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Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors

Appendices

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(a) Associated with McFarland Wrecking Corporation, Seattle, Washington.

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APPENDIX A

DESCRIPTIONS OF THE REFERENCE SITES

Generic sites for the reference research and test (R&T) reactors are described in this appendix. The site descriptions give a reference environment to aid in assessing public safety considerations and potential environmental effects of conceptually decommissioning the reference R&T reactors.

The reference research reactor is assumed to be located on a university campus in an urban environment. The generic site on which the reference research reactor is located is described in Section A.1. The reference test reactor is assumed to be located on a rural site, which is the same as the generic site described in the PWR(1) and BWR(2) decommissioning studies. A summary of this site description is presented in Section A.2.

Individual features of an actual site (for a specific R&T reactor facility) may vary from those of the reference sites. However, it is believed that this approach (i.e., use of reference sites) results in a more meaningful overall analysis of the potential impacts associated with decommissioning most R&T reactor facilities. Site-specific assessments would be required for a similar analysis of a specific R&T reactor.

A.1 RESEARCH REACTOR REFERENCE SITE

The reference site for the research reactor is described in this section. It is based on the safety analysis report (SAR) for the reference research reactor used for this study.(3) The information is representative of a university site in a northwestern location. Expected site radioactivity contamination levels at the time of reactor shutdown are presented in Appendix D.

A.1.1 Site Location and Size

The reference research reactor is located on the campus of a state university. The city in which the university is located is at the base of the foothills of the Pacific Coast Mountain Range, about 90 km from the coast. The site occupies about 15 hectares in a 122-m-square shape.

The Reactor Building is located adjacent to the Radiation Center, which is in the west section of the campus. Land near the reactor facility is used for agricultural purposes except for the Forestry Sciences Laboratory located slightly southeast of the Radiation Center and a water quality laboratory located northwest of the Radiation Center. The nearest residences are located about 300 m north of the Radiation Center and the main shopping center for the city is located 4 km east of the reactor. Long-range plans for the area surrounding the reactor site are for the establishment of a research complex.

A.1.2 Demography

The population of the city is about 45,000 including the university students. Another town of 24,000 population is located 19 km northeast of the reactor. Two larger cities, 50 km from the reactor, have populations of 90,000 and 150,000. About 90 km north of the reactor is a large city of 525,000. The population distribution at various distances from the reactor is given in Table A.1-1.

The population distribution in the immediate vicinity of the reactor is as follows. Within a radius of 100 m of the reactor the only occupants are the staff and students. The peak population is 80 persons. Within a radius

TABLE A.1-1. Population Distribution Around the Reference Research Reactor Site

Distance from Reactor (km)	Total Population in Annulus	Cumulative Population
0 to 2	15 000	15 000
2 to 5	15 000	30 000
5 to 10	25 000	55 000
10 to 20	8 000	63 000
20 to 40	35 000	98 000
40 to 60	300 000	398 000
60 to 80	175 000	573 000
80 to 100	833 000	1 406 000

of 200 m there is a maximum of 300 persons at any one time. Within a radius of 300 m of the reactor the maximum population is expected to be 675. The peak population is expected to be 6,000 people in the area within 700 m of the reactor.

A.1.3 Land Use

The main industry in the city is the state university, with emphasis being placed on research and development. Local industries include sawmill design and manufacture, plywood manufacture, paper pulp products, fiberglass products, concrete products, electronic products (mini-computers, calculators, etc.), and food processing. The surrounding countryside is primarily farm land and federal- and state-owned forests. Agriculture is a significant economic factor in the area. The diversified agriculture includes beef cattle, dairying, seed crops, row crops, flax, berries, nuts, and fruits.

A.1.4 Meteorology

The site has a mild, subcoastal type of weather with wet, open winters and warm dry summers. Electrical storms, hail, and high winds are seldom experienced.

The prevailing wind direction and the monthly mean hourly wind speed are given in Table A.1-2. The maximum wind speed recorded was in excess of 180 km/hr.

TABLE A.1-2. Wind Data for the Reference Research Reactor Site

Month	Prevailing Direction	Mean Hourly Speed (km/hr)
January	S	18.8
February	S	12.1
March	S	15.3
April	S	12.7
May	S	11.1
June	S	10.6
July	S	11.9
August	NW	10.1
September	N	10.1
October	S	8.0
November	S	12.2
December	S	16.4

Small funnel clouds have been obvserved, but no tornados have been recorded in this area. Lightning storms are uncommon, occurring only two to three times a year. Sleet and hail storms rarely occur.

A.2 TEST REACTOR REFERENCE SITE

The reference site described for the test reactor uses some information, namely the meteorological parameters and population distributions, taken from Appendix I of the ALAP study[4] for the river site in the year 2000. Ecological information is derived from the environmental statement for an operating nuclear power plant.[5] The remainder of the information is obtained from a variety of sources. All of the information is representative of potential sites for test reactors in the midwestern or south-mideastern United States.

Individual features of an actual site for a given nuclear facility may vary from those of the reference site. However, it is believed that this generic approach will result in a more meaningful overall analysis of potential impacts associated with most nuclear reactor facilities. Site-specific assessments would be required for individual test reactors.

Naturally occurring radionuclides and radionuclides expected from nuclear weapons testing will be present on any nuclear facility sites consistent with the level generally found in the local region. The magnitude of natural background radiation based on the UNSCEAR report[6] varies with elevation from about 80 mrem per year at sea level to about 170 mrem per year in Colorado. The annual dose from fallout attributed to weapons testing is very small by comparison.[6] Such information on the levels and nature of the radioactive contamination present on the site at the time of decommissioning is necessary, however, in order to make decisions about alternative uses for a site. Expected site radioactivity contamination levels at the time of reactor shutdown are presented in Appendix D.

A.2.1 Site Location and Size

The reference site is located in a rural area with characteristics similar to those found in the midwestern or south-mideastern United States. The

site occupies 4.7 km^2 (4700 hectares) in a rectangular shape of 2 km by 2.35 km. A moderately sized river runs through one corner of the site.

A.2.2 Demography

The site is located in a rural area that has relatively low population density, with highest population densities occurring at distances of 16 to 64 km away, and gradually reducing population densities out to 177 km. The population numbers are given in Table A.2-1. The total population out to 80 km is 3.52 million.

TABLE A.2-1. Population Distribution Around the Reference Site for the Year 2000

Distance From Site Boundary (km)	Population Density (persons/km^2)	Total Population in Annulus[(a,b)]	Cumulative Population
0 to 2	---[(c)]	10	10
2 to 3	136	2 130	2 140
3 to 5	104	5 230	7 370
5 to 6	230	7 940	15 300
6 to 8	133	11 700	27 000
8 to 20	85	89 300	116 000
20 to 30	239	375 000	491 000
30 to 50	175	878 000	1 370 000
50 to 60	298	1 030 000	2 400 000
60 to 80	127	1 120 000	3 520 000

(a) It is assumed that the population in each annulus is uniformly divided in each of 16 uniform 22.5-degree sectors.
(b) Totals are rounded to three significant figures.
(c) Indicates a population density less than 1.0/km2.

A.2.3 Land Use

The use of any part of the total site area of 4.7 km^2 for anything other than for the reactor facility is prohibited. The plant facilities are located

inside a much smaller fenced-in portion (0.12 km^2) of the site, with a minimum distance of 1 km from the point of plant airborne releases to the outer site boundary that is fenced and marked.

In most of the surrounding area, about 80% of the land is used for farming. The main crops in this area, which include all land within 16 km of the site, are soybeans (60%), corn, oats, and other grains (30%), and hay (10%). It is expected that this area will remain largely agricultural and that the population will not change significantly.

A wildlife refuge is located about 14 to 20 km from the site. A state park is located about 10 km in the opposite direction, and a state forest and campground are about 14 km away, near the wildlife refuge.

There are large truck gardens in the area. The nearest dwelling is a farm house located 1.3 km from the site. A milk cow is kept at this farm and is maintained on fresh pasture 6 months of the year. A family garden, with a growing season of 5 months, is kept for vegetables. River water is used for irrigation of this farm.

A.2.4 Water Use

Complete access to natural waterways (except to onsite shorelines) by the public is assumed. The river is used for dispersal of acceptable liquid effluents from the test reactor facility. The river is used for irrigation, fishing, boating, and other aquatic recreational activities. Its shores are used for hunting and for marinas and docks. It is assumed that one-sixth of the average person's diet comes from food crops irrigated by the river water downstream of the site, and one-third of the maximum-exposed individual's food intake is from the same source.

In the vicinity of the plant, it is assumed that a limited number of individuals may spend 50 hr/yr swimming, 50 hr/yr boating, and may spend 70 hr/yr fishing on the shoreline obtaining 20 kg of fish. Essentially no fresh water molluscs or crustacea are taken from the river. Aquatic foods are assumed to be consumed within 24 hours of the time they are caught.

Drinking water is taken from the river by larger towns and cities. Individuals living immediately downstream near the river derive their potable water

from wells and not directly from the river. For development of population radiation dose information, it is assumed that 20% of the residents within 80 km of the site obtain their drinking water from the reference river. Travel time from the site to the location of the urban water consumer is guaged at 48 hours. It is assumed that each person not in the vicinity of the plant will spend on the average 2 hr/yr swimming, 4 hr/yr boating, and 4 hr/yr on the shoreline downstream of the site.[7] The average per-capita fish consumption for this area is estimated to be 6.9 kg/yr.[8] It is assumed that 15% of this consumption is from fish obtained downstream from the site.

The region has no large commercial fishing operations. Most of the fish catch is caught by local recreational anglers for home consumption.

A.2.5 Meteorology

The site has a typical continental climate. It is characterized by wide variations in temperature, modest winter precipitation, normally ample spring and summer rainfall, and a general tendency to extremes in all climatic features. January is the coldest month and July is the warmest. Table A.2-2 gives monthly temperature statistics.

The number of days with maximum temperatures of 32°C and above is estimated to be 12. The number of days with a minimum temperature of 0°C or below and -18°C or below is estimated to be 168 and 40, respectively. The January relative humidities at 7:00 a.m., 1:00 p.m., and 7:00 p.m. are estimated to be 76, 68, and 70%, respectively. The corresponding humidities for July are 86, 55, and 55%. Monthly average humidities are shown in Table A.2-3.

The months of May through September have the greatest amounts of precipitation, with an average rainfall during this period of 432 to 457 mm, 70% of the 610-mm annual rainfall. The maximum 24-hr total rainfall for the period 1894 through 1965 was 127 mm and occurred in May. Thunderstorms have an annual frequency of 36 and are the chief source of rain from May through September. Snowfall in the area averages 1070 mm annually, with occurrences recorded in all months except June, July, and August. The extremes in annual snowfall of record are 152 mm minimum and 2235 mm maximum.

TABLE A.2-2. Monthly Air Temperature Statistics (°C) at the Reference Site

	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sep.	Oct.	Nov.	Dec.
Maximum	-6.1	-4.4	3.3	12.8	20	25	28.3	26.7	22.2	15	4.4	-3.3
Minimum	-16.1	-14.4	-6.7	1.7	7.8	13.3	16.1	15	10	3.9	-4.4	-12.2
Mean	-11.1	-9.4	-1.7	7.2	13.9	18.9	22.2	21.1	16.1	9.4	0.0	-7.8
Extreme Maximum	15	16.1	27.8	32.8	40.6	39.4	41.7	40	40.6	32.2	23.9	17.2
Extreme Minimum	-38.9	-36.7	-34.4	-15.6	-6.7	0.6	5.6	3.3	-5.6	-13.3	-27.8	-33.9

TABLE A.2-3. Mean Monthly Relative Humidity (%) at the Reference Site

Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sep.	Oct.	Nov.	Dec.
74	75	73	66	62	66	68	70	70	66	73	78

Annually, the winds are predominantly bimodal. This bimodal distribution is characteristic of the seasonal wind distributions as well. The average wind speed for spring is 11 km/hr and for the other seasons is about 16 km/hr. The maximum reported wind speed of 148 km/hr was associated with a tornado.

Tornadoes and other severe storms occur occasionally. Eight tornadoes were reported for 1916 through 1967 in the county where the site is located. The probability of a tornado striking a given point in this area is about 5×10^{-4} per year. For design purposes, a wind velocity of 480 km/hr is assumed to be associated with tornadoes.

It is estimated that natural fog restricting visibility to 0.4 km or less occurs about 30 hr/yr. Icing due to freezing rain can occur between October and April, with an average of one to two storms per year. The mean duration of icing on utility lines is 36 hours.

Diffusion climatology comparisons with other locations indicate that the site is typical of the region, with relatively favorable atmospheric dilution conditions prevailing. Frequency of thermal inversion is expected to be about 32% of the year, and the frequency of thermal stabilities is 19% slightly stable, 27% stable, 20% neutral, and 34% unstable.

Data from a number of river sites used for nuclear reactors are used to calculate the "typical" annual atmospheric dispersion pattern in an average 22-1/2-degree sector from the site. Dispersion factors at selected distances for the average sector are determined from joint-frequency distributions for each site. This is done by calculating the dispersion factor, $\overline{X}/Q'$, for each sector at selected downwind distances and then calculating the average dispersion factor at each distance. In other words, the dispersion factors in those sectors corresponding to overland trajectories are added without regard to direction and divided by the number of sectors involved. For river sites, all 16 sectors are used. Thus, an average dispersion factor is obtained for each selected downwind distance for all 16 sectors.(4)

Standard groups of meteorological data are interpolated from the specific site data. The groupings provide four stability classes based on vertical temperature gradient and five wind speed classes based on the Beaufort wind scale.

Vertical temperature gradient classes are based on Regulatory Guide 1.23 (Safety Guide 1.23)[9]; Pasquill classes A, B, and C are classified as B (unstable), Pasquill class D (neutral) and E (slightly stable) are not reclassified, and Pasquill classes F and G are classified as F (moderately stable).

Wind speed data are frequently available at only one height. In this event, the measured values are extrapolated to the 10-m level for building and vent release calculations and to the 100-m level for stack release calculations. Where measurements at two heights are available, the highest is extrapolated to 100 m and the lowest to 10 m, using a standard power law extrapolation procedure.[10]

Calculations are made for a number of specific sites in order to yield average air concentrations and average radiation doses versus distance from the release point (i.e., the values represent the average over all of the wind directions that carry the airborne effluents over land). As previously stated, average sector values are used for this study.

Data on the maximum sector and the average of all 16 sectors are available for 16 sites.[4] These data are used to determine the ratio of the minimum effective dilution (maximum dispersion factor) to the mean sector value, and are applied to the present study.

The value obtained for the ratio of dispersion factor in the maximum sector to the average sector dispersion factor is 2.5, and this value is used for all release heights in this study. Investigation of changes in this ratio with distance from the site determines that it remains essentially constant with distance from the point of release. The dispersion factors for the average sector as a function of height of release and downwind distance are shown in Figure A.2-1.

To assess the effect of increased stack height, atmospheric dispersion factors for stack heights of 150, 200, and 300 m are estimated from the original joint frequency distributions of the 16 reactor sites. These values are also presented in Figure A.2-1.

Where large volumes of heated air are being ejected, the plume rise constant for momentum is estimated to be about 50 m^2/sec. Assuming an annual

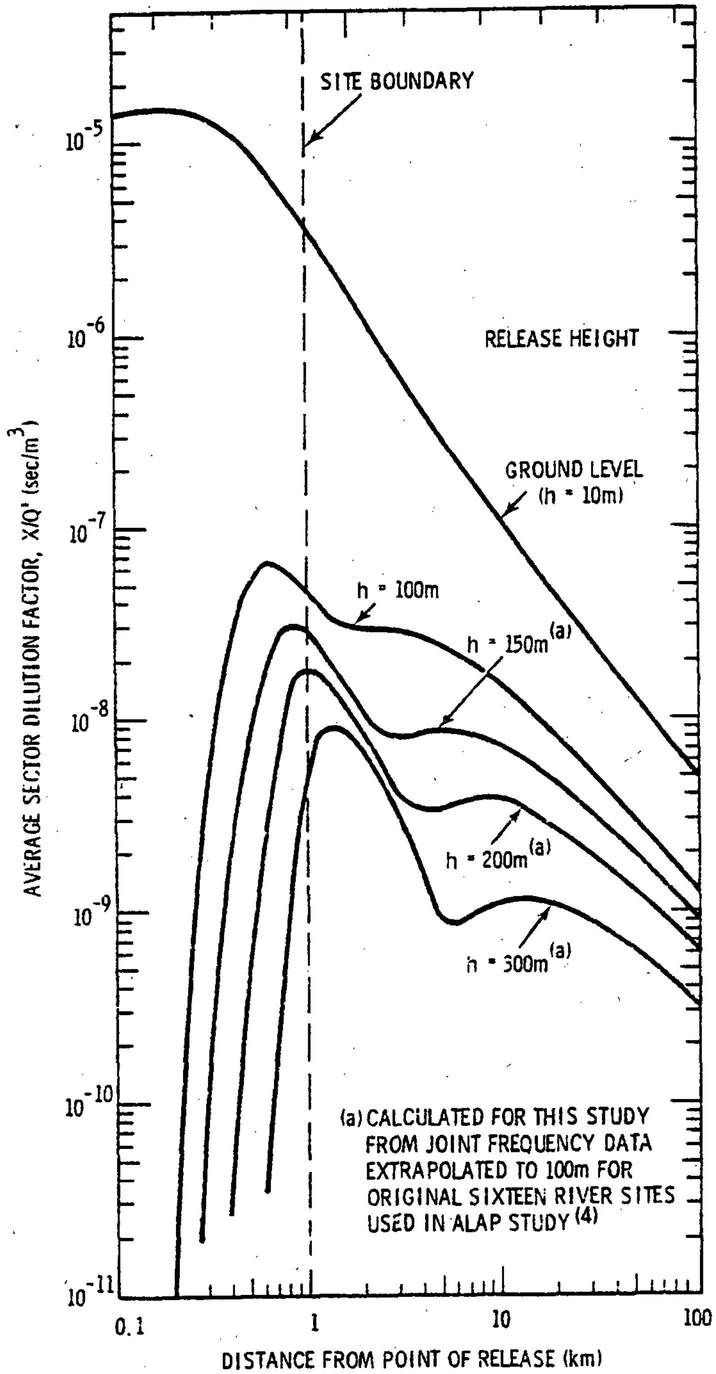

FIGURE A.2-1. Average ($\overline{X}/Q'$) Values Versus Distance in a Sector from the Reference River Site

average wind speed of 2 to 3 m/sec, the increase in effective stack height due to momentum would be 20 to 25 m. Plume rise due to buoyancy (heat effect) would add at least another 25 to 100 m of effective stack height, depending upon the temperature of the exhaust gases. Thus, the $\overline{X}/Q'$ values illustrated in Figure A.2-1 are larger than they would be if credit had been taken for buoyancy and momentum.

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APPENDIX B

REFERENCE RESEARCH REACTOR FACILITY DESCRIPTION

The major features of the reference research reactor are presented in Section 8 of Volume 1. This appendix describes in more detail the portions of the facility that are important to decommissioning. The detailed site description for the reference research reactor is presented in Appendix A.

The Oregon State University TRIGA[a] reactor (OSTR), at Corvallis, Oregon, is the reference research reactor for this study. OSTR is a 1000-kWt, above-ground, open-pool nuclear training and research facility that utilizes a TRIGA-type core and control system. The structures, systems, and components are typical of TRIGA research reactor facilities.

Much of the detailed facility information contained in this appendix is based on the OSTR Safety Analysis Report,[1] on the Hazards Analysis,[2] on the Facility License,[3] on the OSTR Training Manual,[4] on the 1980 Annual Report,[5] and on construction drawings, photographs, and other data supplied by personnel at the OSTR.

B.1 GENERAL FACILITY DESCRIPTION

The reference research reactor site is located on the western edge of the main campus and occupies a surface area of approximately 120 m^2. A security fence surrounds the Reactor Building and its supporting facilities. The reference research reactor site layout is shown in Figure B.1-1, including identification of major structures/areas anticipated to require decontamination activities.

Major structures on the site include: the Radiation Center Building and the Reactor Building, which houses the TRIGA reactor and the support area including the control room and the Cooling Tower; the Annex, the Heat Exchanger, and the Pump House. Collectively, these form a Radiation Center Complex.

(a) TRIGA trademark registered in U.S. Patent Office.

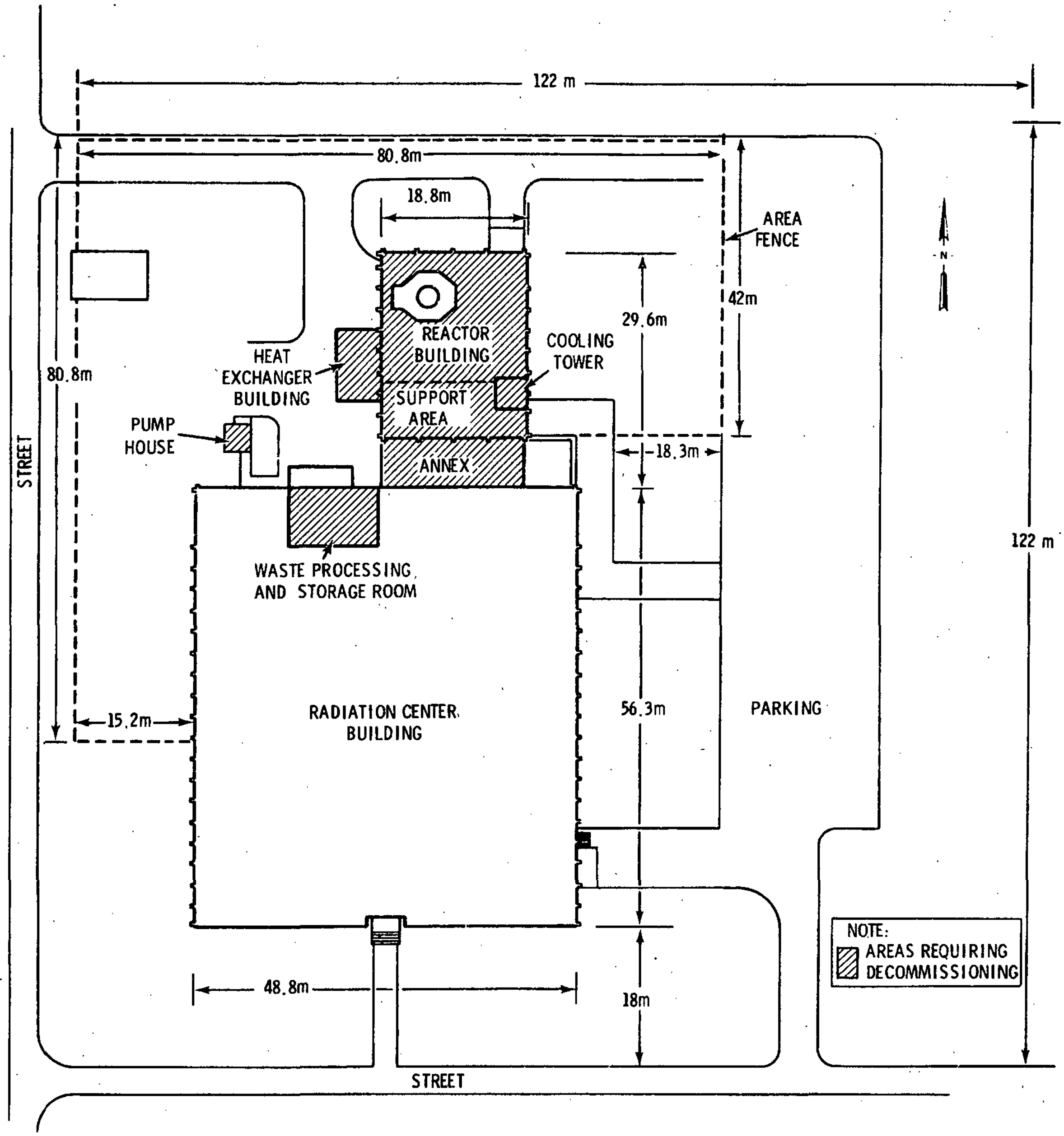


FIGURE B.1-1. Site Layout of the Reference Research Reactor Facilities

B.1.1 Design Criteria

The bases for the design criteria used in this research reactor facility is to first, provide protection for public safety; second, provide for reliable and economic plant performance; and third, provide an attractive appearance.

The essential systems and components of the plant are designed to enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The designs are based upon the most severe of the natural phenomena recorded for the vicinity of the reference site, with margin to account for uncertainties in the historical data.

B.1.2 Research Reactor Operating Characteristics

The maximum steady-state power level for the reference research reactor core is approximately 1000 kWt. A functional diagram of the reference research reactor is shown in Figure B.1-2. The research reactor tank assembly is illustrated in Figure B.1-3.

The reference TRIGA reactor is light-water-cooled and moderated, with solid fuel elements in which zirconium hydride moderator is homogeneously mixed with 20%-enriched uranium hydride. The reactor core consists of a lattice of fuel-moderator elements, graphite dummy elements, and control rods. The fuel elements have 89-mm-long graphite end sections that form the top and bottom reflectors. Core cooling is provided by natural convection of the pool water, which occupies about one-third of the core volume.

Neutron reflection is provided in the radial direction by 260 mm of graphite contained in a leak-tight, welded aluminum container. The height of the graphite in the reflector is about 560 mm. Also in the aluminum container, at the outer perimeter, is 50 mm of lead, which acts as a radiation shield to protect the concrete structure from excessive heating while providing dose reduction outside the concrete shield. The reactor is designed for continuous steady-state, square wave, or pulsed operation. Pulsed bursts yield a prompt energy release of about 36 MW and a peak power of about 3800 MW.

The safety of the TRIGA reactor lies in the large negative temperature coefficient that is an inherent characteristic of the fuel element composition. This results in a prompt shutdown mechanism following any power excursions and has, therefore, permitted the location of TRIGA reactors in urban areas and in structures without the pressure-type containment required for some other types of research reactors.

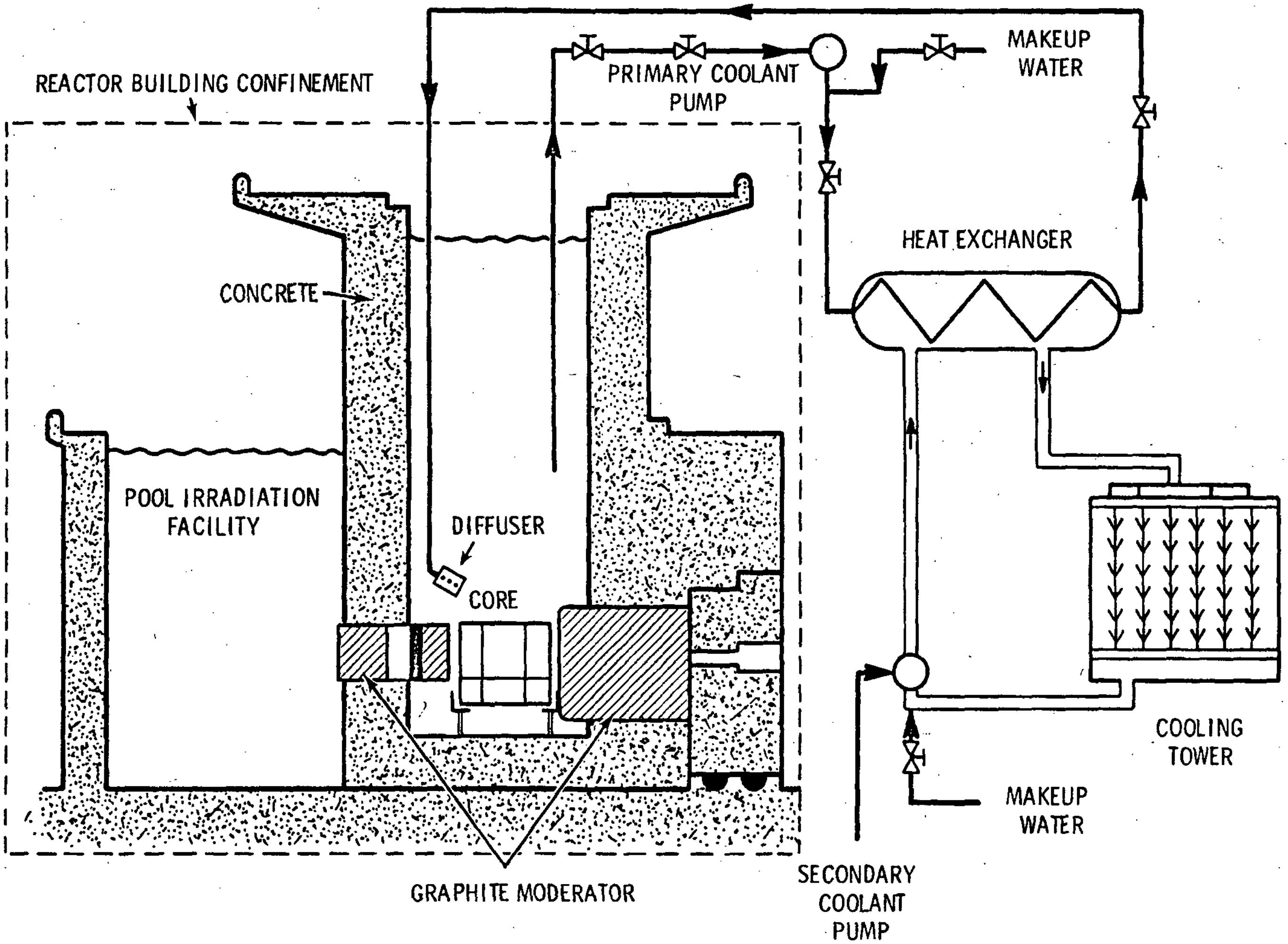
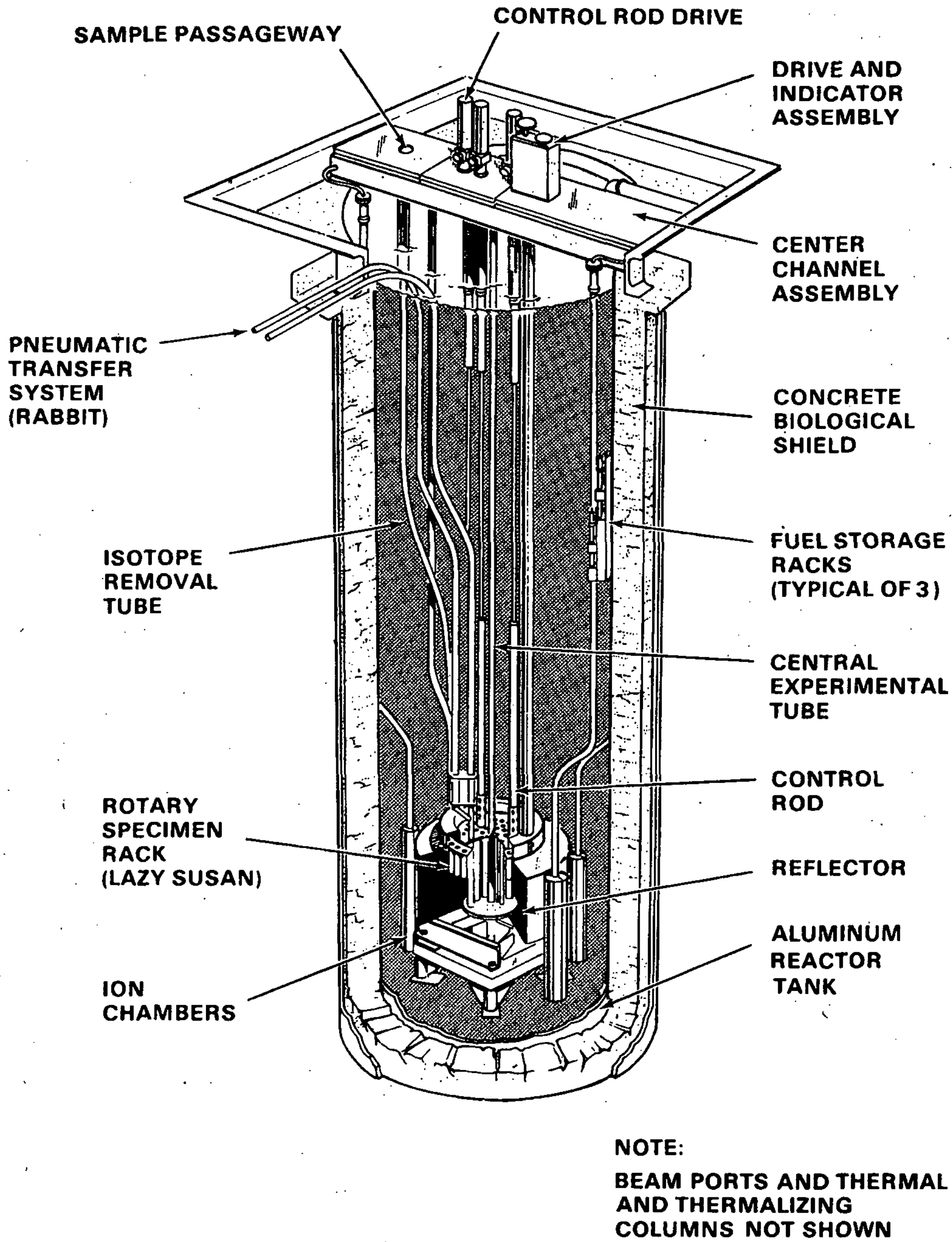


FIGURE B.1-2. Functional Diagram of the Reference Research Reactor



<u>FIGURE B.1-3.</u> Cutaway View, Reference Research Reactor Tank Assembly

During steady-state operation, the reactor is controlled by three motor-driven control rods (one shim rod, one safety rod, and one regulating rod). For pulsing operation, the reference reactor is equipped with a pneumatic drive system which quickly withdraws a poison rod from the reactor core.

Four instrumentation channels monitor and indicate the reactor neutron flux and power level. The bulk water temperature and the reactor tank outlet and inlet water temperatures are indicated on the console. The water conductivity, measured at the inlet and outlet of a demineralizer, is displayed on a panel near the console. A Geiger counter with a count rate meter is used to monitor gamma-emitting radionuclides in the reactor water. Safe visibility of the reactor core is provided by the water moderator, which is maintained at a depth of 6.1 m from the bottom of the aluminum tank.

Experimental Facilities

Operation of the reference research reactor provides a neutron flux to a variety of experimental locations within and around the core. Depending on the mode of operation, both long-term, low-intensity, and very short-term, high-intensity exposures to test specimens can be provided.

Neutron irradiation facilities within the reactor are:

- Pool Irradiation Facility
- Graphite Thermal Column
- Rotary Specimen Rack (Lazy Susan)
- Pneumatic Transfer System (Rabbit)
- Central Irradiation Thimble
- Beam Ports and Shielding.

A number of other experimental facilities exist to coordinate with the irradiation facilities and the research purpose of the reactor. These are:

- short-term hot sample storage
- sample handling and counting facility
- subcritical pile assembly.

B.1.3 Electric Power

Electric power is supplied to the main service center in the Reactor Building from the Radiation Center Building. Emergency electric power is

provided by a propane generator and a storage battery bank. The batteries, inverter and propane generator are located in the Heat Exchanger Building.

B.1.4 Fuel Handling and Storage

Fuel rearrangement takes place under water within the reactor vessel using special long-handled tools. A shielded fuel element transfer cask is used as needed for the safe transfer of irradiated fuel elements or other radioactive material from the reactor tank to the storage pits or to other high-level radioactive storage located in the Annex Building. Three fuel storage racks (see Figure B.1-3), each capable of holding 30 fuel elements, are located under water along the walls of the reactor tank to provide temporary storage of fuel-moderator elements or dummy elements. In addition, five storage pits, located in the reactor room floor, are provided for temporary storage of fuel elements and other highly radioactive materials.

B.1.5 Radioactive Waste Treatment Systems

A small-scale liquid waste treatment system is provided for filtering and demineralizing potentially contaminated water. The concentrated wastes are solidified and either temporarily stored onsite or shipped to a permanent radioactive waste repository. Wastes from the hot sinks and floor drains are sent to the retention tank in the Pump House Building. If found to be releasable (within state and federal concentration limits for radioactivity) after sampling, they are discharged to sanitary sewage. Radioactive wastes from the retention tanks could also be decontaminated further by recirculation through filters, resins, or other media.

B.2 STRUCTURES

This section describes the building and significant structures located at the reference research reactor facility. The principal structure is the Reactor Building. It contains the reactor structure, the fuel storage pits, and the support areas. Other structures include the Cooling Tower, the Annex, the Heat Exchanger Building, the Pump House, and the Radiation Center Building. Refer to Figure B.1-1 for the physical arrangements of the main structures on the reference site. A compilation of the primary construction materials used in these structures is given in Table B.2-1.

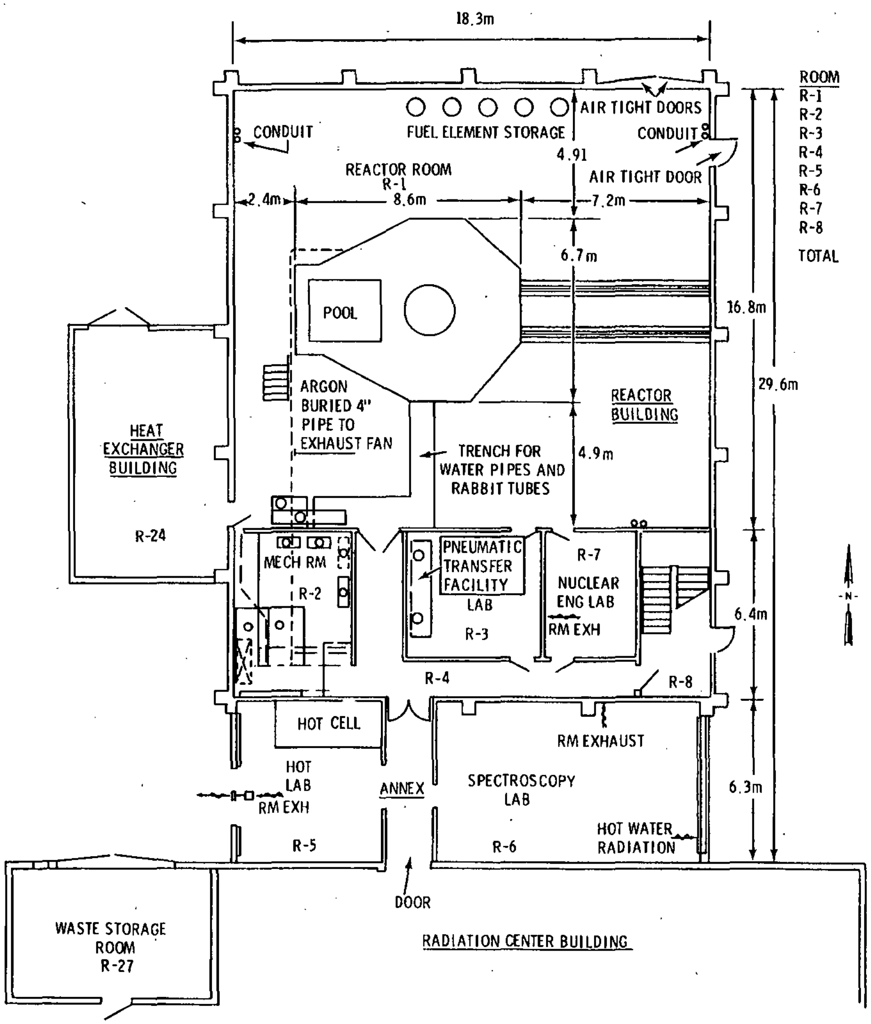
TABLE B.2-1. Estimated Quantities of Structural Material in the Reference Research Reactor Facilities

Structure	Concrete (m^3)	Rebar (mg)	Structural Steel (Mg)
Reactor Building	509	41.9	16.8
Reactor Structure	235	38.6	2.7
Radiation Center Annex	59	4.8	4.0
HX Addition	26	2.1	1.5
Pump House	8	0.6	0.2
Totals	837	88.0	25.2

B.2.1 Reactor Building

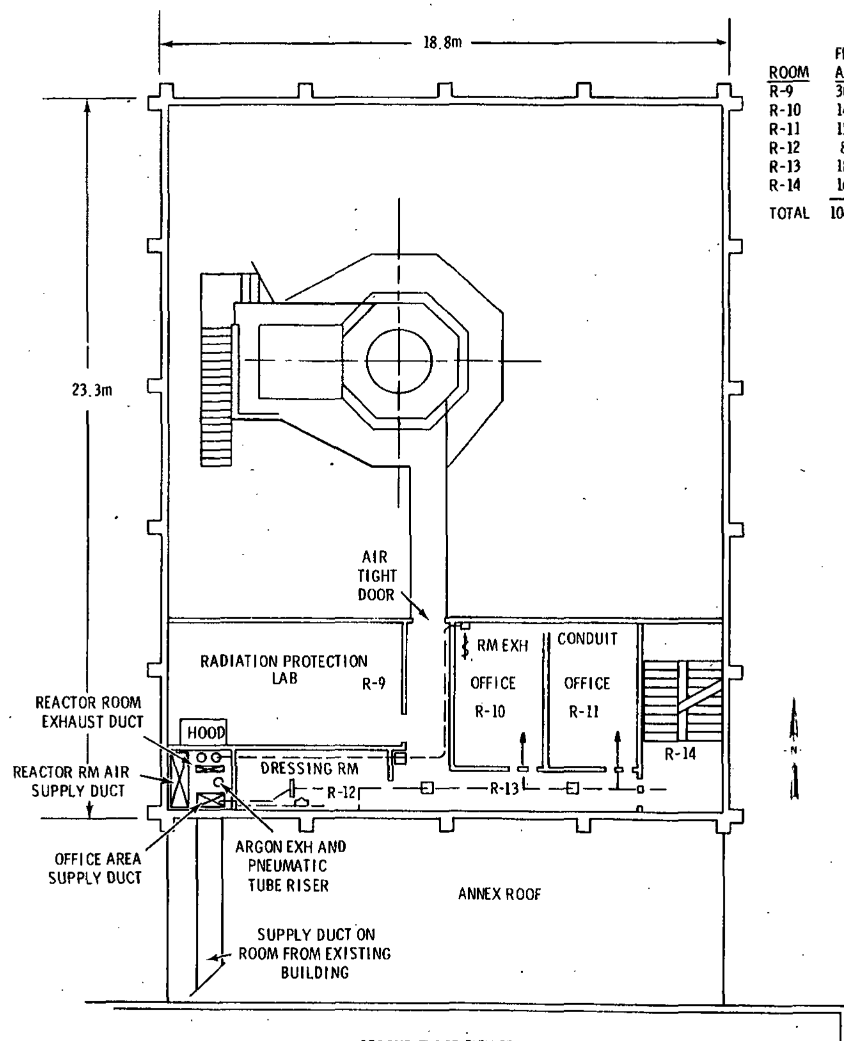
The Reactor Building is a Zone 3 seismic-intensity structure and is illustrated in Figures B.2-1a, B.2-1b, and B.2-2. Rooms and areas in the reference structures are assigned identification numbers, as shown in the figures. A compilation of these numbers is given in Table B.2-2.

The Reactor Building is a concrete structure, rectangular in plan and elevation, 18.3 m wide by 23.2 m long by 13 m high. The building substructure consists of spread footings founded on an 0.9-m depth of rock. The reactor room floor is a 160-mm-thick concrete slab placed on a 160-mm-thick compacted, granular base. The building superstructure consists of precast-prestressed exterior wall panels and poured-in-place pilasters, a structural steel roof frame with metal deck and insulating concrete fill, and a structural steel interior floor frame with metal formed concrete slabs. Structural floors are designed for a superimposed load of 600 kg/m^2 and roofs for 120 kg/m^2. The concrete walls are 160 mm thick. There are 22 reinforced concrete columns which support the tilt-up walls. Each of these columns is 250 mm square. Concrete spandrels, approximately 260 mm square, tie the walls and columns together around the top perimeter of the building. A safety handrail is set into the spandrels around the roof. The roof is constructed of lightweight (1000 kg/m^3) concrete with 24-gauge steel decking. A build-up roof is applied to this base. A 100-mm drainage slope exists from the roof ridge to the walls.



