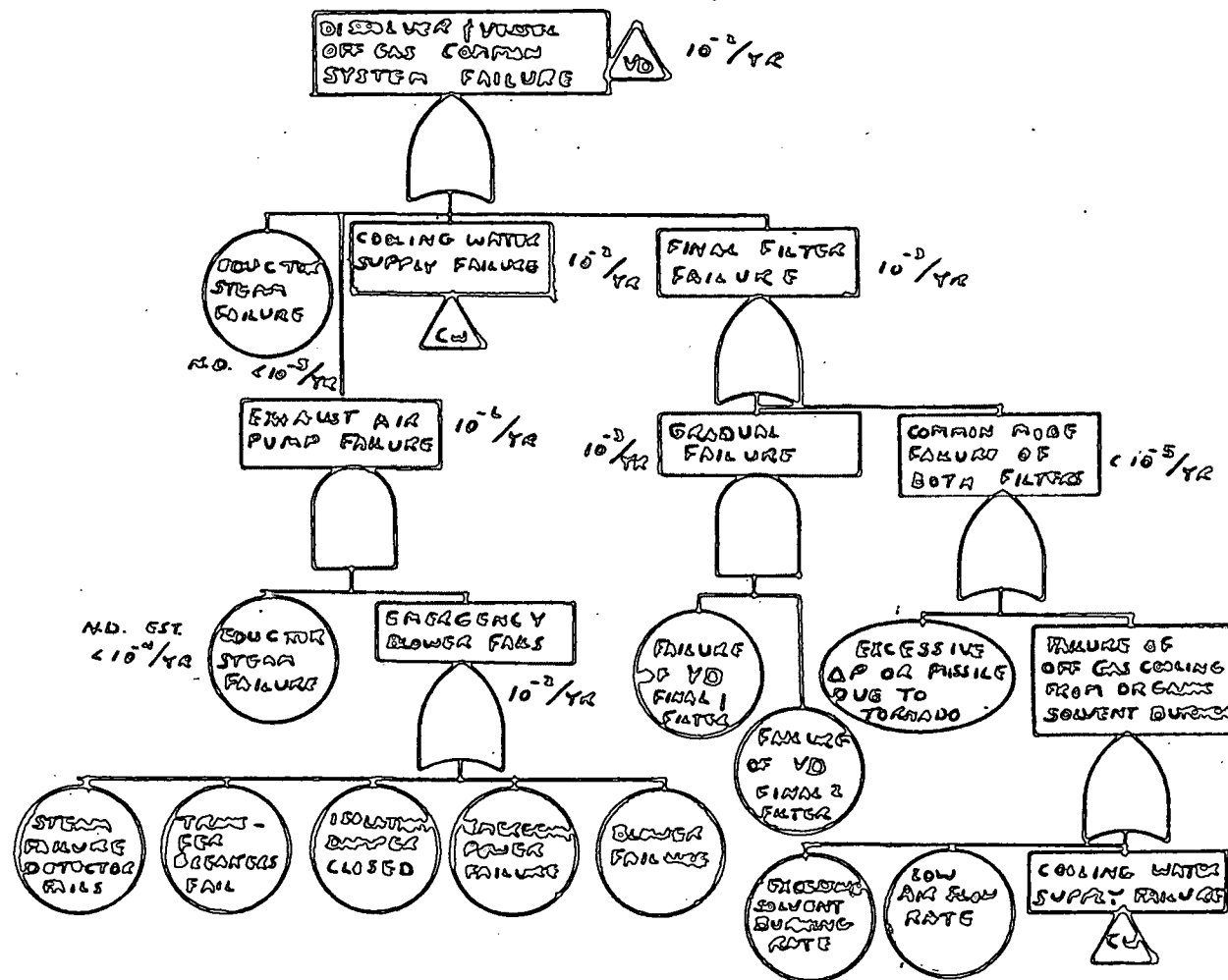
Dissolver and Vessel Off-Gas Common System

URR FROM RESIN REACTION — RRR 10^{-6}/YR

CONTROLLED RELEASE FROM RESIN REACTION — 10^{-6}/YR

EXPLOSIVE RELEASE FROM RESIN REACTION — 10^{-7}/YR

SUCCESSFUL DEPRESS-URIZATION — 10^{-1}/DEMAND

RESIN REACTION — RR 10^{-5}/YR

FAILURE OF DEPRESS-URIZATION — 10^{-2}/DEMAND

EXCESSIVE ACIDITY — 5×10^{-2}/YR

EXCESSIVE TEMPERATURE — 10^{-1}/YR

EXCESSIVE CONTACT TIME — 6×10^{-2}/YR

IMPROPER DILUTION SOLUTION — 10^{-2}/YR

FAILURE OF CONTROL — 10^{-3}/YR

NON-SELF. FAILURE OF VALVE — 3×10^{-2}/D

CLOGGING OF FILTER CHANNELS — 10^{-2}/YR

FAILURE OF STEAM PRESS. REG. — 10^{-1}/YR

FAILURE OF NON CONTROL — 6/YR

PROGRAM FAILURE — 10^{-3}/DEMAND

FAILURE OF AUTOMATIC CONTROL — 1/YR

FAILURE OF PERSONNEL INTER-VENTION — 10^{-2}/DEMAND

FAILURE OF PH SENSORS — 5×10^{-1}/YR

FAILURE OF SERVO CONT. — 5×10^{-1}/YR

Solvent Fire in Partition and Purification

Red Oil Explosion

STEAM EXPLOSION — S

HAW $< 10^{-6}$/YR

SMALL ORGANIC 10^{-1}/YR

PRESSURE BUILD-UP — 4×10^{-3}/YR

FAILURE OF PRESSURE RELIEF — 10^{-2}/DEMAND

FAILURE OF TECHNICIAN INTERRUPTION — 10^{-2}/DEMAND

TEMP. CONTROL FAILURE — 4×10^{-2}/YR

EXOTHERMIC REACTION IN RESIN COLUMN — 10^{-1}/YR

HUMAN FAILURE TO RESPOND — 10^{-2}/DEMAND

WARNING FAILURE — 10^{-3}/DEMAND

COOLING FAILURE — 4×10^{-2}/YR

STEAM HEATER CONTROLLER FAILS — C — 10^{-2}

RESIN COLUMN USED

COOLING FAILURE

PRESS. SENSOR FAILURE

TEMP SENSOR FAILURE

LIMIT DETECTOR FAILURE

COLD WATER FAILURE — 3×10^{-2}/YR

COOLING PUMP FAILURE — P — 3×10^{-2}/YR

CONTROLLER FAILS UNSAFE — C — 10^{-3}/YR

CONTROLLER FAILURE

FAILURE OF COLD WATER SUPPLY

LOSS OF COOLING PUMP — P — 10^{-1}/YR

ALARM FAILURE

INSTR. POWER FAILURE

Steam Explosion

Criticality Accident in Process Cell

Controller Fails Unsafe

Failure of Personnel Intervention

On-Site Power Failure

MONITOR

2×10^{-1}/YR

10^{-5}/YR

FAILURE OF
DECARBONATED
SCRUBBER
WATER
S

FAILURE
OF IODINE
SCRUBBER

STEAM
FAILURE

HEPA
FILTER FAILURE
ACCELERATED
IN HOT AIR

10^{-3}/YR 10^{-1}/YR 10^{-3}/YR 10^{-3}/YR
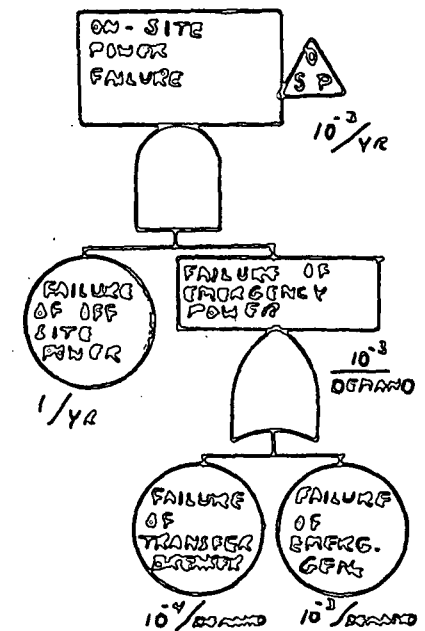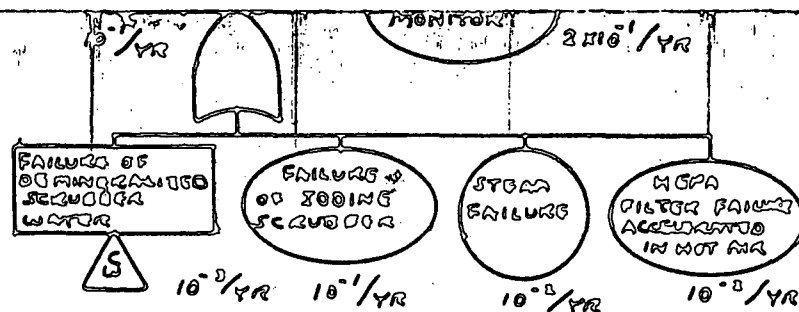
Acid Fraction Overhead

PUMP
FAILURE   P 3×10^{-1}/YR

CONTROL-
LER
FAILURE

MOTOR
FAILURE

PUMP
LOCKS

COOLING
FAILURE

10^{-1}/YR 3×10^{-1}/YR   N.O.   N.O.