ROOM	FLOOR AREA, m^2
R-1	306.7
R-2	29.1
R-3	24.9
R-4	35.3
R-5	33.7
R-6	60.9
R-7	16.5
R-8	16.1
TOTAL	538m^2

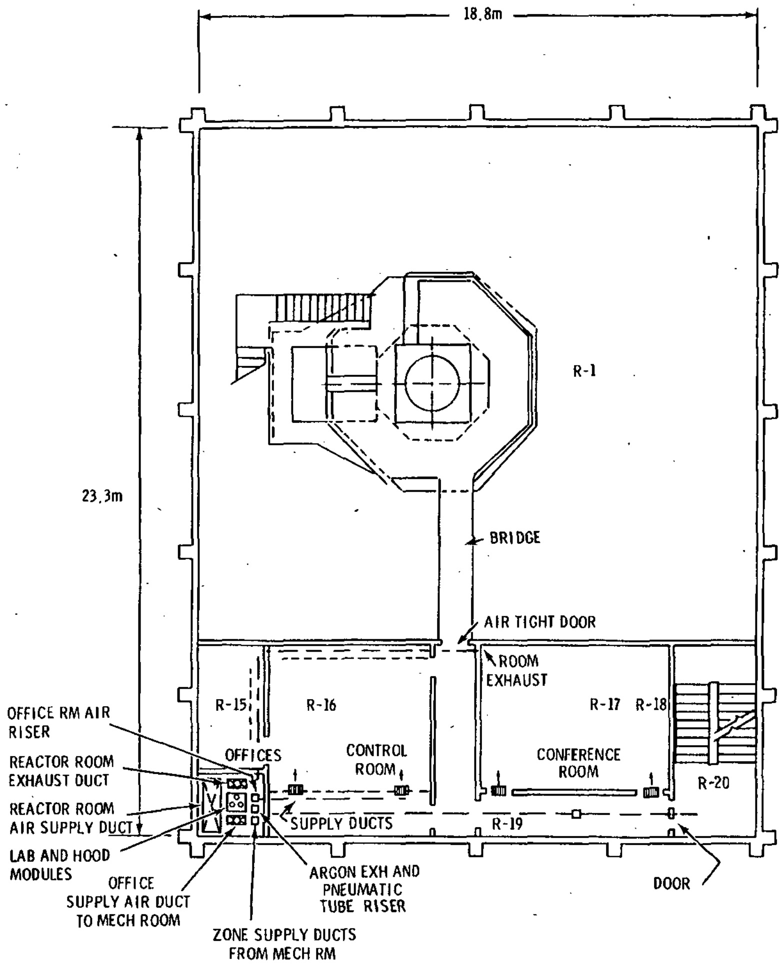
FIRST FLOOR ELEV 74.4m



ROOM	FLOOR AREA, m^2
R-9	30.7
R-10	14.9
R-11	15.2
R-12	8.6
R-13	18.6
R-14	16.1
TOTAL	104.1

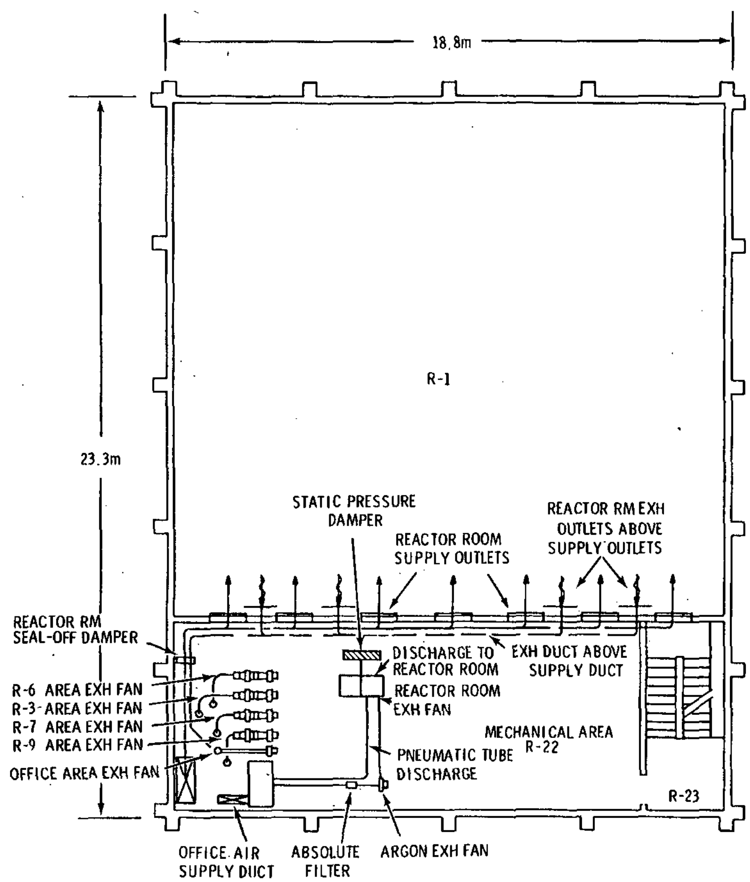
SECOND FLOOR ELEV 77.6m

FIGURE B.2-1a. Reactor Building and Annex Floor Plans



ROOM	FLOOR AREA, m^2
R-15	9.0
R-16	31.6
R-17	14.1
R-18	16.6
R-19	17.9
R-20	16.1
R-21	4.4
TOTAL	117.1 m^2

THIRD FLOOR ELEV. 81.0 m



ROOM	FLOOR AREA, m^2
R-22	100.9
R-25	16.2
TOTAL	117.1 m^2

FOURTH FLOOR ELEV. 84.2m

FIGURE B.2-1b. Reactor Building and Annex Floor Plans

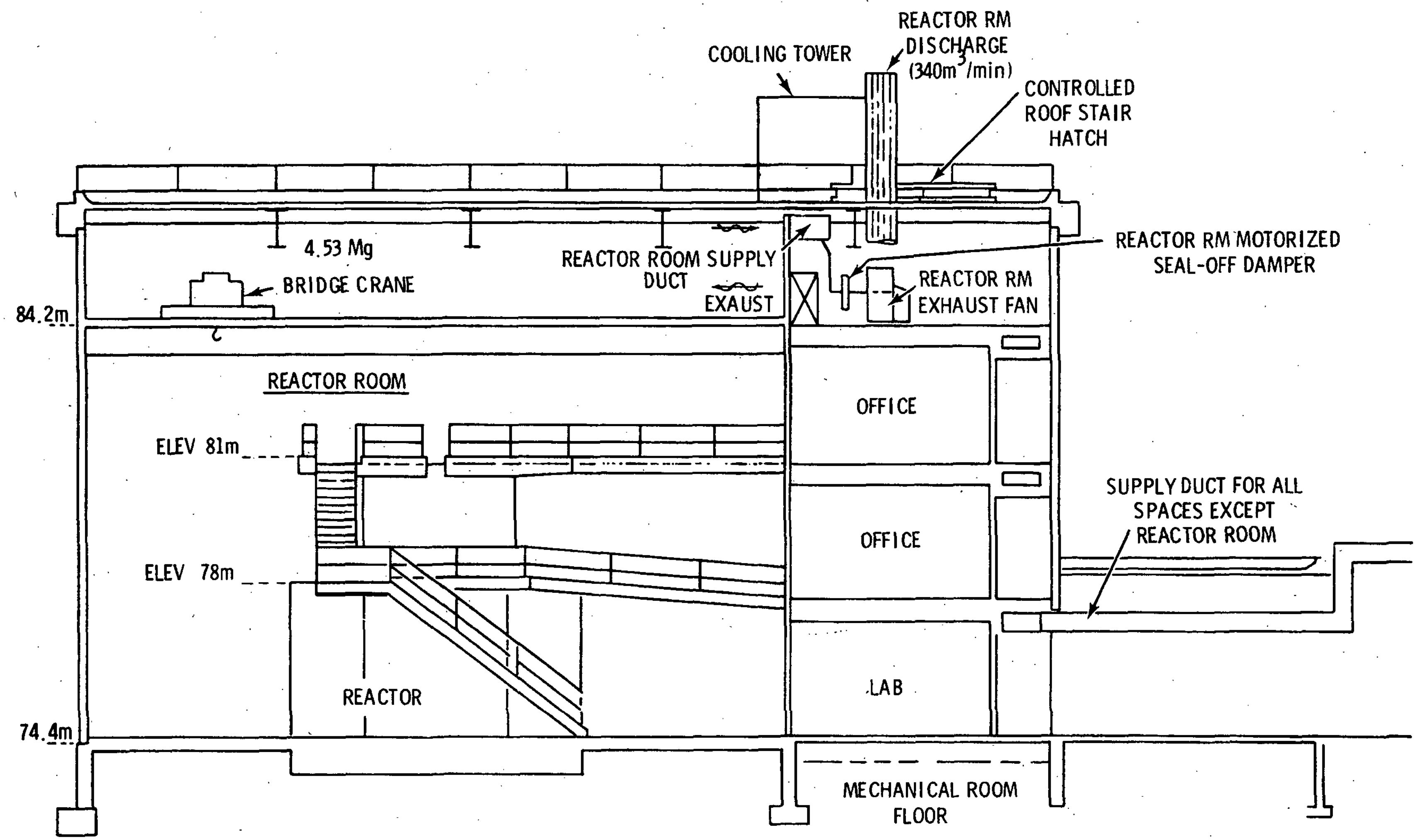


FIGURE B.2-2. Reactor Building--Section View Looking West

TABLE B.2-2. Identification Numbers Assigned to Rooms and Areas Within the Reference Research Reactor

Identification Number	Location
R-1	Reactor Room
R-2	First Floor Mechanical Room
R-3	First Floor "Rabbit" Room
R-4	First Floor Corridor
R-5	First Floor Hot Laboratory and Hot Cell
R-7	First Floor Nuclear Engineering Laboratory
R-8	First Floor Emergency Exit Area
R-9	Second Floor Biology Laboratory
R-12	Second Floor Dressing Room
R-15	Third Floor Control Room Office
R-16	Third Floor Reactor Control Room
R-19	Third Floor Corridor
R-20	Third Floor Stair Stop
R-21	Third Floor Air Supply Area
R-22	Fourth Floor Mechanical Area
R-23	Fourth Floor Stair Stop
R-24	Heat Exchanger Building
R-25	Pump House and Retention Tank
R-26	Cooling Tower
R-27	Waste Storage Room

A 4.5-Mg bridge crane is supported at the 84-m level in the reactor room. It has access to the entire reactor room below the fourth floor. The crane is supported on 0.6-m steel beams along the east and west walls of the reactor room. The bridge is 2 m wide and has a catwalk the entire length of the bridge. It may be operated from atop the structure or by means of a remote control switching system.

The Reactor Building serves as a confinement structure for the reactor. It houses the reactor room (R-1), which contains the reactor structure, fuel storage pits, and a large support area.

B.2.1.1 Reactor Structure

The reference research reactor is built on an independent 0.75-m-thick concrete foundation pad. There are two walkways leading to the upper two levels of the structure. A cantilevered platform surrounded by metal railings is provided for personnel and equipment at the top of the reactor shield. A metal stairway and railing extends from the floor to the cantilevered platform at the top of the shield structure.

The research reactor vessel is surrounded by a concrete biological shield as shown in Figures B.2-3 and B.2-4. The biological shield is a reinforced concrete structure standing 7.8 m above the reactor room floor. Vertically the structure contains a step, 3.7 m above the reactor room floor. The lower octagonal portion is 7 m across.

B.2.1.2 Support Area

A portion of the Reactor Building is allocated to offices, mechanical areas, laboratories, and the Control Room (see Figure B.2-1a,b).

First Floor. The first floor of the Reactor Building is the lowest level of the structure (see Figure B.2-1a). It houses: 1) the "rabbit lab," R-3, including the pneumatic transfer system receiver-sender station; 2) the nuclear engineering lab, R-7, which is used as a neutron radiography preparation room; 3) the mechanical equipment room, R-2, which contains the reactor bay ventilation fan, pneumatic transfer system blower, and the reactor building pressure regulating systems; and 4) the electron spectroscopy lab, R-6, which houses the chemistry department's electron spectrometer.

Second Floor. The second floor of the Reactor Building (see Figure B.2-1a) contains: 1) two offices, a shower, and a dressing room (R-10 and R-11); 2) a rest room (R-12); and 3) a radiation protection lab (R-9). A walkway is provided from the second floor hallway to the second level of the reactor structure.

Third Floor. The third floor of the Reactor Building (see Figure B.2-1b) contains: 1) a conference room (R-17 and R-18) with windows that overlook the

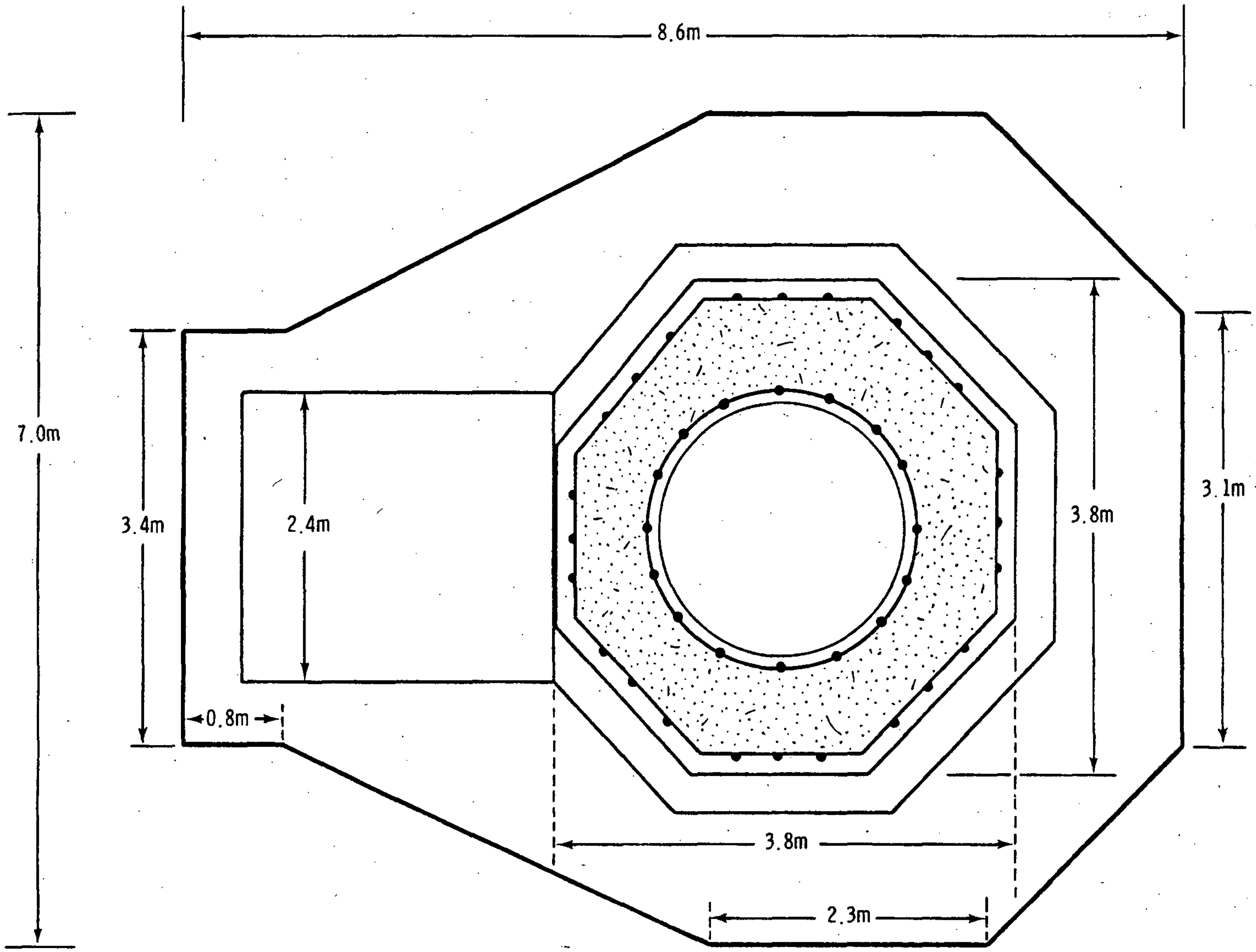


FIGURE B.2-3. Top View of TRIGA Reactor

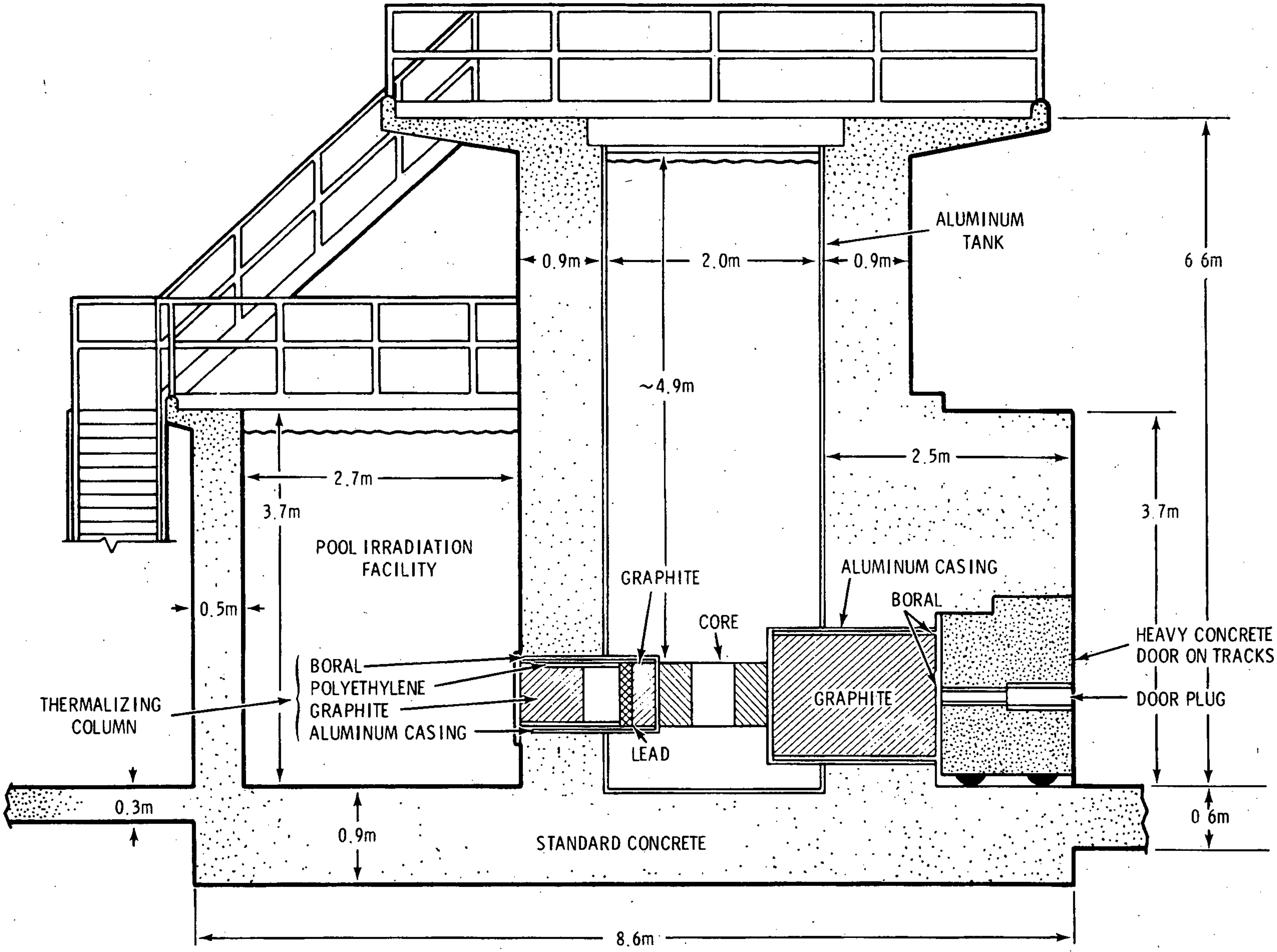


FIGURE B.2-4. Vertical Section View of the Reference TRIGA Reactor Looking North

reactor bay; 2) the reactor control room (R-16); and 3) the reactor supervisor's office (R-15). A walkway is provided from the third floor hallway to the top of the reactor structure.

Fourth Floor. The fourth floor of the Reactor Building (see Figure B.2-1b) contains: 1) an office (R-23); and 2) a mechanical equipment room (R-22), which houses the fume hood fans, main reactor room exhaust fan, Argon system fan, hot drain exhaust fan, and the stack monitor equipment.

Roof. The Reactor Building has a built-up, hot mop, gravel roof (see Figure B.2-2). The reactor secondary system cooling tower is located on the southeast end of the roof adjacent to the 6.7-m-high reactor room exhaust stack.

B.2.2 Cooling Tower

A cooling tower structure (see Figure B.2-2) is located on the roof at elevation 81.3 m. It is used to cool the secondary water from the reactor coolant heat exchanger. The tower is approximately 2.7 m wide by 3.7 m long by 3 m high and rests on two I-beams placed on the roof. The tower is made of galvanized steel, weighs 3.8 Mg, and has a filled working weight of 5.8 Mg.

The tower utilizes two 16-hp (220-V) 3-phase motors, each turning squirrel-cage fans. Two 3.6-kW heaters are incorporated into the system to prevent the water from freezing during cold weather.

B.2.3 Annex

A single-story building, 18 m by 6.3 m by 4.2 m, connects the Reactor Building and the Radiation Center Building (see Figure B.2-1a). An interconnecting corridor and two laboratories (R-5 and R-6) are located in the annex. One laboratory, R-5, is utilized as a radioactive hot lab with a built-in Hot Cell. The cell structure is a concrete cube approximately 2-1/2 m in all dimensions and has 0.5-m-thick walls.

B.2.4 Heat Exchanger Building

The single-story Heat Exchanger Building is attached to the Reactor Building and houses a heat exchanger system and equipment that are used in the reactor water cooling systems (see Figure B.1-1).

Reactor water is cooled by passage through a heat exchanger located in the Heat Exchanger Building and then returned to the reactor vessel. The heat exchanger is connected to a secondary cooling system for heat dissipation. A line diagram of this system is shown in Figure B.2-5.

The secondary cooling system utilizes a base mounted centrifugal pump which delivers 2.60 m^3/min of water from the cooling tower through the heat exchanger at an operating head of 27.5 m. The pump is directly coupled to a 25-hp (208-V) motor, rotating at 1765 rpm. The secondary system piping is black iron, schedule 40, with arc-welded joints. Piping connecting the tower and pump inlet is 0.20 m in diameter and the remaining piping is 0.15 m in diameter.

B.2.5 Pump House

A sheet-metal Pump House, approximately 2.6 m square and 3 m high, is located about 3 m from the northwest corner of the Radiation Center Building (see Figure B.1-1). The Pump House (with associated piping and valves) is above an underground liquid retention tank whose capacity is 11.1 m^3. Liquid wastes from contaminated areas of the Reactor Building, the Annex, and the Radiation Center Building are collected in this retention tank.

B.2.6 Radiation Center Building

A large, single-story concrete building (see Figure B.1-1), the Radiation Center Building is located south of and connected to the Annex. It incorporates a large attic loft area which contains a HVAC system that is shared with the Annex and support area of the Reactor Building. In addition, this building houses a 7-m by 10-m by 3-m-high rectangular waste processing room.

The Waste Processing and Storage Room, an area within the Radiation Center Building (R-27) and adjacent to the Reactor Building, is used for solidification of liquid waste, compaction of solid dry waste and solid-waste storage (see Figure B.1-1). It is a single-story structure with a concrete floor and walls.

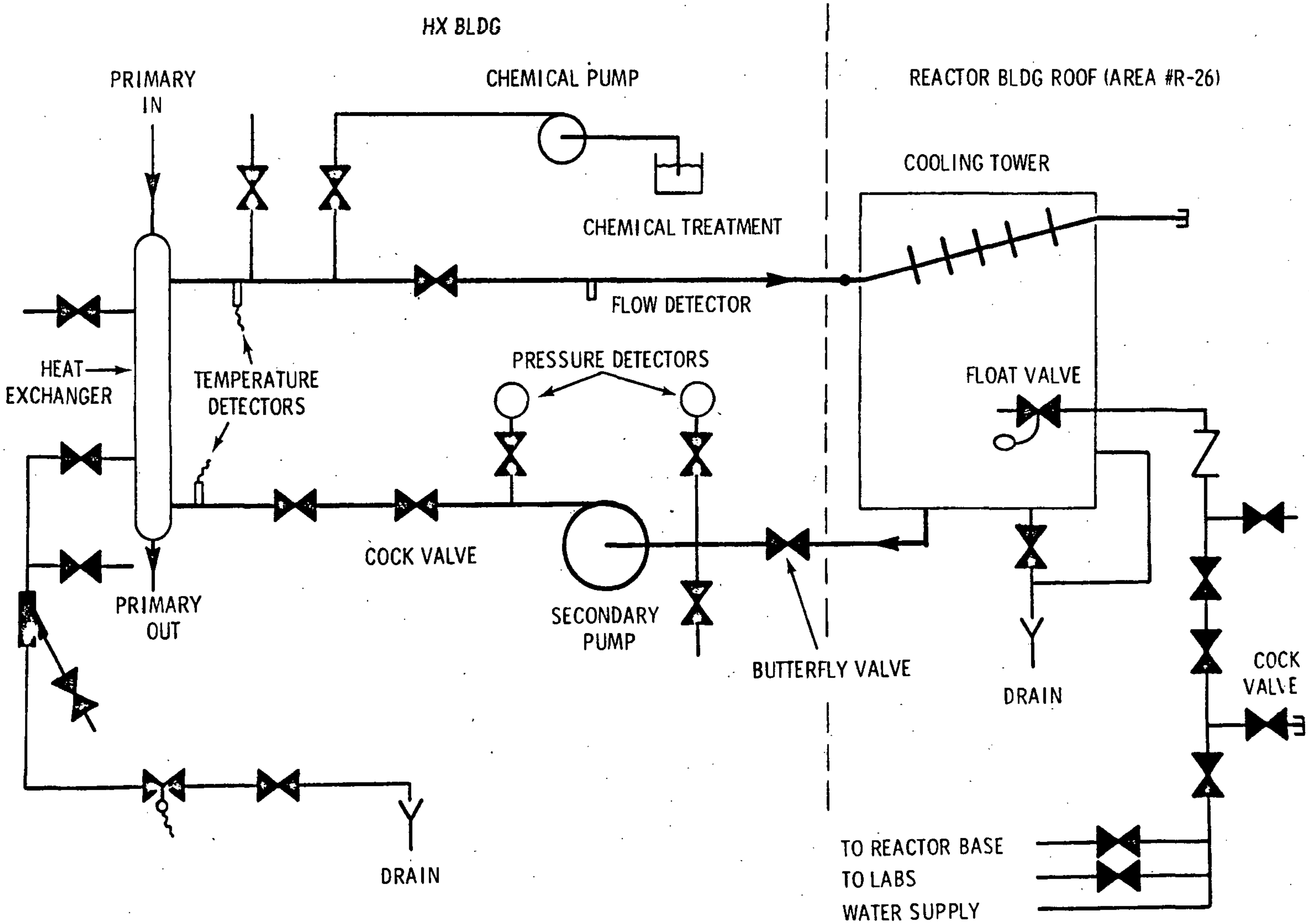


FIGURE B.2-5. Secondary Water Cooling System

B.3 CONTAMINATED EQUIPMENT

Descriptions of the contaminated equipment located in the Reactor Buiding, the Heat Exchanger Building, the Pump House, and the Radiaton Center Building are presented in this section.

B.3.1 Reactor Building Equipment

The Reactor Building houses the reference research reactor and most of the support equipment for operation and maintenance of the reactor. The Reactor Building contains most of the contaminated materials assumed to require disposal during decommissioning. Descriptions of their design, construction and location appear in the following sections.

B.3.1.1 Reactor Vessel and Internals

The reference research reactor appearance and overall dimensions were discussed briefly in Section B.2.1 (see Figures B.2-3 and B.2-4). All concrete in the biological shield structure has a nominal density of 2350 kg/m^3, except for the concrete in the thermal column door, which has a density of 3500 kg/m^3. The concrete shield is pierced radially by the beam tubes, the thermal column, and the thermalizing column, shown in Figure B.3-1. Embedded in the concrete are 13-mm pipes that lead from the thermal column, thermalizing column, and each of the beam ports through valves and into a manifold mounted on the side of the shield structure. An exhaust line is connected to this system through a 13-mm pipe connection to the building exhaust. During operation, this system permits the venting of ^{41}Ar and other radioactive gases from these penetration areas. A 13-mm-diameter conduit, buried in the shield, leads from each of the beam tubes and terminates in the upper platform trench. This conduit contains wiring necessary for indicating the beam plug position. (See Figure B.1-3 for a cutaway view of the reactor illustrating the many experimental facilities.)

Reactor Vessel. The aluminum tank that serves as the reactor vessel has an outside diameter of 2 m, a depth of 6.3 m, and a minimum thickness of 6.4 mm. The aluminum tank is pierced by the four beam tubes, the thermal column, and the thermalizing column.

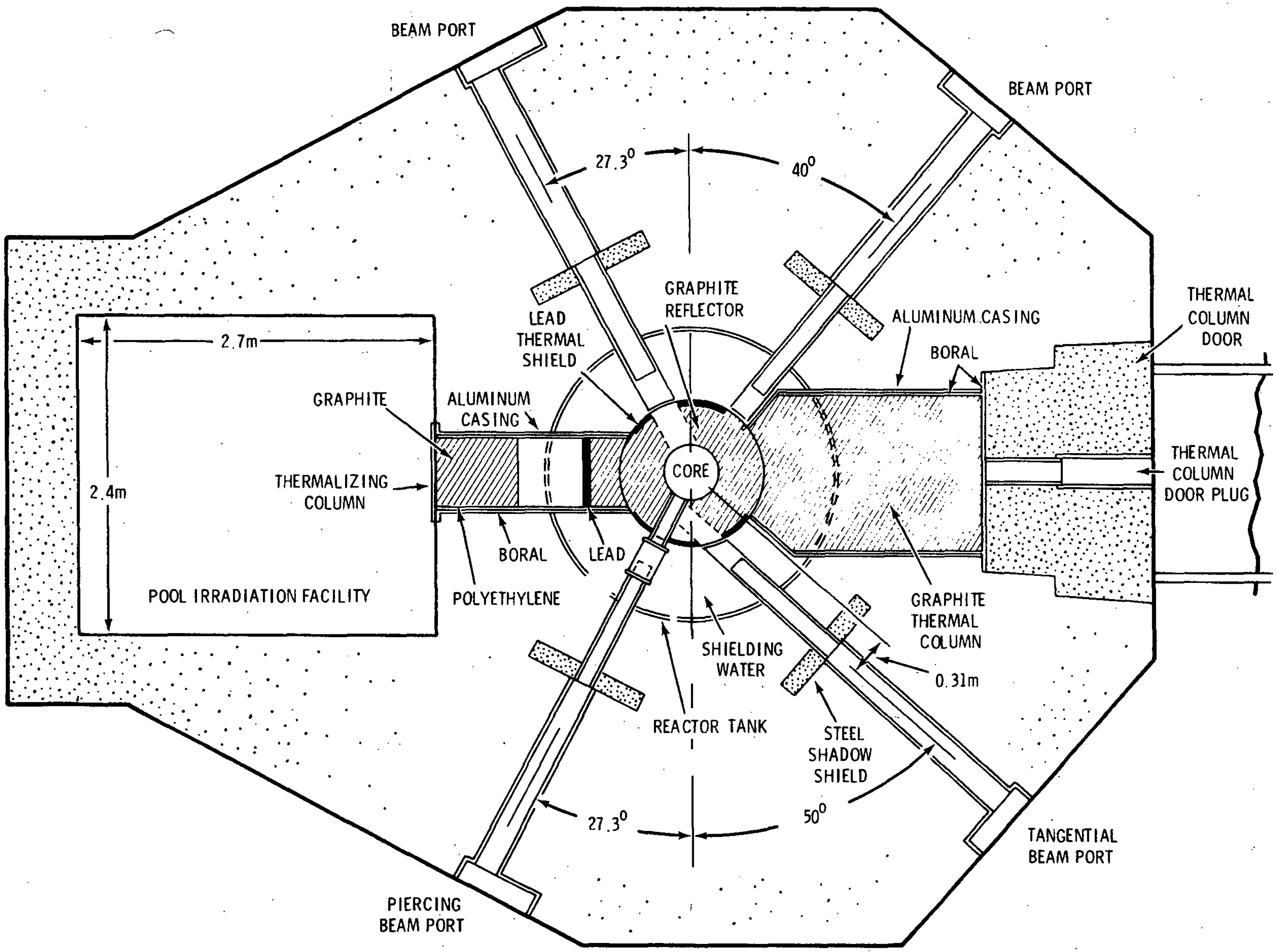


FIGURE B.3-1. Horizontal Section of Reference Research Reactor

The experimental facilities are assembled into inner and outer sections and are separated by a small gap at the exterior surface of the reactor tank. The inner sections of these facilities are welded to and form an integral part of the reactor tank. The outer sections are embedded in the concrete biological shield on the same horizontal centerline as the inner sections. The gap between the inner and outer sections relieves stresses resulting from thermal expansion of the aluminum tank during reactor operation.

The reactor tank and inner sections of the experimental facilities are made leakproof by the use of continuous welded joints. The outside of the tank is coated with an adhesive primer followed by two layers of polyethylene tape.

A 50-mm by 50-mm aluminum channel used for mounting the ion chambers and underwater lights is welded around the top circumference of the tank.

<u>Center Channel Assembly</u>. Support for the central thimble and rotating specimen rack (lazy susan) irradiation facilities, the control-rod drive mechanisms, and the tank covers is provided by the center channel assembly at the top of the reactor tank. The assembly consists of two 0.2-m structural steel channels with three 0.4-m-wide by 16-mm-thick steel cover plates bolted to the flanges of the channels. The assembly has the shape of an inverted U which is 2.6 m long and positioned directly over the center of the reactor tank. The assembly is attached to the recessed tank top by two steel angle brackets. The channel assembly is designed to support a shielded isotope cask weighing 3.2 Mg, placed over the specimen removal tube.

<u>Reactor Vessel Covers</u>. The top of the reactor tank is closed by six aluminum grating covers that are hinged and installed flush with the floor. Lucite plastic, 60 mm thick, is attached to the bottom of each grating section to prevent foreign matter from entering the tank while still permitting visual observation. Each cover has two flush lifting handles to facilitate its movement. The center channel assembly provides support for the unhinged end of the covers when they are closed.

<u>Reactor Core</u>. The reactor core and the reflector assembly surrounding and supporting the core are situated upon an aluminum pedestal about 0.46 m

above the bottom of the vessel. The reflector rests on a platform which raises it 0.6 m above the vessel bottom. The internal arrangements of the reactor are shown in Figure B.3-2.

The core assembly is a right-circular cylinder, 1.1 m in diameter and 0.6 m high, consisting of a compact, concentric array of cylindrical fuel elements, a central thimble, a neutron source, and control rods, all positioned vertically between two grid plates fastened to the reflector assembly.

The top grid plate has 126 positions for fuel elements and control rods, arranged in six concentric rings around a central thimble used for high flux irradiations. The outer ring of the core contains some graphite dummy elements. The reflector surrounds the core and is composed of graphite and lead encased in an annular aluminum can with a radial thickness of about 0.25 m.

The control rods pierce both grid plates. The fueled follower control rods are guided by the grid plates and the transient rod is guided by a guide tube. A restraining safety plate is welded to the reflector assembly beneath the bottom grid plate to prevent a disconnected or loose control rod from dropping out of the core. Most of the reactor internals are constructed of aluminum to minimize the production of long-lived activation products.

Reflector Platform. The reflector platform, shown in Figure B.3-3, is a square, all-welded aluminum frame structure. It rests on four legs that are held down by aluminum anchor bolts welded to the bottom of the aluminum tank. Four pads with spherical indentations are mounted on top of the platform to serve as receptacles and supports for the reflector assembly.

Graphite Reflector. The reflector surrounding the core (see Figure B.3-3) consists primarily of a ring-shaped, aluminum-clad, block of graphite having an approximate inside diameter of 0.55 m, a radial thickness of 0.26 m and a height of 0.56 m. Included in this container, at the perimeter and adjacent to the water, is a 51-mm-thick lead "donut" which serves as a thermal shield to protect the concrete structure from excessive nuclear heating. The lead has been flame-sprayed to a molybdenum coating on the inner surface of the aluminum can. Provision for the rotary-specimen rack is made in the form of a ring-shaped well in the top of the reflector. (The rotary-specimen-rack

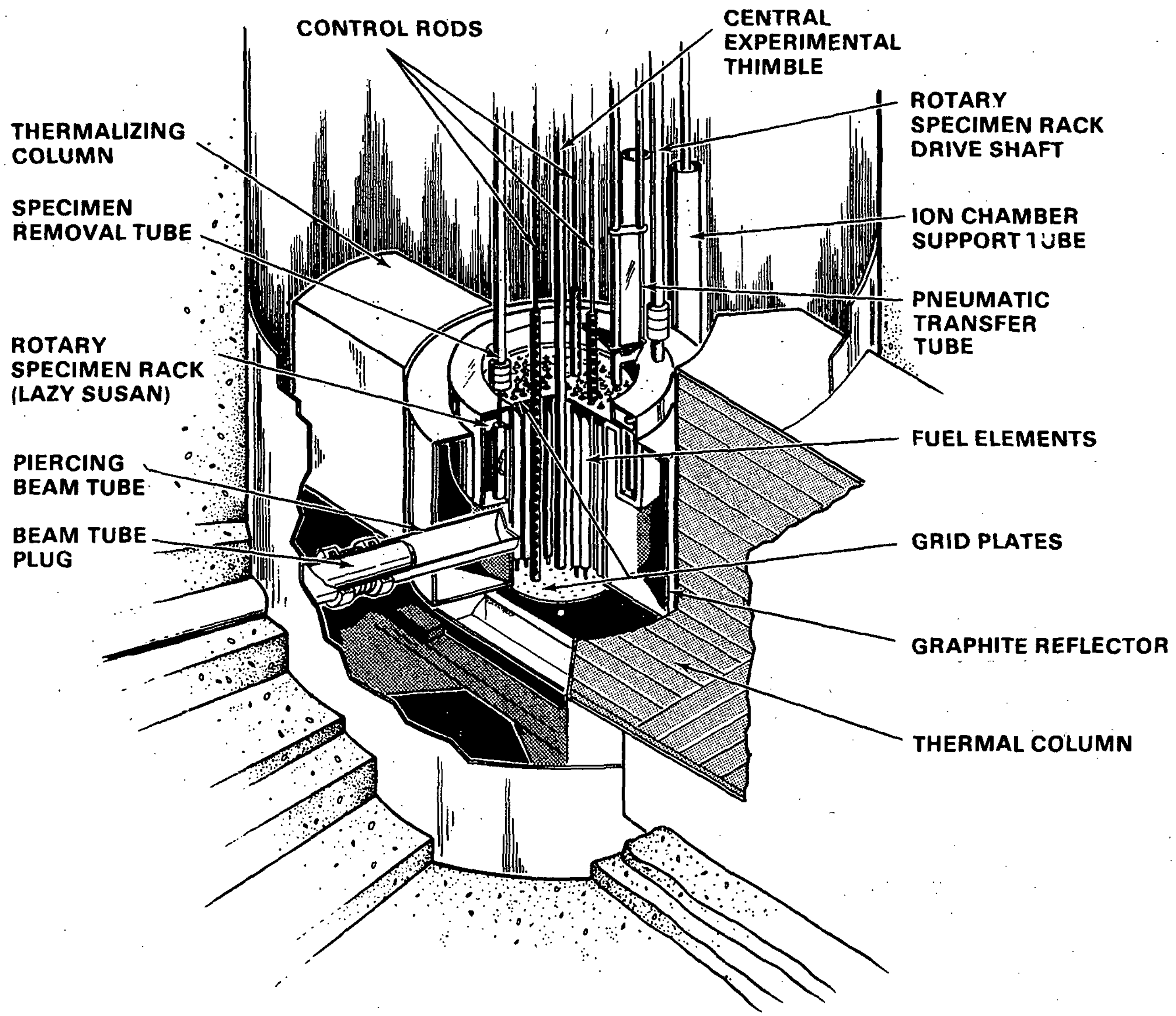


FIGURE B.3-2. Cutaway View of the Reference Research Reactor Core Structure

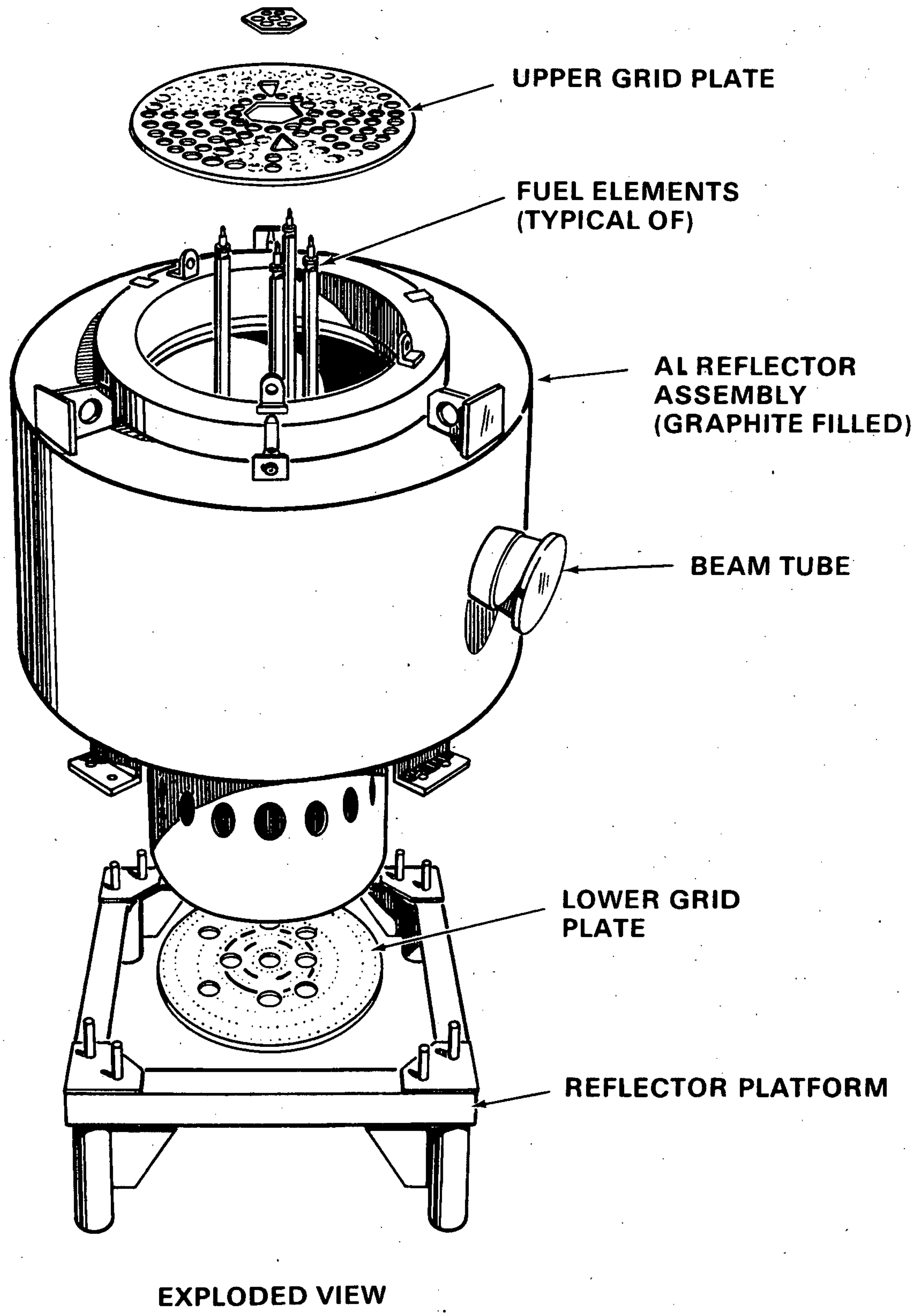


FIGURE B.3-3. Reactor Core and Support Design

container seats in the aluminum-clad well and does not penetrate the sealed reflector assembly at any point. On the top surface of the container are three aluminum blocks which provide for rotary-rack-centering screws and holddown clips.