POWER
FAILURE 10^{-3}/YR

FAILURE
OF OFF
SITE
POWER

FAILURE
OF EMGR.
POWER

$^{-1}$/YR 10^{-3}/DEMAND
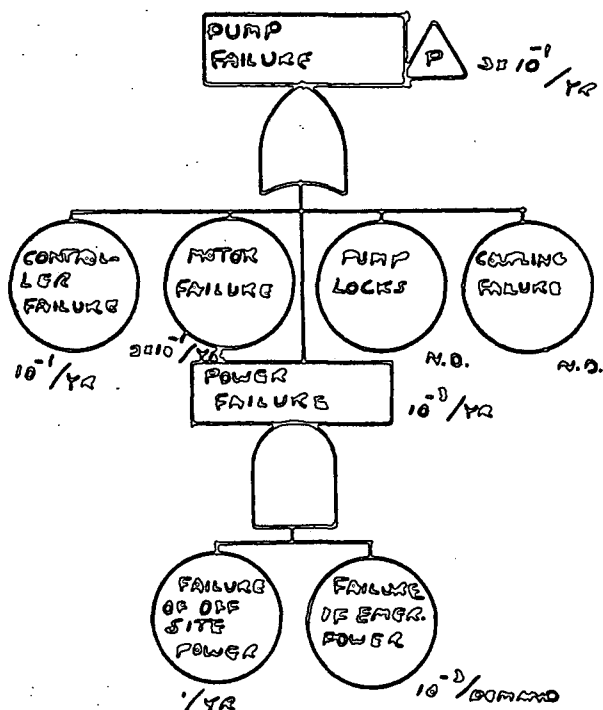
Pump Failure

APPENDIX C

Descriptions of Accidents Experienced in the Nuclear
Energy Field and Chemical Industry Relating to Anticipated
Credible Events at a Fuel Reprocessing Plant.

This Appendix reviews the types of accidents which have occurred in AEC operational activities relating to operations performed at a nuclear fuels reprocessing plant. The information is drawn from AEC field office reports as described in USAEC Report WASH-1192[34] and supplemental information supplied by the USAEC Division of Operational Safety.[54] There is a tendency to emphasize exposures and contamination from criticality incidents in discussions concerning radiation hazards because of the AEC's involvement. These occurrences, however, are in the minority in the overall picture of potential environmental impact. By far, the majority of potential incidents which could occur during activities at a fuel reprocessing plant, that might have an environmental impact are comparable, in kind, to those occurring in the chemical industry. To broaden the statistical base for accident probability estimates for this study, available relevant accident case histories from the chemical industry are also included in this Appendix.[21] These case histories, covering the period 1951 through 1972 are voluntarily submitted to the Manufacturing Chemists Association for publication in an endeavor to improve plant safety in the industry. This compilation, to be sure, is not complete. It does, however, indicate the types of accidents that have occurred in the operation of a reprocessing plant or related facility and was utilized in selecting the hypothetical accidents considered in this study.

The abbreviations used for the USAEC field offices are as follows:

AL   Albuquerque Operations Office
BH   Brookhaven Office

CH	Chicago Operations Office
GJ	Grand Junction Office
HA	Hanford Operations Office
ID	Idaho Operations Office
LAR	Lockland Aircraft Reactors Office
NV	Nevada Operations Office
NY	New York Operations Office
OR	Oak Ridge Operations Office
PNR	Pittsburgh Naval Reactors Office
RL	Richland Operations Office (Formerly HA)
SAN	San Francisco Operations Office
SNR	Schenectady Naval Reactors Office
SR	Savannah River Operations Office
SNPO-C	Space Nuclear Propulsion Office-Cleveland
SNPO-N	Space Nuclear Propulsion Office-Nevada

1.   Criticality Accidents

a.   Oak Ridge, Tenn., June 16, 1958

A nuclear accident occurred in a 55-gallon stainless steel
drum in a processing area in which enriched uranium is re-
covered from various materials by chemical methods in a
complex of equipment. This recovery process was being re-
modeled at the time of the accident.

The incident occurred while they were draining material
thought to be water from safe 5-inch storage pipes into
an unsafe drum.

Eight employees were in the vicinity of the drum carrying
out routine plant operations and maintenance. A chemical
operator was participating in the leak testing which inad-
vertently set off the reaction. He was within three to
six feet of the drum, while the other seven employees were
from 15 to 50 feet away.

Using special post hoc methods for determining the neutron
and gamma exposures of the employees involved, it was
estimated that the eight men received: 461 rem, 428 rem,
413 rem, 341 rem, 298 rem, 86 rem, 86 rem, and 29 rem.

Area contamination was slight, with decontamination costs
amounting to less than $1,000.

During this incident 1.3 x 10^{18} fissions occurred.

b.   <u>Los Alamos, N. Mex., Dec. 30, 1958</u>

The chemical operator introduced what was believed to be a dilute plutonium solution from one tank into another known to contain more plutonium in emulsion. Solids containing plutonium were probably washed from the bottom of the first tank with nitric acid and the resultant mixture of nitric acid and plutonium-bearing solids was added to the tank containing the emulsion. A criticality excursion occurred immediately after starting the motor to a propeller type stirrer at the bottom of the second tank.

The operator fell from the low stepladder on which he was standing and stumbled out of the door into the snow. A second chemical operator in an adjoining room had seen a flash, which probably resulted from a short circuit when the motor to the stirrer started, and went to the man's assistance. The accident victim mumbled he felt as though he was burning up. Because of this, it was assumed that there had been a chemical accident with a probable acid or plutonium exposure. There was no realization that a criticality accident had occurred for a number of minutes. The quantity of plutonium which actually was present in the tank was about ten times more than was supposed to be there at any time during the procedure.

The employee died 35 hours later from the effects of a radiation exposure with the whole-body dose calculated to be 12,000 rem ±.

Two other employees received radiation exposure of 134
and 53 rem, respectively. Property damage was negligible.
(See TID-5360, Suppl. 2, p. 30; USAEC Serious Accidents
Issue #143, 1-22-59.)

c.   Idaho Falls, Idaho, Oct. 16, 1959

A nuclear incident occurred in a process equipment waste
collection tank when an accidental transfer was made of
about 200 liters of uranyl nitrate solution, containing
about 34 kilograms of enriched uranium (91 percent U^{235}),
from critically safe process storage tanks to a geometric-
ally unsafe tank through a line formerly used for waste
transfers.

Limited visual inspections and test indicated that no sig-
nificant property damage or loss resulted beyond the
approximately $60,000 cost to recover contaminated uranium
solution resulting from the incident.

Of the 21 personnel directly involved in this incident,
seven received external exposures to radiation. The ex-
posures were 8, 6, 3.95, 1.50, 1.38, 1.17, and 1.17 rem.
Two individuals also received external exposures to the
skin of 50 rem and 32 rem. No medical treatment was re-
quired for the 21 personnel involved.

d.   Idaho Falls, Idaho, Jan. 25, 1961

A nuclear excursion of approximately 6×10^{17} fissions
occurred in a first-cycle product evaporator at a chemical
processing plant. The criticality accident resulted when

a solution of enriched uranyl nitrate accidentally surged
from a geometrically safe section of the evaporator into
the upper critically unsafe, vapor disengagement section.
The accident occurred behind thick concrete walls in a
processing cell which is part of the first cycle for pro-
cessing highly radioactive spent-fuel elements.

Personnel response to the radiation alarms and the evacua-
tion signal was prompt and orderly.

Analyses of badges from 65 individuals indicated a maximum
exposure of 55 millirem gamma and 0 beta. The maximum
thermal neutron exposure detected in the badges analyzed
was less than 10 millirem. Analyses of nuclear accident
dosimeters indicated that there was negligible fast neutron
flux associated with personnel exposures.

The radioactivity released to the atmosphere as a result
of the accident was about twice normal background when
it left the area. Loss of $6,000 resulted from cleanup of
the incident.

e.   Richland, Wash., Apr. 7, 1962

An unplanned nuclear excursion occurred in a plutonium
processing facility because of the inadvertent accumulation
of approximately 1500 grams of plutonium in 45-50 liters
of dilute nitric acid solution in a 69-liter glass transfer
tank. The sequence of events which led to the accumulation
of the plutonium in the tank cannot be stated positively.
However, it is believed that, when a tank valve was opened,
the solution from another process vessel overflowed to a

sump and was drawn into the transfer tank through a temporary line between this tank and the sump.

When the excursion occurred, radiation and evacuation alarms sounded. All but three employees left the building immediately, according to well-prepared and well-rehearsed evacuation plans. Fortunately, they were not in close proximity to the involved system nor in a high radiation field.

The course of the nuclear reaction involved initial criticality (10^{16} fissions); a subsidence; one or more later peaks; and after approximately one-half hour, a declining rate of fission, which terminated in a subcritical condition 37 hours later. The total number of fissions was approximately 8×10^{17}.

Of the 22 persons in the building at the time, only four employees, those who were in the room with the system, were hospitalized for observation. Three of them were the system operators, who were in close proximity to the excursion, and who received estimated radiation doses of 110, 43, and 19 rem. None of them showed symptoms definitely referable to their radiation exposures. The fourth was sent to the hospital only because he was in the room at the time of the incident.

Some fission product activity, airborne via the vent system and the exhaust stack, was detected in the atmosphere for a brief period after the accident. The physical damage amounted to less than $1,000.

f.   <u>Wood River Junction, R.I., July 24, 1964</u>

Because of startup difficulties an unusually large amount
of highly enriched uranium-contaminated trichloroethane
(TCE) had accumulated at United Nuclear's Scrap Recovery
Plant. The recovery was by mixing the TCE with a sodium
carbonate solution. On the day of the incident the pro-
cess was shifted to an 18 in. dia. by 25 in. deep tank
to try to catch up with the backlog. The plant evaporator
failed resulting in a plug of uranium nitrate crystals in
the converting line. This plug was dissolved with steam
and the concentrated solution was drained into a polyethy-
lene bottle. This bottle was mistaken for trichlorethane
and the operator poured it into the tank of solution. Cri-
ticality was reached in a burst of 10^{17} fissions creating
a flash of light. 1/5 of the solution was ejected and the
operator knocked to the floor. He ran to the emergency
building 200 yards away but having received about 10,000
rad, died 49 hours later.