The inner extension of the piercing beam tube (which is an integral part of the reflector assembly) is an aluminum tube that penetrates through the graphite to the inner surface of the reflector container. This tube is 168 mm in outside diameter and has a wall-thickness of 6.5 mm. The tube extends 140 mm beyond the outer surface of the reflector, where it ends in a flange for attachment with a special bellows assembly. The bellows assembly is in turn clamped to the section of the piercing beam tube that is welded to the aluminum reactor tank. A blank plate is welded to the inner end of the extension tube in the reflector.

One beam tube and the tangential beam tube penetrate the reflector graphite. These tubular holes form air voids in the graphite but do not pierce the aluminum reflector can. The reactor tank sections of these beam tube facilities terminate outside the reflector assembly and abut against the reflector can.

The reflector assembly rests on the reflector platform. Support is provided by two aluminum channels welded to the bottom of the reflector container. Four tapped holes in the lower flanges of the channels accept the leveling screws, which transmit the weight of the reflector and core to the reflector platform. The inner surface of the reflector housing is 0.53 m ID and extends about 0.51 m below the graphite. It serves as a core shroud providing support for the two grid plates. A safety plate, which is welded to the bottom of the shroud, prevents the possibility of a loose control rod from dropping out of the core.

Three lugs with 51-mm-diameter holes are provided for lifting the reflector assembly. Its weight is approximately 1.37 Mg.

Grid Plates. Both the upper and lower grid plates are made of anodized aluminum. The upper grid plate (see Figure B.3-3) is mounted on an aluminum adapter ring, which is welded to the reflector container. Two 80-mm-diameter

stainless steel dowel pins, which fit tightly in the adapter ring and loosely in the plate, position the grid. Four 10-mm captive stainless steel bolts secure the upper grid plate to the ring. The top plate is 0.54 m in diameter and 16 mm thick. The diameter of the central thimble and fuel element holes is 38 mm. Two triangular holds, occupying the space of three fuel elements in the upper grid plate, are filled by stainless steel spacers.

The lower grid plate (see Figure B.3-3) is 19 mm thick and supports the core. It is positioned upon support pads by means of two 80-mm-diameter stainless steel dowel pins. Four 9.5-mm stainless steel bolts secure the grid to the support pads.

Neutron Source Holder. A specially designed source holder houses the americium-beryllium neutron startup source. The dimensions of the holder permit it to be installed in any fuel element location in the core, but it generally occupies one of the outermost positions.

Graphite Dummy Elements. Aluminum- or stainless steel-canned graphite dummy elements occupy some grid positions not filled by fuel elements and other core components. These graphite dummy elements have aluminum end fixtures and are completely filled with graphite.

Transient Control Rod Guide Tube. The transient rod is guided by a 35-mm ID aluminum guide tube which is held in position by the grid plates. The guide tube may be unscrewed from its location with a special handling tool.

Ion Chambers. Four ion chambers are mounted around the reflector in the reactor vessel. Each chamber is seal-welded in aluminum containers. The electrical connections for each chamber are contained in a 19-mm aluminum pipe. The length of each assembly is 5.8 m.

Fuel Storage Racks. Three fuel storage racks, each capable of holding 30 fuel elements, are located underwater along the walls of the reactor tank to provide temporary fuel storage (see Figure B.1-3). Each rack is 0.5 m high and 0.57 m wide, with 41-mm-diameter cutouts, and is made of 16-gauge aluminum. Each rack is suspended from two aluminum rods, 19 mm in diameter by 4.6 m long.

Control Rods. The three motor-driven control rods are 36-mm OD, stainless steel clad, and pass through and are guided by the grid plates.

For pulsing operation the reactor is equipped with a pneumatic drive system which rapidly withdraws a transient poison rod from the core. The rod is a 32-mm OD aluminum tube with plug ends. The upper end of the transient rod is connected to an aluminum transition piece which is pinned in place by 6.4-mm (SS) pins to a 22-mm aluminum extension shaft. A small stainless steel retaining pin is also used to assure linkage. The transient rod is guided by the thin-wall aluminum guide tube described previously.

The control rod and transient rod drive assemblies are fastened to a mounting plate located on the center channel assembly, as shown in Figure B.2-1.

B.3.1.2 <u>Experimental Irradiation Facilities</u>

The Experimental Irradiation Facilities are located both internal to and external to the reactor vessel. Often, even the external portions are welded to and form an integral portion of the vessel exterior. The experimental facilities are described in this section.

<u>Rotary Specimen Rack</u>. The rotary specimen rack, shown in Figure B.3-4, surrounds the core and consists of an aluminum rack for holding 40 specimens during irradiation. This rack is located inside a ring-shaped, seal-welded aluminum housing, and is positioned by centering screws attached to the reflector assembly. The rack is rotated on a stainless steel ball-bearing assembly. It supports 40 evenly spaced tubular aluminum containers, open at the top and closed at the bottom, which serve as receptacles for specimen containers. The tubes are 0.275 m long and spaced 0.053 m apart, center to center, on a 0.337-m radius from the center of the core.

The specimen-removal chute is an aluminum tube that begins in a funnel just below the top plate of the center-channel assembly. This funnel aids the insertion of the specimen container. Loading and unloading of the 40 specimen tubes in the rack takes place through the specimen-removal chute, which has an internal diameter of 34 mm. The tube is offset by approximately 457 mm by means of large-radii tube bends to provide water shielding.

The tube-and-shaft assembly, located 180° from the specimen-removal chute, is a sealed straight tube. It encloses the drive and locking shafts and connects the rotary-specimen-rack housing with the drive-and-indicator assembly

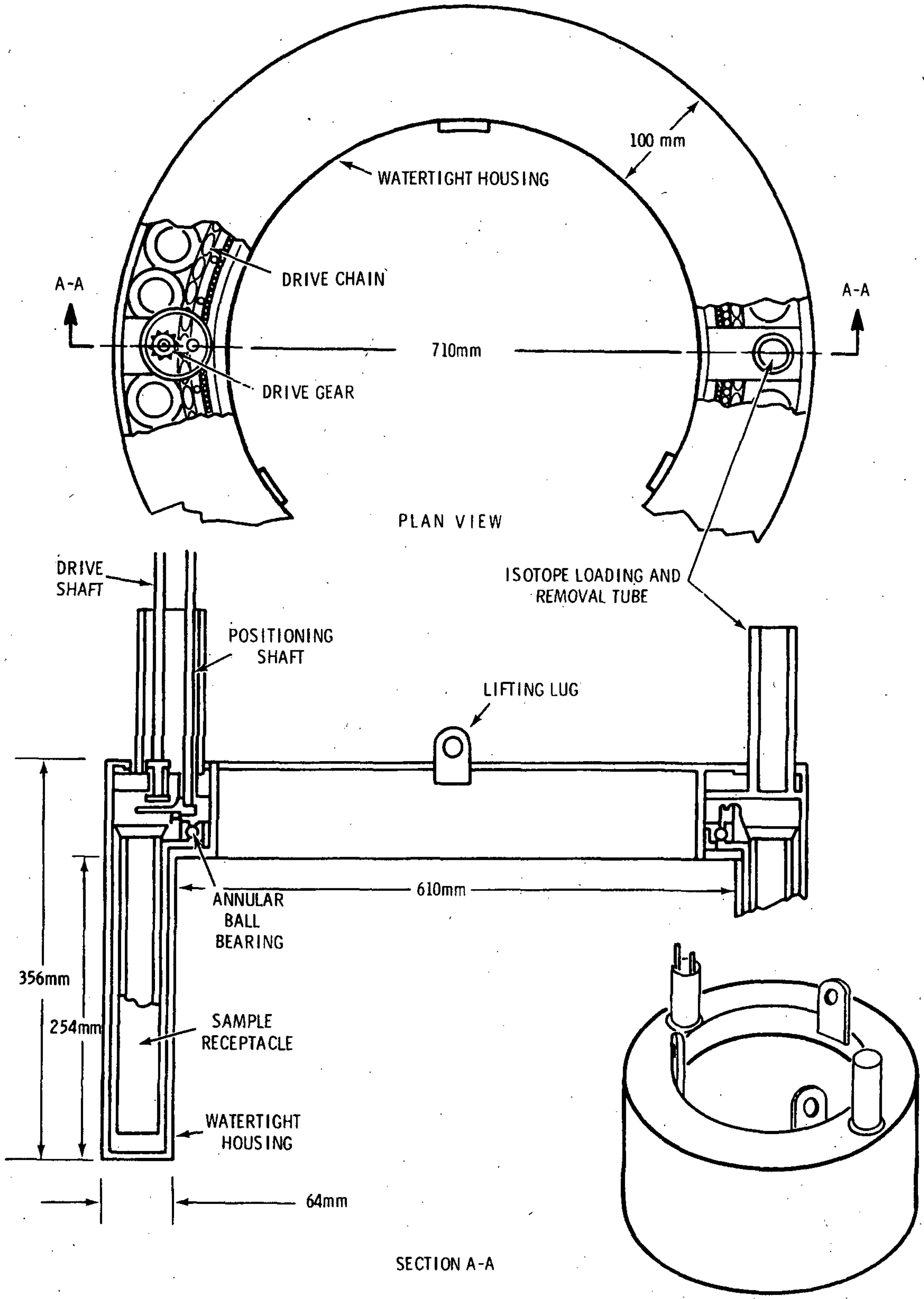


FIGURE B.3-4. Rotary Specimen Rack (Lazy Susan)

on the center channel assembly. Since this tube is in a straight line from the reflector, radiation shielding is provided by 1.5 m of polystyrene enclosed within the tubing.

The drive-and-indicator assembly is located on the center-channel cover at the top of the reactor tank. The assembly includes an indicator dial with 40 divisions, a crank for rotating the specimen-rack gear train, and a locking rod handle. The motorized drive permits continuous rotation and consists of a fractional horsepower motor, a worm gear, and a slip clutch located inside the drive-and-indicator assembly box.

The specimen lifting device (a modified fishing pole) is used for inserting specimen containers in, and removing them from the rotary rack. A solenoid-operated specimen pickup tool is located at the end of the electrical wire fishing line.

Pneumatic Transfer System (Rabbit). Most components of this system are located within the support area of the Reactor Building. The pneumatic transfer system is shown schematically in Figure B.3-5. It consists of the following major components:

- specimen capsule (polyethylene rabbit)
- blower (1.5 hp) and filter (0.3 m x 0.3 m x 0.15 m)
- four solenoid-operated valves and 57-mm tubing
- terminus
- receiver-sender
- control assembly (timer, electrical switching box)
- tube fittings.

The terminus assembly is located in the reactor tank. The terminus support is shaped like a fuel element and can fit any fuel location. The tubes are 32-mm OD aluminum and have an overall length of 3.9 m. The aluminum receiver-sender assembly is located behind a swing door cover. The length of the swing door access port is about 300 mm.

The rabbit travels within 32-mm aluminum tubing; however, many of the long runs in the return air system are 57-mm copper tubing.

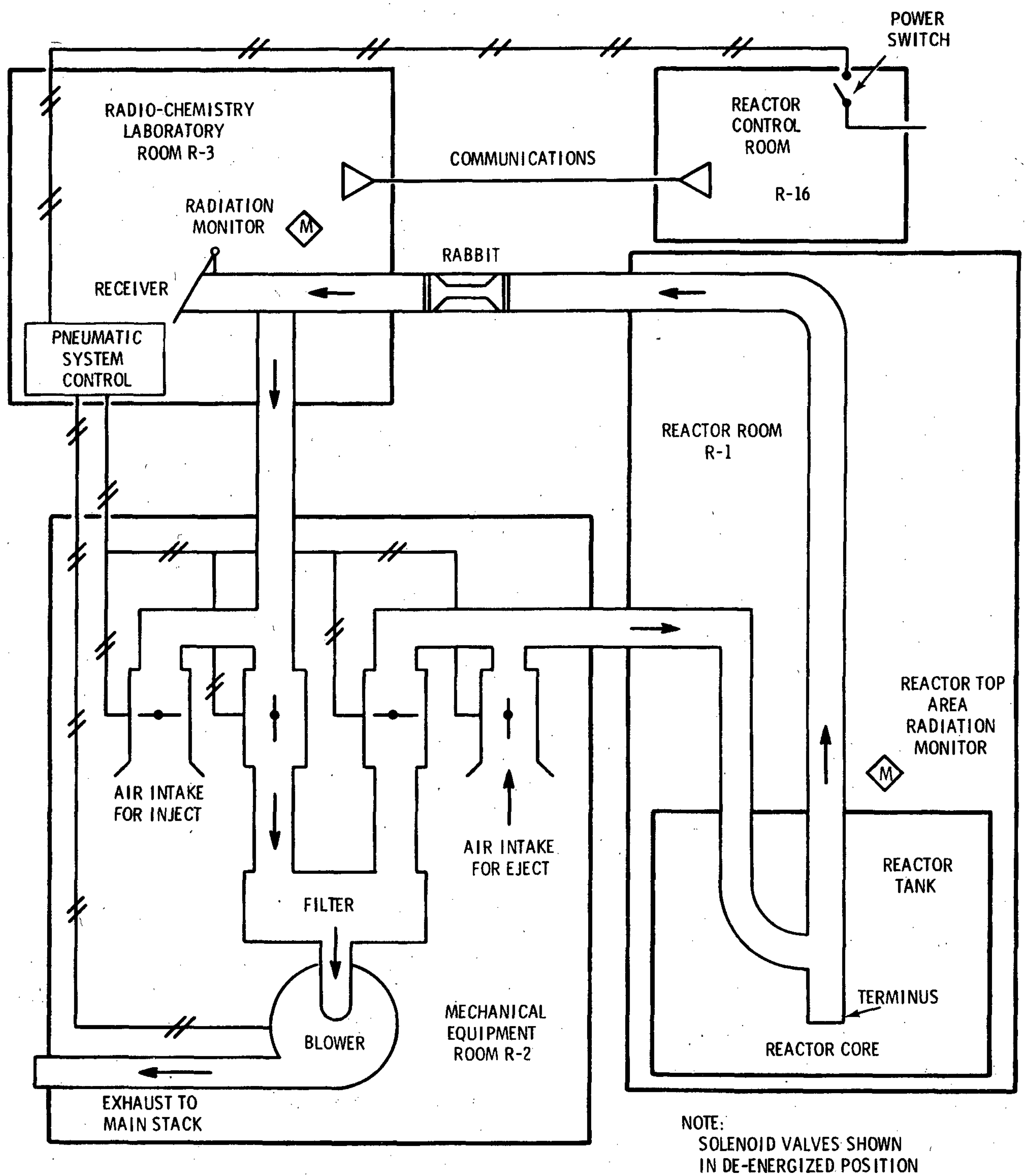


FIGURE B.3-5. Pneumatic Transfer System Schematic Diagram

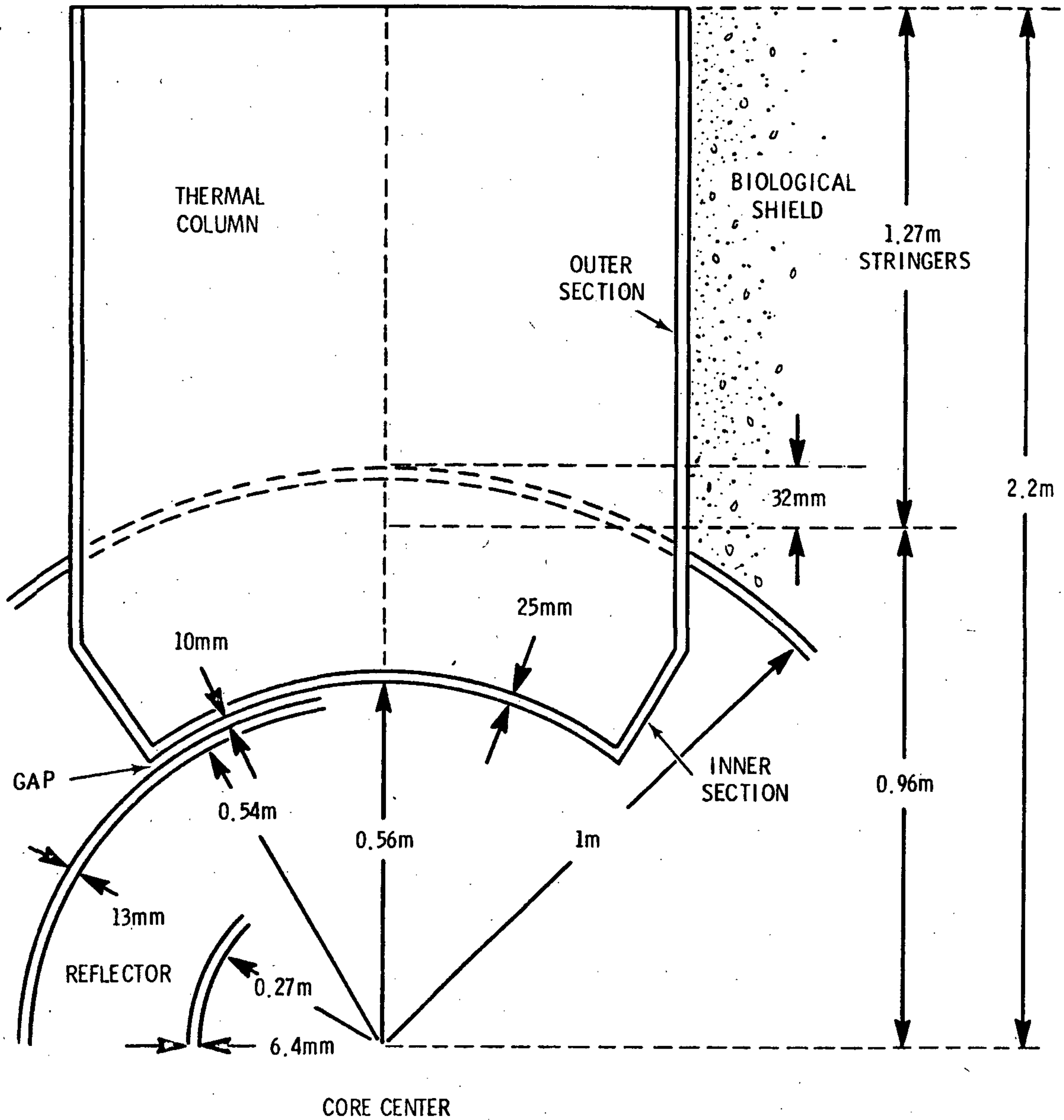
Central Thimble. The central thimble is located in the center of the core. It is a 32-mm aluminum tube with a wall thickness of 2.1 mm. It extends from the reactor bridge down through the central hole of the removable hexagonal section in the top grid plate and through the lower grid plate. It terminates with a plug and is supported by the safety plate situated 0.43 m beneath the lower grid plate. The thimble is made in two sections, 3.7 m and 3.4 m long, and is supported by a fixture at the bridge.

Thermal Column. The thermal column, shown in Figure B.3-6, is a large, boral-lined, graphite-filled aluminum container. The thermal column is installed in the lower portion of the biological shield structure; it penetrates the shield and terminates at the outer edge of the reflector. Its outside dimensions are 1.2 m by 1.2 m in cross-section by approximately 1.6 m in length.

The thermal-column liner is a seal-welded container fabricated in two sections from 13-mm aluminum plate. The outer section is embedded in the concrete shield and the inner section is welded to and is an integral part of the aluminum tank. The exterior surfaces of the outer section (which are in contact with the concrete) are coated with plastic tape for corrosion protection. Surrounding the graphite on the inside of the aluminum casing are 3.2-mm boral sheets. The inner section (welded to the aluminum tank) extends to the graphite reflector and matches the contour of the reflector over a 100° angle. The horizontal centerline coincides with that of the active core lattice. In a vertical plane, the thermal column extends approximately 0.330 m above and below the reflector, with the centerlines of the column and the reflector coinciding.

The aluminum container is open toward the reactor room. Blocks of nuclear-grade graphite occupy the entire volume. The individual blocks are approximately 0.100 by 0.100 m in cross section, the longest being 1.3 m in length. To gain access to the graphite, the thermal-column shield door must be rolled back on its tracks (see Figure B.3-1).

The outer face of the thermal column is shielded by a track-mounted, stepped door approximately 1.3 m thick. The door is recessed into the concrete biological shield and is flush with the biological shield structure when



NOTE: WATER GAP BETWEEN REFLECTOR AND THERMAL COLUMN CANNING IS 10mm. THE STRINGERS ARE MADE OF GRAPHITE AND THE REFLECTOR CONTAINS 0.26m OF GRAPHITE.

FIGURE B.3-6. Thermal Column and Core Alignment

closed. The door is filled with heavy-aggregate concrete with a density of 3500 kg/m^3. Its total weight is about 17 Mg. A four-wheeled carriage supports the door and rolls on two steel rails which are flush with the floor.

On the surface facing of the thermal column, a 3.2-mm-thick boral sheet 1.3 m by 1.3 m is attached to the door with an epoxy adhesive. A 0.17-Mg, 0.23-m-diameter plug penetrates the door near its midpoint. Small-diameter steel pipes embedded in the concrete are connected to the thermal column and each of the beam ports. The steel pipes terminate in a manifold at the base of the biological shield and are connected to a venting system for radioactive argon gas.

Pool Irradiation Facility. A water-filled neutron moderating pool is located at the end of the thermalizing column next to the reactor core (see Figure B.2-4). This rectangular shaped pool irradiation facility is 2.7 m long, 2.4 m wide, and 3.7 m deep. It is waterproofed with five applications of epoxy coating and is filled with approximately 23.7 m^3 of water.

The thermalizing column between the reactor and the pool irradiation facility is constructed in two sections similar to the thermal column, but smaller. Its dimensions are 0.6 m by 0.6 m in cross-section by approximately 1.3 m long and is fabricated from 13-mm-thick welded aluminum plates. Its outer sections extend from the pool through the concrete shielding to the aluminum reactor tank. The inner section of the column is welded to and is an integral part of the tank and extends inward to the reflector assembly and matches its contour.

The horizontal centerline coincides with the centerline of the active fuel lattice. The exterior surface of the outer section (which is in contact with the concrete) is coated with plastic tape.

An aluminum cover plate, 16 mm thick, is held in place by 36 aluminum bolts. This cover plate seals the water out of the thermalizing column. The bolts are threaded into captive stainless steel inserts.

In the region adjacent to the biological shield, the aluminum container is lined with 3.2-mm boral sheets that extend 1 m inward from the pool irradiation facility.

At the inner end (the end nearest the reactor core), the column is filled with graphite blocks to an axial thickness of 0.020 m. All the blocks are made from machined blocks of nuclear-grade graphite 0.1 m by 0.1 m in cross-section. This 0.2-m wall of graphite is backed by a 0.050-m-thick lead slab which stands on its edge and is held in place by two aluminum angles fastened to the aluminum inner section.

In the outer section of the container, where the aluminum is lined with boral, 26-mm-thick polyethylene sheets line the boral. The polyethylene extends into the column to the supporting brackets for the lead slab and overlaps the boral by approximately 20 mm.

Graphite blocks are stacked 0.6 m thick from the outer edge of the column.

Beam Port Facilities. Four beam ports penetrate the concrete shield and the aluminum tank and pass through the reactor tank water to the reflector (see Figure B.3-1). They provide irradiation facilities for large specimens (up to 0.15 m in diameter). The outer, embedded portions of the beam ports are 0.2 m in diameter. They are constructed of cadmium plated steel. A 13-mm pipe leads from each of these outer port sections to an argon gas vent system.

The beam port inner sections are made of aluminum and divided into two pieces, one part with a 0.154-m ID tube embedded in the concrete shield, and the other with a 0.16-m ID tube welded to the aluminum tank. A flexible-joint bellows assembly connects the two tube sections. The joint consists of a stainless steel bellows and flange on each end. The embedded sections are coated on the outside with plastic tape. The beam ports are 3 m in length.

A 100-mm-thick steel shadow shield (see Figure B.3-1) is placed around each beam port. Each one is approximately 1 m^2 and surrounds the 0.16-m tube portion, adjacent to the 0.2-m section. Shielding is also provided at the ends of each beam port by an aluminum plug filled with sections of concrete, boral, lead, and borated concrete.

The outer section of beam-port shielding is a wooden plug about 1.3 m in length, depending upon the port, and built to fit the 0.2-m-diameter outer beam tube. The outer end of the beam-port is equipped with a lead-filled safety shutter and door. The welded steel frame of the shutter is 0.230 m square and 0.114 m thick. The shutter assembly is mounted on horizontally sliding shafts.

A recess area is provided within the concrete shield for operation of the shutter. An additional 10-mm steel side-hinged door, lined with 32 mm of lead, is the final port barrier.

B.3.1.3 Beam Port Shielding

Biological shields surround two of the four beam ports for safety purposes. These shields are constructed from building-block-type materials and the structures house experimental arrangements utilizing the beam tubes. They are built so that personnel access is indirect, thereby preventing accidental beam exposure. The building blocks are typically 0.25 m long by 0.063 m wide by 0.025 m deep. The materials used in these blocks are normal concrete, high-density concrete, parafin, and lead. Typically, normal-density concrete blocks constitute 80% of the structures which are built up from the concrete pad surrounding the reactor. Their dimensions vary slightly, depending upon the particular application. However, the shielded area is approximately 2.3 m long by 1.8 m wide with a height of over 2 m. Overhead shielding of the roof is usually greater than 0.25 m thick.

B.3.1.4 Short-Term Hot Sample Storage

Two small lead shields are provided, one on the reactor bridge and the other in the reactor room sample processing area to allow sample radiation decay. These are made of standard lead bricks and are approximately 0.5 by 0.7 by 0.7 m in dimension.

B.3.1.5 Sub-Critical Pile and Experimental Apparatus

Special experimental apparatus are assumed to be part of the reference research reactor's equipment. One such apparatus is a small graphite-moderated subcritical nuclear pile located in the reactor room. The approximate overall dimensions of the pile are 2 m long by 2 m wide by 2-1/2 m high. Natural uranium rods make up the core of the facility.

B.3.1.6 Storage Pits

Five storage pits, located in the reactor room floor, provide for storage of radioactive materials. Location of the storage pits in the reactor room floor is shown in Figure B.3-7. Each storage pit is 0.25 m in diameter and 3.7 m deep and is provided with:

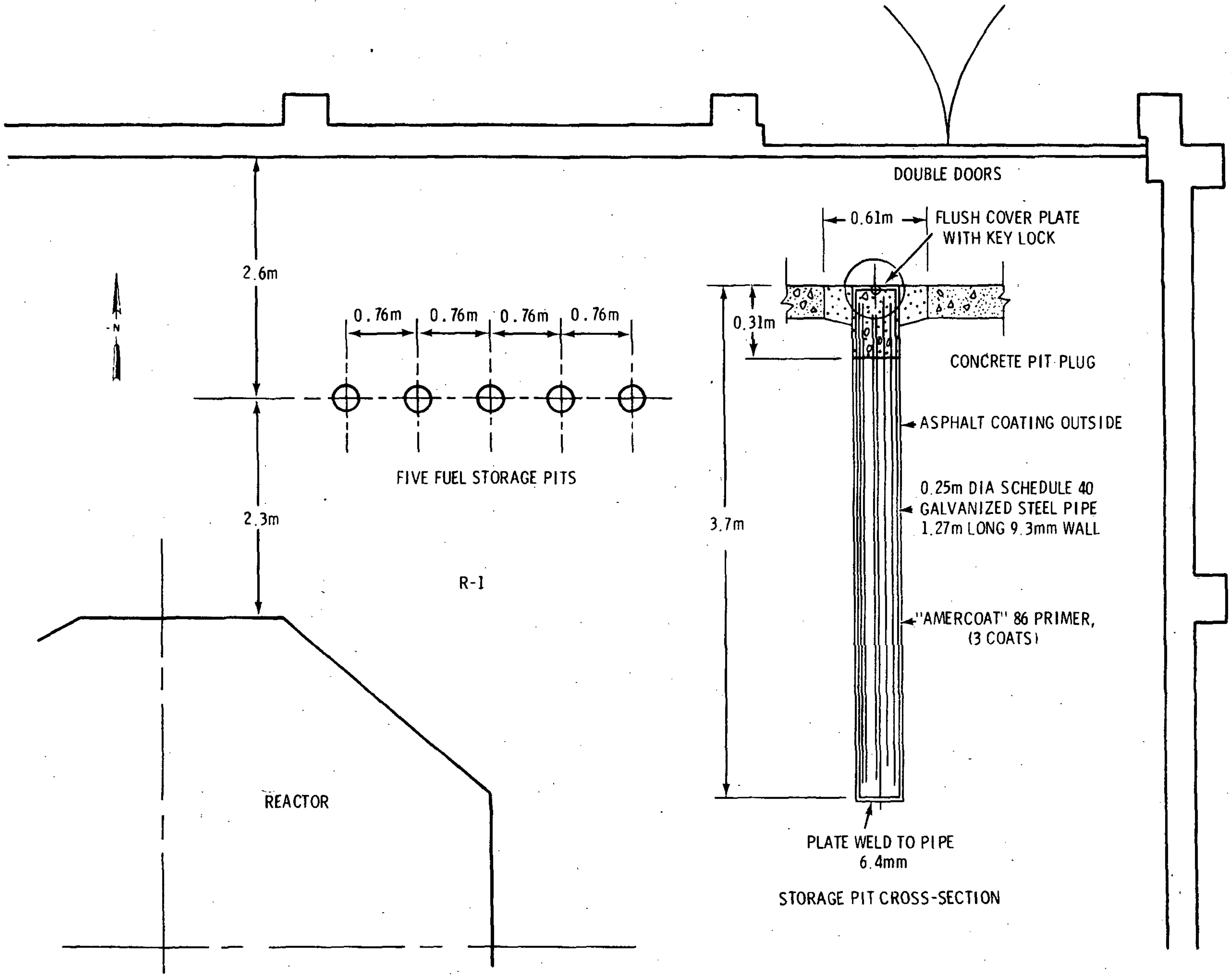


FIGURE B.3-7. Storage Pits

- a 0.3-m-thick concrete shielding plug installed at the top of the pit
- a cover plate that fits flush with the floor
- a key lock on the cover plate to limit pit access.

For storage of irradiated fuel elements, the pits can be filled with water to aid in the cooling process.

B.3.1.7 Support Area

Within the Reactor Building, several areas are allocated for the handling and processing of radioactive materials. Brief descriptions of these areas follow.

First Floor. Within the radiochemistry laboratory, R-3 (see Figure B.2-1a), where the receiver-sender assembly of the "rabbit" facility is located, there are two work hoods and a sink-drain appliance. The reactor room floor has a radioisotope work area containing a shielded hood, an isotope storage area, and a sink-drain appliance.

Second Floor. The radiation protection laboratory, R-9 (see Figure B.2-1a), on the second floor of the Reactor Building contains a hood and a sink-drain appliance.

Third Floor. None of the third floor facilities is expected to be contaminated or require removal of any radioactive materials.

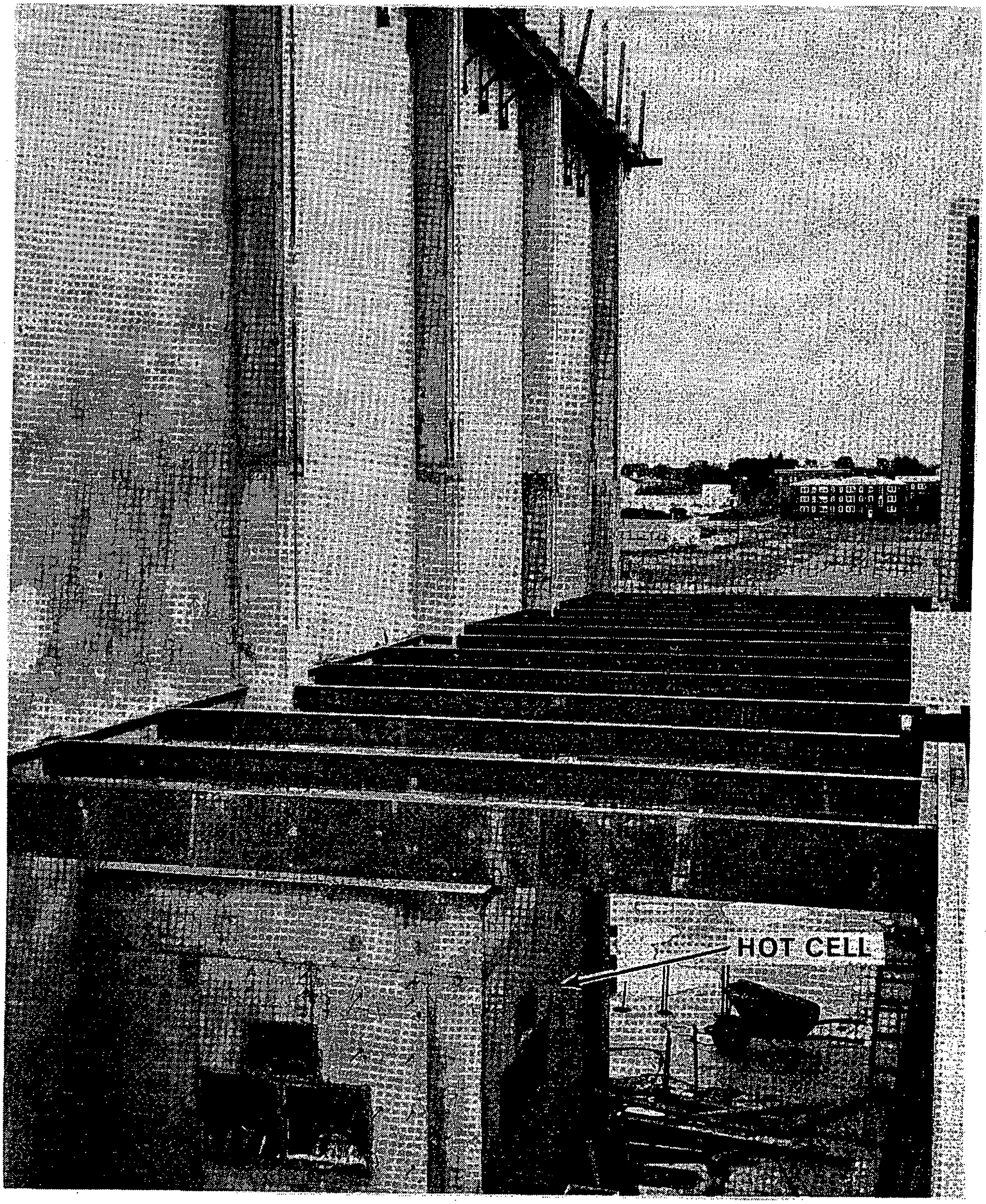
Fourth Floor. A mechanical area, R-22 (see Figure B.2-1b), located on the fourth floor of the Reactor Building, contains most of the air handling equipment for the reactor room and support areas within the Reactor Building.

B.3.2 Annex Building

A single-story building connects the Radiation Center and the Reactor Building (see Figure B.2-1a). A hot laboratory area, R-5, and hot cell are located in this annex. A construction photograph, shown in Figure B.3-8, clearly illustrates the hot cell. Locations for two manipulator appliances are built into this cell.

B.3.3 Heat Exchanger Building

The Heat Exchanger Building, R-24, contains equipment for operation of three water pumping systems used in reactor operations and an air compressor



<u>FIGURE B.3-8.</u> Hot Cell

system for transient rod operation. The three pumping systems are the water purification system, the primary water pumping system, and the secondary water pumping system. The first two systems are expected to be contaminated and to require decommissioning activities. These are discussed in the following section.

B.3.3.1 Reactor Water Purification and Recirculating Systems

The primary water cooling and purification system is shown schematically in Figure B.3-9. The primary water cooling system pumps water from the reactor pool through a heat exchanger and back to the pool. The centrifugal pump has a capacity of 1.33 m^3/min. Piping in the primary water cooling system is 100-mm-diameter schedule 40 aluminum pipe.

Essential features of the reactor water purification system for radioactive waste removal are a filter and a demineralizer. The filter removes small particulate contamination while the demineralizer essentially eliminates solution ions. The demineralizer is a mixed bed ion exchanger which uses HOH nuclear grade resin. It contains 0.085 m^3 of resin and can process 0.037 m^3/min of liquid.

In addition, the purification system consists of a pump; a monitor vessel that contains probes for measuring temperature, radioactivity, and conductivity; a fiber cartridge filter; and a flow meter. The piping material is 25-mm ID aluminum, except in the long trench/runs where 25-mm ID plastic piping is used. The demineralizer contains 0.08 m^3 of resin and is shielded with concrete bricks.

A surface skimmer is also provided for cleaning the reactor pool water. It is connected to the main water suction line by 25-mm ID piping. The skimmer is a 200-mm-diameter plastic right cylinder with an internal particulate filter basket.

Heat Exchanger. A shell and tube heat exchanger shown in Figure B.3-10 is provided for the removal of heat from the reactor. The steel shell is 4.3 m long by 0.56 m OD. It contains 72 U-shaped stainless steel tubes that are welded into a removable stainless steel tube bundle. All parts of the

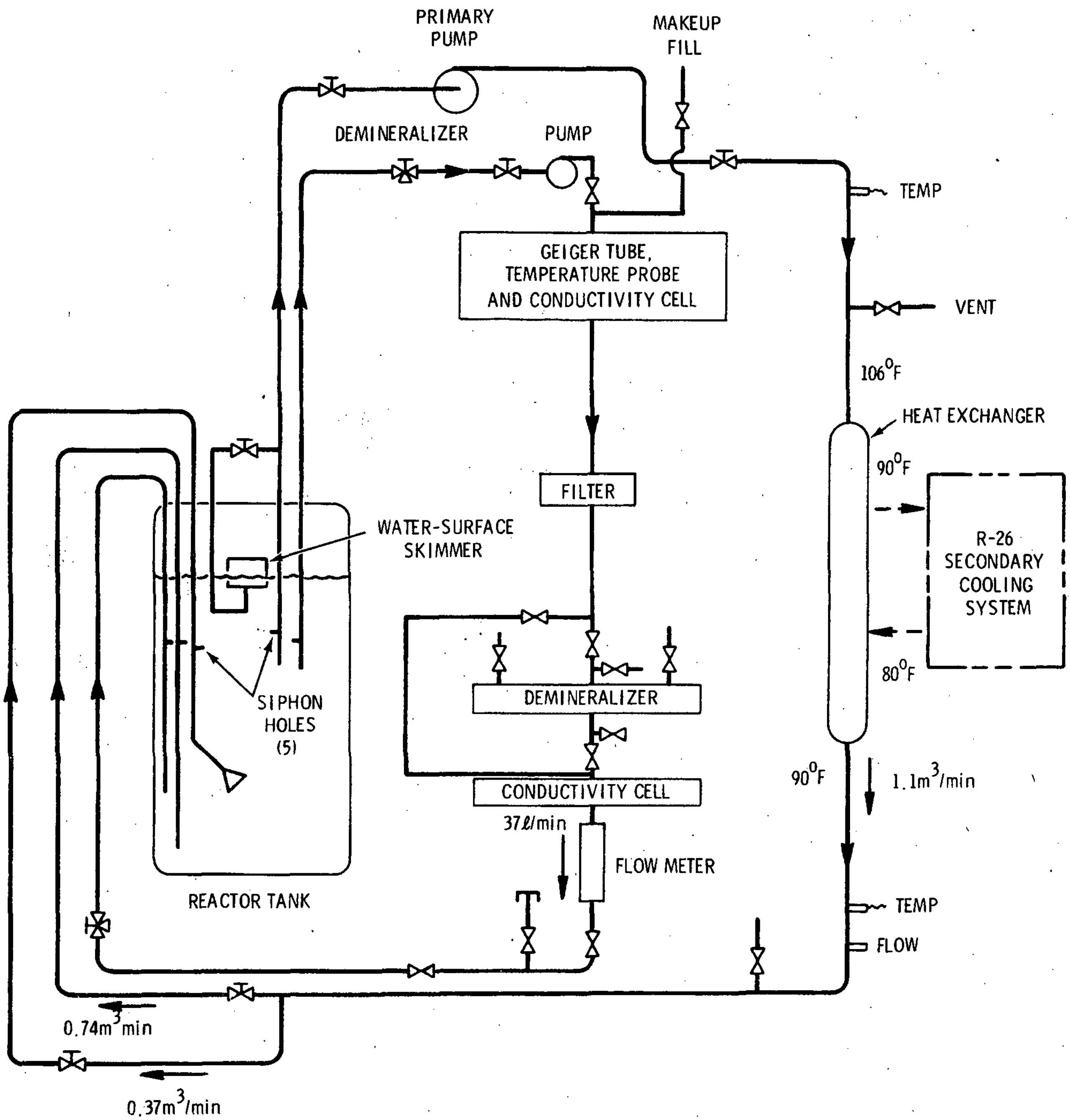
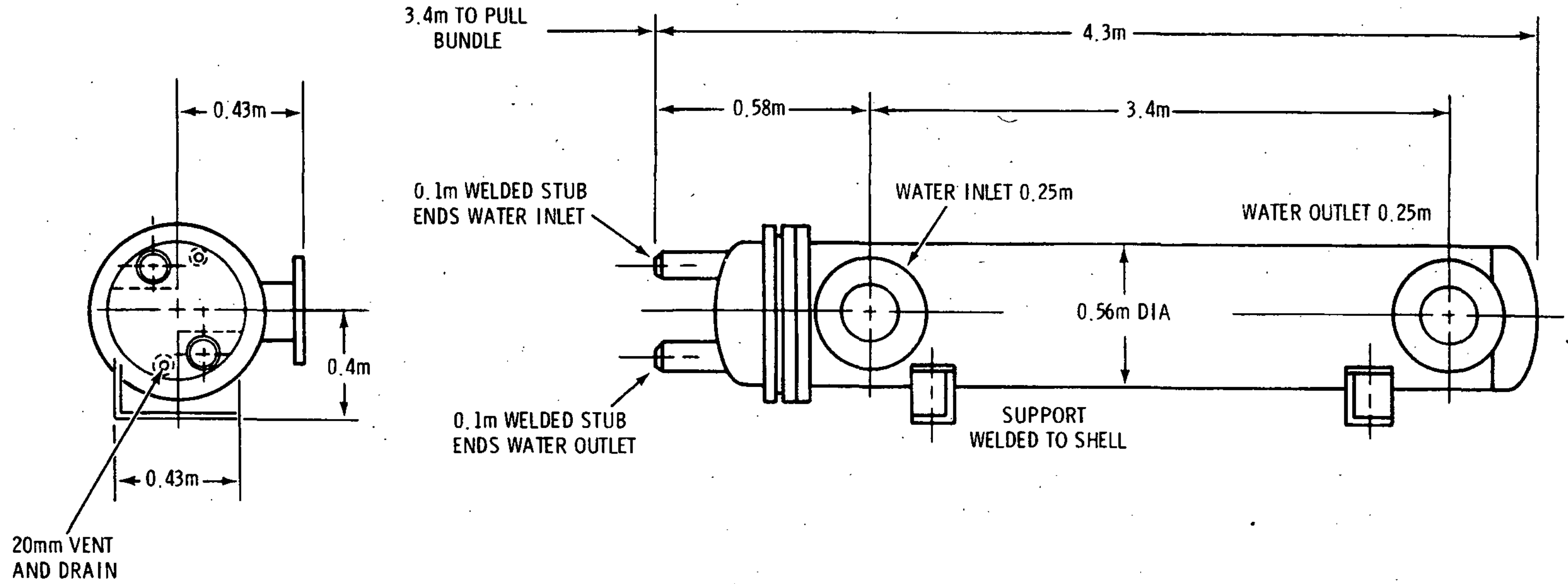


FIGURE B.3-9. Reference Research Reactor Primary Water Cooling and Purification System



NOTE:

TUBESIDE NO. 304 STAINLESS STEEL: SHELLSIDE CARBON STEEL

$79 m^2$ OF 19mm O.D. x 18 BWG 304 STAINLESS STEEL WELDED AVERAGE WALL TUBING

FIGURE B.3-10. Heat Exchanger

heat exchanger contacting the demineralized primary water are made of Type 304 stainless steel. The water in the secondary cooling system flows on the shell side of the heat exchanger.

B.3.3.2 Transient Rod Drive System

The transient rod may be either electromechanically or pneumatically withdrawn from the reactor core. For pneumatic operation compressed air is supplied to a cylinder in the rod-drive mechanism from an air compressor located in the Heat Exchanger Building. The compressor motor drive is approximately 10 hp and the interconnecting air lines are 25-mm ID aluminum pipe.

B.3.4 Liquid Waste System

The liquid waste system for handling contaminated wastes is shown schematically in Figure B.3-11. Radioactive liquid wastes from the facility drains are directed to a retention tank (11.1 m^3) for temporary storage. Two pumps (0.37 m^3/min) direct the liquid to permanent disposal in either the sanitary public sewage system or to offsite shipping appliances. Piping and valving allow liquid sampling and diversion to the proper depository.

B.3.5 Waste Processing and Storage Room

A liquid waste processing and solid waste storage room, R-27 (see Figure B.2-1a), is located along the north wall of the Radiation Center Building (see Figure B.1-1) and has direct outside access. The floors and walls are concrete. Radwaste compaction and solidification equipment are located within this room.

B.4 HEATING, VENTILATING, AND AIR CONDITIONING SYSTEMS

This section presents information concerning the air supply and exhaust systems used to provide heating, ventilation, and air conditioning to the Reactor Building.

B.4.1 Reactor Building

Air systems for the reactor room and support areas are separate and distinct. The following sections describe each system.

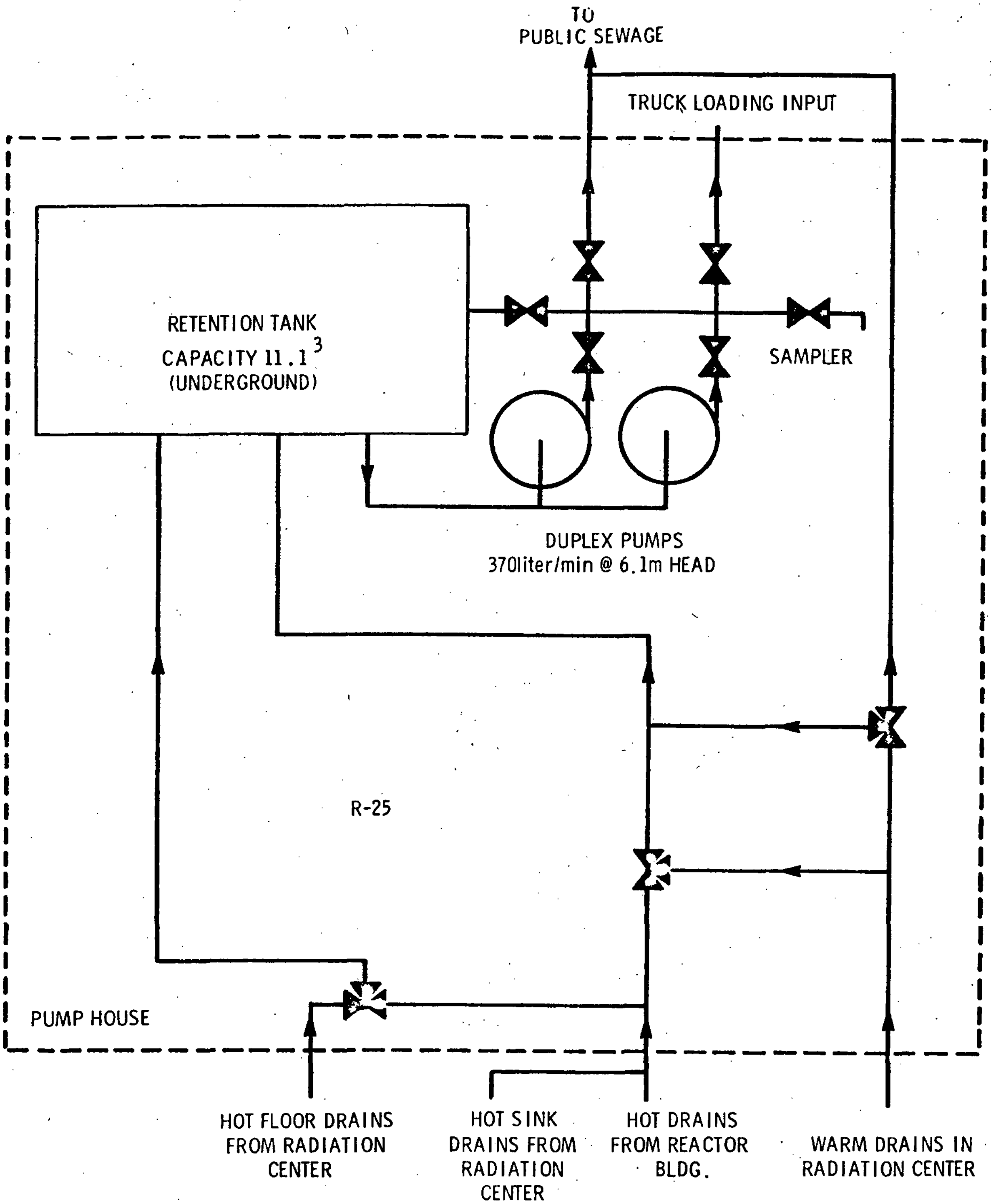


FIGURE B.3-11. Liquid Waste Handling System Schematic Diagram

B.4.1.1 Reactor Room

A separate ventilation and heating system is provided for the reactor room, R-1 (see Figure B.2-1a). It supplies fresh air to the reactor room at the rate of 340 m^3/min. The fresh air intake enters through a 2.3-m by 1.8-m pneumatic damper-filter-fan system and into the reactor room through four ducts near the ceiling.

The exhaust effluents leave the reactor room through four outlet ducts; three near the ceiling carry one-half the air, while the one near the floor exhausts the rest. All of the reactor room exhaust is discharged through a 6.7-m stack located on the Reactor Building roof. A schematic of this ventilation system is shown in Figure B.4-1.

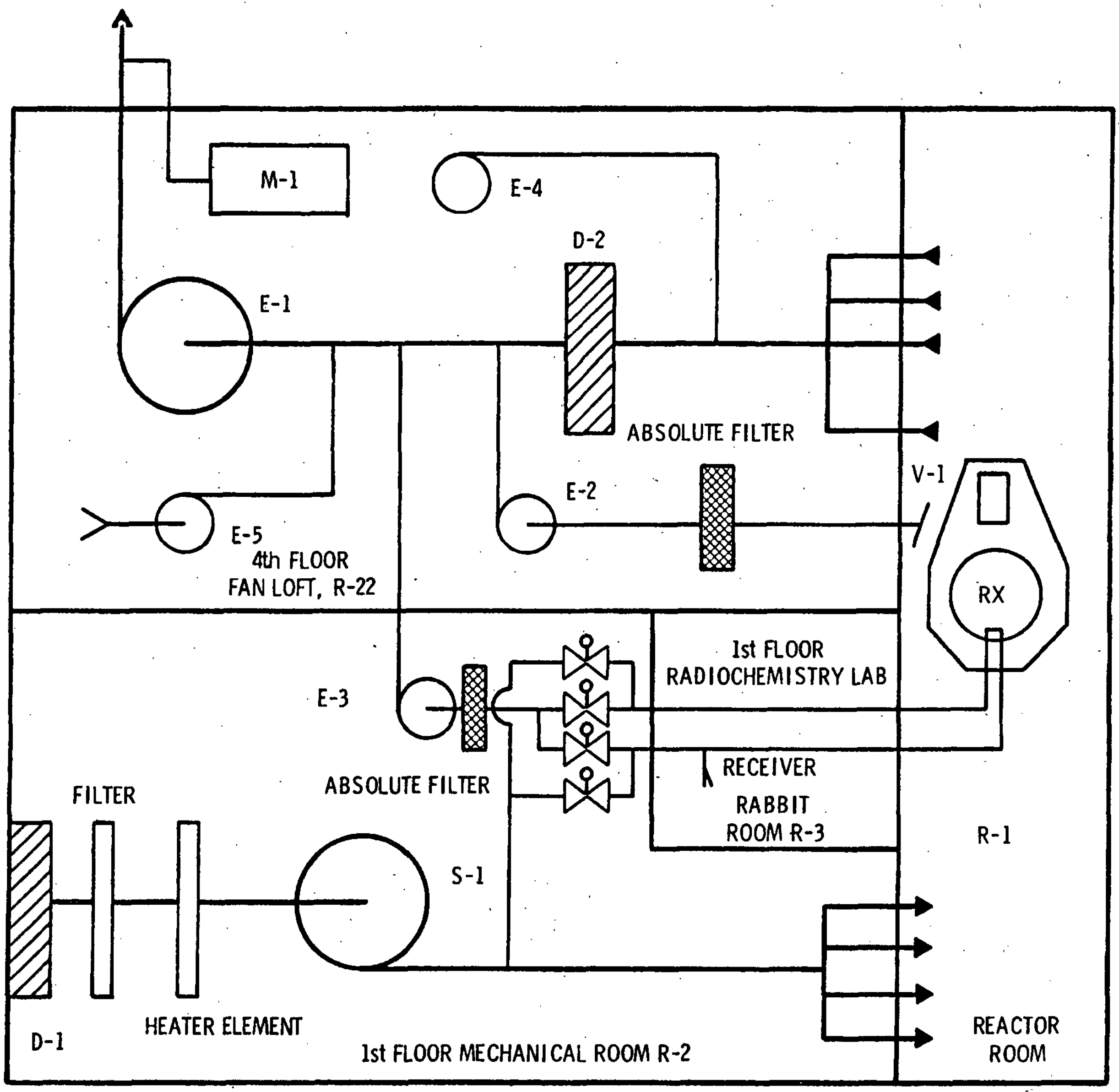
The reactor room effluent passes through a static pressure regulating damper before being exhausted to the main stack. When necessary, positive sealing dampers, located in the ductwork, automatically close and effectively isolate the reactor room.

B.4.1.2 Reactor Room Ventilation Pneumatic and Control System

A schematic representation of the ventilation pneumatic system is shown in Figure B.4-2. The exhaust line from pneumatic system blower discharges into the reactor room ventilation exhaust.

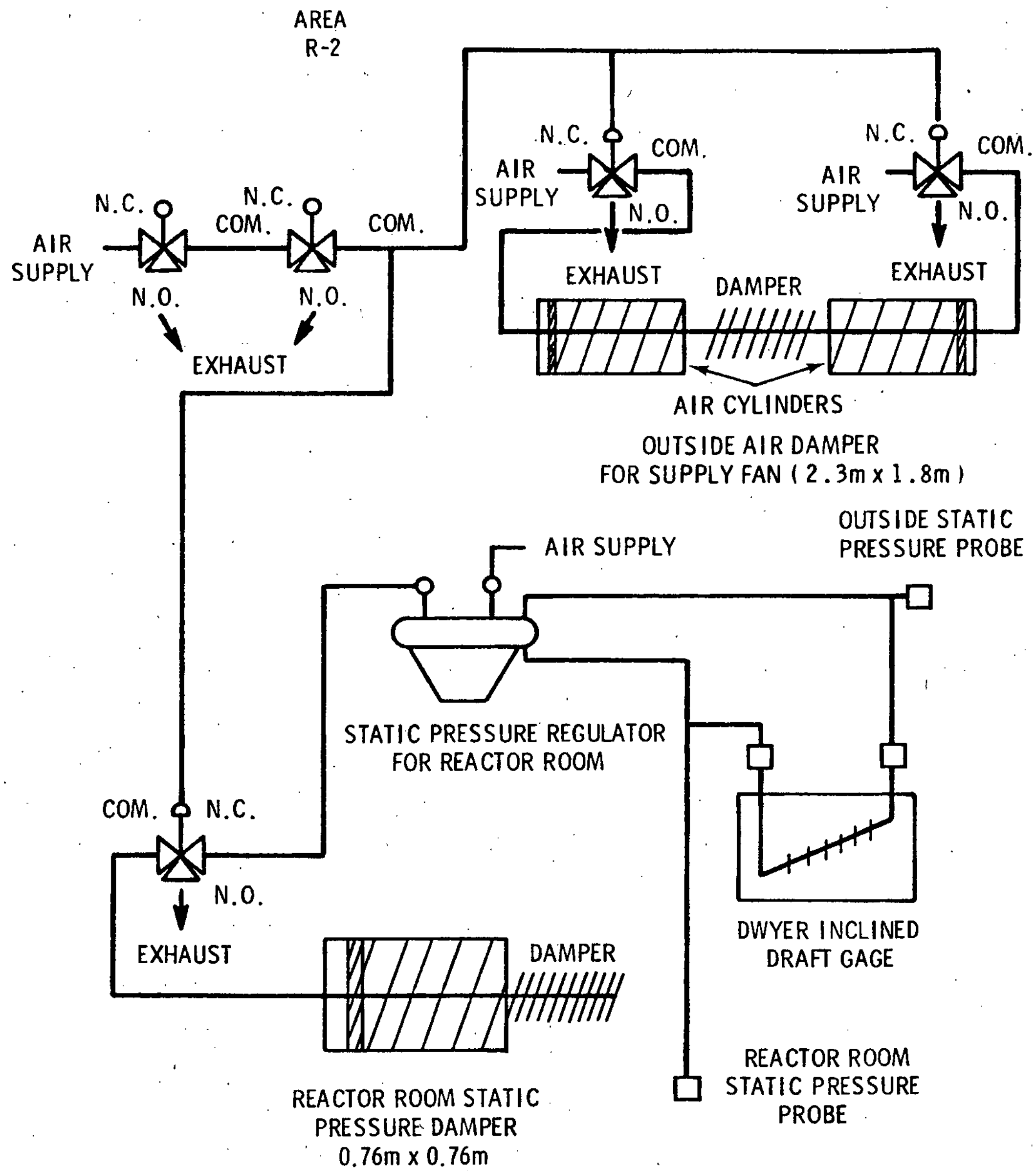
B.4.2 Support Area

Fresh air is supplied to the laboratories, offices, and corridors in the Reactor Building and the original Radiation Center Building and Annex from the main heating and cooling system in the mechanical room of the Radiation Center Building. A single duct provides conditioned air to the control room and to all offices and hallways. Air to the laboratories is supplied from the hallways. Laboratory exhaust will be withdrawn from the individual laboratories through fume hoods and discharged above the roof of the building through individual exhaust fans. A common exhaust fan is provided for offices, conference rooms and control rooms.



KEY			
S-1 -	SUPPLY FAN	E-5 -	CONTAMINATED WATER DRAIN VENT FAN
E-1 -	EXHAUST FAN	D-1 -	SUPPLY DAMPER
E-2 -	ARGON EXHAUST FAN	D-2 -	STATIC PRESSURE CONTROL DAMPER
E-3 -	PNEUMATIC SYSTEM EXHAUST	M-1 -	STACK GAS PARTICULATE RADIATION MONITOR
E-4 -	STACK GAS AND PARTICULATE MONITOR BLOWER	V-1 -	ARGON 41 MANIFOLD

FIGURE B.4-1. Reactor Room Ventilation System Schematic Diagram



NOTE: ALL COMPONENTS SHOWN
TO BE DE-ENERGIZED

FIGURE B.4-2. Schematic Diagram of the Ventilation Pneumatic System in the Reactor Room

B.5 RADWASTE SYSTEMS

This section describes the techniques and equipment utilized in the reference research reactor for handling the gaseous, liquid, and solid wastes generated during normal operations. Because of the relatively small reactor and scale of operations, the radwaste systems are basic in character.

B.5.1 Gaseous Radwaste System

The gaseous radwaste system is designed to provide a controlled route for radioactive gases from the production site to the point of mass dilution before release to the enviornment through the plant exhaust gas stack.

Only two radioactive gases are normally produced by the reactor operation (non-volatile tritium production excluded). These are nitrogen-16 and argon-41. The short half-life of the nitrogen isotope renders it unimportant. Essentially all of the ^{41}Ar production comes from the experimental facilities. A 76-mm-diameter vent line is provided from the reactor beam tubes, rotating rack, and thermal column to the reactor room ventilation exhaust. A separate fan (2.8 m^3/min) and absolute filter are incorporated into the vent line before the exhaust enters the reactor room ventilation system. The contaminated drains from the Reactor Building laboratories are vented by an exhaust fan. The drain exhaust fan discharges into the reactor room main exhaust plenum. (See Figure B.4-1 for gas flow schematics.)

B.5.2 Liquid Radwaste System

Liquid contaminated laboratory wastes are discharged to an 11-m^3 underground retention tank under the Pump House. The piping is arranged so that a pump can discharge and dilute the waste water to the sanitary sewer system or to other suitable storage containers (see Figure B.3-11).

B.5.3 Solid Radwaste System

Solid radioactive waste products are stored in metal drums. The drums are approximately 0.75 m high by 0.5 m in diameter and have a capacity of 0.15 m^3. A waste disposal room, described in Section B.3.5, is centrally located adjacent to the north wall of the Radiation Center Building (see Figure B.1-1).

Facilities are available to convert liquid radwaste to a solid form for shipment to a repository.

B.6 ACRONYMS AND ABBREVIATIONS

Table B.6-1 lists the acronyms and abbreviations that are used in Appendix B.

TABLE B.6-1. Acronyms and Abbreviations in Appendix B

ASA	American Standards Association	min	minute
ASME	American Standard Mechanical Engineer	Mg	Mega Gram
BLDG	Building	mm	millimeter
BWG	Birmingham Wire Gage	mR	milliroentgen
COM	communications	MWt	Mega Watt Thermal
DECON	Decontamination and Dismantlement	N	North
DIA	Diameter	N.C.	Normally Closed
DWN	Down	N.O.	Normally Open
EXH	Exhaust	OD	Outside Diameter
g	grams	OP	Operating Pressure
GA	gauge	OSU	Oregon State University
hp	Horse Power	RM	Room
HP	High Pressure	rpm	Revolutions per Minute
hr	Hour	RX	Reactor
HX	Heat Exchanger	SAR	Safety Analysis Report
ID	Inside Diameter	sec	Second
kg	kilogram	SS	Stainless Steel
kW	kilowatt	Temp	Temporary
ℓ	liter	TP	Total Pressure
LAB	Laboratory	TYP	Typically
m	meter	V	Volt
MECH	Mechanical		

REFERENCES

1. J. C. Ringle, T. V. Anderson and A. G. Johnson, Safety Analysis Report for the Oregon State University TRIGA Research Reactor, Corvallis, Oregon, August 1968.

2. Hazards Analysis for the Oregon State University TRIGA Mark II Nuclear Reactor, Corvallis, Oregon, August 1965.

3. Technical Specifications and Bases for the Oregon State University TRIGA Reactor, Appendix A to Facility License No. R-106, Docket No. 50-253, Amendment No. 3, July 21, 1976, and Amendment No. 4, December 18, 1979.

4. T. V. Anderson, Oregon State TRIGA Reactor Training Manual, Vol. 1 and 2, Corvallis, Oregon, October-November 1973.

5. T. V. Anderson, et al., Oregon State University TRIGA Reactor Annual Report, to the U.S. Nuclear Regulatory Commission, Corvallis, Oregon, August 1980.

APPENDIX C

REFERENCE TEST REACTOR FACILITY DESCRIPTION

The major features of the reference test reactor are presented in Section 8 of Volume 1. The portions of the facility that are important to decommissioning are described in more detail in this appendix. The detailed site description for the reference test reactor is presented in Appendix A.

The National Aeronautics and Space Administration's (NASA) Plum Brook Reactor Facility (PBRF), at Sandusky, Ohio, is the reference test reactor facility for this study. A test reactor and a research reactor are colocated at the PBRF site and are an integral part of the PBRF. The test reactor, the Plum Brook Reactor (PBR), is a 60-MWt materials test reactor, light-water-moderated and cooled, used to test materials for space flight applications. The research reactor, the Plum Brook Mock-Up Reactor (MUR), is a low-power (100-kWt) swimming pool-type research reactor, used as an experimental tool to assist in the operation of the PBR. Both reactors at the PBRF have been shut down since January 1973. Both reactors, however, are conceptually decommissioned in this study as if they had been recently shut down.

Much of the detailed facility information contained in this appendix is extracted from the PBRF Environmental Reports,[1,2] from the PBRF Dismantling Plans,[3,4] and from the PBRF Final Hazards Summary.[5] Other information has been obtained through consultations with the staff at the reference facility.

C.1 GENERAL FACILITY DESCRIPTION

The reference test reactor site, which occupies a surface area of approximately 4.7 km^2 near a large body of water, is described in detail in Appendix A. The reference test reactor exclusion area itself requires an area of 0.12 km^2 within the larger plant site. A well-labeled perimeter fence exists around the total site to exclude the public. Another fence surrounds the smaller reactor exclusion area and includes a security entrance to the plant.

The reference test reactor exclusion area layout is shown in Figure C.1-1, including identification of major structures and areas anticipated to require decontamination activities.

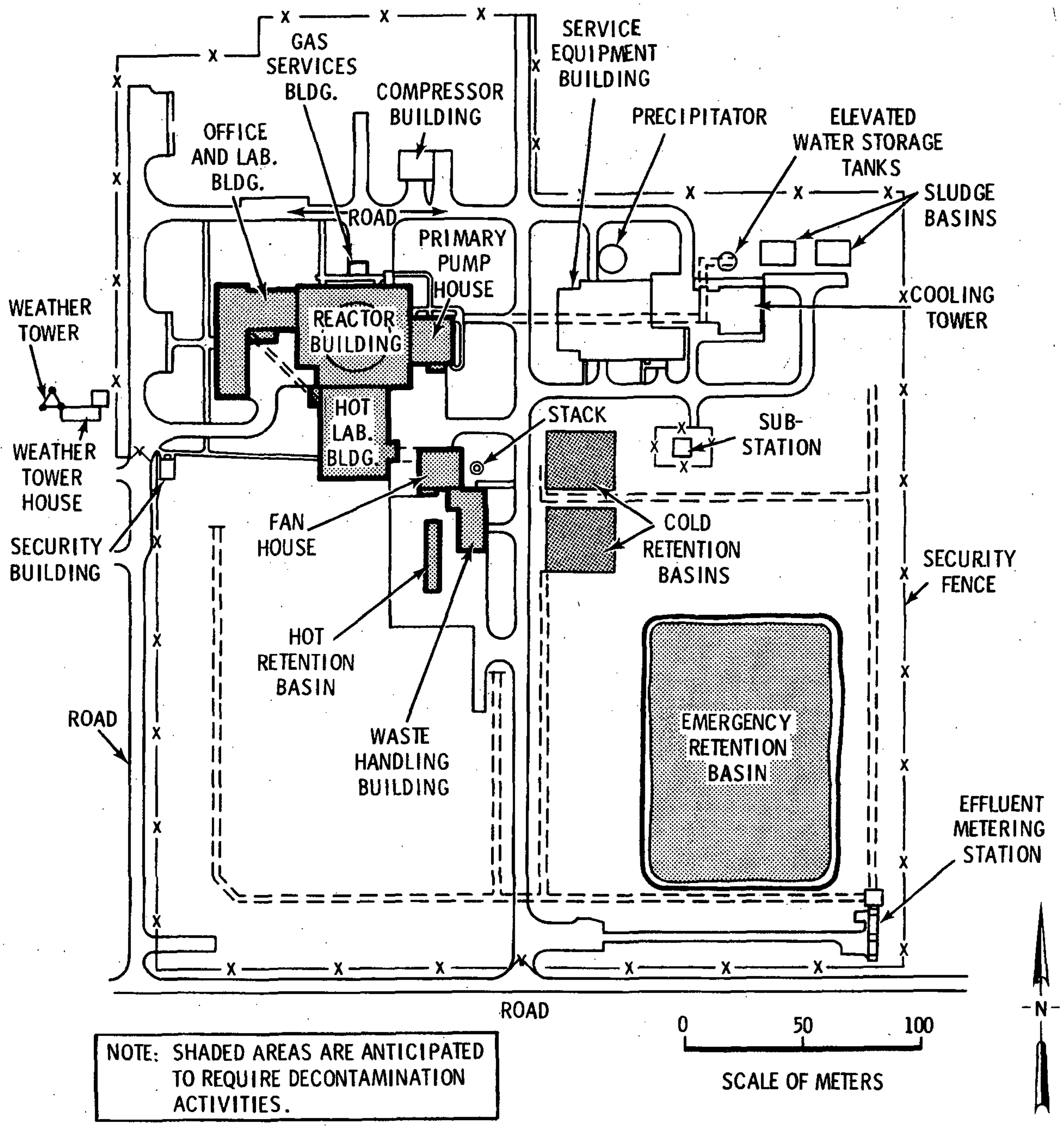


FIGURE C.1-1. Site Layout of the Reference Test Reactor Facilities

Major structures in the exclusion area include: The Reactor Building, which houses the reactors, reactor control room and Containment Vessel; the Hot Laboratory Building with seven hot cells; the Primary Pump House; the Office and Laboratory Building; the Fan House; the Hot Retention Area; and the Waste Handling Building.

C.1.1 Design Criteria

The bases for the design criteria used in this materials test reactor facility is to first, provide protection of public safety; second, provide for reliable and economic plant performance; and third, provide an attractive appearance.

The essential systems and components of the plant are designed to enable the facility, without loss of the capability to protect the public, to withstand the additional forces that might be imposed by natural phenomena. The plant is designed to withstand the most severe of the natural phenomena recorded for the vicinity of the site, with margin to account for uncertainties in the historical data.

C.1.2 Test Reactor Operating Characteristics

The design and licensed power level for the reference materials test reactor core is 60 MWt. A functional diagram of the reference test reactor is shown in Figure C.1-2. A cutaway view of the test reactor tank assembly is illustrated in Figure C.1-3.

The nuclear test system includes a forced-circulation materials test reactor that is light-water-cooled and moderated with a primary beryllium reflector and a secondary water reflector. It is designed to produce a high neutron flux for the irradiation of fueled and unfueled experiments for space program applications. Since the core is very compact, 10 tubes exist to convey neutrons outside the core for more convenient use by several experiments at a time, if required. Experiments are inserted by means of two horizontal through tubes, six horizontal beam tubes and two vertical experiment tubes, all of which are of aluminum alloy construction. The upper elliptical top of the reactor vessel is flanged so that it can be removed. In addition, a hatch is provided in the

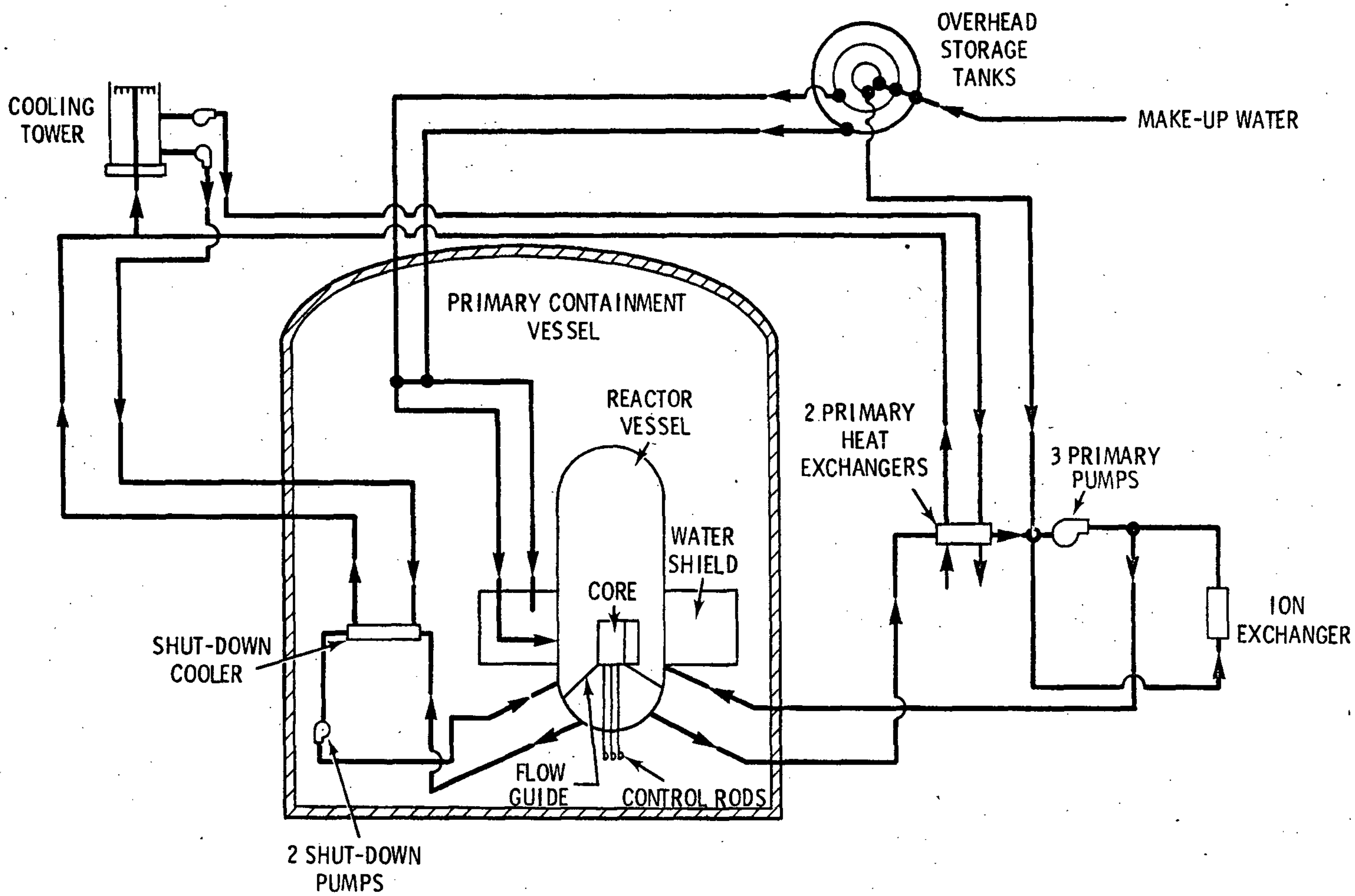


FIGURE C.1-2. Functional Diagram of the Reference Test Reactor

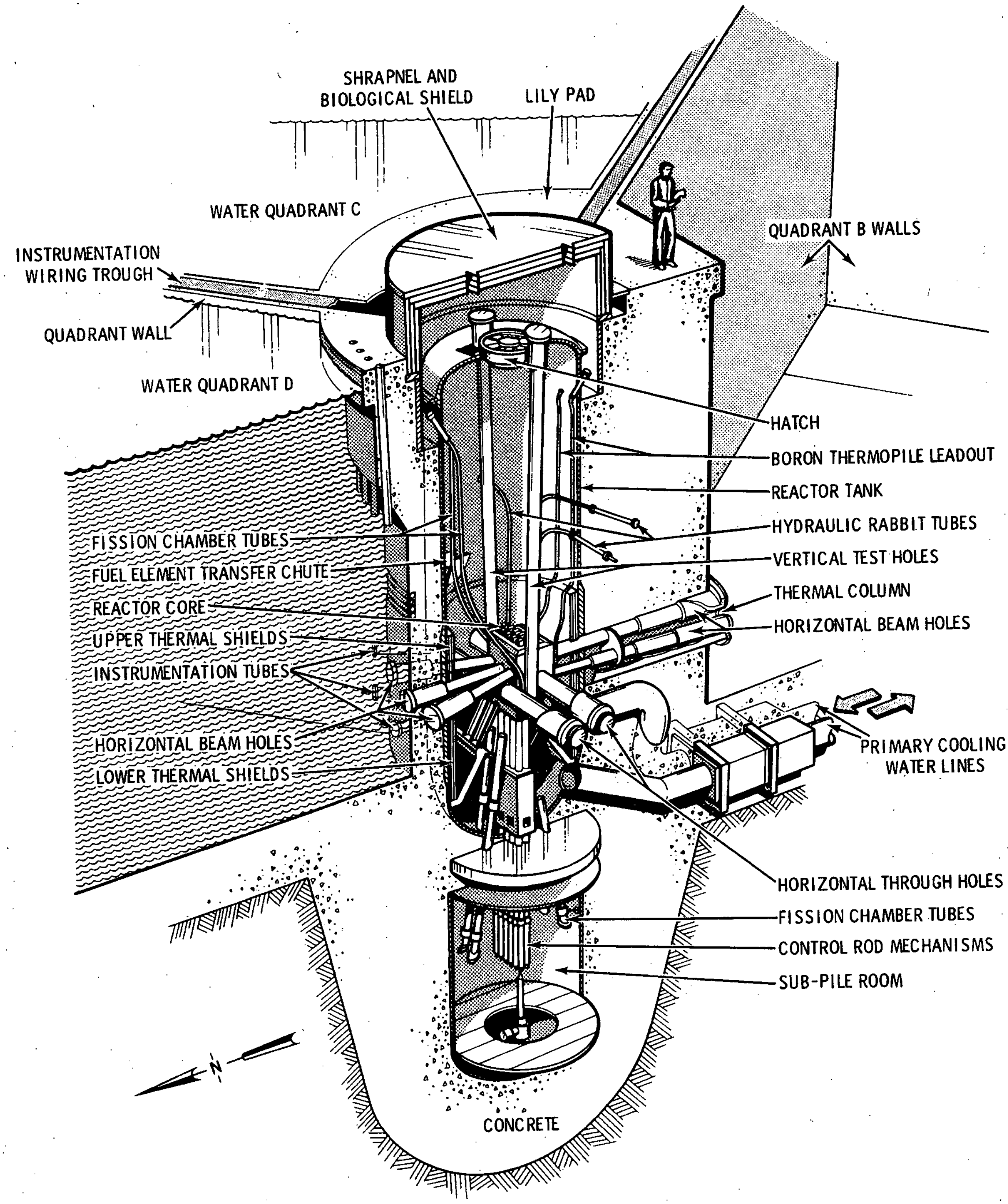


FIGURE C.1-3. Cutaway View, Reference Test Reactor Tank Assembly

top to facilitate changing fuel elements and inserting or withdrawing experiments. Various hydraulic rabbit and instrument thimble assemblies are also present in-tank.

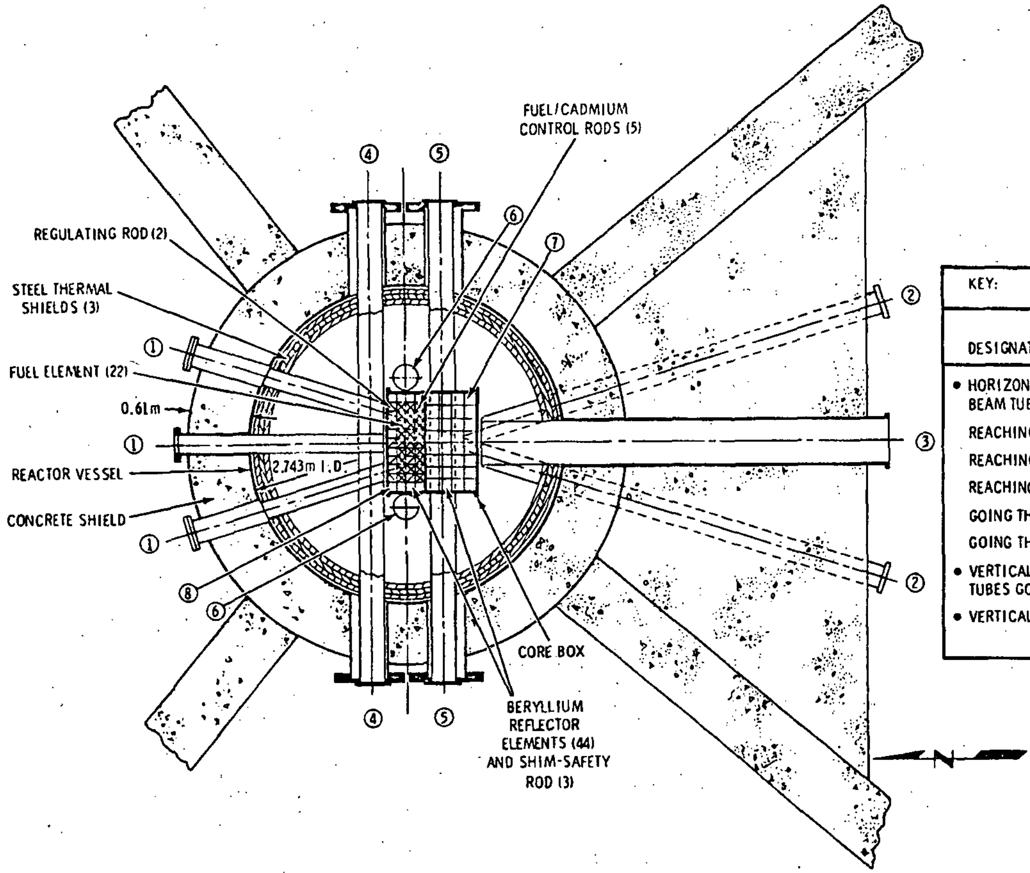
The test reactor core is composed of 22 uranium/aluminum alloy fuel elements clad with aluminum alloy and arranged in a 3 by 9 lattice. Five fueled/cadmium control rods are located in the center row of the lattice. Forty-four beryllium reflector pieces surround the fuel eccentrically along with two stainless steel cadmium/beryllium regulating rods and three shim safety rods (cadmium/beryllium). This 810-mm by 860-mm array is housed in a "core box" with three aluminum alloy side plates, one beryllium side plate and aluminum alloy top and bottom grids. A Lockalloy (beryllium-aluminum alloy) flow divider plate is also part of the core box. The test reactor core rests on a stainless steel support structure within a stainless steel-clad reactor tank approximately 2.7 m in diameter by 9.4 m high. Three concentric stainless steel thermal shields protect the reactor tank wall in the "near core" region and two concentric thermal shields are located below the core region. A horizontal section view at reactor level is shown in Figure C.1-4. The reactor core box assembly is illustrated in Figure C.1-5.

The reactor is controlled by neutron absorbing rods which enter from beneath the core. Heat removal from the core occurs when coolant water entering from the primary coolant inlet flows up the outside of the core box from above the vessel flow guide, down through the core fuel elements, and out the primary coolant exit below the flow guide. Primary coolant system heat is then transferred to the secondary coolant system by heat exchanger. The secondary system heat load is discharged to the atmosphere via a cooling tower.

C.1.2.1 <u>Facility Energy Supply</u>

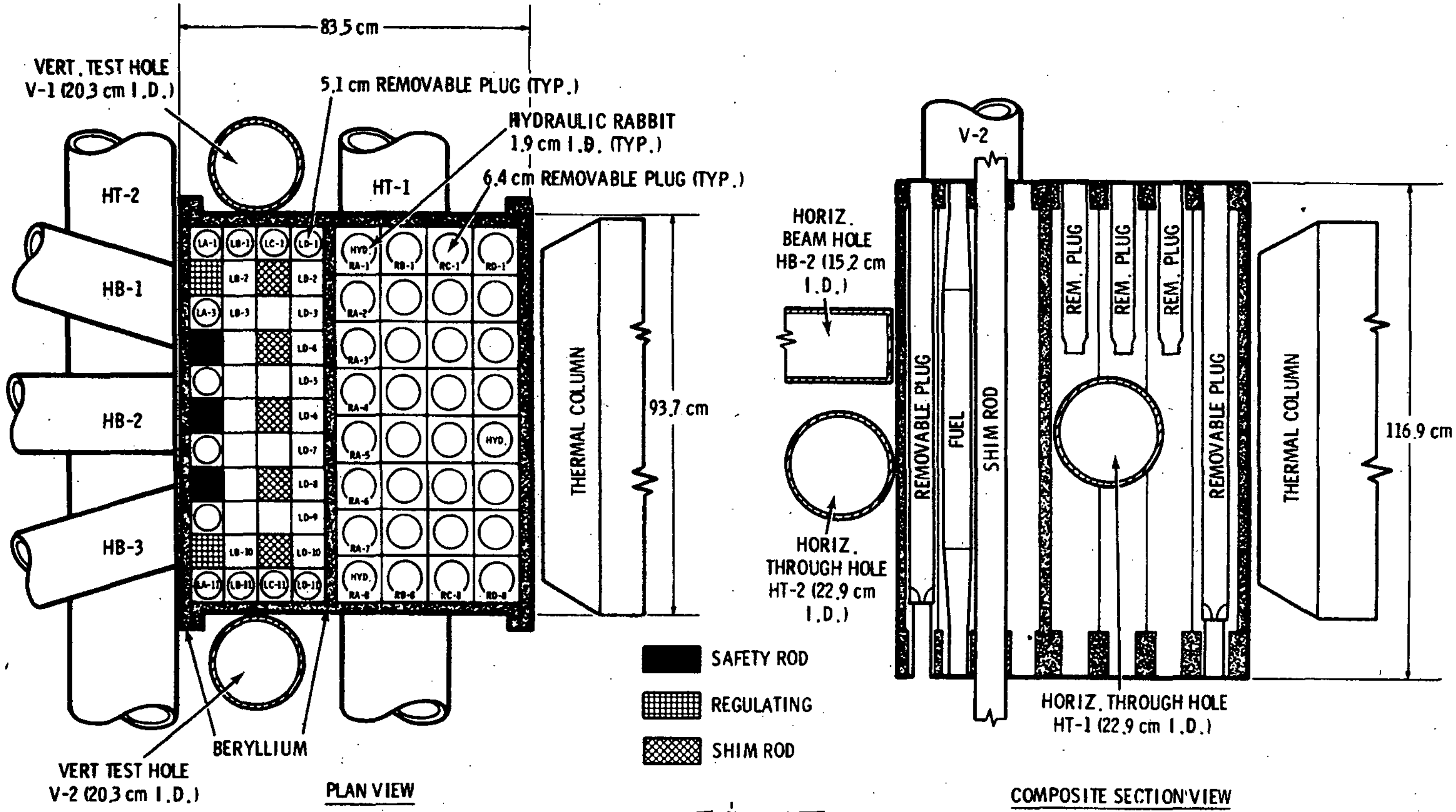
Both electrical and natural gas service are supplied to the site. A substation is located just south of the service equipment building and east of the cold retention basins. Natural gas is provided to each building as required.

Two diesel generators are provided as standby sources of emergency power at 41.6 kV each in the event of a loss of normal and preferred AC power. Each generator has sufficient capacity to operate the equipment necessary to prevent



DESIGNATION	NO.	POSITION*	INSIDE DIAMETER DIMENSIONS, cm
• HORIZONTAL NEUTRON BEAM TUBES:			
REACHING CORE	3	①	15.24
REACHING REFLECTOR	2	②	15.24
REACHING REFLECTOR	1	③	38.10
GOING THROUGH	1	④	30.48
GOING THROUGH	1	⑤	22.86
• VERTICAL NEUTRON BEAM TUBES GOING THROUGH	2	⑥	20.32
• VERTICAL HOLES IN REFLECTOR	32	⑦	6.35
	12	⑧	5.08

FIGURE C.1-4. Reference Test Reactor Core--Horizontal Section



KEY:

IRRADIATION FACILITY	ACCESSIBLE FROM	IRRADIATION FACILITY	ACCESSIBLE FROM
HORIZONTAL THROUGH HOLE, HT-1 HORIZONTAL THROUGH HOLE, HT-2	QUADRANT[(a)]	HYDRAULIC RABBIT TUBES	OUTSIDE OF QUADRANT WALL
HORIZONTAL BEAM HOLE, HB-1 HORIZONTAL BEAM HOLE, HB-2 HORIZONTAL BEAM HOLE, HB-3		REFLECTOR TEST HOLES "L" PIECES, (12) "RA", "RB" AND "RC" PIECES, (22) "RD" PIECES, (6)	TOP OF REACTOR[(a)]
VERTICAL TEST HOLES, V-1, V-2	TOP OF REACTOR[(a)]	THERMAL COLUMN	QUADRANT

[(a)] ACCESSIBLE ONLY DURING REACTOR SHUTDOWN PERIODS

FIGURE C.1-5. Core Box Assembly

undue risk to the public in an emergency situation. In addition, storage batteries provide guaranteed AC power through an inverter in the event of a loss of guaranteed power.