Later two men entered the area to drain the solution into
safe containers. When the stirrer was turned off, the
geometry change resulted in a second criticality of 2-3x
10^{16} fissions and these men received 60 and 100 rads.

g.   <u>Oak Ridge, Tenn., Jan. 30, 1968</u>

Unexpected criticality was achieved in a volume of an
aqueous solution of a salt of U^{233} during a series of rou-
tine critical experiments in progress in a well-shielded
assembly area of a critical experiments facility. The

criticality-radiation alarm system functioned as designed, the evacuation of personnel from the building was prompt and orderly, and the excursion was terminated expeditiously by a negative coefficient of reactivity and was prevented from recurring by the action of the safety devices. The fission yield was 1.1 x 10^{16}. Gamma-ray sensitive personnel dosimeters read immediately following the excursion showed no direct exposure greater than 5 mr to any person present. There was no property damage or loss of fissile materials. An estimated 100 cm^3 of solution (15 g of U) were spilled when a rubber-stoppered connection immediately above the sphere was dislocated.

The purpose of the particular experiment in progress was to establish the critical concentration of a sphere of the solution of uranyl nitrate surrounded by a thick water reflector. In the course of approaching criticality by incremental additions of solution, a small volume of air was observed entrapped in a flexible transparent tube. Supercriticality occurred during an attempt, by remote manipulation of liquid levels, to remove the air.

2. <u>Fires</u>

<u>Year</u>	Reporting AEC <u>Field Office</u>	
1959	OR	Multiple circuit breaker fail-ure led to severe electrical fire. Property damage $86,000. No exposures.
	LAR	Electrical fire due to severe arc-ing on the lineside of heater break-ers. Property damage $30,000. No exposures.
1960	HA	Fire and explosion in pyrophoric metal contents of a chemical dis-solver, off-gas filter, and rela-ted process equipment. Contamina-tion spread to cell, canyon and crane. Cause(s) of the accident not established. Property damage $250,443. No exposures.
1962	OR	Fire occurred in ventilation system, probable cause electrical spark. Property damage $24,700. No expo-sures.
	HA	Air ventilation equipment failure. Property damage $10,000. No expo-sures.
	OR	Explosion and fire in cell. Pro-perty damage $2,900,000. No expo-sures.
1963	OR	Fire (definite cause undetermined) originated in building exhaust system and was confined to labora-tory hoods and exhause system; smoke damaged building. Property damage $43,400. No exposures.
	RL	Fire (definite cause undetermined) in plutonium purification facility. Pu contamination in immediate area of fire. Firemen received slight skin contamination, readily removed. Costs related directly to fire $85,400; decontamination costs $251,300; overhead related to direct losses $60,300.

2. <u>Fires (Continued)</u>

<u>Year</u> | Reporting AEC
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1964   SR   Fire (definite cause undetermined) occurred around an anion exchange column in hot canyon. Fire caused airborne contamination to crane used for remote maintenance. Water to quench fire damaged electric motors. Property damage $21,000.

1965   RL   During an aluminum jacket dissolution in a dissolver, an exothermic reaction involving ammonia and/or hydrogen occurred with an electric heater, through which these gases were accidentally vented. The reaction, which continued for 3 hours, totally destroyed the heater. Property damage $7,200. No exposures.

1966   SR   A fire occurred when a drying oven overheated. Faulty loading blocked the thermostat sensing element, causing it to indicate erroneously low temperature and call for additional heat. Property damage ($6,000) was confined to the room of origin. No exposures.

          OR   A fire, of undetermined cause, occurred in a laboratory. It was confined to one hood and a section of ductwork because of the successful operation of a sprinkler head, a fire damper in the exhaust system and other fire protection controls. Property damage $5,500. No exposures.

1968   RL   An electrical short circuit and the resulting power arc in the main electrical switchgear damaged two breakers extensively, when the lights went out and the building ventilation stopped. Emergency actions were taken to preclude any contamination spread. Operations were curtailed for two and one-half days while repairs were being made. Property damage $34,000. No exposures.

2.   Fires (Continued)

Case History No. - Manufacturing Chemists Association

41	"Static Spark Flashes "Empty" Styrene Drum"
129	Fire Due to Static Spark - Benzene
141	Toluene Vapors - Flash Fire
150	Escape of Vapor From Condenser
255	Ignition of Solvent Vapors - Employee Burned
341	Solvent Fire
348	Electric Mixer Fires Solvent
612	Waste Solvent Fire
643	Flash Fire in Exhaust Duct
699	Toluene - Static Fire
701	"Boil Over" - Flash Fire
1025	Flammable Solvents - Electric Motor
1217	Kerosene Vapor Flash - Synthesis Kettle
1234	Zirconium Fines Flash Fire
1966	"Static" Ignition of Flammable Solvent
1970	Solvent-Vapor Flash Fire