C.1.2.2 Fuel Handling System

The reference test facility is designed to permit the handling of spent fuel under water from the time the fuel leaves the reactor vessel until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat from the spent fuel. This system also has the capability of receiving, handling, and storing new fuel. The system of canals that permits the transfer of irradiated material and spent fuel elements from within the reactor containment vessel to various storage canals in the Reactor Building and in the Hot Laboratory is shown in Figure C.1-6. The canal system is described in greater detail in Section C.2.1.2.

Unloading spent fuel elements from the reactor vessel is accomplished through a stainless steel discharge chute, an isolation valve, and an unloading piston unit (see Figure C.1-7). The spent fuel elements are transferred under water from the Quadrant C area to a set of stationary underwater storage racks located in Canal G, using an underwater cart. The storage racks and underwater cart are cadmium-lined and are designed to allow adequate free-convection cooling of the fuel elements. When the afterheat has reduced to the point where it can be handled by convective cooling within the shipping casks, the spent fuel elements are transferred underwater and loaded into the shipping casks in Canal G.

C.1.2.3 Radioactive Waste Treatment Systems

The radioactive waste treatment systems collect, process, monitor, and discharge radioactive liquid, gaseous wastes, and solid wastes produced during reactor operation.

Liquid wastes potentially containing radioactive material are collected, filtered, and concentrated or demineralized, as required. The decontaminated water from the demineralizers or the concentrator distillate may be recycled

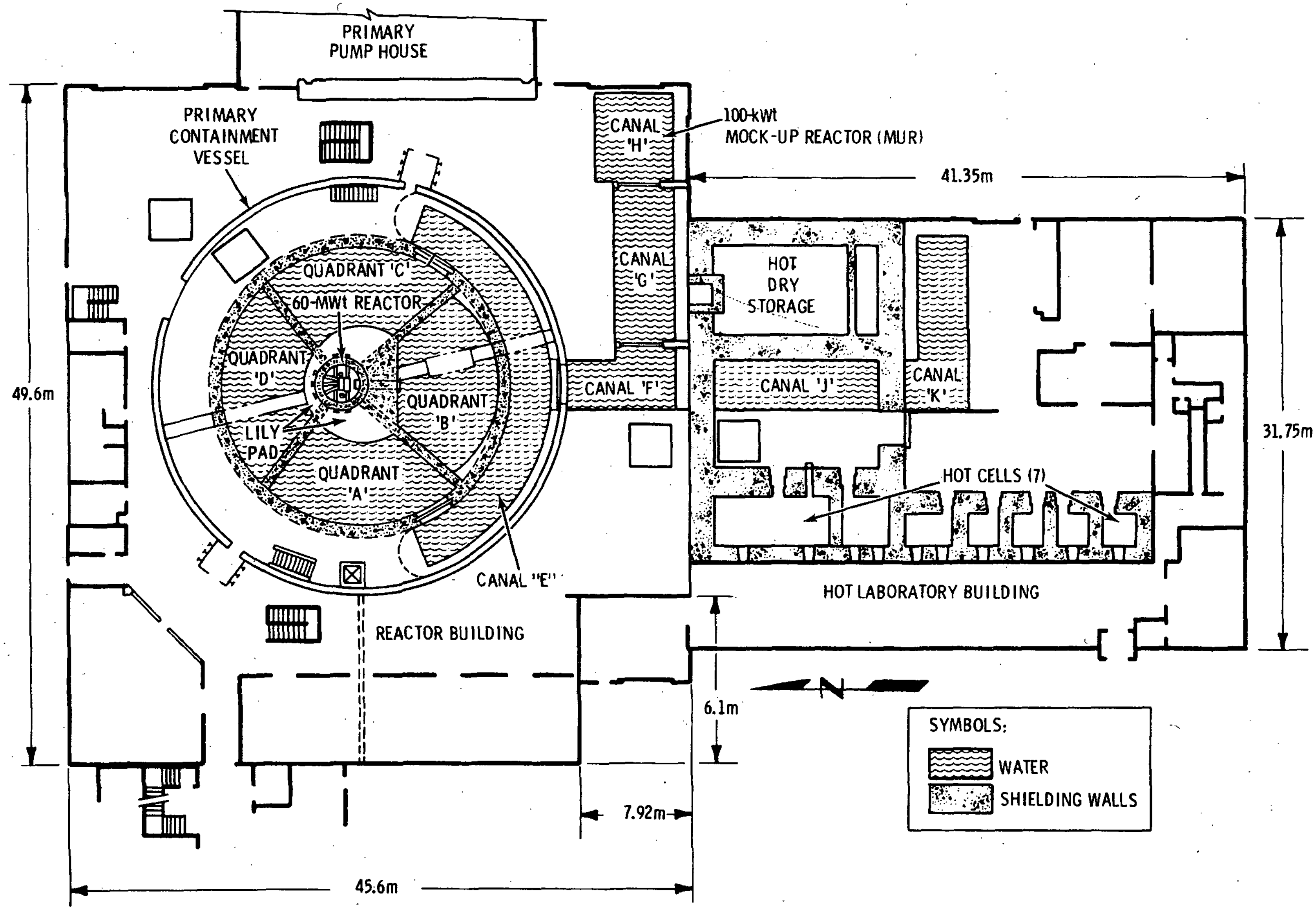


FIGURE C.1-6 . Reactor Building and Hot Laboratory Building Canal System--Plan View

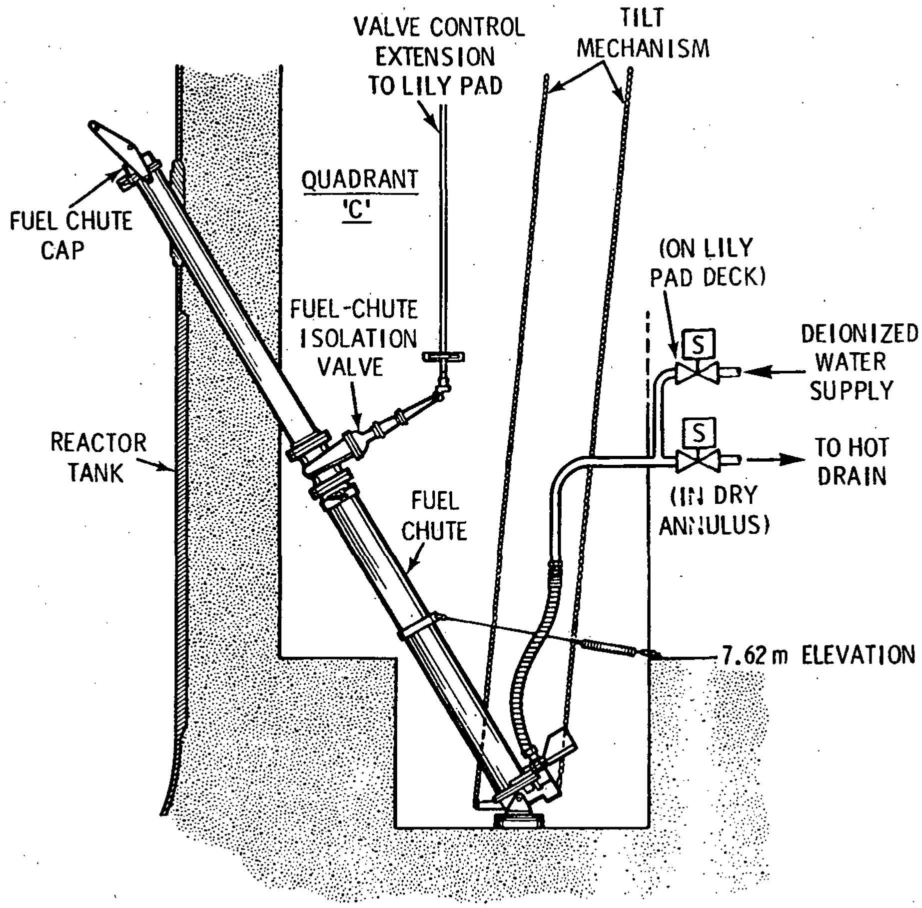


FIGURE C.1-7. Fuel Chute

for use in the plant or may be discharged to the nearby river. The concentrator bottoms, spent demineralizer resins, and spent filters are solidified, packaged, and shipped to a licensed disposal site. Gaseous wastes are collected and discharged to the environment after filtration.

C.1.3 Mock-Up Reactor (MUR) Operating Characteristics

The Mock-Up Reactor (MUR) is located in Canal H, inside the Reactor Building but outside of the Primary Containment Vessel (see Figure C.1-6). The MUR is used as an experimental tool to assist in the operation of the reference materials test reactor. The purpose of the MUR is to supply information necessary to properly design the experiments and the core loadings proposed for the reference test reactor and to evaluate the effects of irradiation on

experiments subjected to the high radiation fluxes of the test reactor. It is a realistic physical and neutronic mock-up of the test reactor core, including the major beam tubes. A vertical section view of the MUR is shown in Figure C.1-8.

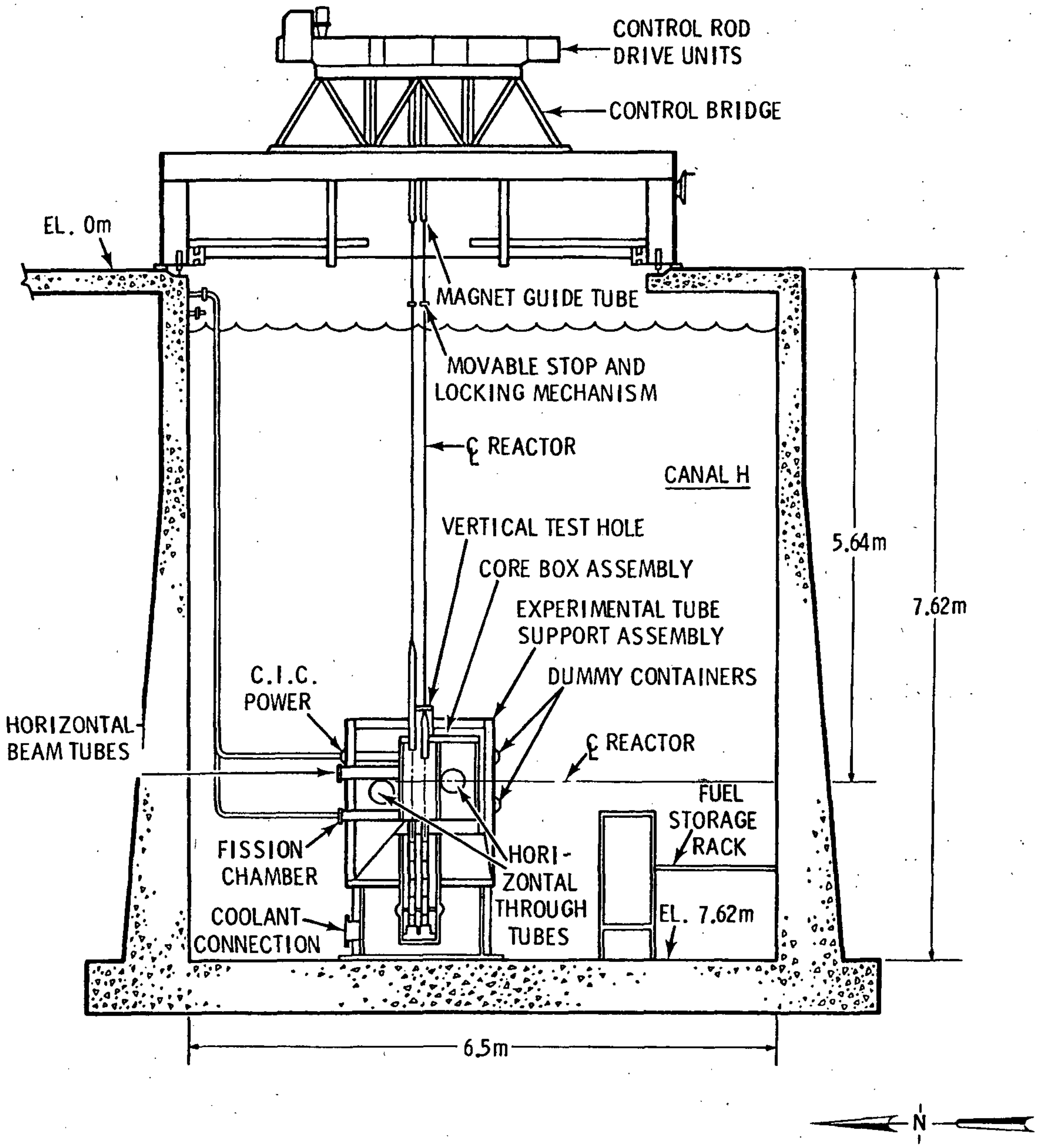


FIGURE C.1-8. MUR Facility--Vertical Section View

C.1.4 Hot Laboratory Building

The Hot Laboratory Building is attached to the south wall of the Reactor Building, and contains a hot dry storage pit, seven hot cells, and two storage canals with massive concrete shielding. Transfers of radioactive materials occur from the Reactor Building to the Hot Laboratory Building by means of a canal (see Figure C.1-6).

C.2 STRUCTURES

This section describes the structures of the reference test reactor. Refer to Figure C.1-1 for the arrangement of the structures on the plant site. The structural materials used in the various plant structures are listed in Table C.2-1.

TABLE C.2-1. Estimated Quantities of Structural Material in the Reference Test Reactor Facility

Structure	Concrete (m^3)	Rebar (Mg)	Structural Steel (Mg)
Reactor Building[a]	2554	353	542
Hot Laboratory Building	3395	235	45
Primary Pump House	1193	83	54
Fan House	272	19	16
Waste Handling Building	191	13	20
Service Equipment Building	482	33	94
Office & Laboratory Building	382	26	110
Hot Retention Area	325	22.5	254
Cold Retention Basins	365	12.6	41
Cooling Tower	153	5	9
Water Tower	76	5	108
Utility Tunnels	347	48	93
Auxiliary Structures	57	4	12
Totals	9792	859.1	1398

(a) Includes the Reactor Vessel and the Mock-up Reactor.

C.2.1 Reactor Building

The Reactor Building (Figures C.2-1 through C.2-4) is a flat-roofed, metal-frame building 45.6 m by 49.6 m which completely surrounds the primary containment vessel (CV) up to an elevation of 8.2 m. The primary structural unit of the Reactor Building is the CV itself, which is described in detail in Subsection C.2.1.1. The first floor elevation is the zero elevation reference point for all elevations given in Figure C.2-1 and in all subsequent figures. Grade level is -0.3 m. The CV continues below-grade and under the concrete pedestal supporting the test reactor and the quadrant pools.

The center of the reactor tank is displaced from the center of the CV by approximately 1.4 m. The quadrant walls radiate from the center of the reactor and are of different lengths. The arrangement of shield concrete and the quadrant walls is shown in Figure C.2-3. The thickness of the reactor concrete shield is approximately 0.7 m minimum. The concrete immediately under the reactor vessel contains a lead shield in the form of a disc to provide shielding for the sub-pile room (see Figure C.2-4). The disc is approximately 3 m in diameter and 0.3 m thick.

Outside the CV, basement foundations with footings are approximately 0.5 m thick at the side walls, with 0.25-m-thick floors. An exception to this is the east wall of the building, where the basement side wall thickness is approximately 1.5 m to provide a support for the Primary Pump House shielding wall (also 1.5 m thick). Since there is no sub-basement level on this side of the building, the thick wall ends at -4.6 m elevation and is supported on pilings. To the south, the heavy concrete shield of the Hot Laboartory Building forms a common wall shared with the Reactor Building.

The Office and Laboratory Building share a common wall of ordinary light frame-wall construction on the west side of the Reactor Building.

In addition to support systems and transfer and storage canals, the Reactor Building houses the MUR swimming pool reactor, work space for setting up experiment assemblies, restrooms, a personnel decontamination facility, a tool and small-parts decontamination facility, a change room, and a control room for remote operation of experimental rigs. The reference test reactor control

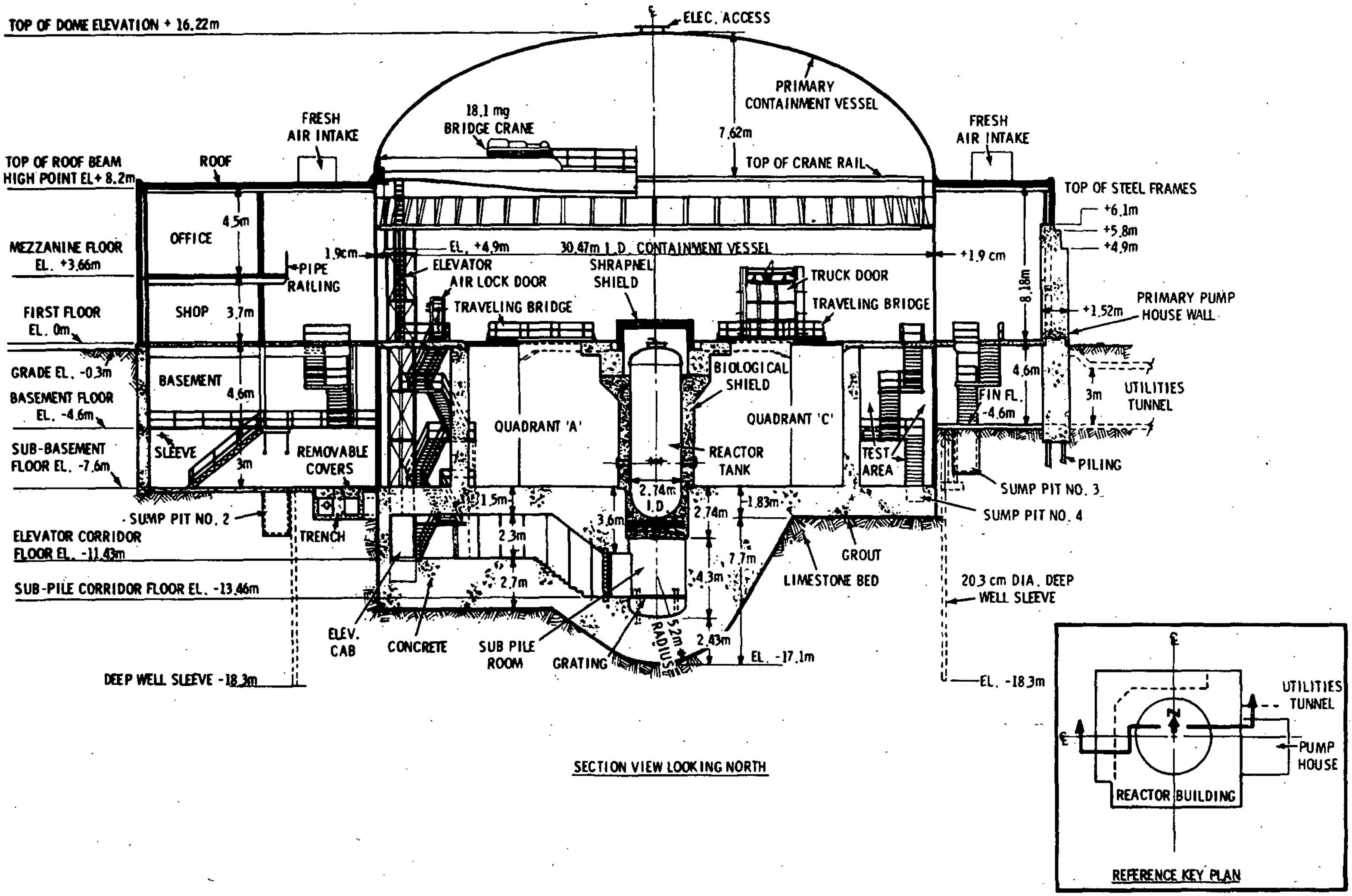


FIGURE C.2-1. Reactor Building Section--Sheet 1

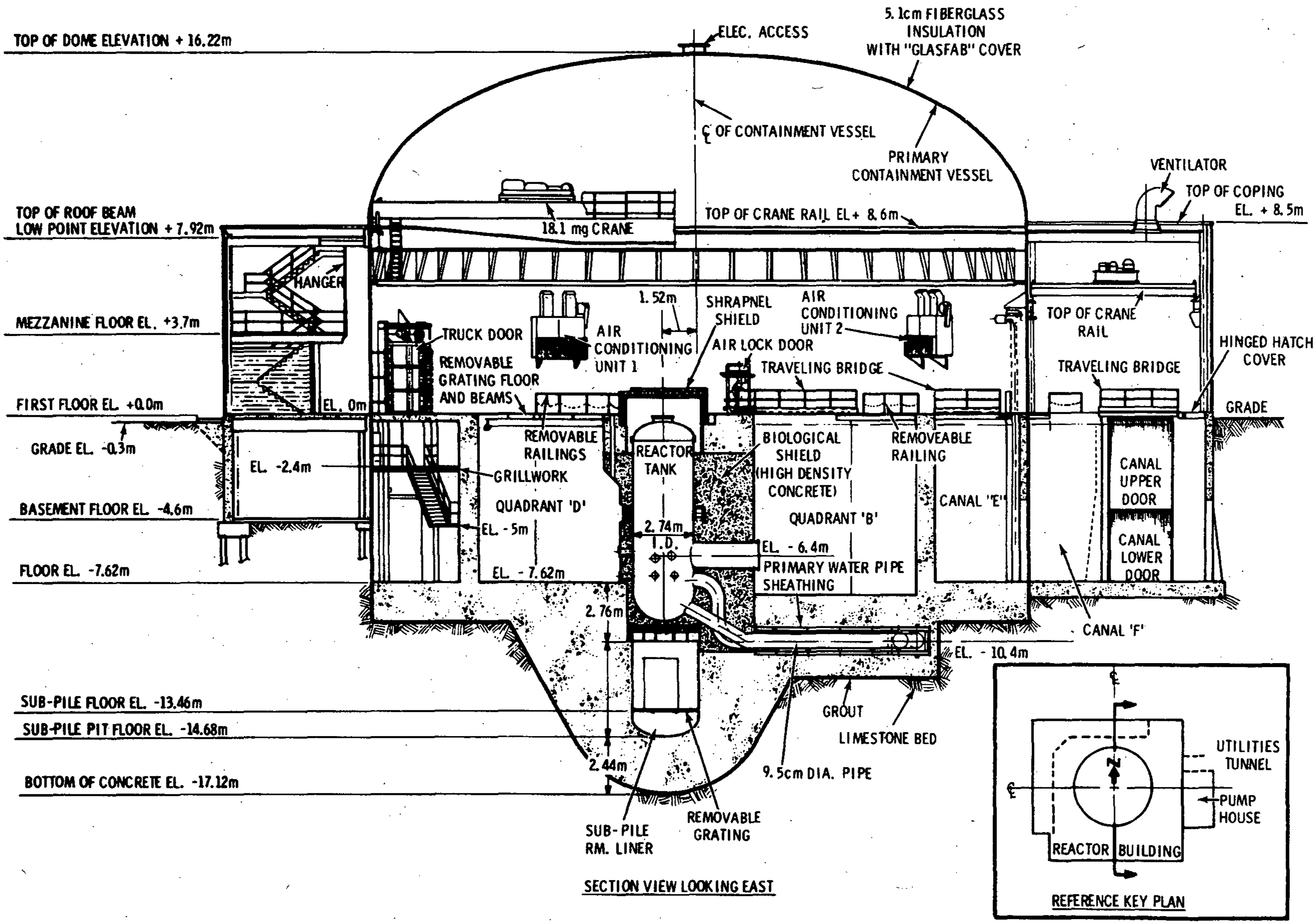


FIGURE C.2-2. Reactor Building Section--Sheet 2

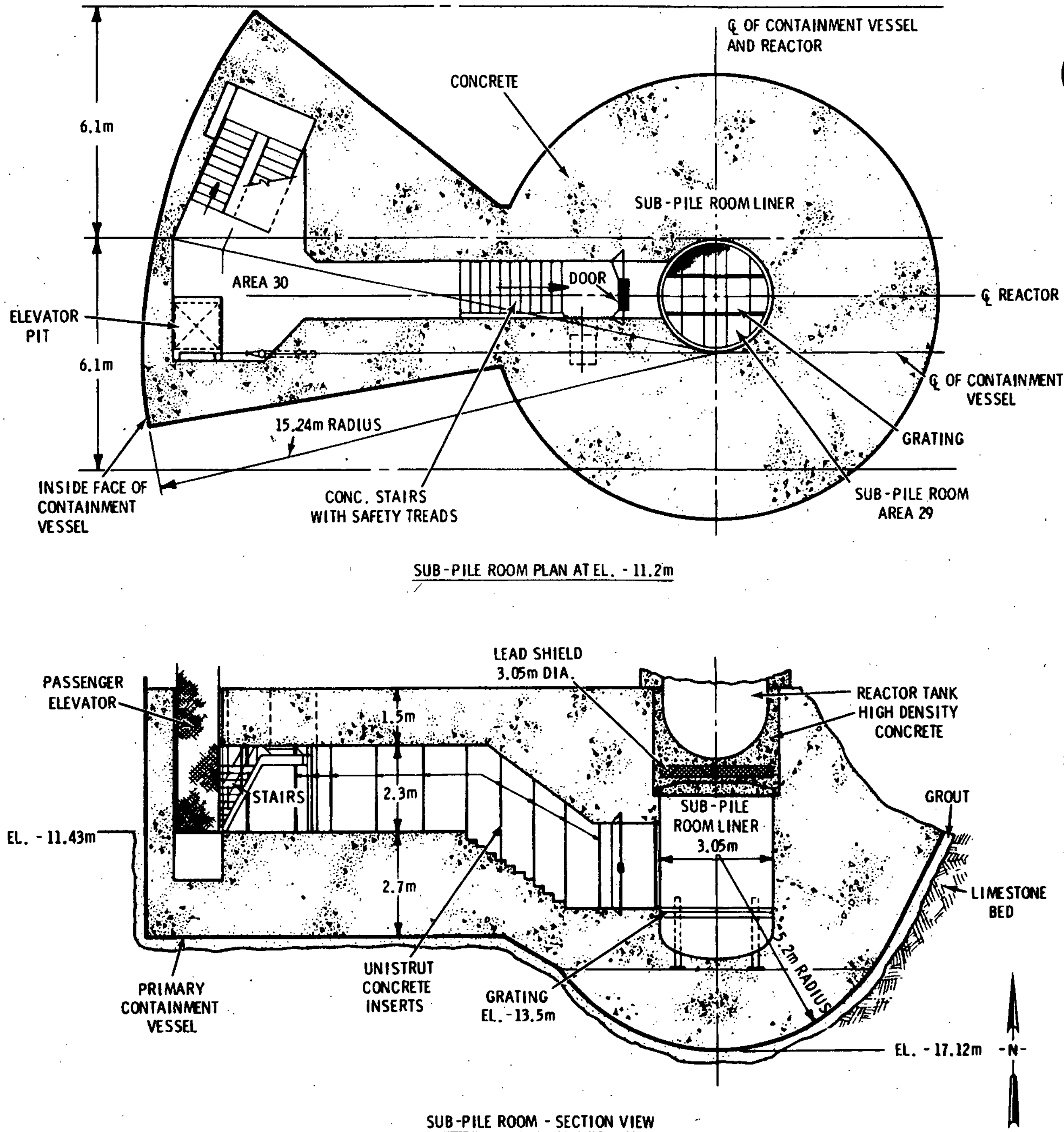


FIGURE C.2-4. Reactor Building Sub-Pile Room

room and offices are on a mezzanine extending along the north and west walls (see Figure C.2-3). Basement areas are accessible by stairways and provide access to the Reactor Office and Laboratory Building, the Hot Laboratory Building, and the Service Equipment Building by means of tunnels.

C.2.1.1 Primary Containment Vessel

The primary containment vessel (CV) houses the reactor vessel and is located within the Reactor Building, as shown in Figures C.2-1 and C.2-2. The CV consists of a cylindrical steel tank having an elliptical dome. The bottom dish is of an irregular shape and is pressure grouted for full support. The structure has a diameter of 30.48 m, a height above grade of 16.76 m, and extends 17.1 m below grade. The wall and dome structure are 190-mm plate, while the bottom dish is welded 9.5-mm plate. The elliptical dome is insulated by 51 mm of urethane foam to prevent heat loss and minimize condensation inside the CV. The free volume within the CV is 14,160 m^3. The design over pressure capability of the CV is 34.5 kPa at 0°C.

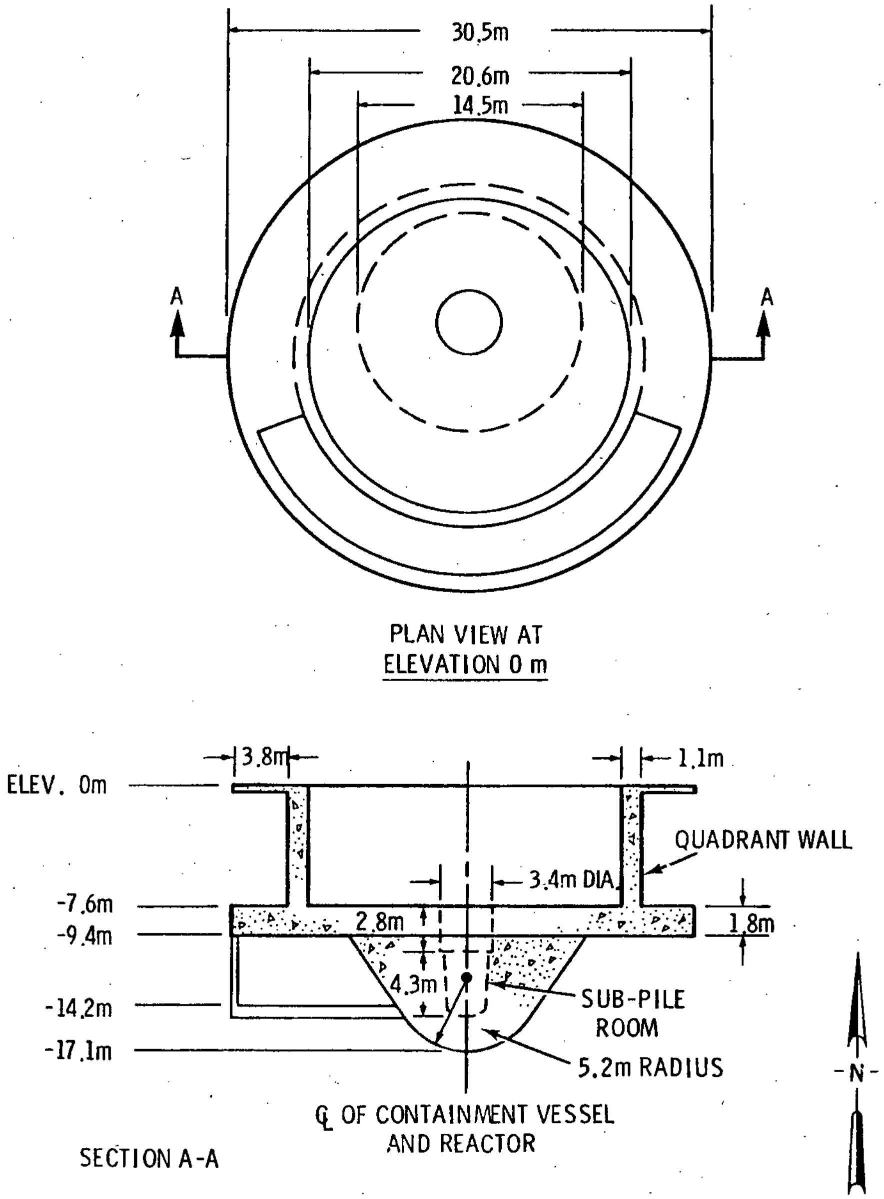
There are approximately 80 electrical penetrations and approximately 110 piping penetrations. All piping penetrations are seal-welded except for approximately 30 bulkhead fittings. The electrical penetrations are potted at two points. There are two sets of double air-lock personnel doors, one set of double air-lock experiment transfer doors, one truck door and one underwater canal door. The double air-lock doors are interlocked so that only one may be opened at a time. Pressures are maintained so that leakage is always inward during use of the air lock. The truck door is doubly gasketed. The underwater door is singly gasketed and is mechanically secured in place.

C.2.1.2 Primary Containment Vessel Internals

The major structures inside the primary containment vessel (see Figures C.2-1 through C.2-4) are described in the following subsections.

Concrete Foundation Mat. This structure is in the shape of a flat disc (see Figure C.2-5). It is 1.8 m thick and 30.5 m in diameter, with a 7.7 m downward dome in the center to house the sub-pile room. A rectangular rib extends to the west to house the staircase and access passage to the sub-pile room. An annular wall extends above the top of the disc to form the outer wall of the quadrants surrounding the biological shield.

Biological Shield and Quadrants. The reactor tank is surrounded by high-density concrete to provide part of the biological shield (see Figure C.1-3 and Figures C.2-1 through C.2-4). The remainder of the shield is provided by



SECTION A-A

FIGURE C.2-5. Primary Containment Vessel Foundation

water contained in a circular pool surrounding the reactor tank. This pool is approximately 21 m in diameter, and its vertical centerline is offset 1.5 m from that of the reactor tank. The walls of the pool are made of reinforced concrete and are 0.91 m thick. The pool is divided into four quadrants (designated A, B, C, and D) by water-tight partitions (quadrant walls). The concrete

surrounding the reactor tank has a minimum thickness of 0.61 m in quadrants A, C, and D. In quadrant B, which contains the thermal column, the concrete is thickened to 2.74 m (minimum) to provide adequate shielding without water in the quadrant. The water in quadrants A, C, and D is 7.6 m deep. In quadrant B the water is 8.2 m deep. This additional depth increases the distance from the centerline of the thermal column to the pool floor and provides a better geometry for shield testing. Each quadrant contains about 680 m^3 of water.

No important direct generation of activity occurs in the quadrant water because of the concrete biological shield interposed between the reactor tank and the water. The main source of contamination occurs through corrosion products from irradiated materials stored in the quadrant or canal areas.

All surfaces in the quadrant areas are painted with an epoxy resin for waterproofing and decontamination purposes.

A concrete platform (lily pad), cantilevered in quadrants A, C, and D, surrounds the reactor tank near the top, at approximately grade level, to provide work space for changing fuel elements, and for handling experimental equipment that is inserted from above the reactor. Three moveable bridges are provided to permit access, from the annular space around the shielding pool, to the concrete platform above the reactor tank.

There are a total of 71 penetrations in the biological shield, with nominal penetration diameters ranging from 20 mm to 1050 mm. Eight penetrations are for the control and circulation of coolant, with the remaining penetrations used for experiment instrumentation lead-ins and reactor control.

Shrapnel Shield. Above the reactor tank is a heavy steel cover (see Figure C.1-3) that serves as both a missile shield and biological shield. It consists of three pieces. each piece weighing 18.1 Mg. The shield is approximately 4.1 m in diameter, 1.2 m high, and 300 mm thick with a flat disc top cover.

Canal System. The canal system, shown in Figure C.2-6, is provided for underwater transportation of radioactive materials, equipment, experiments, and

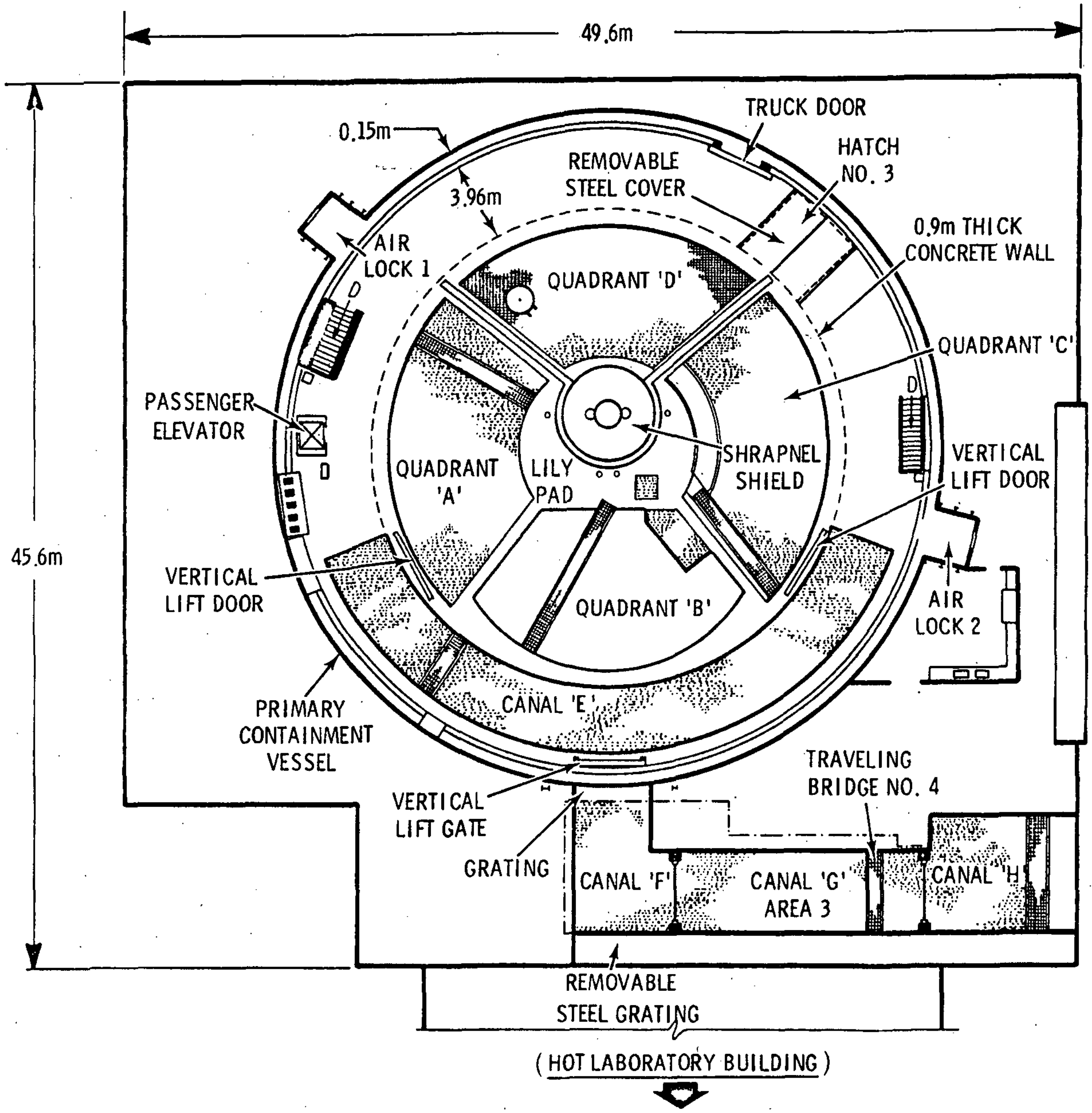


FIGURE C.2-6. Reactor Building Canal System--Plan View

fuel elements between the two reference reactors or from either of the reactors to storage, or to the Hot Laboratory Building. As shown in Figure C.2-6, the portion of the canal inside the CV extends completely around the outer wall of quadrant B, and partially around quadrants A and C. It is 3.96 m wide and 7.62 m deep. Walkways are provided on both sides. Direct access from

quadrants A and C to the canal is provided by means of underwater doors, designed to be water tight. There is a small underwater door connecting quadrants B and C. There is no access from quadrant D to canal E. Each canal contains about 830 m^3 of water.

All surfaces in the canal areas are painted with an epoxy resin for waterproofing and decontamination purposes.

One underwater canal door penetrates the CV between canal E on the inside and canal F on the outside to permit equipment and experiments to be moved underwater from within the CV to the Reactor Building and the Hot Laboratory. This door also permits that portion of canal E inside the CV to be drained independently of Canal F.

C.2.2 Hot Laboratory Building

The Hot Laboratory Building is a combination concrete and mill-type structure measuring approximately 31.2 m by 41.5 m, attached to the south wall of the Reactor Building. Transfer of irradiated materials and equipment from the Reactor Building to the Hot Laboratory is via canal. The Hot Laboratory Building, shown in Figure C.2-7, houses seven hot cells, controlled (and generally clean) work areas, an office, a manipulatory repair shop, a decontamination room, and storage and repair shop areas. Hot Laboratory Building elevation section views are shown in Figure C.2-8. A labryrinth change room and personnel decontamination area connect the cell operating area with the controlled areas behind the cells. The 12-m by 23-m hot handling room nearest the Reactor Building serves as a radiation shield for Cell 1 and Cell 2 transfer operations as required. A 3-m by 3-m, 72.5-Mg lead-filled sliding door separates the hot handling room from the controlled work area.

In general, irradiated equipment, materials and experimental loops are transferred underwater from the reactor through canals to the Hot Laboratory. At this point, the material can be transferred to the hot wet storage area (see Figure C.2-7, Area 20), hot dry storage area (see Figure C.2-7, Room 19), or to the hot cutting and dismantling Cell 1.

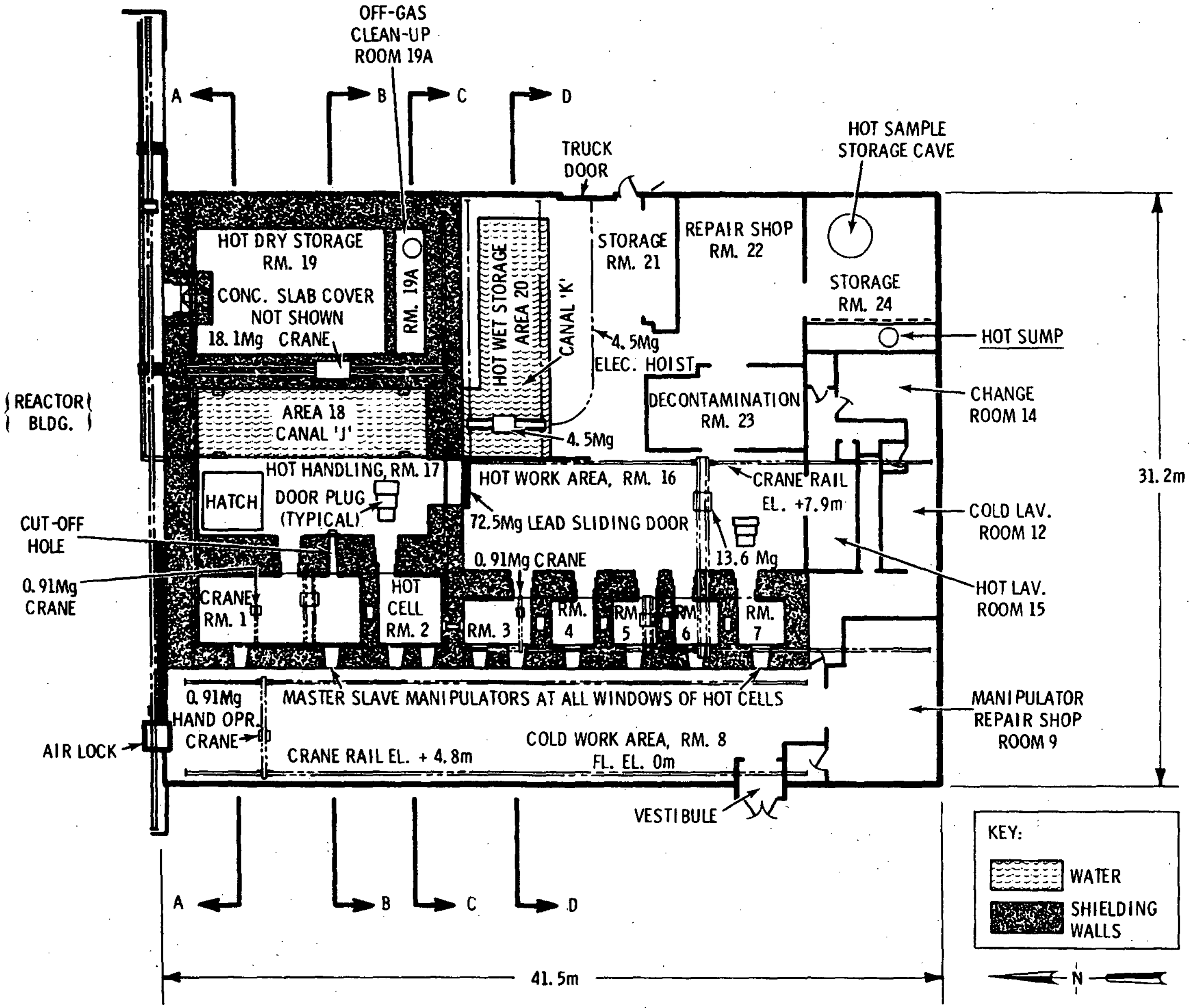
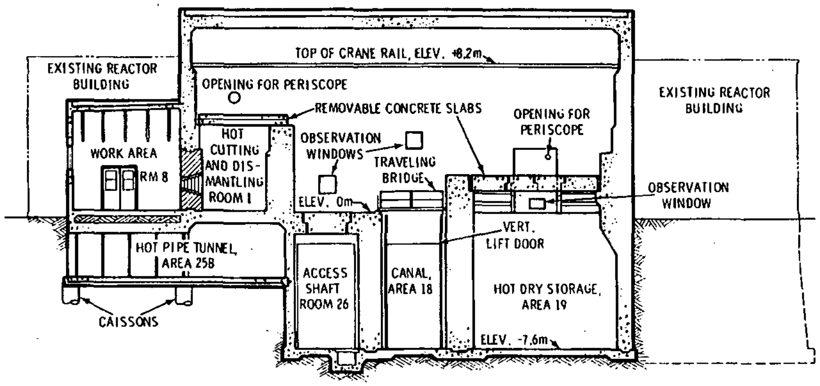
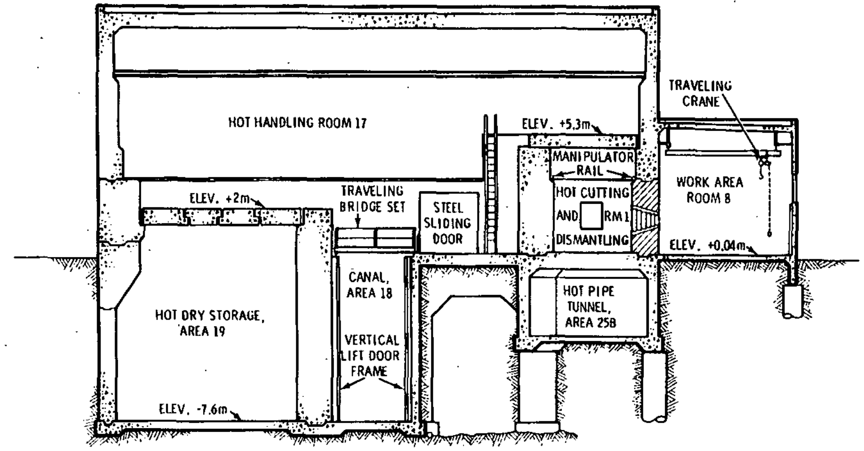


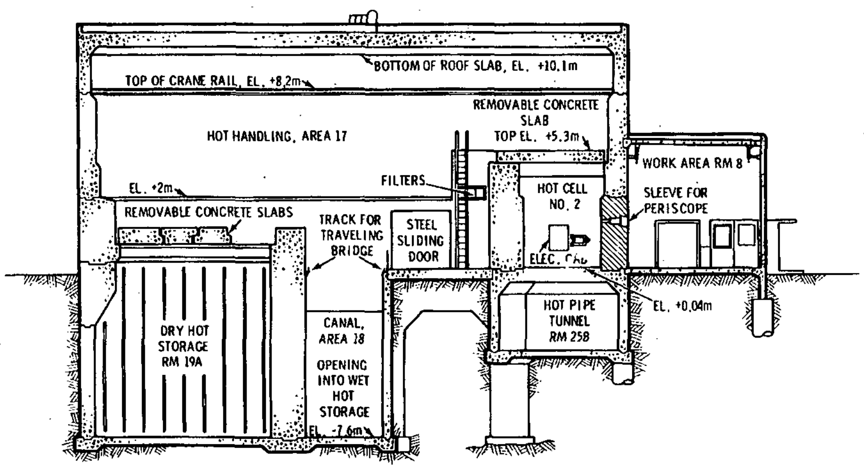
FIGURE C.2-7. Hot Laboratory Building-Plan View



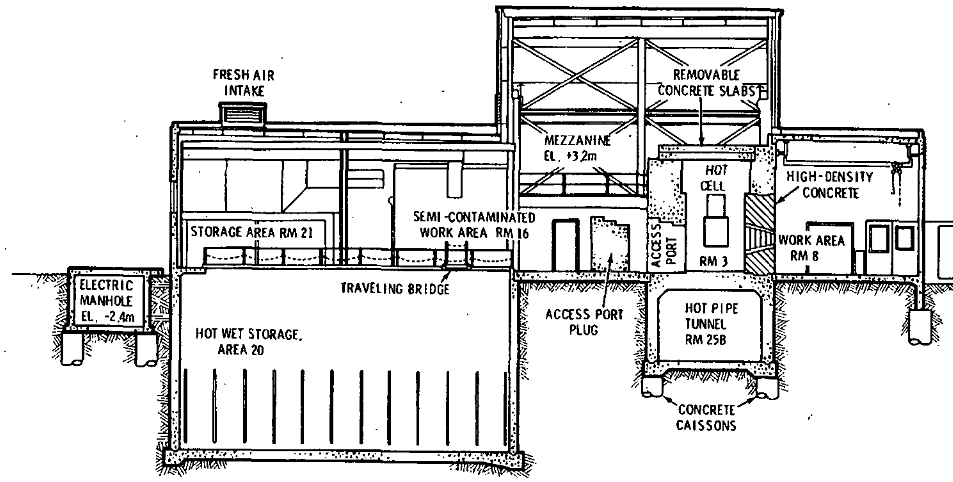
LOOKING NORTH - ELEVATION SECTION A-A



ELEVATION SECTION B-B



ELEVATION SECTION C-C



ELEVATION SECTION D-D

FIGURE C.2-8. Hot Laboratory Building--Elevation Sections

Hot cell structural details, the hot handling room, the controlled work area, the cell operating area, the hot dry storage room, and the hot pipe tunnel are discussed in the following subsections. Hot cell equipment is discussed in detail in Section C.3, Contaminated Equipment.

C.2.2.1 Hot Cell Structural Details

The hot cells are arranged for initial transfer of irradiated experiments into Dismantling Cell 1 or the Machining Cell 2. A partly counterbalanced, screwjack-operated, steel and concrete barrier door separates Cell 1 from Cell 2. Similar intercell barrier doors provide shielding equivalent to 0.91 m of high-density concrete (3.5 Mg/m^3) between Cells 3 through 7. A 1.1-m-thick, solid high-density concrete wall between Cell 2 and Cell 3 permits personnel entry into Cell 3 while high-level radioactive material is being prepared in Cell 2. Specimens are transferred from Cell 2 to Cell 3 solely through a 150-mm by 200-mm by 510-mm long transfer drawer across which an air pressure differential exists to reduce the spread of contamination from Cell 2 to Cell 3. Each cell is clad with 3.2-mm stainless steel sheet to a height of 3 m for ease of cell decontamination. The 1.2-m-thick high-density (3.5 Mg/m^3) magnetite aggregate wall at the front face of the hot cells extends to an elevation of 3 m. The rear walls of all cells are 1.6-m-thick ordinary-density concrete, as are the cell door plugs. Removable slabs comprising the roof of each hot cell are two 0.3-m thicknesses of concrete over Cells 3 through 7 and a 0.6-m thickness of concrete covering Cells 1 and 2. The slabs can be removed for installation or removal of heavy equipment or experiments.

Lead glass windows 1.1 m thick are installed for observation within the areas as follows:

- two windows penetrate the Hot Handling Room--Reactor Building wall
- two windows penetrate each of the operating faces of Hot Cells 1, 2, and 3
- one window penetrates the Hot Dry Storage Room wall
- one window penetrates each of the operating faces of Hot Cells 4 through 7.

All lead glass windows are 0.81 m by 0.56 m at their smallest (outside) faces. Lead glass windows provide gamma shielding power approximately equal to that of steel.

A feature of the Hot Laboratory Building design criteria is versatility in rearrangement and interchangeability of equipment. Few through-the-wall mechanical linkages or fixed optical relays are employed, in order to provide flexibility and, more importantly, to eliminate the use of a multiplicity of stepped plugs, thereby substantially reducing the chance of radiation streaming to the front-of-cell work area.

Cell 1 is the largest of the seven hot cells. Cells 2 and 3 are the next largest in size, followed by Cells 4 through 7, which are all equal in size. The inside dimensions of the hot cells are listed in Table C.2-2.

Electrical connections which terminate at the operating face of the hot cells pass through labyrinth conduits under the floor. There are no cell face penetrations other than manipulator and periscope ports and observation windows.

TABLE C.2-2. Inside Dimensions of the Hot Cells

	Inside Dimensions (m)[a]	
Cell	Front Face Length	Depth
1	8.69	3.66
2	3.5	3.66
3.	3.66	2.44
4	2.44	2.44
5	2.44	2.44
6	2.44	2.44
7	2.44	2.44

(a) All of the cells are 4.72 m in inside height.

In addition, each cell is provided with outside services, including hot and cold water, compressed air, deionized water, a vacuum header to vent in-cell enclosures, an intercommunication system, stepped roof plugs (permitting introduction of gravity feed lines), floor plugs for passage of gravity feed lines (into shielded vessels, as required, in the tunnel below), and floor drains to remove liquid wastes.

C.2.2.2 Hot Handling Room

The hot handling room (see Figure C.2-7, Room 17) serves as a radiation shield and air lock for most hot laboratory operations. It is shielded and equipped to provide for safe handling of highly radioactive, fueled experiments. The wall separating the Hot Laboratory Building from the Reactor Building is 1830-mm-thick ordinary-density concrete (2.6 Mg/m^3) to an elevation above grade of 7.92 m. Both the wall and room above 7.92 m are at least 380-mm-thick ordinary-density concrete. All other walls of the room have decreased thickness above an elevation of 3.4 m, as schematically indicated in Figure C.2-8, warranted by lesser shielding requirements.

There is a crane control station located near each of the three observation windows shown in Elevation Section A-A, Figure C.2-8. The heavily shielded wall surrounding the window wells is of magnetite aggregate and/or lead-shot construction. Minimum shielding at grade elevation and at the control stations is sufficient to shield a nominal one million curies source of 1 MeV gammas, based on a nominal 2 mr/hr dose rate at the control stations.

The hot dry storage area (see Figure C.2-8, Elevation Section A-A, Room 19) provides 650 m^3 of shielded volume and has a shielded observation window tilted for maximum visibility. The transfer between canal and storage area is generally by remote control. At the south end of the hot dry storage is a 2-m by 8.2-m room (see Figure C.2-8, Elevation Section C-C, Room 19A) separated from the dry storage area by a 0.45-m-thick concrete wall. This room houses the Off-Gas Cleanup Systems.

An 18.1-Mg crane with a rotatable 4.5-Mg auxiliary hoist services the hot dry storage area, canal area 18, hot handling deck, and Cells 1 and 2. The hot handling room crane can be remotely controlled from outside any of the three

crane control stations mentioned above. Other hot handling room equipment includes two hot cell transfer carts and a closed-circuit television system with a non-browning zoom lens to supplement viewing capabilities. Cell 1 and Cell 2 door plugs are motorized for remote control. The entire hot handling area is a limited access area to be entered, as are the hot cells, only with proper precautions and actions taken with regard to the radiological hazards.

C.2.2.3 Controlled Work Area

The controlled work area (see Figure C.2-7, Rooms 16, 21, 22, and 24) is used primarily for assembly or repair of special apparatus and equipment prior to their insertion into the appropriate hot cell. The decontamination room is a limited access area within the controlled work area. A truck entrance is located in the east wall of Room 16 and is serviced by a 4.5-Mg monorail-type hoist. This hoist can also be transferred to an overhead crane bridge which runs the length of the hot wet storage canal area (Room 20). A 13.6-Mg crane traverses the "high-bay" section over the controlled work area and Cells 3 through 7.

C.2.2.4 Cell Operating Area

The corridor in front of the hot cells houses the equipment to support and control the in-cell operations. A hand-operated, two-trolley, 0.91-Mg crane services the area. It can be used for the removal of periscopes and other heavy control apparatus and mechanical equipment. This area is considered to be a nonradioactive area of low contamination potential, except when manipulators must be withdrawn for repair or replacement.

C.2.2.5 Hot Dry Storage Room

The hot dry storage area is a pit 10 m by 6.7 m by 7.6 m deep which is covered by stepped concrete slabs. It provides a storage volume of 650 m^3 for highly radioactive materials which do not require cooling. The facility is serviced by an 18-Mg-capacity overhead crane with a 4.5-Mg rotatable auxiliary hoist.

C.2.2.6 Hot Pipe Tunnel

The Hot Pipe Tunnel (HPT) is located directly under the row of hot cells. It contains the contaminated drain pipes from the low-level chemistry laboratories in the Reactor Office and Laboratory Building, and from the Hot Laboratory Building itself. In addition, the HPT contains contaminated air handling systems piping. The role of the HPT in relation to contaminated air handling systems is discussed in detail in Section C.5.2.

C.2.3 Fan House

The Fan House, shown in Figure C.2-9, is located to the southeast of the Reactor Building. The building is approximately 17.1 m (north-south direction) by 19.1 m (east-west direction) by 4.7 m high. It consists of two levels, a basement level and a first-floor level. It is of light mill-type construction, except for the concrete shielding walls (0.65 m thick) of the deionizer room (see Figure C.2-9, Room 4).

During operation, air from potentially contaminated areas is filtered and discharged at rates up to 6 m^3/sec via a 30.5-m-high stack east of the Fan House. The Fan House contains the CV ventilating compressors, tanks, and monitoring system and ventilating fans, both feed and discharge, for the reference test facility. It also houses various waste cleanup deionizers, filters, and sumps. These facilities have low-to-moderate levels of surface and internal contamination.

C.2.4 Waste Handling Building

The Waste Handling Building (see Figure C.2-9) is located south of the Fan House. It includes a boiler room annex on the northwest corner of the building. The Waste Handling Building is of mill-type construction and measures approximately 31 m by 15.4 m by 6.2 m high. The boiler room annex is about 7.9 m by 3.3 m by 4.7 m in height.

The Waste Handling Building contains the liquid waste evaporator system with associated boiler, condenser, sumps, filters, and pumps. It also contains contaminated laundry equipment, a gantry room, waste packaging equipment, and waste storage facilities. The facilities have low-to-moderate levels of surface and internal contamination.

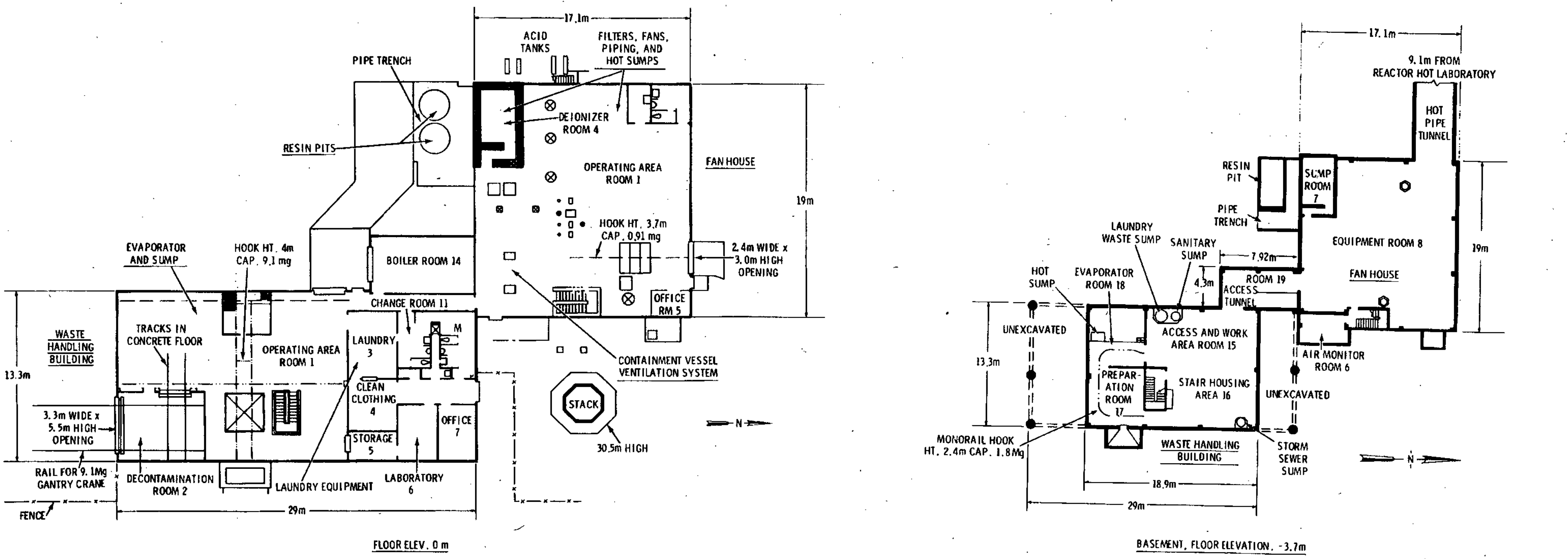


FIGURE C.2-9. Fan House and Waste Handling Building--Plan Views

C.2.5 Primary Pump House

The Primary Pump House, shown in Figure C.2-10, is attached to the east side of the Reactor Building and shares a thick concrete shield wall with it. The overall outside dimensions are approximately 21.3 m by 22.1 m by 6.1 m high.

The outer north and east walls are of mill-type construction on a concrete slab floor. The shielded portion of the building, which is inside this shell, is 16.8 m by 15.9 m. The concrete walls are 1.2 m thick.

The building has six internally shielded cells which house the primary heat exchanger, three primary process water pumps, deionizer tanks, and a tank room for process water additives. The roof of the shielded area is a 1.1-m-thick concrete slab.

On the south side, outside the building, are two hot spent resin pits approximately 2.5 m in diameter. The primary pump electric motors are outside the shielded area. The electric motors are connected to the primary pumps by shafts which pass through the concrete walls. Valve controls are also located outside the shielded area.

C.2.6 Hot Retention Area

The Hot Retention Area is located south of the Fan House and contains a rectangular concrete pit 13.7 m wide, 27.4 m long, and covered by 1.2 m of earthen shield. Within the pit are eight tanks, each with a capacity of 240 m^3 and fed with waste liquids from the Hot Drain System. Each tank has a discharge pump which discharges via an underground pipe tunnel to the waste cleanup system. The waste cleanup system is located within shielded cells in the Fan House.

Under the eight tanks is a steel plate floor cover with side sections and 0.3-m-high divider plates sectioning each tank area. These sections serve as collection dishes which are monitored for tank leakage collection. The arrangement of the tanks is shown in Figure C.2-11. The tanks are double-wall and epoxy-coated on the inside of the inner tank.

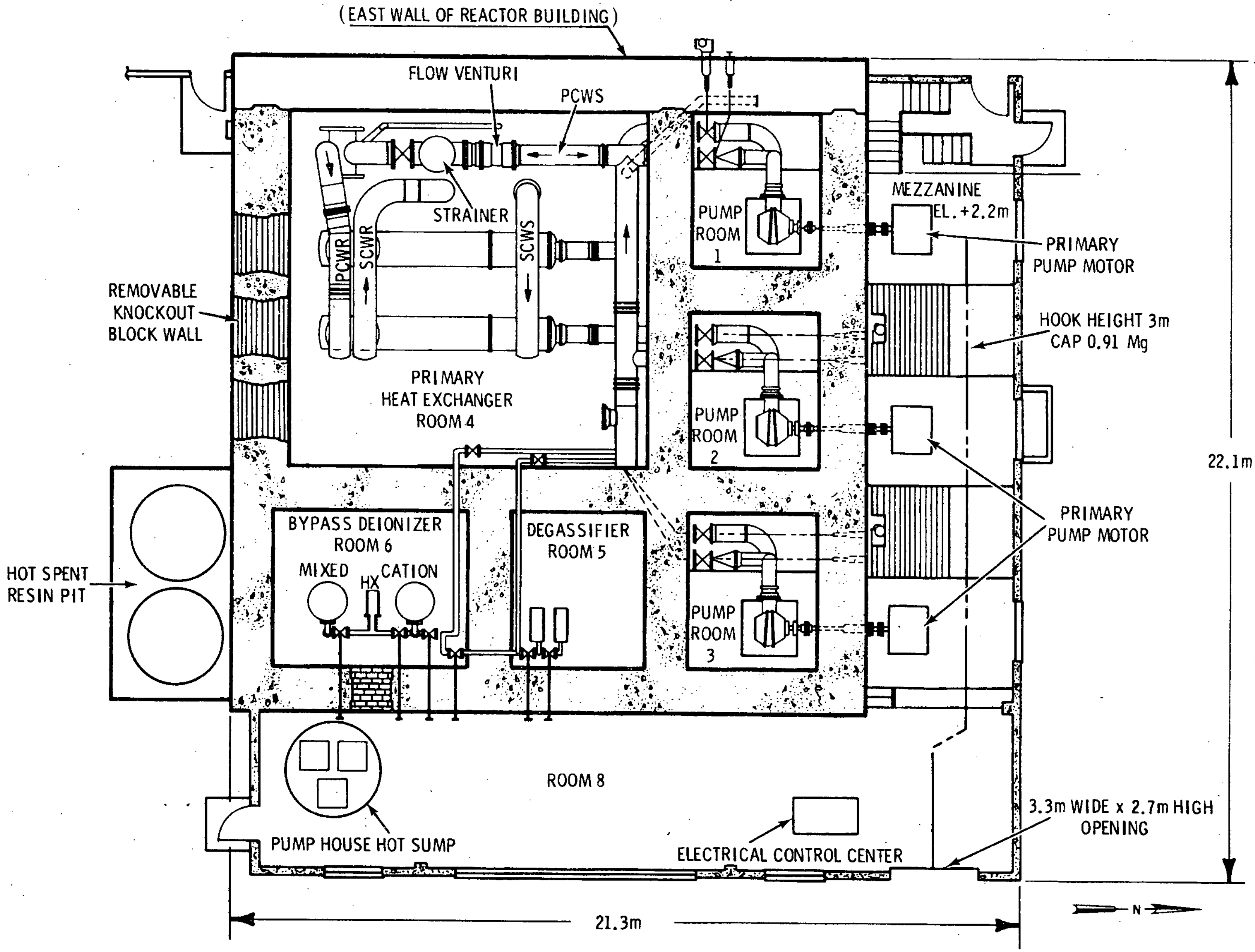


FIGURE C.2-10. Primary Pump House--Plan View

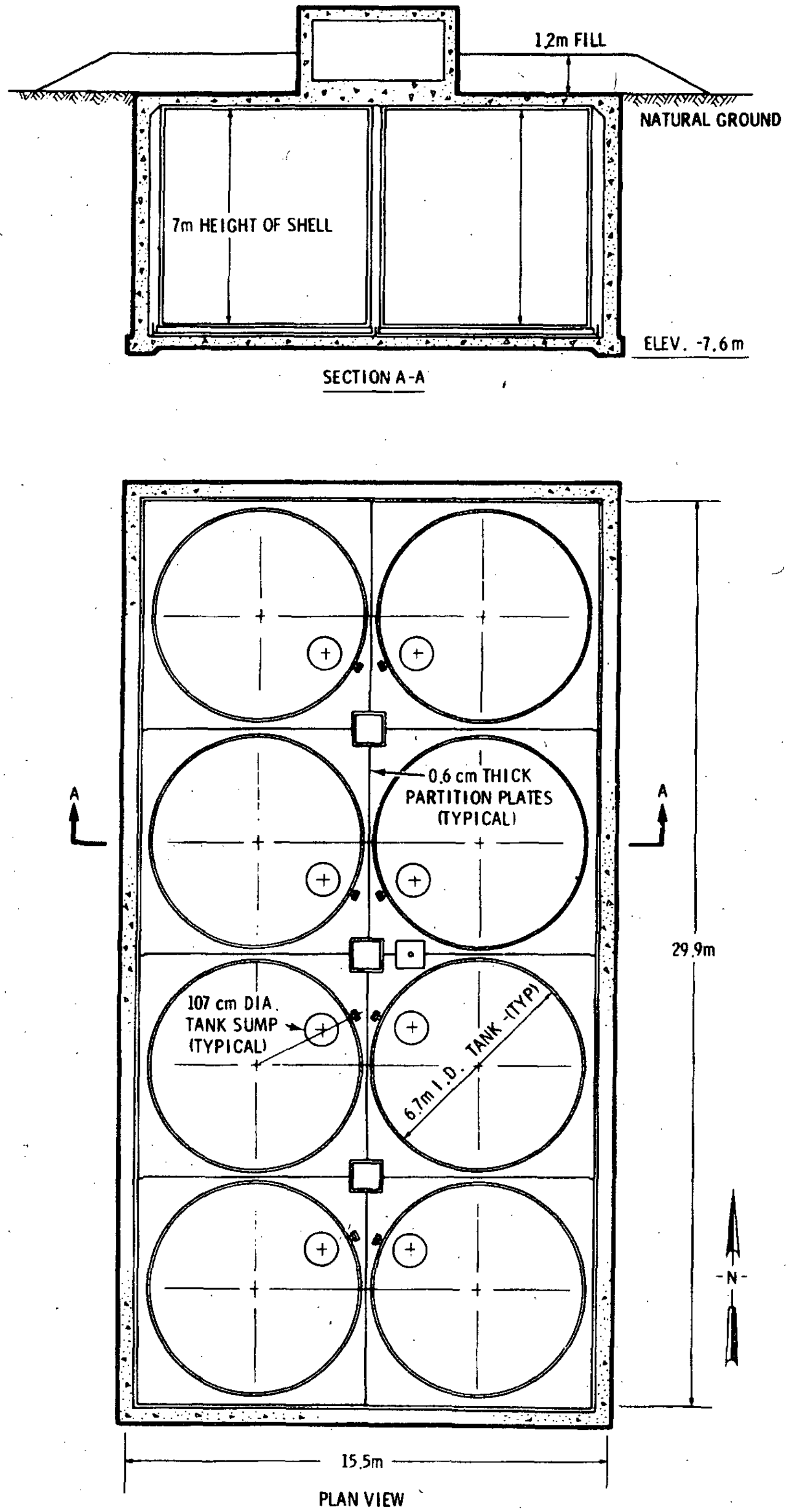


FIGURE C.2-11. Hot Retention Area--Plan and Section Views

Immediately north of the concrete pit are four 28.4-m^3 tanks (not shown in Figure C.2-11). These tanks are interconnected and have one 5-hp discharge pump. They are fed by return waste water from the waste cleanup system and serve as a monitoring storage point in the system. Liquid from these tanks may be transferred to the Cold Retention Area, into the quadrant and canal system, or diluted with uncontaminated waste water for disposal. These tanks are anticipated to contain low-to-moderate levels of internal contamination after draining.

C.2.7 Cold Retention Area

The Cold Retention Area is located east of the Fan House and consists of two 1,900-m^3 basins. The basins are 5.5 m deep with above-grade covers approximately 28.6 m square. These basins are used primarily for storage of water pumped from the quadrant and canal water system. Also, they serve doubly as retention basins for the waste water from the hold tanks in the waste cleanup system. Water from the basins may be transferred for reuse in the quadrants and canals. Plan and section views of the basins are shown in Figure C.2-12. Each basin has a concrete floor, lined with plastic.

C.2.8 Emergency Retention Basin

The Emergency Retention Basin (refer to Figure C.1-1) is a 37,800-m^3 above-ground earthen-diked basin, approximately 130 m by 96 m, located at the southeast corner of the reference site. It provides for the emergency storage of water for the facility. Very low radioactivity levels exist in this area. This area will be decontaminated by soil removal.

C.2.9 Office and Laboratory Building

The Office and Laboratory Building (OLB) is attached to the west side of the Reactor Building and consists of one basement level and two floors above grade. It houses offices, electronics repair shops, health physics offices, a first aid facility, and low-level radiochemistry laboratories. The radiochemistry laboratories are located on the second floor and are equipped with special chemical hoods of the design developed by Oak Ridge National Laboratory. There are 22 OLB laboratory hoods that exhaust to the building roof.

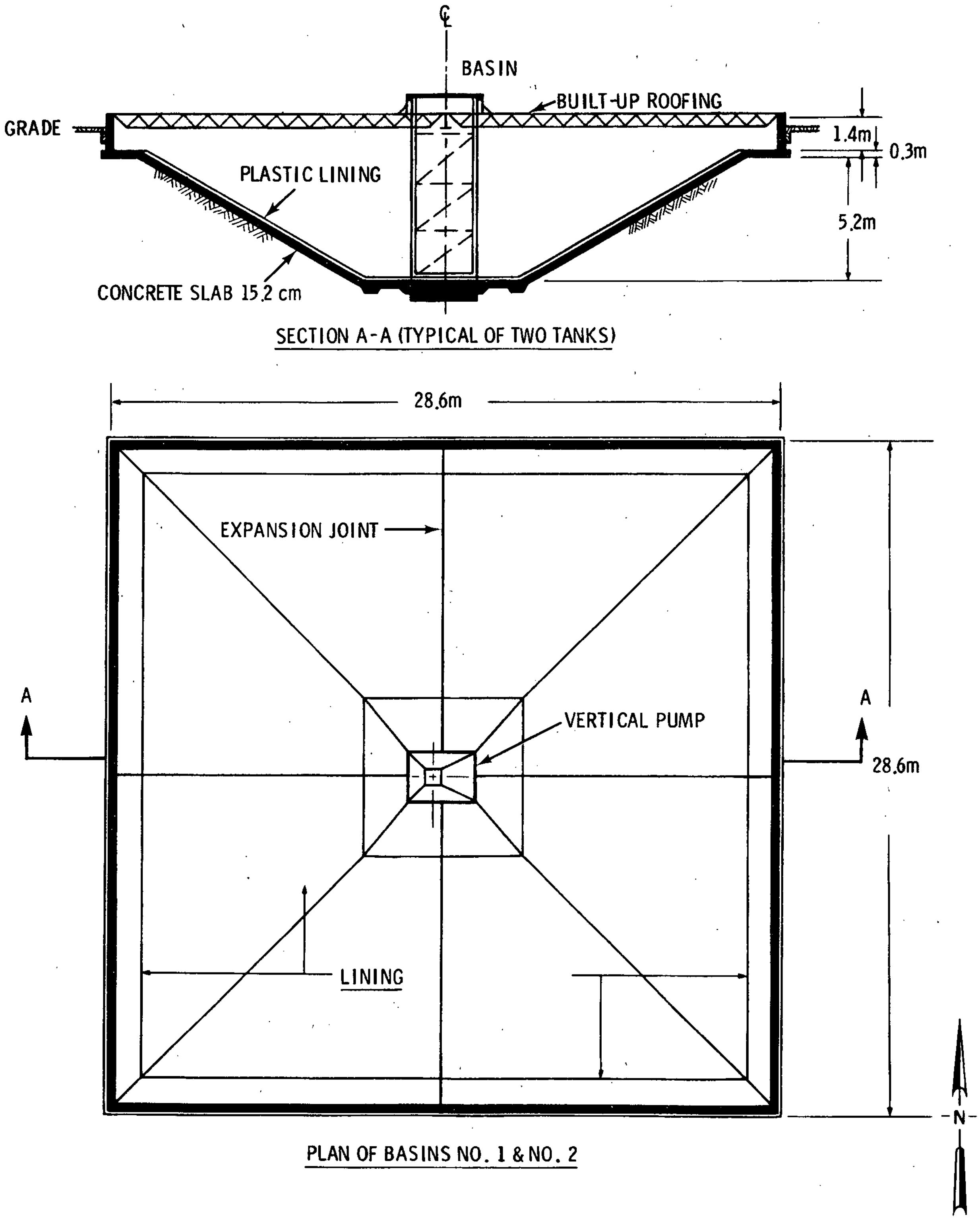


FIGURE C.2-12. Cold Retention Area--Plan and Section Views

Each hood has a 610-mm by 610-mm by 305-mm absolute filter in a sealed housing above the hood. The exhaust fans on the roof servicing the hoods are sized to maintain a hood face velocity in excess of 30 m per minute.

The majority of the chemical and radiochemical analyses is performed in these laboratories. Dissolution and chemical decontamination steps are performed in the Hot Laboratory Building when required.

There are both "hot" and "cold" drains in the laboratories; however, the radiochemistry laboratory has only hot drains. These drains lead to either hot or cold monitored sumps and then to hot or cold retention areas for ultimate disposal. A utility tunnel connects the hot sump in the basement of the Office and Laboratory Building to the Hot Laboratory Building.

C.2.10 Service Equipment Building

The Service Equipment Building, east of the Primary Pump House, contains the raw water processing equipment, three large air compressors, electrical control equipment, two steam boilers, and two diesel electric generators for emergency electrical power. It also houses the health physics radiochemistry/analytical laboratory. No radiological involvement of any significance takes place in this building.

C.2.11 Cooling Tower

The cascading flow-type cooling tower is about 24.5 m by 21.3 m by 13.1 m high. The redwood plates are highly impregnated with the various water conditioners--algacides, fungicides, and corrosion control chemicals--used to treat process water. The main structural material is wood frame with process water distribution manifolds.

C.2.12 Auxiliary Structures

This section contains structural information about the other buildings and structures on the reference test reactor site shown in Figure C.1-1. These structures are generally presumed to be nonradioactive.

Security Building. This building is located off the west perimeter fence boundary. It is 8.2 m by 6.1 m and is 2.8 m high. It is of frame construction and houses the security personnel who control vehicular and personnel access to the facility.

Gas Service Building. This building is located just north of the Reactor Building. It is 6.1 m by 7.6 m and is 3 m high. It is of steel construction. It contains storage of specialty gasses in steel cylinders.

Compressor Building. This building is located due north of the primary pump house. It is 12.8 m by 15.2 m and is 3.7 m high. It is of steel construction.

Weather Tower and Building. A three-sided, steel meteorology tower and an associated single-story instrument building are located on the extreme west side of this facility, outside the perimeter fence and near the main entrance. The weather tower is 3 m by 3 m by 3 m by 46 m high.

Effluent Water Monitoring Station. This facility is located in the extreme southeast corner of the reference facility site. It consists of a series of flumes through which flows all facility surface and waste water collected by a series of open ditches and covered culverts. A small structure at the site houses the monitoring instruments. It is of steel construction and is 3.7 m by 3.7 m by 2.4 m high.

Water Tower. The water tower, located to the east of the Service Equipment Building and north of the Cooling Tower, is 56.8 m high. Two storage tanks, one directly above the other, are visible. In fact, the upper stainless steel tank contains an inner stainless steel tank. Both of these upper tanks contain demineralized reactor-grade water. The combined total capacity of both these upper tanks is about 284 m^3, with the inner tank designed to hold 41.6 m^3. The lower tank (378.5 m^3 capacity) is made of carbon steel and is the same inside diameter as the upper tank (approximately 8.5 m). The tanks are supported by a tubular steel frame resting on a structural concrete foundation.

Substation. An electrical substation is located east of the Cold Retention Basins. It occupies an area of approximately 7.5 m by 7.5 m. The perimeter fence surrounding the substation measures about 17 m by 24 m.

Sludge Settling Basins. Two concrete-lined sludge basins are located northeast of the cooling tower. They are approximately 9.1 m by 15.2 m, and are used as part of the reference facility's water treatment capability.

Exhaust Stack. The exhaust stack is a 30.5-m-high, 1.5-m-diameter, vertical steel pipe with a concrete support stand and a vortex plenum at the base. The air flowing through the exhaust stack is monitored and contamination levels are recorded. The flow through the stack is measured and recorded at the Fan House.

C.3 CONTAMINATED EQUIPMENT AND MATERIALS

Descriptions of the contaminated equipment and materials located in the buildings and areas at the reference test reactor site are presented in this section. In general, contaminated equipment or materials and piping and valves associated with a particular building or area are described in one of two ways: 1) in the subsection for that building or area; or 2) as part of a major test reactor system.

C.3.1 Reactor Building Equipment

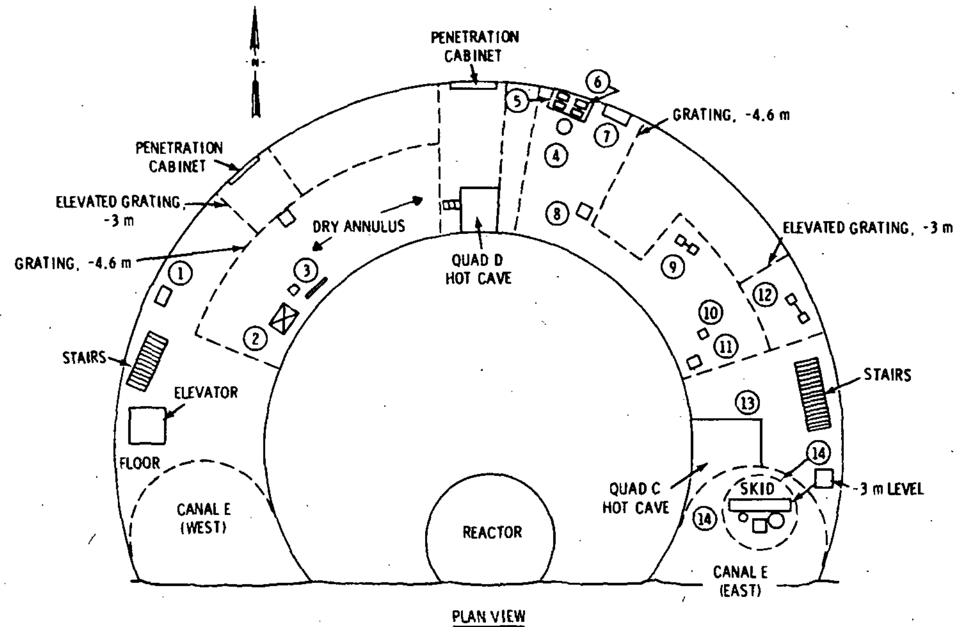
The Reactor Building houses the reference test reactor and primary coolant water system (PCW), the shutdown cooling sytem, refueling equipment, spent fuel, the Mock-up Reactor (MUR) facility and various experimental rigs which are potentially radioactive due to direct neutron activation and/or the spread of activation products.[a] Table C.3-1 gives a list of contaminated equipment in the Reactor Building. It is assumed that this equipment is contaminated since it contacts (either directly or indirectly) the contaminated reactor water. The majority of the fixed equipment at the various levels in the Dry Annulus, shown in Figures C.3-1 and C.3-2, is also assumed to be contaminated.

(a) It is assumed that no failure of clad fuel or fueled experiments occurred throughout the operating lifetime of the reference test facility.

TABLE C.3-1. Contaminated Equipment: Reactor Building[a]

Location	Equipment Piece (Quantity)	Estimated Mass, (each) kg	Estimated Volume, (each) m^3	Estimated, Overall Dimensions, L x W x H, m
First Floor	Experimental Consoles (10)	1190	0.75	2 x 0.61 x 0.61
Quadrant A	HT-1 Insertion Machine			
	HT-2 Insertion Machine			
	Experimental Gas Storage Tank	6820	23	3 dia. x 3
	Hydraulic Rabbit Equipment	68	0.06	--[b]
	Miscellaneous:			
	Steel Cask A	2730	0.37	0.72 dia. x 0.92
	Steel Cask B	2730	0.37	0.72 dia. x 0.92
	Steel Cask C	640	0.1	0.28 dia. x 1.58
	Steel Cask D	455	0.09	0.36 dia. x 0.83
	Concrete Drum Shield	500	0.39	0.64 dia. x 1.22
	Concrete-Lined Drums (2)	400	0.36	0.56 dia. x 2.31
	Snout Shield with Transfer Table	730	2.9	--
	Track	955	0.14	--
	Exposed Shield	2455	0.4	--
Quadrant B	Shim Rod Drive Actuators (10)	114	0.2	0.23 dia. x 5.2
	Regulating Rod actuators (3)	115	0.2	0.23 dia. x 5.2
	Double-Stacked 0.21-m^3 Drums of Exposed Equipment (4)	140	0.62	0.58 dia. x 0.76
	Experimental Consoles (3)	1190	0.75	2 x 0.61 x 0.61
	Drums of Exposed Hardware (3)	140	0.62	0.58 dia. x 0.76
	Centrifuges (2)	160	0.71	--
	Neutron Radiography Equipment Housing	230	0.77	--
	Underwater Vehicle	910	3.6	2.75 x 2.6 x 0.5
	Underwater Lights (15)	9	0.03	--
	UWBR Window Box	230	0.77	--
	Dummy Fuel Elements (25)	4.6	0.007	1.08 x 0.08 x 0.08
	Fission Chamber Drive Units (2)			
	Filter Housings (4)			
	Neutron Shielding Blocks & Foundation			
Quadrant C	Hot Cave Insertion Machine, Rails, and Foundation	1140	2.17	2.26 x 1.17 x 0.82
	Hot Gas Storage Tanks (3)	435	0.73	0.61 dia. x 2.49
	Fuel Chute Liners ( )			
	Fuel Chute			
	HT-1 Insertion Machine			
	HT-2 Insertion Machine			
	Hydraulic Rabbit Equipment			
Quadrant D	HB-1 Insertion Machine			
	HB-2 Insertion Machine			
	Control Panels (3)			
	Control Panels, Hydraulic (3)			
	Decking and Support Steel			
Canal E	Fuel Trucks (2)			
	Exposed Air Lock Conveyor			
Canal F	(c)			
Canal G	Fuel Storage Racks (permanently mounted)			
	Gamma Irradiation Facility			
	3 Fuel Shipping Casks (KNAPP-MILLS casks @ 18.2 Mg/each)			
Canal H	MUR[a]			

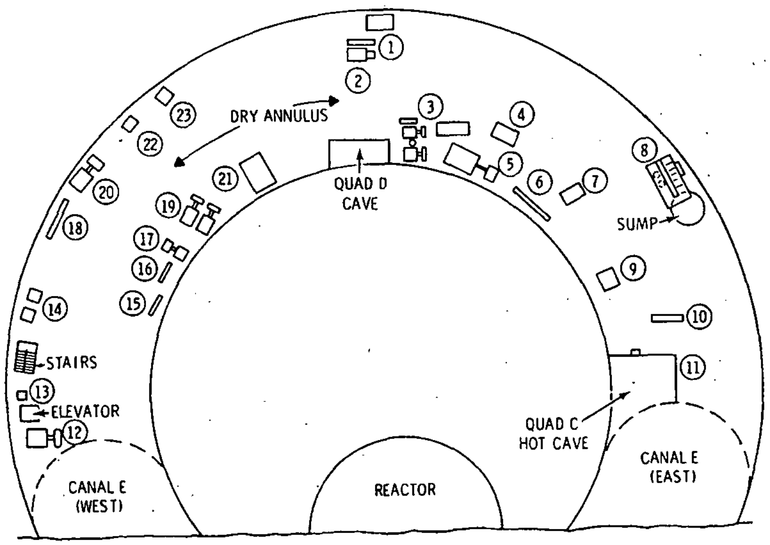
(a) Does not include the MUR and its associated hardware (see Section C.3.1.2 for MUR and associated equipment).
(b) Data not available.
(c) This canal is empty.



Position	Equipment	Overall Dimensions, (Each) L x W x H, m	Estimated Volume (Each), m³
1	Vacuum Services Systems Tanks (3)	0.61 dia. x 0.86	0.26
2	Pre-Fab Monitoring Cabinet; Contains 1 Vacuum Pump & Shielded Tubing, Lead Brick Lined (~300 Lead Bricks)	1.52 x 1.52 x 1.02	2.4
3	Meter Mount Panel	0.91 x 1.98 x 0.003	0.01
4	Tank	0.61 dia. x 2.44	0.71
5	Nash Vacuum Pumps (2)	0.28 dia. x 0.28	0.02
	Motors (2)	0.28 dia. x 0.23	0.02
6	Tank	0.61 dia. x 2.44	0.71
7	Neutron Poison Injection System	1.22 x 1.22 x 0.61	0.91
8	Console	0.64 x 1.96 x 0.61	0.76
9	Compressor:		
	Heads	0.28 dia. x 0.36	0.03
	Crank	0.36 dia. x 0.28	0.03
	Motor (7.5 hp)	0.30 dia. x 0.36	0.03
	Tank	0.61 dia. x 1.63	0.33
10	Console	0.56 x 1.21 x 0.43	0.30
11	Console	0.61 x 1.27 x 0.56	
12	Compressor:		
	Tank	0.48 dia. x 1.62	0.28
	Motor	0.25 dia. x 0.28	0.02
	Compressor	0.28 dia. x 0.38	0.02
13	Blower (Top of Cave)	(a)	0.15
14	Compressor:		
	Motor	0.23 dia. x 0.25	0.01
	Tank	0.33 dia. x 0.46	0.04
	Compressor	0.23 dia. x 0.25	0.01

(a) Only total volume data available.