3.   Explosions

Year	Reporting AEC Field Office	
1959	SR	Gasket on head of secondary condenser in unit failed. Relief valve vented open due to overpressure. 4-5T of H_2S gas released to atmosphere. $7,000 property damage. No exposures.
	HA	Plutonium glovebox explosion. $9,500. property damage. No exposures.
	HA	Autoclave explosion. Property damage $4,000. No exposures.
	AL	Drybox explosion. Property damage $1,933. No exposures.
	OR	Chemical explosion in innercycle evaporator. Property damage $350,000. No exposures.
	OR	An explosion occurred in a digester. Property damage $10,000+. No exposures.
1960	OR	Hydrogen gas explosion occurred in gas furnace enclosure in metal plant. Property damage $5,000. One employee suffered serious injuries.
	OR	Explosion occurred in a uranium sintering furnace located in a foundry. Major structural damage to furnace and buildings. Property damage $20,000. No exposures.
	AL	The accidental discharge of radioactive material into a room occurred as a result of pressure buildup in a drybox. This was due to an inlet solenoid being locked in the open position and a venting solenoid being closed due to a malfunction. The pressure built up to a point that one of the drybox gloves blew out, thereby releasing radioactive particulate material into the room. Property damage $31,360. Eleven persons received minor exposures.

3.   Explosions (Continued)

Year	Reporting AEC Field Office	
1961	AL	Pressure buildup in closed caustic scrubber system forced airborne radioactive material into room. Property damage $4,016. One employee received 71 rem exposure to bone. Area contaminated.
1963	ID	Low-level spread of plutonium contamination from glovebox. Property damage $25,451. No exposures.
1964	AL	Chemical explosion in metal hood when methanol vapors reached flashpoint. Two sets of gloves were shredded by the explosion. Contamination spread in operating area. Property damage $34,922. Three employees received slight contamination.
1965	RL	An explosion occurred in a boiler during an attempt to relight the oil-fired burner with a kerosene torch after the automatic re-ignition system failed to function. Property damage $75,000. No exposures.
	RL	Plutonium contamination spread following an explosion and fire occurring in a glovebox when cleaning fluid ignited. Ten employees left the building immediately. Prompt showering easily removed all skin contamination. One employee received 10% of a maximum permissible body burden (bone) of Pu-239 by inhalation. Contamination did not spread outside the building. About 90% of the cost ($76,800) was incurred for decontamination.

3.   Explosions (Continued)

Year	Reporting AEC Field Office	
1965	AL	An explosion and fire occurred when acetone fumes from a "cocoon" used in a glovebox paint-stripping operation, contacted a hot muffle furnace in another part of the glovebox line. Plutonium contamination spread to adjacent rooms and the second floor. 12 employees required skin decontamination; none received internal radiation exposures. Property damage costs ($23,253) was for decontamination of facilities.
	CH	An explosion resulting from the ignition of a hydrogen-air mixture, the hydrogen apparently evolved from nickel-iron batteries, occurred in the equipment airlock joining a reactor building and a fuel cycle facility. No radioactive material was involved. Property damage $22,600.
	AL	An undetermined small quantity of Pu-238 was released when a double-contained vessel, nearly full of drybox seived material exploded, dispersing a quantity of the waste material into the laboratory. Property damage $19,100. No exposures.

3. **Fires (Continued)**

Case History No. - Manufacturing Chemists Association

103	Nitrating Operation Explosion
116	Storage Battery Explosion
128	Nitrogen Peroxide - Cyclohexane Mix Explosion
131	Nitric Acid - Waste disposal Explosion
163	Reaction in Solvent Recovery Tank
223	Laboratory Explosion - No Injuries
258	Explosion - Ignited by Vacuum Cleaner
347	Hydrogen dessicator - Drainage Trench Explosion
569	Runaway Nitration Reaction
576	Hydrogen Compressor - Explosion
611	Oil Vapor Explosion
678	Explosion in Nitrobenzene Recovery Kettle
679	Unsafe "Fail Safe" Flame Safety Device
703	Explosion in Vent Stack - Static Generation
859	Spilled Four Gallons Solvent on Laboratory Floor - Fire
880	Chemical Fire - Azido Compound
976	Silver Complex Detonation
987	Explosion and Fire - Lead Azide
988	Tank Explosion
1048	Explosion - Silver Oxide

3. Explosions (Continued)

Case History No. – Manufacturing Chemists Association

1068	Gas Explosion – lighting Burner
1097	Explosion – Hydrogen gas vent stack
1105	Americium Solution Shipping Container Explosion
1188	Hazardous Solvent Causes Explosion in a Plutonium Fuels Laboratory Glovebox (Furnace)
1310	Flammable Vapor Explosion – Slurry Mix Tank
1311	Nitration Explosion – Organic Intermediate Mixed with Nitric and Sulfuric Acids
1496	Drums burst from internal pressure – Accumulation of Hydrogen within
1499	Tank Rupture – Organic Material Nitrated with Nitric Acid
1554	Ammoniacal Silver Nitrate Explosion
1733	Laboratory Explosion – Silver Nitrate, Ammonium Hydroxide, Alcohol Silvering Solution
1957	Disposal of Deposits of Metal Azides
1958	Mix Tank Explosion