FIGURE C.3-1. Contaminated Equipment: Containment Vessel Dry Annulus, -3 m and -4.6 m Levels



PLAN VIEW

Position	Equipment	Overall Dimensions, (Each) L x W x H, m	Estimated Volume (Each), m³
1	Control Panel	0 49 x 0.36 x 0 77	0 13
2	Motor-Generator Set	0.31 dia. x 0.84	0 06
3	Compressors (2) Motor (100 hp) Control Panel	(a)	1.46
4	MCC	2 33 x 1.02 x 0.51	1 2
5	Q&C System:		
	Motor	0.56 dia. x 0 69	0 :7
	Pump	0.56 dia. x 0.46	0 12
6	Pre-Fab Panel (Aluminum Stock)	2.34 x 2.1 x 0.006	0 13
7	Experimental Console No. 63-04	2 x 0.61 x 0.61	0 75
8	Utility Air System Tanks (4)		
	Tank A	0.26 dia. x 0 41	0 02
	Tank B	0.33 x 0.41 x 0.61	0.09
	Tank C	0.31 dia. x 1.4	0 02
	Tank D	0.31 dia. x 1.4	0 02
9	Experimental Cabinet: Compressor Vacuum Pump (2) Panel (meters & valves)	1.53 x 1 53 x 1.07	2.5
10	Pre-Fab Panel (Aluminum Stock)	1.83 x 0.61 x .006	0 03
11	Miscellaneous Equipment	(b)	
12	Q&C System:		
	Motor	0.56 dia. x 0.69	0.17
	Pump (100 hp)	0.64 dia. x 0.36	0.12
13	Q&C Pump Control Panel	2.29 x 1.02 x 0.51	1 19
14	Hydraulic Pump & Storage Tanks (3):		
	Pump (7.5 hp)	0.32 dia. x 0.41	0.04
	Tank A	0.69 x 0.61 x 1.13	0.48
	Tank B	0.23 dia. x 0.43	0 02
	Tank C	0.24 dia. x 0.71	0.04
15	Valve Panel	(b)	0.16
16	Pre-Fab Panel	(b)	0.29
17	Pump & Motor	0.23 dia. x 0.28	0 02
18	Control Panel	1.53 x 2.34 x 0.52	0 86
19	Q&C Recirc. Pumps & Motors (2), 25 hp/each		
	Motor	0.36 dia. x 0.51	0.05
	Pump	0.31 dia. x 0.15	0 02
20	Vacuum Motor & Pump		
	Motor	0.46 dia. x 0.61	0 10
	Pump	0.26 dia. x 0.26	0 02
21	Neutron Shield Tanks & Stand		
	Tank	1.59 dia. x 1.88	4 35
	Stand	(b)	0.19
22	Metering Panel	2 x 0.61 x 0.28	0 34
23	Noble Gas Purifier	0.61 x 0.61 x 0.64	0 24

(a) Each compressor plus control panel is skid-mounted.
(b) Only total volume data available.

FIGURE C.3-2. Contaminated Equipment: Containment Vessel Dry Annulus, -7.6 m Elevation

After draining the primary coolant water (PCW) system, contamination is assumed to be concentrated at the inlet end of the heat exchangers, with only millicurie quantities of radioactiviity in the interior of the PCW piping.

Descriptions of the reference test reactor vessel and its internals and the MUR follow.

C.3.1.1 Test Reactor Vessel and Internals

The separation of the test reactor vessel and internals into individual parts is illustrated in Figure C.3-3. The reactor core horizontal and section views, shown in Figure C.3-4, are included to clarify core box sections and relative locations of the various parts shown in Figure C.3-3.

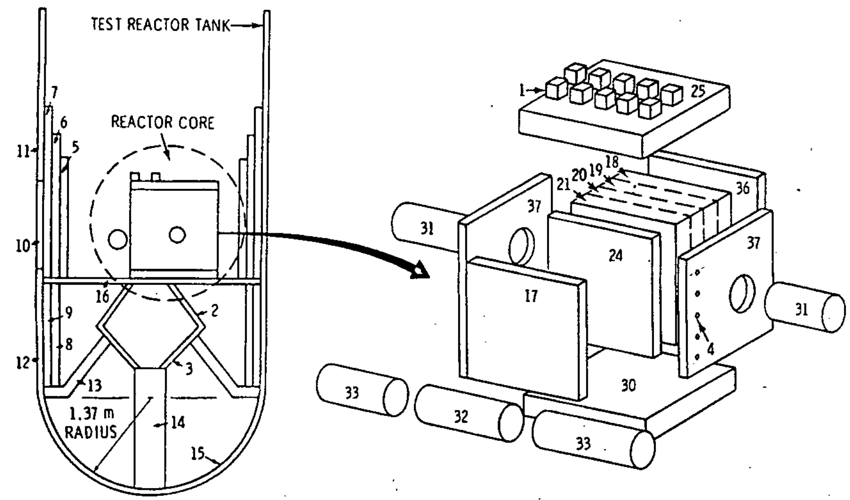
Test Reactor Vessel. The test reactor vessel, shown in Figure C.3-5, contains the core and supporting structures, beam tubes, control rod guide tubes, in-core instrumentation, and other components.

The reactor vessel is 2.743 m inside diameter by about 9.48 m in height and weighs approximately 35.5 Mg, including all appurtenances that are welded to the vessel. The elliptical top of the tank is at grade level. The center of the reactor core is 6.4 m below grade. The reactor vessel wall thickness varies from top to bottom, as shown in Figure C.3-6.

The reactor vessel is a vertical, cylindrical ASME code pressure tank with a welded hemispherical bottom head and an elliptical top head that is flanged and gasketed so that it can be removed. A hatch is also provided to facilitate changing fuel elements and inserting or withdrawing experiments.

The reactor vessel is fabricated of A-201 steel and internal surfaces that are in contact with primary coolant are clad with type 304 stainless steel. The exterior of the reactor vessel is surrounded by 30 mm of insulation. Access to reactor vessel side insulation is limited by the surrounding concrete.

There are a total of 89 penetrations in the reactor tank, ranging in diameter from 19 mm to 1,041 mm. Eighty-one penetrations are for experimental instrumentation lead-ins, reactor control, and fuel discharge. Eight penetrations are for control and circulation of coolant. All penetrations are of



NOTE: NOT SHOWN ARE BEAM TUBES HB-1 THRU HB-6 NOR VERTICAL TUBES V-1 AND V-2. SEE FIGURE C.3-4 FOR OTHER DETAILS.

Item No.	Description	Material	Mass, kg	Approximate Overall Dimensions L x W x H, m
1	Control Rod Roller Guides (10)	304-SS	12.5	0.7 x 0.2 x 0.05
2	Upper Flow Guide	304-SS	123	1.55 x 1.55 x 0.5
3	Lower Flow Guide	304-SS	123	1.55 x 1.55 x 0.5
4	Miscellaneous Bolts (75)	304-SS	1.94	(a)
5	Inside Upper Thermal Shield	304-SS	3760	2.6 dia. x 1.3
6	Middle Upper Thermal Shield	304-SS	4700	2.65 dia. x 1.5
7	Outer Upper Thermal Shield	304-SS	5650	2.70 dia. x 1.7
8	Inside Lower Thermal Shield	304-SS	3800	2.65 dia. x 1.3
9	Outer Lower Thermal Shield	304-SS	4100	2.70 dia. x 1.3
10	Reactor Vessel Near Core	304-SS	3410	
11	Reactor Vessel Above Core	304-SS	2910	2.74 dia. x 9.45
12	Reactor Vessel Below Core	304-SS	2910	
13	Flow Guide (Part of Tank)	304-SS	2120	2.74 dia. x 0.9
14	Rod Drive Box	304-SS	704	0.7 x 0.2 x 1.3
15	Reactor Vessel Bottom	304-SS	2300	2.74 dia. x 1.37 m radius
16	Metering Plate	304-SS	592	2.6 dia. x 0.01
17	North Core Box Plate	Be[b]	42	1 x 1 x 0.025
18	RD Pieces with Plugs (8)	Be	163	0.109 x 0.109 x 1 (each)
19	RC Pieces with Plugs (8)	Be	163	0.109 x 0.109 x 1 (each)
20	RB Pieces with Plugs (8)	Be	113	0.109 x 0.109 x 1 (each)
21	RA Pieces with Plugs (8)	Be	113	0.109 x 0.109 x 1 (each)
22	LI, LII Pcs. w/o Plugs	Be	54.4	(a)
23	LA 2 thru 10 Pcs. w/o Plugs	Be	61.2	0.76 x 0.76 x 1
24	Flow Divider Plate	Lockalloy	49.1	1 x 1 x 0.025
25	Upper Grid	6061 T-6 Al	93.7	0.7 x 0.2 x 0.05
26	V-2 Near Core	6061 T-6 Al	15	0.203 dia.
27	V-2 Above Core	6061 T-6 Al	13.9	0.203 dia.
28	HB-4 First 710 mm	6061 T-6 Al	17.2	0.152 dia.
29	HB-5 & 6 First 710 mm	6061 T-6 Al	16.9	0.152 dia. & 0.381 dia.
30	HT-1 In Core Portion	6061 T-6 Al	24.7	0.229 dia. x 4.14
31	HT-1 Ends Outside Core	6061 T-6 Al	43.2	
32	HT-2 In Core Portion	6061 T-6 Al	31.4	0.229 dia. x 4.14
33	HT-2 Outside Core Box	6061 T-6 Al	54.4	
34	HB-1 & 3 First 0.61 m	6061 T-6 Al	11.3	0.152 dia. x 1.68 (each)
35	HB-1 & 3 0.61 m	6061 T-6 Al	10.7	
36	Far South Core Box Plate	6061 T-6 Al	.62	1 x 1 x 0.025
37	Core Box Side Plates (2)	356-T Al	100	0.2 x 1 x 0.025 (each)
38	Lower Grid	356-T Al	168	0.7 x 0.2 x 0.12

(a) Data not available.
(b) Be is Beryllium.

FIGURE C.3-3. Schematic Diagram of Test Reactor Tank and Internals

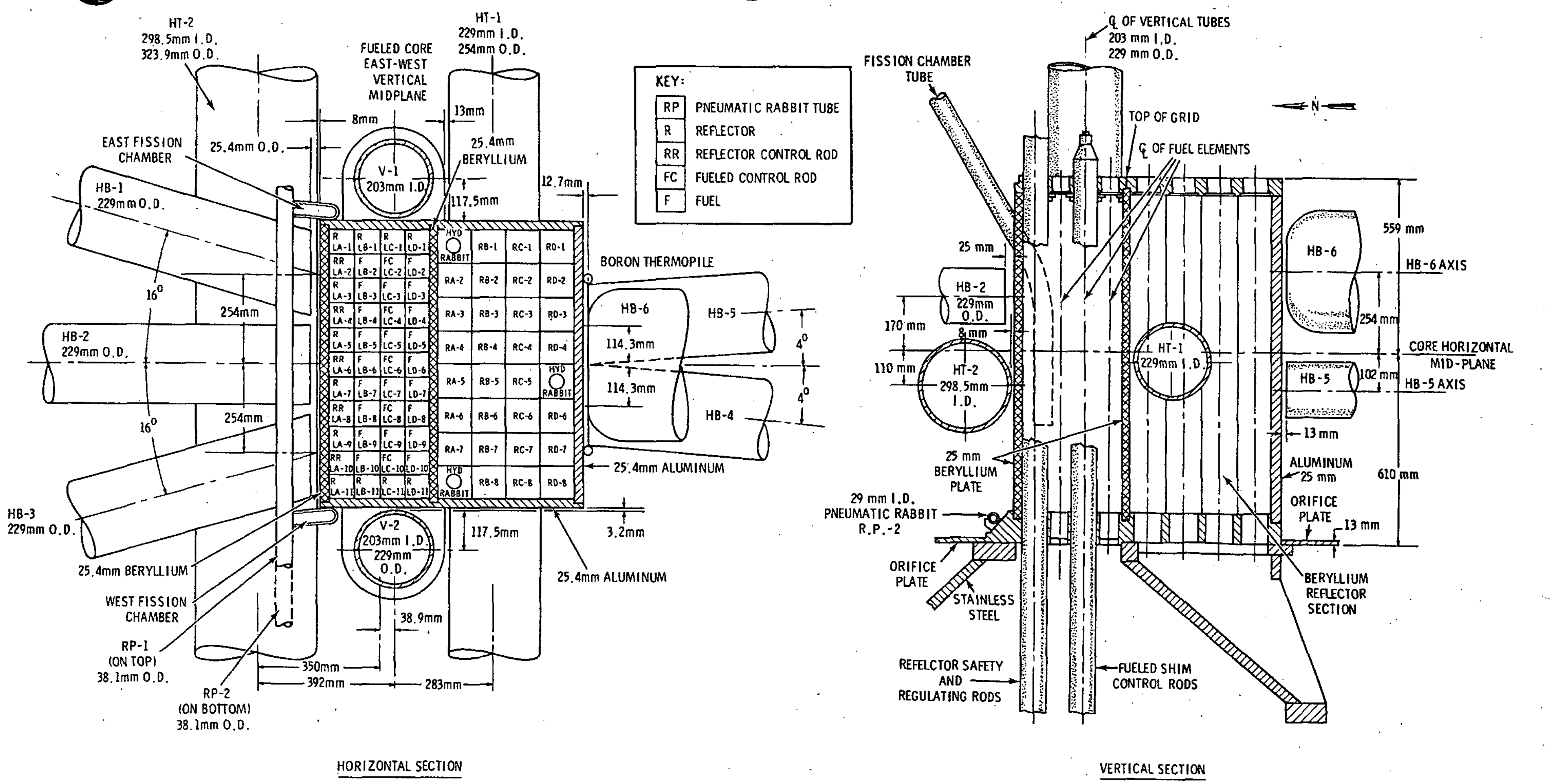


FIGURE C.3-4. Test Reactor Core--Horizontal and Section Views

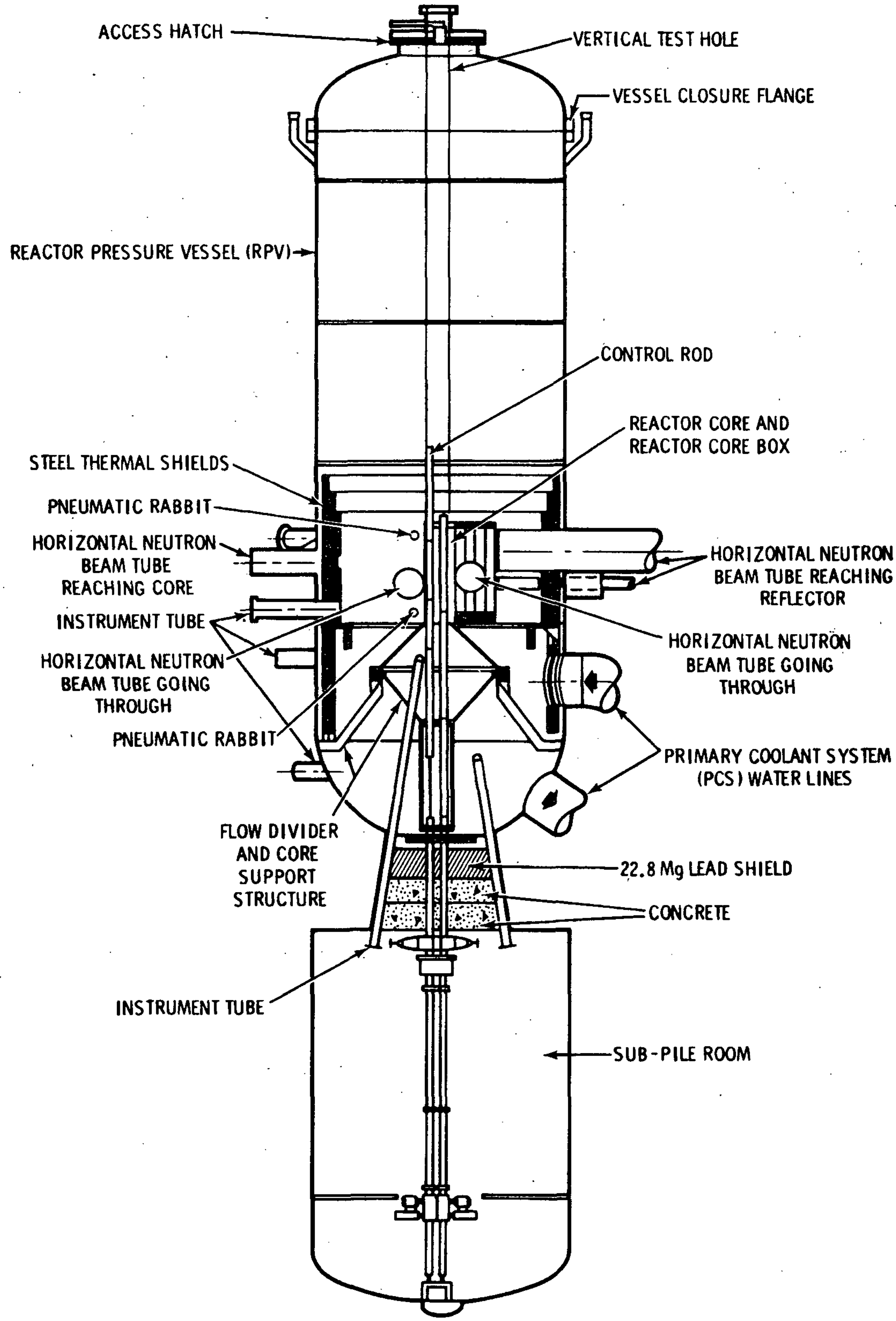


FIGURE C.3-5. Test Reactor Vessel and Internals

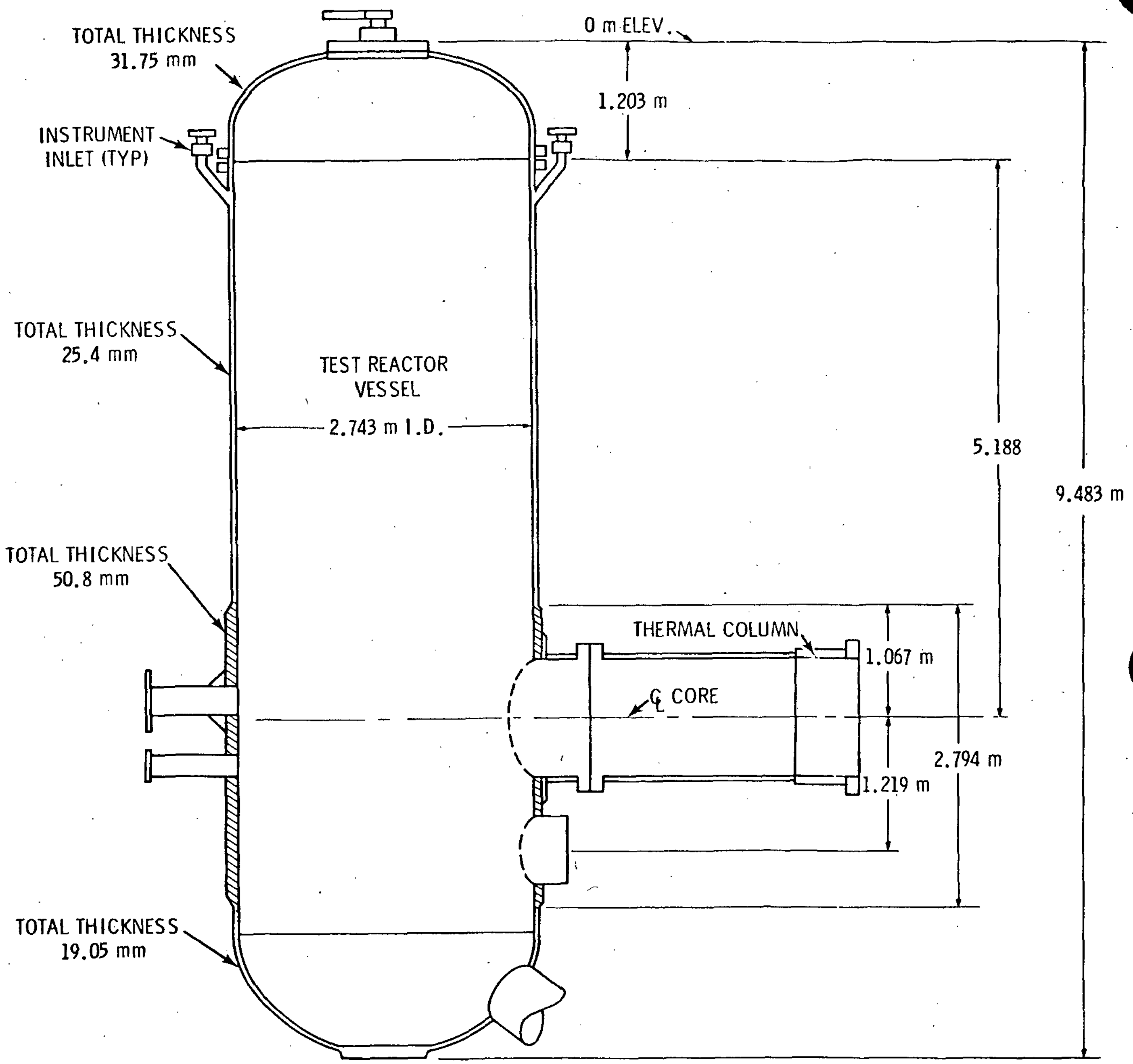


FIGURE C.3-6. Test Reactor Vessel Dimensions

welded construction at the reactor tank. External connections to the penetrations are welded, flanged, or threaded. The penetrations are made in four locations, as summarized in Table C.3-2: 1) head (dome), 2) thin-wall cylindrical (25.4 mm) above the core, 3) heavy-wall cylindrical (50.8 mm) surrounding the core, and 4) bottom hemisphere. Locations 2 and 3 are further divided into quadrants.

TABLE C.3-2. Summary of Reference Reactor Vessel Penetrations

Location	Number of Penetrations	Nominal Outside Diameter (mm)
Head (dome)	18	51 to 915
Wall Section Above Core		
Near Top	24	26 to 51
Quadrant A	2	38
Quadrant B	1	19
Quadrant C	3	38 to 159
Quadrant D	1	203
Wall Section Surrounding Core		
Quadrant A	6	38 to 343
Quadrant B	2	610 to 1041
Quadrant C	6	38 to 343
Quadrant D	9	179 to 241
Bottom Hemisphere	17	26 to 610
Total	89	

Test Reactor Tank Internals. The test reactor tank internals support the core, maintain fuel alignment, limit fuel assembly movement, direct flow past the fuel elements, provide gamma and neutron shielding, and guide in-core instrumentation (see Figure C.3-3). The tank internals include (exclusive of fuel, control rods, and in-core nuclear instrumentation) the following components:

1. Thermal Shields. There are five annular thermal shields (three above the core and two below the core), as shown in Figure C.3-3. There are four centering tabs on the inner shield. The middle and outer shields are each held in place by eight hold-down bolts. The lower two thermal shields rest on the lower core support and are held in position by eight support brackets on the top.

2. Core Support. The core box is supported by an upper core support structure, which consists of the upper flow guide and the lower flow guide. These two structures are bolted to the lower core support flange. The flange junction is secured on its outer edge with a welded 3.2-mm by 19-mm steel perimeter strap. The lower core support is welded to the tank. The control rod drive box is attached to the lower flow guide. The orifice plate is bolted to the upper core support and is also attached to the thermal shield support brackets at the outer edge.

3. Core Box. The core assembly vertical and horizontal section views are shown in Figure C.3-4. The upper grid is attached to the core box. The core box sides are formed by the north core plate, which is beryllium metal; the south core plate, which is aluminum; and the two aluminum side plates. The bottom of the core box is formed by the lower grid plate.

   The north side of the core is a 4 by 11 array of beryllium pieces, control rods, and fuel elements on an approximate 76.2-mm pitch. The beryllium flow divider plate separates this fueled core area on the north side from the south reflector side, which is composed of an array of 4 by 8 beryllium pieces on a 111-mm pitch.

4. Test Holes. The locations of the various test holes, with respect to the reactor core, as shown in Figure C.3-4, are described below:

   - Horizontal Test Holes. Two horizontal test holes (designated HT-1 and HT-2) lie parallel to the long horizontal axis of the core. HT-1 passes through the beryllium reflector blocks, and HT-2 is outside of the core box structure. The 229-mm I.D. aluminum thimbles pass completely through the reactor tank, penetrating the

tank wall through 343-mm I.D. flanged penetrations. A remotely operable sealing mechanism is therefore provided to seal the clearance space, between the test hole wall and the tube containing the experiment, from water in the shielding pool and surrounding the reactor. Connections are provided so that water can be circulated from the primary cooling water system through the annular clearance space, to remove the heat from the experiment in the thimble.

- Horizontal Beam Holes. There are three horizontal 153-mm I.D. test holes (designated HB-1, HB-2, and HB-3) which terminate at the north face of the reactor. These holes are located above HT-2, and pass through 242-mm I.D. penetrations in the pressure tank walls similar to that for the horizontal through-holes.

- Vertical Test Holes. Two 203-mm I.D. vertical test holes, designated V-1 and V-2 are provided, one at each end of the core. The walls of these holes are aluminum tubes which penetrate the top cover of the reactor tank and terminate near the bottom of the core.

- Thermal Column. Provision is made for a thermal column in the hole designated TC, on the same side of the core as HT-1, through a 1041-mm-diameter penetration in the tank wall. Horizontal beam (HB) tubes, numbers 4, 5, and 6, are located inside the TC.

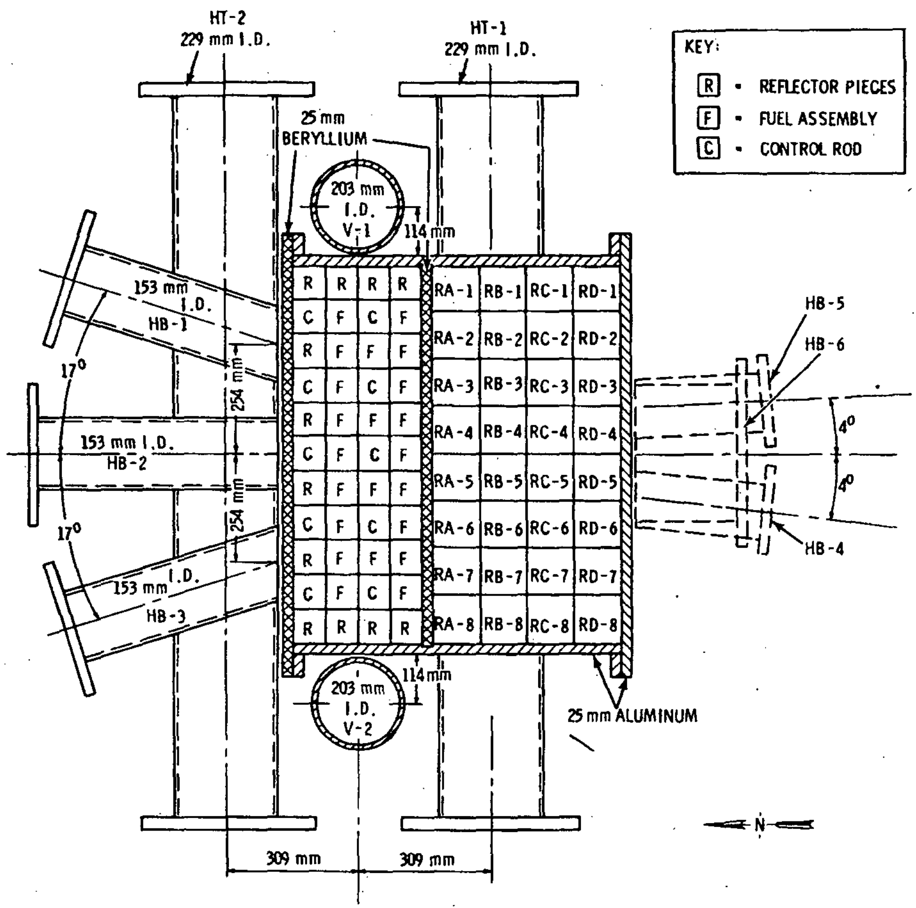
- Rabbit Tubes. Two pneumatic rabbit tubes, designated RP-1 and RP-2, are illustrated in Figure C.3-4. The carrier tubes are 19-mm I.D. and pass the core horizontally and parallel to HT-2. One is located below HT-2 and the other just above the horizontal beam holes. These are through-tubes and penetrate the pressure tank wall on both ends. There are three hydraulic rabbit tubes, designated RH-1, RH-2, and RH-3, which have a carrier tube inside diameter of 19 mm and terminate in the beryllium blocks surrounding HT-1. They are located in positions RA-1, RA-8, and RD-5.

C.3.1.2 Mock-Up Reactor (MUR)

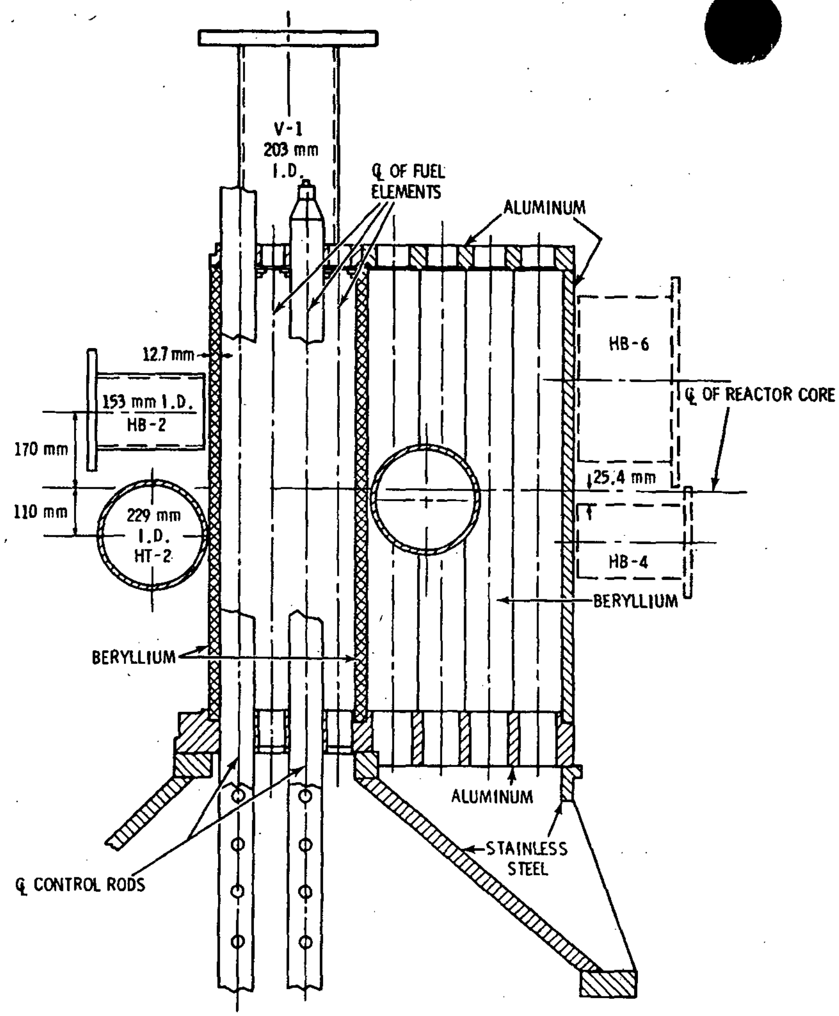
The MUR is a low-power, highly enriched, water-cooled and moderated, beryllium-water reflected reactor. The reactor core is submerged in Canal H, a demineralized light-water pool located in the southeast corner of the main Reactor Building, just outside of the Containment Vessel. Figure C.1-6 shows the location of the MUR in relation to the reference test reactor. The layout of the core and experimental facilities is shown in Figure C.3-7, with a vertical section view of the MUR shown in Figure C.3-8. The core is elevated above the floor of the canal to allow water to circulate under the core for convective cooling. The support stand provides the core support base. The control rod connections are made to one of two movable control bridges which straddle the canal. The MUR is controlled from an enclosed control room which overlooks the canal.

The MUR water pool is constructed of reinforced concrete and is backed by earth to grade level. The pool can be drained into either the Hot or Cold Retention Basins. The walls of the pool are made of reinforced concrete and are epoxy coated to minimize leaching and make decontamination somewhat easier. Walkways are provided on three sides of the pool. A 19.1-Mg overhead crane, with a maximum hook clearance of 4.88 m above grade, services this section of the canal system. Two traveling bridges service the MUR pool surface area. The platform of one bridge is raised several feet above grade level and supports the reactor control rods. The platform of the other bridge can be dropped to near the pool surface and provides working space above the reactor. The difference in height between the control bridge and the working platform of the personnel bridge provides access to the core with a minimum of bridge interference. Full access to the core can be achieved by moving the submerged instrumentation clear of the core, unlatching the control rods, and then moving the control bridge to another section of the pool.

Canal H, 6.4 m by 6.4 m by 7.62 m deep, has its own water-circulating and deionizing system. The system consists of two mixed-bed deionizers, two afterfilters, a Y filter, heat exchangers for heating or cooling the pool water, and a 10-hp pump, which recirculates 0.38 m^3 per minute from the pool.



HORIZONTAL SECTION



VERTICAL SECTION

FIGURE C.3-7. Mock-Up Reactor Core--Horizontal and Section Views

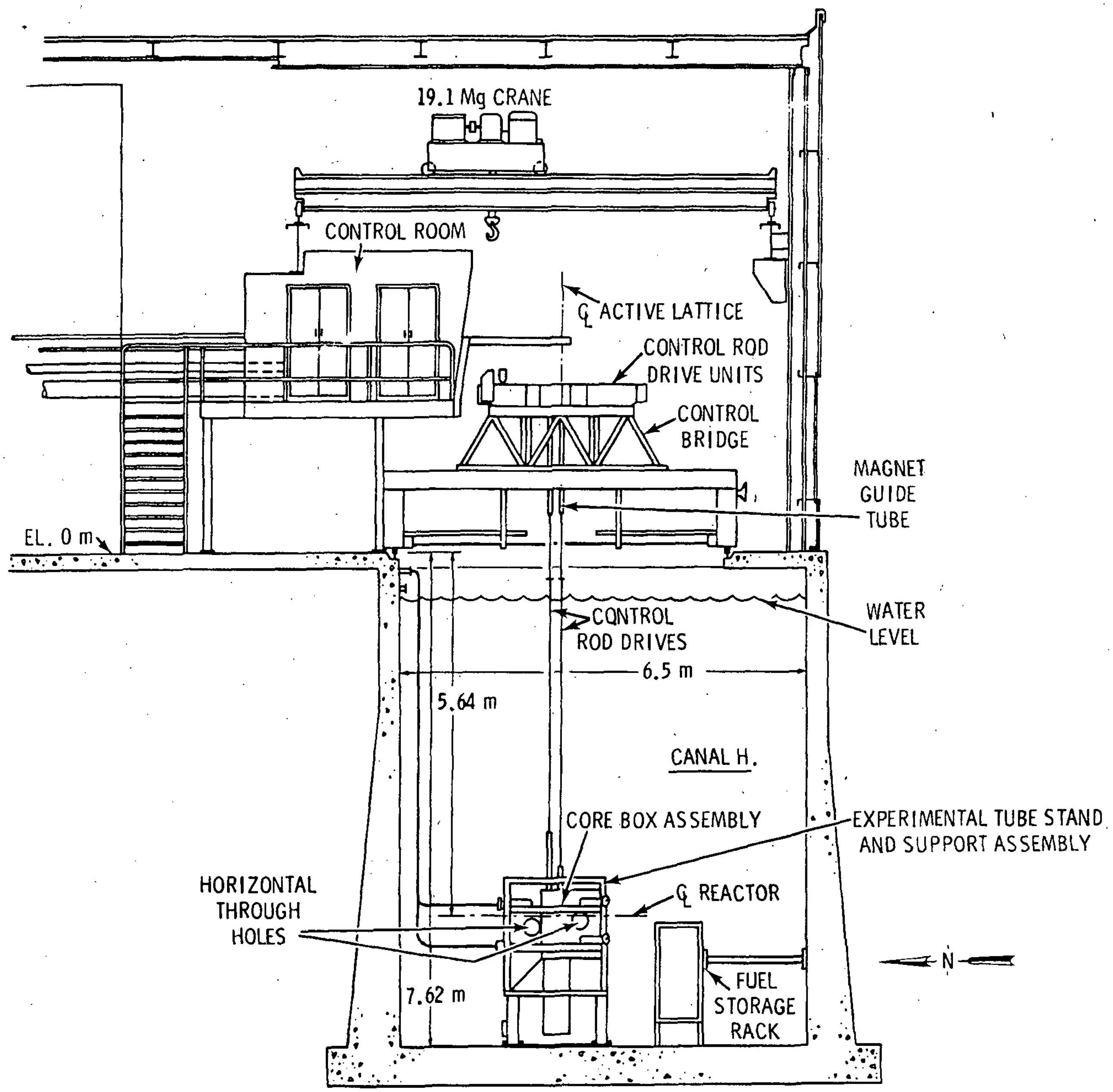


FIGURE C.3-8. MUR--Vertical Section View

The core structures of the reference test reactor and the MUR are identical in the nuclear sense (when the same fuel and control rod elements are used in each). The beam and test holes are located in the same places but provision is made to load all the MUR test holes by manipulating the specimens from the

surface of the pool. As discussed in Section C.2.1.2, the two reactors are connected via a system of canals to facilitate the transfer of irradiated experiments or specimens.

MUR Core Box. The core box, which is essentially identical with the core box in the reference test reactor, is contained within the experimental tube support assembly. There is a mock-up of the flow guide and the control rod box immediately under the core. The core box bottom is the lower grid plate. The sides are completely analogous to the reference reactor core: north plate, south plate, and two end plates with an internal flow divider plate. The upper grid plate completes the core box assembly. All plates are secured with bolts.

The entire core box with beryllium, beam tube mock-ups, flow guide, rod box, and support frame is estimated to weigh less than 4,550 kg.

Core Support Structure. The core support is composed of two welded structures attached together and mounted on a base plate. The top weldment is rigidly attached to the lower weldment and to the lower grid plate of the core box structure. The bottom weldment is mounted on a base plate which is supported by the Canal H floor. The base plate which establishes the reactor vertical axis is leveled, accurately positioned, and securely fixed to the floor to assure rigidity and performance of mounting position. Ten control rod guide tubes with dashpot-type shock absorbers extend below the core.

Fuel Storage Rack. A fuel storage rack consisting of a 4 by 8 array of cells on 127-mm spacings, is located on the south side of Canal H. Each cell is lined with stainless steel and is surrounded on all four sides by 0.5 mm of cadmium. Openings for adequate cooling are provided on the top and bottom of the storage rack. The rack rests on the floor of Canal H and is bolted to the south wall of the pool. The rack is provided with a lid which is fastened shut when fuel transfers are not being made.

Experiment Storage Rack. An experiment storage rack for underwater storage of irradiated experiments and components is located east of the fuel storage rack on the south side of Canal H. The storage rack consists of an array of compartments of various sizes. The rack contains two 305-mm by 305-mm, two 152-mm by 812-mm, eight 102-mm by 102-mm, and eight 76-mm by 76-mm storage compartments. Each storage compartment is about 0.91 m in length. The rack rests

on the floor of Canal H and is bolted to the south wall of the pool. The rack has openings so that when the pool is drained the rack is emptied of water.

C.3.2 Hot Laboratory Building Equipment

The Hot Laboratory Building (HLB) houses the equipment used for handling radioactive experiments, equipment, and materials coming from the reference test reactor. The major pieces of radioactively contaminated equipment in the HLB are found in the hot cells. Experience and test requirements define the type of equipment used in the hot cells. The purpose of each cell, along with the contaminated equipment in each cell and its use, are listed in Table C.3-3, including estimated dimensional and weight data. Contaminated items of potential salvage value are highlighted in Table C.3-3.

Equipment common to all cells includes:

- Oil-filled non-browning glass window (shielding equivalent to the operating face wall, i.e., 2 mR/hr radiation level for a nominal 1×10^6 Ci source of 1 MeV gammas) for each cell.
- ANL Model 8 master slave manipulators and stepped plugs to accept periscopes or optical relays. One set of General Mills manipulators, shown in Figure C.3-9, and a 0.91-Mg crane services Cells 1 and 2. A second General Mills manipulator and 0.91-mg crane services Cells 3 through 7.

C.3.3 Fan House Equipment

The Fan House ventilation equipment (compressors, tanks, fans, and monitoring equipment) is contaminated by filtration, monitoring, and venting air from potentially contaminated areas of the reference test facility. The contaminated ventilation equipment is described in detail in Section C.5.

The waste cleanup system is located in the concrete shielded room in the southwest corner of the Fan House (see Figure C.2-9). The system is composed of a Warner Lewis water filter rated at 0.565 m^3 per minute and a 1.13-m^3 mixed-bed deionizer. The tank is 304 stainless steel and is 1.05 m in diameter by 1.83 m high. In the resin pit just outside the Fan House are two 7.1-m^3 contaminated resin storage tanks.