4.   Fuel Receiving & Storage

Year	Reporting AEC Field Office	
1960	SR	During shipment of irradiated fuel elements, 30 to 40 gallons of contaminated water leaked from the cask. $24,000 cost due to decontamination of area. No exposures.
1964	CH	Broken valve on autoclave, housed in lead shipping cask, allowed contaminated water to seep out of cask during transit, contaminating container and truck floor. No property damage.
1967	RL	A diesel locomotive collided with a cask car during coupling operations, due to the inattention of the diesel's engineer. The cost of $5,124 was for repairing the locomotive; the cask car was not damaged. No exposures.

5.   Waste and Product Storage

Year	Reporting AEC Field Office	
1959	SR	Leaking compression fitting. $20,000 damage. No exposures.
	SR	Solution leaked from the loosened flange during maintenance work on a waste evaporator in hot canyon, vaporized and contaminated crane. Property damage $129,324. No exposures.
	SR	Loose contaminated particles on the lid of a waste burial box were scattered by the wind, contaminating the ground, locomotive and spacer car. Property damage $5,200. No exposures.
1960	SR	Contaminated cooling water discharged from canyon onto floor. Property damage $250,000 (due to decontamination). No overexposures.
1961	HA	Uranyl nitrate (1355 lbs. of depleted uranium) lost to ground when tank trailer was overfilled due to misunderstanding between regular operators and their lunch relief. Property damage $13,000. No exposures.
	HA	Approximately 1,089 pounds of depleted uranium lost to chemical sewer in plant. Property damage $9,000. No exposures.
1963	AL	Leak in line carrying high-level plutonium solution caused contamination of building and equipment. Property damage $8,364. No exposures.
	AL	Spill of contaminated nitric acid solution. Property damage $5,662. No exposures.

5.   Waste and Product Storage (Continued)

Reporting AEC
Field Office

1964   CH   Clean water, being used to test two
                          new waste tanks, was contaminated by
                          condensation from contaminated vent
                          line connected to one tank. Water
                          subsequently drained onto asphalt
                          surface, contaminating it and drain-
                          age ditch. Property damage $6,075.
                          No exposures.

            ID   During steam flushing to remove rad-
                          ioactive contamination from pipelines
                          to permit tie-in to new lines, leak
                          developed in hose coupling. Contam-
                          inated fluid and steam issuing from
                          leak were rapidly dispersed by high
                          wind over approximately 10 acres.
                          Majority of $12,884 cost due to
                          cleanup.

            SR   Leaking nitric acid corroded canyon
                          cell equipment beyond repair. Pro-
                          perty damage $6,000. No exposures.

1965   AL   While attempting to activate a prod-
                          uct transfer line, contaminated acid
                          solution was sprayed out of a flanged
                          union that had not been tightened.
                          Three contaminated employees were
                          readily decontaminated. Property dam-
                          age $7,557 for decontamination and for
                          replacing contaminated equipment.

            SR   Process water (2400 lbs) was lost when
                          it leaked through an unseated sleeve.
                          Property damage $33,600. No exposures.

            SR   Process water (700 lbs) was lost when
                          it leaked through an unseated sleeve.
                          Property damage $9,800. No exposures.

            SR   A cooling coil in a vessel developed
                          a leak and allowed contaminated solu-
                          tion from a tank to enter the cooling
                          water system when the coil was pres-
                          surized. The cost for cleaning the
                          system and associated work was $19,500.
                          No exposures.

5.   Waste and Product Storage (Continued)

Year	Reporting AEC Field Office	
1966	RL	An estimated 420 lbs. of uranium solution were lost to radioactive waste through a milling tank overflow, caused by the failure of a normally closed supply line valve. Property damage $7,200. No exposures.
	RL	During the repair of an air circulation valve, approximately 10 gallons of high-level radioactive waste solution were spilled onto the floor. Three employees, wearing protective clothing, were sprayed with droplets of the solution, but were readily decontaminated. Property damage $19,746.
	RL	Less than 5 grams of concentrated plutonium nitrate solution spilled on the elevator floor when a product receiver assembly overturned and the lid of the inner container came off. Cost for decontamination was $13,443. No exposures.
1967	AL	Abandoned storage vessels inside a stainless steel glovebox were being flushed with 7-9N nitric acid to recover plutonium nitrate. During this operation, the air monitor alarmed, and the odor of nitric oxide fumes was detected. Shortly thereafter, a puddle of dark liquid (plutonium nitrate solution) was seen on the floor under the glovebox. The solution had leaked from one of the storage vessels into the glovebox well and thence onto the floor of the room. 3 contaminated personnel were readily decontaminated by showering; the cost for decontamination of the storage vessel area was $16,465.

5.   <u>Waste and Product Storage (Continued)</u>

<u>Year</u>   Reporting AEC
       <u>Field Office</u>

1967   SR   Radioactive liquid waste was stored
                   in an underground tank. The pipeline
                   for the concentrate entered the stor-
                   age tank through a shielded riser,
                   extending from the top of the tank to
                   approximately one foot above ground.
                   Access plugs were sealed with mastic
                   compounds. The inlet pipe entered
                   the riser horizontally below ground
                   and terminated with a valve near the
                   center of the riser. When the riser
                   became plugged with concentrate crys-
                   tals below the inlet line, the liquid
                   flow reversed and forced its way
                   through the access plugs. Approximately
                   13 curies of radioactive liquid waste,
                   primarily cesium, were released to plant
                   streams but sampling showed that radio-
                   activity concentration standards were
                   not exceeded in streams beyond the
                   plant boundary. The cost for decontam-
                   ination of ground in the vicinity of
                   the tank was $49,179. No exposures.

1969   SR   Approximately 20 millicuries of air-
                   borne radioactive contamination
                   (mostly curium-244) were released via
                   an exhaust stack and were spread by a
                   northeasterly wind across the roof of
                   a building and along a line leading
                   from the main entrance of the building
                   to a parking lot. The level of radio-
                   activity on the roof was 4×10^7 d/m/100cm^2
                   and on vehicles inside the area fence
                   1.5×10^4 d/m/100cm^2. The highest level
                   of activity outside the area was approx-
                   imately 5000 d/m/100cm^2. All activity
                   was contained within the plant's boun-
                   dary. The cost for decontamination
                   was $37,506. No radiation exposures.

         SR   Acidic waste solution (approximately 8200
                   pounds), which was being processed for
                   neptunium-237 and plutonium-238 recovery,
                   was lost when inadvertently transferred
                   to an underground waste system due to a
                   leaking valve in the stream supply to a
                   transfer jet. Property damage $32,000.
                   No exposures.

5.   Waste and Product Storage (Continued)

Year   Reporting AEC
        Field Office

1970   SR   A solution containing 20 grams of
                            curium-244 and americium-243 was
                            transferred by mistake to the waste
                            system. Property damage $124,523.
                            No exposures.

            RL   Minor cerium-cesium contamination
                            resulted from a routine change of
                            a stack filter in the 300 area.
                            Over 200 employee's shoes were
                            checked and none were found to be
                            contaminated. Walkways and roadways
                            were washed down; no radioactivity
                            was found in surveys beyond the 300
                            area. No property damage; no ex-
                            posures.

5.   <u>Waste and Product Storage (Continued)</u>

<u>Case History No. - Manufacturing Chemists Association</u>

254	Pressure Build-Up in Pfandler Kettle Operation
298	Collapse of 20,000 gallon S.S. Tank
1088	Implosion in Still During Cleaning Operation
1498	Gross Leakage of Plutonium Nitrate Solution from Stored Polyethylene Bottles
1716	Dry Radioactive Waste Unloading Incident - Localized Radioactive Dust Escape

6.   Natural Phenomena

Year	Reporting AEC Field Office	
1959	OR	Lightning damaged transformer. Property damage $6,500. No exposures.
	SR	Lightning damaged two 750 KVA transformers. Property damage $13,750. No exposures.
	LAR	Wind damage to aluminum side wall of building. Property damage $7,500. No exposures.
	SR	Hurricane damaged water dam. Property damage $50,000. No exposures.
1960	SR	During an electrical storm, lightning struck two 200-hp pump motors in an out-of-door pump pit. $6,000 cost for rewinding of burned out motors. No exposures.
	OR	During violent storm, severe power system disturbance caused oil circuit breaker failure. Property damage $18,132. No exposures.
	AL	High-velocity winds caused circuit breaker failure in substation, resulting in fire readily controlled by fire extinguisher. Property damage $8,200. No exposures.
1963	PNR	Severe winds during electrical storm damaged roofs, stacks, ventilation ducts, trees and fences. Property damage $9,400. No exposures.
1965	CH	Four transformers were damaged by lightning. Property damage $35,000. No exposures.
	CH	Lightning caused the destruction of a breaker and the burning of a cubicle. Property damage $8,000. No exposures.
	AL	Repeated lightning strikes damaged transformers. Property damage $5,400.

6. Natural Phenomena

Year	Reporting AEC Field Office	
1966	AL	Roofing destroyed by high winds. Property damage $47,000. No exposures.
1967	AL	A severe wind and hailstorm, with winds in the range of 80 to 100 miles per hour and hailstones the size of oranges, caused extensive roof and other structural damage to numerous buildings, disrupted utilities, demolished a warehouse wall, leveled security fencing and caused severe vehicle damage. Property damage $1,872,000. No exposures.
1970	SR	Water supply lines, drainlines and traps, water-jacketed equipment, heating and cooling coils, instruments, gages, and fire sprinkler lines froze in numerous plant locations during a period of extremely low temperatures, unusual and unexpected in the area. Property damage $38,200. No exposures.

U.S. ENVIRONMENTAL PROTECTION AGENCY

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