TABLE C.3-3. Hot Laboratory Building: Hot Cells Contaminated Equipment(a)

Cell Identification Number	Cell Purpose	Equipment and Use	Estimated Overall Dimensions (each), L x W x H, m	Estimated Volume (each) m^3	Mass (each), kg	Salvagable(b)	Remarks
Cell 1	Cutoff and Dismantling	A Model 302 Allicon Campbell abrasive cutoff machine. Used for preparing metallographic specimens and performing severing and milling operations	0.51 x 0.76 x 0.92	0.36	140	No	Commercial Unit, one-piece, floor mounted. Power and piping to be disconnected. Be certain tank is empty before moving
		Portable power tools. Modified for manipulator usage by the addition of grips and fittings	---(c)	---(c)	4.5 to 6.8	No	Size varies. Small, hand-carried
		A Kollmorgen periscope which has 356 mm of lead shielding within the through-tube to provide collimation of streaming radiation. Used for postirradiation examination of experiment components	2 pieces: 0.22 dia. x 1.35 and 0.15 dia. x 3.28	0.11	90	Yes	
Cell 2	Machining	Engine lathe, modified for electromechanical control from a console in the cold work area. A chip collection unit is used in conjunction with the lathe	0.91 x 0.46 x 1.22	0.51	680	Yes	Assumed to be attached to the cell floor. Except for the mounting frame it can be dismantled into portable-size parts
		A Rockwell hardness tester modified for use by the master slave manipulator is used to determine material hardness prior to machining	0.3 x 0.46 x 0.61	0.08	34	No	Portable as a single unit
Cell 3	Physical Testing	An electromechanically controlled tensile testing facility consisting of two testing assemblies. Servo drives provide guage length and rapid return control, extension and/or compression cycling control, and establishment of initial guage length by manual operation, all from an out-of-cell control console	0.92 x 0.61 x 1.83	1.03	1810 to 4530	Yes	Tensile frame is mounted above. Unit can be partially dismantled
		A Bausch and Lomb Stereomicroscope has been sized to be interchangeable with a Kollmorgen scanning periscope in a stepped optical relay sleeve. The apparatus is inserted in the stepped plug located midway between viewing windows of Cell 3. The stereomicroscope has a range of 1/2 to 30 power and is used for inspection and photography of specimens	0.15 dia. x 0.02	0.06	45	Yes	
Cell 4	Metallagrapnic Preparation	Three Syntron vibratory polishing units(d)	0.3 x 0.3 x 0.3	0.03	30	No	
		Buehler ultrasonic cleaning unit(d)	0.2 x 0.2 x 0.3	0.02	3	No	
		Buehler Powermet mounting press(d)	0.3 x 0.3 x 0.46	0.04	23	Yes	—
		Beuhler electropolisher and etching unit(d)	0.46 x 0.3 x 0.61	0.09	16	No	Recirculating fluid and storage tank is used during operation. Be sure all fluid is drained before dismantling
		A Kollmorgen periscope facilitates placement and inspection of specimens	2 pieces: 0.22 dia. x 1.35 and 0.15 dia. x 3.28	0.11	90	Yes	

TABLE C.3-3. (contd)

Cell Identification Number	Cell Purpose	Equipment and Use	Estimated Overall Dimensions (each), L x W x H, m	Estimated Volume (each) m^3	Mass (each), kg	Salvagable[b]	Remarks
Cell 5	Metallography	Modified version of a Bausch and Lomb metallograph	0.61 x 0.3 x 0.3	0.06	114	Yes	The metallograph is equipped with an L-shaped optical relay which mates with an out-of-cell viewing unit. Electromechanical control of the metallograph at a front-of-cell console is accomplished through use of 10 to 12 selsyn drive motors and flexible mechanical linkages. Optical lenses and other accessories are valuable
		Modified version of a Wilson Model LR Tukon microhardness tester	0.46 x 0.3 x 0.46	0.07	36	No	The microhardness tester is equipped with a "U" type optical relay and is electromechanically controlled from a front-of-cell console
		Modified version of a NUMEC cathodic etcher	0.61 x 0.61 x 1	0.04	114	No	An out-of-cell console provides for complete electromechanical control of the in-cell cathodic etcher unit which consists of the etching apparatus and vacuum system
Cell 6	Chemical Analysis	Equipment for chemical analysis includes: a dual delivery unit Beckman titrator, two Brinkman remote control burettes, two Research Equipment Company powered rod-runner units on remote control rotating ring stands, magnetic stirrers, Atkins electronic thermometers, centrifuge and a modified analytical balance.	---(c)	---(c)	---(c)	No	In addition, tailor-made glassware comprises much of the Cell 6 equipment
Cell 7	Analytical Measurement	A thin crystal scintillation chamber is used for x-ray spectrometry and a quartz monochromator setup is used for x-ray diffraction analysis.	1 x 1 x 1	1	500	Yes	A single power supply provides excitation for both x-ray tubes. Equipment can be disassembled except for HV transformer and mounting base. Equipment consists of goniometer, sample holder, crystal relay, large power supply (50 kV) and sophisticated detector recorder system

(a) Technical data estimates are based on same or similar equipment, which are considered typical for these hot cells.
(b) Based on engineerng judgement. For each component the final determination is based on a cost-benefit analysis.
(c) Data not available.
(d) These equipment pieces have been modified for remote operation from a front-of-cell console.

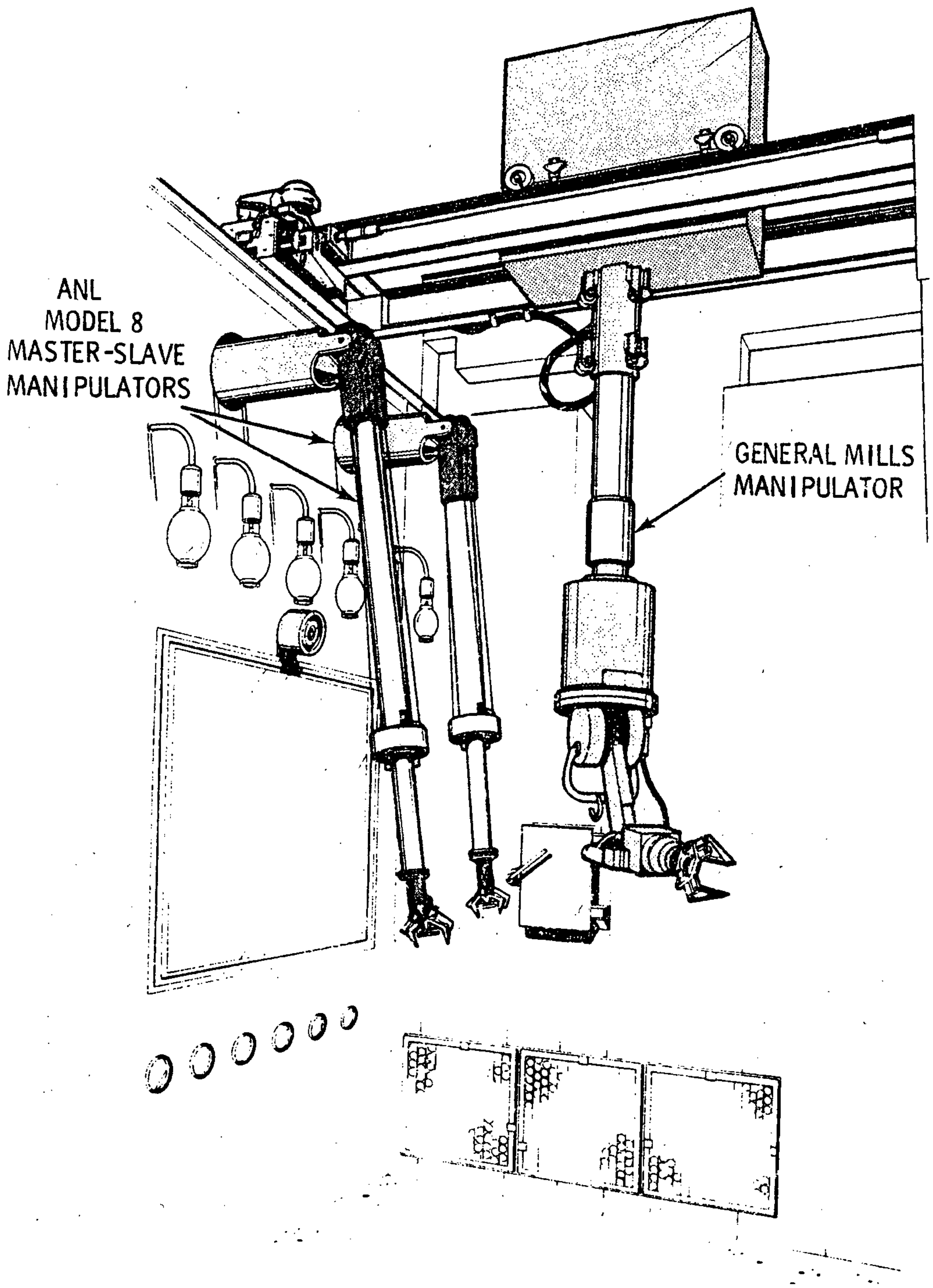


FIGURE C.3-9. General Mills Manipulator Serving Cells Number 1 and 2

Quadrant and canal cleanup equipment--filters and deionizers--are located in the basement of the Fan House.

C.3.4 Waste Handling Building Equipment

The Waste Handling Building (see Figure C.2-9) contains the liquid waste evaporator system with associated boiler, condenser, sumps, filters, and pumps. In addition, it contains the contaminated laundry equipment, gantry room, waste packaging equipment and waste storage facilities. The process equipment is designed to treat low-level liquid wastes from the reference test facility. The major items of contaminated equipment are listed in Table C.3-4.

TABLE C.3-4. Contaminated Equipment: Waste Handling Building

Equipment Piece	Estimated Mass (kg)	Estimated Volume (m^3)	Estimated Overall Dimensions L x W x H (m)
Solid Waste Hydraulic Baler	900	2.8	1.3 x 0.8 x 2.7
Washing Machine	90	1	1.0 x 1.0 x 1.0
Dryer	70	1	1.0 x 1.0 x 1.0
Concrete Mixer	---[(a)]	---	---
Waste Canner	---	---	---
Evaporator (skid-mounted)	3400	$\sim$21[(b)]	1.82 x 3.1 x 6.1

(a) Data not available.
(b) Volume indicated is packing volume after dismantlement of the evaporator unit.

There are two hot sumps in the Waste Handling Building. The laundry area drains into one and the floor drains in the building drain into the other. These sumps and associated contaminated equipment are described in detail in Section C.3.7.

C.3.5 Primary Pump House Equipment

The Primary Pump House (see Figure C.2-10) contamination results from pumping primary coolant water (PCW) from the reactor core. The PCW contains dissolved and suspended activation products. The reference facility design, therefore, places the pumps, heat exchangers, deionizers, and degassifier

system in thick-concrete shielded rooms. The Primary Pump House contaminated equipment is listed in Table C.3-5. With the exception of the sump and the Hot Spent Resin Pit, all of the equipment listed in this table is located in one or more of the six heavily shielded equipment cells.

The PCW system piping and valves and other ancillary equipment are described in Section C.3.7.

TABLE C.3-5. Contaminated Equipment: Primary Pump House

Equipment Piece (quantity)	Estimated Mass (each), kg	Estimated Volume (each), m^3	Estimated Overall Dimensions L x W x H (m)
Primary Pumps (3)[(a)]	4 090	3.52	1.52 x 1.52 x 1.52
Primary Heat Exchanger (2)	9 090	5.8	1.07 dia. x 6.4
Deionizers (2)	1 500	2.4	1 dia. x 3.05
Strainer	430	1.5	1.12 dia. x 1.53
Flow Venturi	200	0.27	0.61 dia. x 0.91
Degassifier System			
Degassifier Recirculation Pump Steam Generator Superheater Recombiner Condenser Vacuum Pump Moisture Separator Water-Cooled Compressor Gas Storage Tank Condensate Pump Filter Unit	17 160 (total)	29 (total)	---[(b)]

(a) Does not include the primary pump motors; they are not anticipated to be contaminated since they are located outside the thick concrete shielded rooms that house the primary pumps (see Figure C.2-10).
(b) These pieces of equipment are of various shapes and sizes and are estimated to require a total of about 29 m^3 of container volume for transport and subsequent burial.

C.3.6 Office and Laboratory Building Equipment

The low-level radiochemistry laboratories, located on the second floor of the Office and Laboratory Building (OLB), contain 22 contaminated laboratory hoods. Typical exterior dimensions of a radiochemical fume hood are 1.5 m wide by 0.9 m deep by 2.1 m high. Each hood has a 610-mm by 610-mm by 305-mm absolute filter in a sealed housing above the hood. Exhaust fans on the roof service the hoods. The hoods are discussed in greater detail in Section C.3.8. In addition, there are two hot collection sumps in the basement of the OLB which lead to hot or cold retention areas for ultimate disposal. The equipment associated with the sumps is described in Section C.3.7.3.

C.3.7 Contaminated Water Systems Equipment, Piping, and Valves

Upon final shutdown of the reference test reactor, radioactive contamination is expected in certain piping and equipment systems in the Reactor Building, the Primary Pump House, the Fan House, and the Hot and Cold Retention areas. This contamination (both deposited on the internal surfaces and included in the contents of these systems), if allowed to remain, would pose a hazard to decommissioning personnel and the environment and would deter effective completion of decommissioning activities.

Brief descriptions of the contaminated water systems follow, including estimates of piping and valves in plant systems that handle contaminated contents.

C.3.7.1 Primary Cooling Water System (PCWS)

The PCWS is sub-divided into four loops: 1) the main loop which circulates through the reference test reactor, 2) the bypass cleanup loop, 3) the instrument and test hole cooling loop, and 4) the shutdown loop.

The main loop is a closed loop containing 98.4 m^3 of deionized water. The bypass cleanup loop is a secondary loop on the main loop and is used to control the purity of the water in this system. There are two mixed-bed deionizers with an auxiliary heat exchanger and two pumps for circulation during shutdown. The instrument cooling loop is a secondary loop on the main and

shutdown loops which supplies cooling water to instrument thimbles and experimental test holes within the reactor vessel. The shutdown loop is an auxiliary loop on the reactor vessel which circulates the water through a heat exchanger and two pumps during reactor operation and cooldown and provides sufficient capacity for decay heat removal after shutdown. Flow from the shutdown loop also supplies the instrument cooling loop.

The primary main system supply and return lines from the reactor vessel are 610-mm-diameter, 12.7-mm-wall, 304 stainless steel pipes. The individual primary pumps have 457-mm-diameter, 9.53-mm-wall, 304 stainless steel suction lines and 406-mm-diameter, 9.53-mm wall, 304 stainless steel discharge lines.

The estimated amounts of piping and valves in the PCWS are listed in Table C.3-6.

C.3.7.2 Quadrant and Canal Systems (Q&CS)

There are two Q&CSs: 1) the quadrant and canal recirculation system (Q&C-Recirc.), and 2) the quadrant and canal pumpout system (Q&C-Po.). The Q&C-Recirc. system is used to maintain water quality; also, it includes provision for cooling the water. The Q&C-Po. system provides a method of pumping water from the quadrants and canals to the cold retention basins or to the hot retention tanks.

The estimated amounts of piping and valves in these systems are listed in Table C.3-7.

C.3.7.3 Hot Drain System, Retention, and Cleanup Systems

The hot drain system is a collection system for all waste water drainage which originates directly or indirectly from radioactively contaminated areas. There are 12 collection sumps: two each in the Fan House and Office and Laboratory Building, one each in the Primary Pump House and the Hot Laboratory Building, four in the Reactor Building, and two in the Waste Handling Building. All of the connecting lines of this system are doubly contained as protection against leakage, with leakage indication. This is accomplished by pipe tunnels, sheath piping and drains, and dual-wall tanks.

TABLE C.3-6. Estimate of Contaminated Piping and Valves: Primary Cooling Water System

Material	Primary System Loop	Nominal Outside Diameter (mm) 61 to 115	168 to 219	407 to 610	Total
Stainless Steel	Primary Main Loop	30/180/20[a]	---[b]	118/12 425/9[c]	148/12 605/29[d]
	Bypass Cleanup Loop	25/210/70	---	---	25/210/70
	Instrument & Test Hole Loop	20/168/114	---	---	20/168/114
	Shutdown Loop	10/84/10	15/300/9	---	25/384/19
Totals		85/642/214	15/300/9	118/12 425/9	218/13 367/232

(a) The data are presented in the following format: Piping length, m/piping mass, kg/number of valves.
(b) Dashed line represents insignificant amount.
(c) These PCW inlet and outlet pipes are routed to the reactor tank from the Primary Pump House through a pipe tunnel in the concrete foundation mat. The tunnel ends at the concrete biological shield. These pipes are then grouted into the concrete of the biological shield until they enter the reactor tank.
(d) The total number of valves for a particular system does not include small-bore valves 51 mm or less in diameter.

TABLE C.3-7. Estimate of Contaminated Piping and Valves: Quadrant and Canal Systems

Material	Q&C System	Nominal Outside Diameter (mm) ≤61	≤168	Total
Stainless Steel	Q&C-Recirc.	80/288/59[a]	305/4 240/54	385/4 528/113[b]
	Q&C-Po.	---[c]	155/2 155/37	155/2 155/37
Totals		80/288/59	460/6 395/91	540/6 683/150

(a) The data are presented in the following format: Piping length, m/piping mass, kg/number of valves.
(b) The total number of valves for a particular system does not include small-bore valves of 51 mm or less in diameter.
(c) Dashed line represents insignificant amount.

The Hot Retention Area, HRA (discussed in Section C.2.6), is a tank farm for the radioactive decay storage from the hot drain system. The Cold Retention Area, CRA (discussed in Section C.2.7), is a double basin for the storage of water from the quadrant and canal water systems.

The waste cleanup system (located in the Fan House) is composed of a water filter and a mixed-bed deionizer which operates on water from the HRA, CRA, or returns to the quadrant and canal water system, depending on the degree or types of radioactivity remaining in the effluent water.

The following tables list the contaminated equipment and piping and valves in these systems: Table C.3-8, Hot Drain Equipment, Table C.3-9, Hot Drain System Piping and Valves; Table C.3-10, HRA and CRA Equipment; Table C.3-11, HRA and CRA Piping and Valves; and Table C.3-12, summary of the metal piping and valves of the entire plant. Contaminated soil and concrete piping are discussed in Section C.3.4.

C.3.8 Details of Selected Facility Components

As previously discussed, the operations related to chemical and radiochemical analysis are performed in fume hoods in the Office and Laboratory Building (OLB); master-slave manipulators are used to transfer and handle irradiated materials in the shielded hot cells of the Hot Laboratory Building (HLB); and various decontamination areas utilize fume hoods at the reference test reactor. These facility components are described in this section.

C.3.8.1 Fume Hoods

A radiochemical fume hood is shown in Figure C.3-10. Typical exterior dimensions are 1.5 m wide, 0.9 m deep, and 2.1 m high. The distance from floor to working deck is approximately 0.9 m. The fume hoods used for work in the OLB exhaust through an absolute filter in a sealed housing above the hood. The location of the filter makes it readily replaceable with little possibility of contaminating the work area.

A variety of materials are used for fume hood construction. Older hoods may be of wood or steel frame construction with masonite or asbestos board

TABLE C.3-8. Estimate of Contaminated Equipment: Hot Drain System

Location	No. of Sumps[a]	Sump Size (m) or Capacity (ℓ)	Sump Pump Parameters: No. of Pumps	hp	gpm	ℓpm	Mass (kg)	Volume (m^3)	Overall Dimension L x W x H (m)
Reactor Building									
Sub-Pile Room	1	378 ℓ	1	1/2	9	34			
			1	1/2	9	34			
			1	1-1/2	40	151			
Process Piping Pump Room (-7.62 m Elev)	1	2.14 I.D. x 2.29 dp	1	7-1/2	200	757			
			1	15	500	1892			
-4.6 m Elev	1	---[b]	1	1	50	189			
			1	1	50	189			
Test Corridor or Dry Annulus	1	---	1	1	---	---			
			1	1	---	---			
Office And Laboratory Building									
Room 11	2	---	1	1	50	189			(b)
			1	1	50	189			
Fan House	2	378 ℓ to 750 ℓ/each	1	1	50	189			
			1	1	50	189			
Primary Pump House	1	214 I.D. x 2.29 dp	1	2	100	378			
			1	5	200	757			
Waste Handling Building	2	---	1	1	50	189			
			1	1	50	189			
			1	1	100	378			
			1	1	---	---			
Hot Laboratory Building	1	---	1	3/4	50	189			
			1	3/4	50	189			

(a) All sumps are assumed to be lined with stainless steel.
(b) Data not available.

TABLE C.3-9. Estimate of Contaminated Piping and Valves: Hot Drain System

Location	Nominal Outside Diameter (mm) <61	61 to 115	168 to 219	≤324	Total
Reactor Building	30/108/6[a]	200/897/23	145/2 564/3	75/3 735/6	450/7 304/38[c]
Office & Laboratory Building	90/324/9	---[b]	---	---	90/324/9
Fan House	---	30/108/8	---	---	30/108/8
Primary Pump House	90/324/18	45/378/0	30/600/0	---	165/1 302/18
Waste Handling Building	---	45/162/10	---	---	45/162/10
Hot Laboratory Building					(d)
Totals	210/756/33	320/1 545/41	175/3 164/3	75/3 735/6	780/9 200/83

(a) The data are presented in the following format: Piping length, m/piping mass, kg/number of valves.
(b) Dashed line represents insignificant amount.
(c) The total number of valves for a particular system does not include small-bore valves 51 mm or less in diameter.
(d) Insufficient data available to make a determination.

TABLE C.3-10. Estimate of Contaminated Equipment and Materials: Hot Retention Area and Cold Retention Area

Contaminated Item	Dimensions (each), m	Surface Area (each), m	Estimated Weight (each), kg	Tank Pump Parameters (each): Number of Pumps/Tank	hp	gpm	pm	Mass, kg	Volume, m	Overall Dimensions, L x W x H, m
Hot Retention Area										
Storage Tanks[a]										
Nos. 1 through 8	6.7 dia. x 7	217.85	16 404	1	7.5	200	53		(b)	(b)
Nos. 9 through 12	2.44 dia. x 6.1	56.11	4 225	1 (total)	5	100	26		(b)	(b)
Miscellaneous Structural Steel[c]										
Floor Plate	29.9 x 15.24	456	22 272	N.A.[d]						
Partition Plate	0.91 x 45.1	41	2 045	N.A.						
Cold Retention Area[e]										
Storage Basins										
Nos. 1 and 2	5.5 dp x 28.6 m^2	--[f]	--	1	25	1 500	396		(b)	(b)

(a) Tank wall thickness is 9.525 mm, double wall and epoxy-coated on the inside of the inner tank.
(b) Insufficient data available to make an estimate.
(c) Partition plate and floor plate thickness is 6.35 mm.
(d) N.A. means not applicable.
(e) Each basin has a concrete floor lined with plastic.
(f) The estimated amount of contaminated concrete is given in Table D.2-3 of Appendix D.

TABLE C.3-11. Estimate of Contaminated Piping and Valves: Hot Retention Area and Cold Retention Area

Location	Nominal Outside Diameter (mm) 89	115	168	219	Total
Hot Retention Area					
Buried Underground	60/678/0[a]	38/612/0	30/849/0	120/5 112/0	248/7 251/0
In Concrete Bunker	80/905/8	25/403/0	120/3 396/14	130/5 538/16	355/10 241/38
Cold Retention Area	---[b]	---	185/5 236/7	---	185/5 236/7
Totals	140/1 582/8	63/1 015/0	335/9 481/21	250/10 650/16	788/22 728/45

(a) The data are presented in the following format: Piping length, m/piping mass, kg/number of valves.
(b) Dashed line represents insignificant amount.

TABLE C.3-12. Summary of Estimates of Contaminated Piping and Valves

Location or System	Nominal Outside Diameter (mm) ≤61 to 115	168 to 219	≤324	407 to 610	Total
PCW Systems	85/642/214[a]	15/300/9	---[b]	118/12 425/9	218/13 367/232[c]
Q&C Systems	80/288/59	460/6 395/91	---	---	540/6 683/150
Hot Drain System	530/2 301/74	175/3 164/3	75/3 735/6	---	780/9 200/83
HRA & CRA	203/2 597/8	585/20 131/37	---	---	788/22 728/45
Totals	898/5 828/355	1235/29 990/140	75/3 735/6	118/12 425/9	2326/51 978/510

(a) The data are presented in the following format: Piping length, m/piping mass, kg/number of valves.
(b) Dashed line represents insignificant amount.
(c) The total number of valves for a particular system does not include small-bore valves 51 mm or less in diameter.

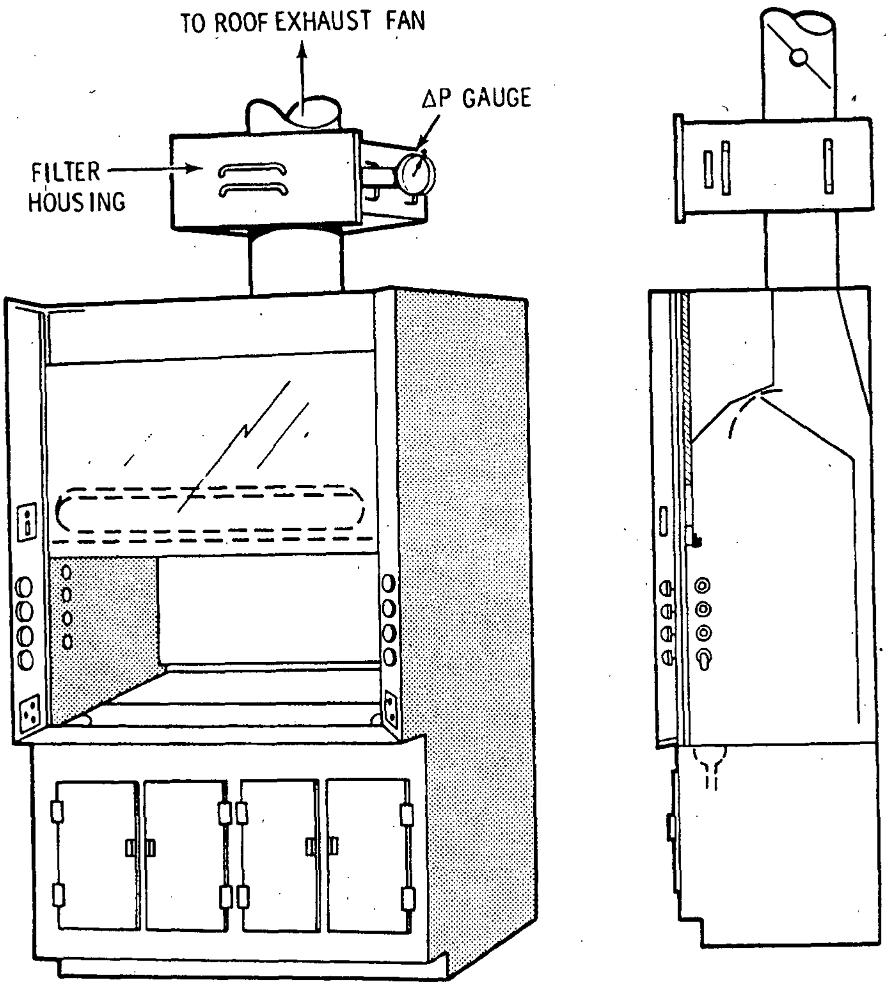


FIGURE C.3-10. Radiochemical Fume Hood

floors and walls. Masonite and asbestos board surfaces are usually painted with a chemical-resistant plastic-base paint. New hoods are usually of steel construction with interior surfaces of stainless steel, aluminum, or masonite. The hood window may be fabricated from an acrylic plastic.[a]

The air velocity through the hood face is typically 0.5 m/sec with the window open 610 mm. Air flow control is maintained by an exhaust fan and by manually operated and/or motor-driven louvers located in the exhaust ductwork

(a) Lucite and Plexiglass, both trade names, are two commonly available acrylic plastic materials.

above the hood. When the hood window is lowered, air is drawn into the hood through auxiliary vents above the window. Baffling is required to direct the air flow and to ensure uniform face velocity. This usually consists of a back baffle with a slot opening at the bottom, one at the top, and frequently one in the middle. Slot widths are usually adjustable and are set to maintain slot velocities from 5 to 10 m/sec.

In some radiochemical fume hoods, the hood face design is modified as shown in Figure C.3-11. The top portion of the hood face consists of a viewing panel made of glass or acrylic plastic. The bottom portion consists of acrylic plastic panels that slide horizontally to permit access to the interior of the hood.

Service outlets for gas, water, steam, vacuum, and compressed air are located along the side walls or the back of the hood and are controlled through knobs at the front of the hood. Electrical outlets and switches are located on the outside of the hood. Hoods are usually designed with a raised sash along the front edge to contain material that is spilled on the hood floor.

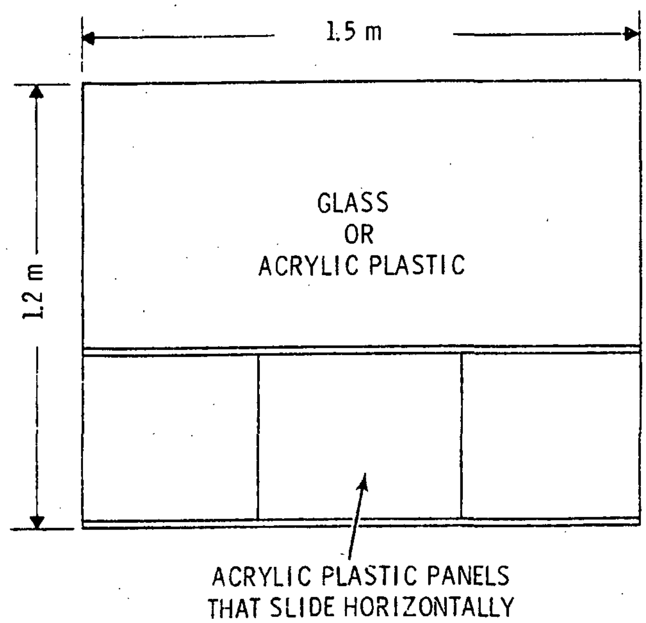


FIGURE C.3-11. Hood Face for Radiochemical Fume Hood

Exhaust ductwork may be of stainless steel, glass fiber, asbestos pipe, iron, aluminum, or sheet metal. Some surfaces (e.g., iron or aluminum) may be finished with several coats of corrosion-resistant plastic-base paint. Ducts fabricated from stainless steel have the advantage of not requiring a supplemental coating for corrosion resistance, although they are susceptible to attack from halogens and halogen acids.

C.3.8.2 Hot Cells

There are seven hot cells in the Hot Laboratory Building. Each hot cell is a shielded enclosure used for work with high levels of beta and gamma radioactivity where personnel protection from penetrating radiation is required.

Hot cells are usually custom built to conform to the requirements of a particular chemical or metallurgical operation. The walls may be constructed with ordinary-density or high-density concrete (either poured in place or as concrete blocks), with lead bricks, or with magnetite aggregate and/or lead shot construction.

Viewing windows are fabricated from lead glass, cerium glass, lime glass, or laminated glass with the space between glass panes filled with a special oil. If the salvage value of the hot cell shielding windows is to be preserved, it should be recognized that both the proper equipment and skilled personnel should be used for their decontamination and extraction. Technique is extremely important with these operations. For example, the surfaces of shielding slabs are easily scratched, chipped, etched or stained.

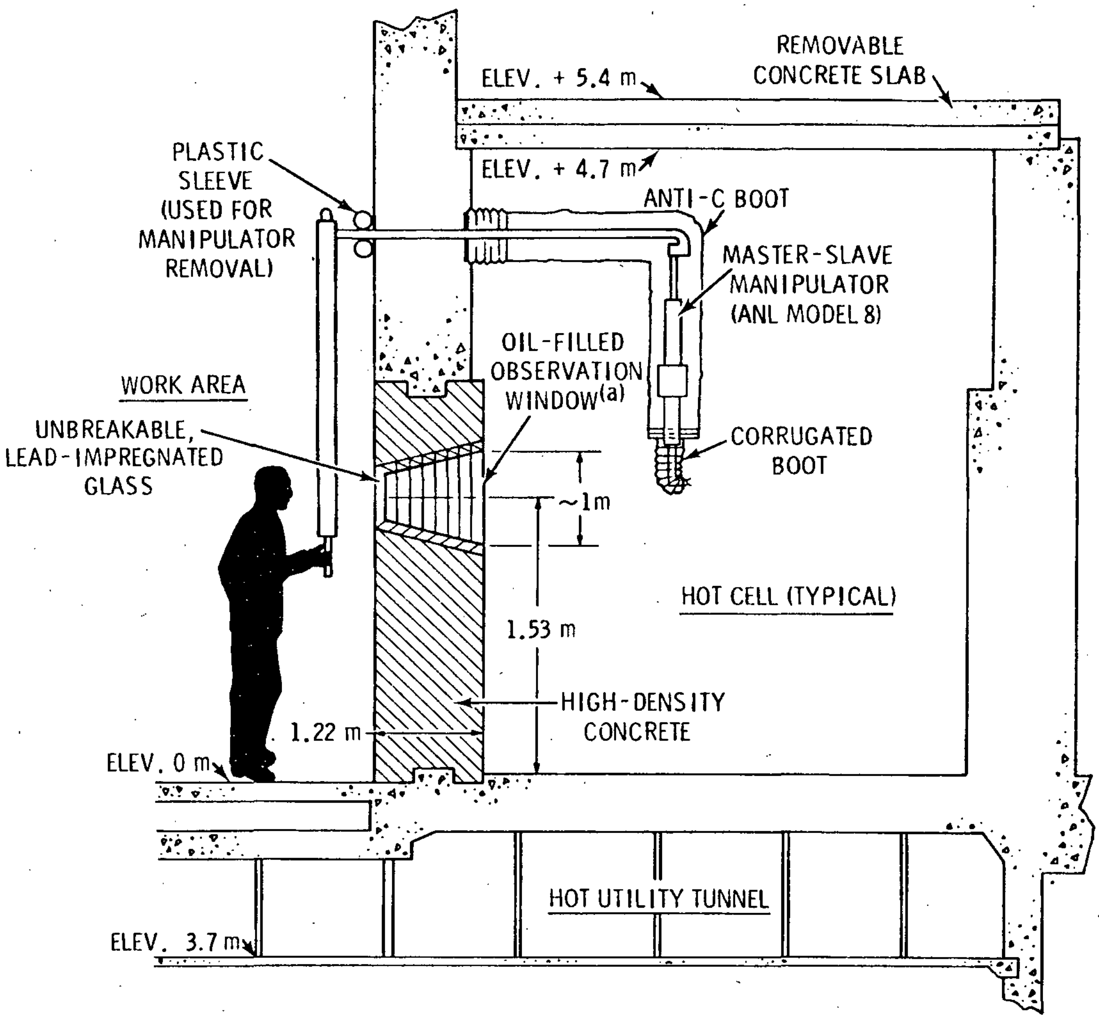
Hot cell shielding windows may be either hot-side loaded or cold-side loaded. If the window is a hot-side load (as is the case for the reference test reactor hot cells), the window is extracted into the hot cell when it is removed. Conversely, a cold-cell loaded window permits extraction into the working gallery without first requiring cell decontamination.

Operations inside a cell are performed with the aid of master-slave manipulators. (Master-slave manipulators are described in Section C.3.8.3.)

The reference hot cell for this study is shown schematically in Figure C.3-12. The cell structural details are described in Section C.2.2.1 and are not repeated here.

C.3.8.3 Master-Slave Manipulators

A master-slave manipulator is a mechanical arm used by chemists and metallurgists to operate equipment and to execute processing operations safely



(a) HOT-SIDE LOADING WINDOW IS ILLUSTRATED (TYPICAL OF HOT CELL WINDOWS IN THE REFERENCE TEST REACTOR'S HOT LABORATORY), NEWER VERSIONS ARE COLD-SIDE LOADING WINDOWS, WHICH MEANS THEY CAN BE REMOVED AND/OR INSTALLED FROM THE WORK AREA.

FIGURE C.3-12. Hot Cell

within the confines of hot cells. In general, motions of the slave end of the manipulator correspond directly with motions performed by the operator at the hand end. Descriptions of several master-slave manipulators are given in Reference 6. A manipulator that is typically used in the reference hot cells is shown in Figure C.3-13.

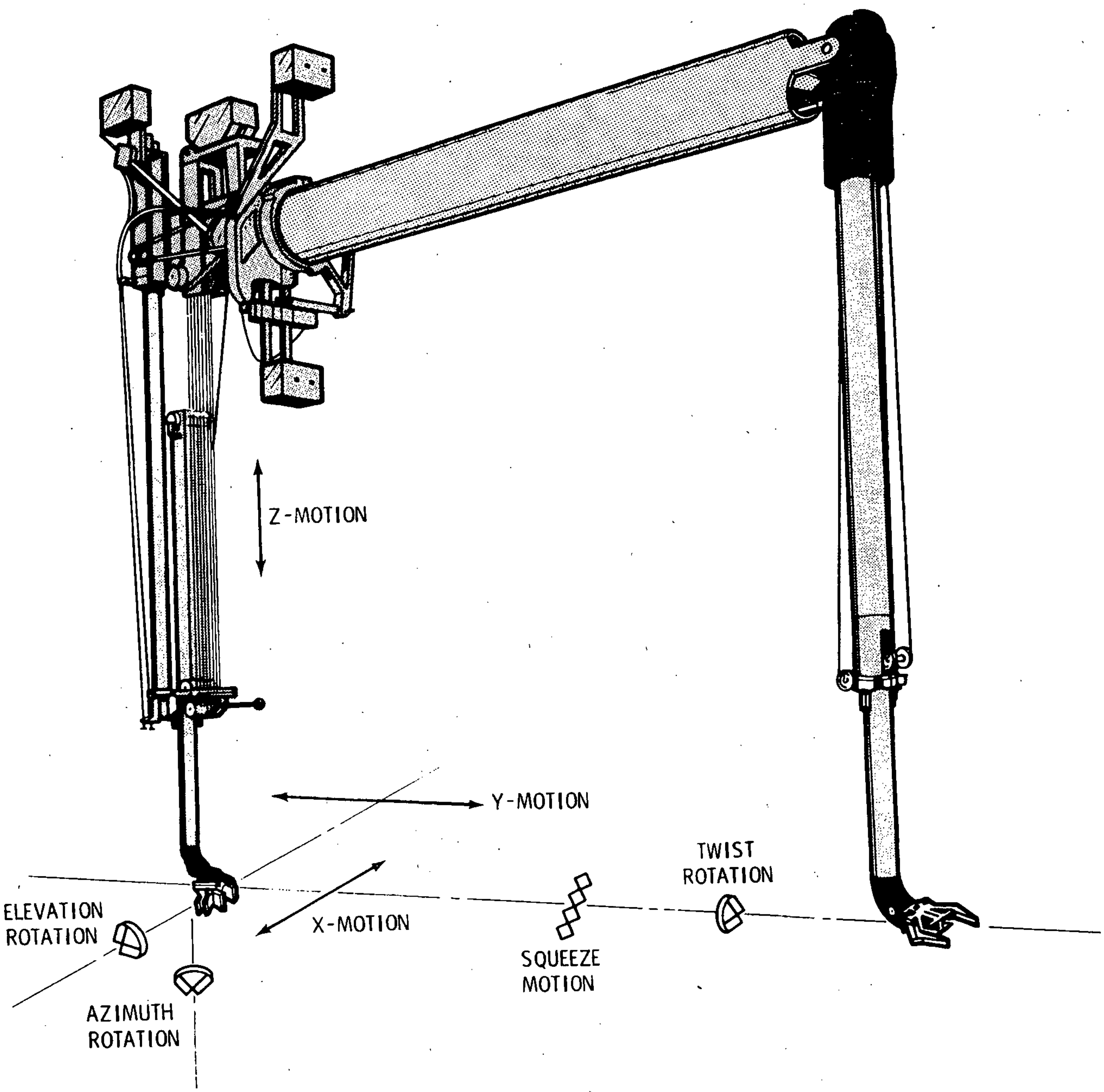


FIGURE C.3-13. Master-Slave Manipulator

The manipulator has seven independent motions: three for rotational orientation, three for movement in space, and a squeeze motion. The positional movements are achieved by parallelogram coupling of the master and slave arms. Vertical motion is provided by steel tapes. The wrist rotational motions and the tong squeeze use stainless-steel cables. Lateral motions are provided by rollers that support the manipulator. Moderately low friction is achieved by using ball bearings for all the major movements. To operate the manipulator, the operator simply grasps the handle with one hand, using the palm and lower fingers for general movement of the manipulator in its six motions and the thumb, index, and middle fingers for the squeeze motion.

Manipulators are generally used in pairs having right- and left-hand handles. One manipulator can cooperate with and aid the other in getting a suitable grasp or orientation on an object or in lifting a heavy object.

All master-slave manipulators are booted to retard the spread of contamination. The boots can be slipped off in-cell before transfer of the manipulator out of the cell.

C.4 CONTAMINATED SOIL, CONCRETE PIPING, AND ASPHALT

Three areas of the reference test reactor are anticipated to require the removal of contaminated soil. The first area is the contaminated drainage ditches, the second area is the Emergency Retention Basin (see Figure C.1-1), and the third area is the soil beneath the concrete floor slabs of the Cold Retention Basins (see Figure C.2-12). Nominal amounts of contaminated soil and asphalt, as well as significant amounts of concrete piping, are postulated to be removed from the contaminated ditches. On the other hand, significant amounts of contaminated soil are postulated to be removed from the Emergency Retention Basin and the Cold Retention Basins. These areas are described in the following subsections.

C.4.1 Contaminated Drainage Ditches

There are approximately 1,040 m of ditches at the reference test reactor facility that contain contaminated concrete piping, asphalt, and soil. Originally, these ditches consisted of open V-shaped asphalt-lined trenches. Later,

three different sizes of concrete pipes were laid on the asphalt in the various ditches and the entire system was covered with soil. The concrete piping from the areas listed in Table C.4-1 is removed. In addition, a nominal amount of asphalt and soil is anticipated to be contaminated and also is removed. During packaging, the latter two materials are packaged with the concrete piping to provide a reduction in the overall total burial volume for this decommissioning activity.

TABLE C.4-1. Estimate of Contaminated Concrete Piping in Drainage Ditches

	Parameters				
Classification of Concrete Pipe	Length (m)	Length per Section (m)	Weight per Section (kg)	Wall Thickness (mm)	Type
Culvert	122	0.92	136	51	Tongue & Groove
	488	0.92	232	77	Tongue & Groove
Well or Man-Hole Pipe	427	1.22	1000	≤102[a]	Tongue & Groove, Straight Wall

(a) Varies depending on the amount of reinforcing bar used in the construction of the pipe.

C.4.2 Contaminated Soil in the Emergency Retention Basin

The Emergency Retention Basin is a 37,800-m^3 above-ground earthen-diked basin, approximately 130 m by 96 m, located at the southeast corner of the reference site. Very low radioactivity levels exist in the soil in this area. It is estimated from samples (see Appendix E) that the soil is contaminated to a depth of 153 mm to 305 mm, depending on location. Soil to an average depth of 229 mm is assumed to be removed with a backhoe equipped with a wide grading bucket and packaged as LSA material for subsequent burial.

C.4.3 Contaminated Soil in the Cold Retention Area

The Cold Retention Area consists of two 1.9 million liter basins, located east of the Fan House. The basins are 5.5 m deep with above grade covers approximately 28.6 m square. Very low radioactivity levels exist in the soil beneath the plastic-lined concrete floor slabs of the basins. It is

postulated that the soil is contaminated to a depth of 153 mm to 305 mm, depending on location. Soil to an average depth of 229 mm is assumed to be removed with a backhoe and bucket.

C.5 HEATING, VENTILATING, AND AIR CONDITIONING SYSTEMS

This section presents information about the various heating, ventilating, and air conditioning (HVAC) systems in the reference test reactor facilities. These systems are designed to minimize the spread of airborne radioactive contamination.

In general, areas throughout the reference test facility that require heating or air conditioning are office areas, general work areas, or uncontaminated operational areas and are handled by the normal ventilation system. If these areas are in close proximity to contaminated areas, the systems are designed to force ventilating air from the uncontaminated areas into the contaminated areas by air pressure differences. Building areas where airborne contamination may be reasonably expected to occur at some time discharge their ventilating air through roughing filters followed by high-efficiency absolute filters.

Descriptions of the normal HVAC systems and of the contaminated air-handling systems for the reference test facility follow. The former are included for illustration only and are not essential to the study results, while the latter are postulated to be integrated into the overall decommissioning plan to ensure safety during its removal, since the facility air-handling systems are complex to operate.

C.5.1 Normal HVAC for the Reference Test Facility

The normal HVAC systems for the Reactor Building, the Office and Laboratory Building, the Hot Laboratory Building, the Primary Pump House, the Fan House, the Service Equipment Building, and the tunnel areas are described in the following subsections.

C.5.1.1 Reactor Building

The Reactor Building (RB) first floor and mezzanine areas are heated by seven steam unit heaters and two fresh-air steam-heated roof intake units. The latter units bring in 680 m^3 per minute of air and distribute the warmed air throughout the area above the 0-m elevation. Exhaust is through a roof ventilator on the south end of the building. Each of the offices, as well as the reference test reactor control room and the MUR control room, is heated or cooled by unit air conditioning. The basement area ventilation and the containment vessel ventilation are described in Section C.5.2.

C.5.1.2 Office and Laboratory Building

The Office and Laboratory Building (OLB) has individual room air-handling units for heating and cooling within the office areas on the first and second floors. The basement has a 127-m^3/min heated fresh air makeup to a multi-zone unit. The laboratory area is supplied with forced air at 170 m^3 per minute from the second floor office area surplus. There are four roof ventilators, one from the basement, one from the washrooms, one from the first and second floor office area, and one with a filter unit ventilator for surplus from the laboratory work area. Additional ventilation in the laboratory area is handled by the OLB laboratory hood exhaust system.

C.5.1.3 Hot Laboratory Building

The Hot Laboratory Building (HLB) has a circulating multi-zone air-handling unit with a 65-m^3/min capacity for heating and cooling the work corridor. In the contaminated area the ventilation is handled by the HLB cell ventilating system (see Section C.5.2) equipped with a fresh air heater unit with 395- to 850-m^3/min-capacity air intake.

C.5.1.4 Primary Pump House

The Primary Pump House (PPH) has three exhaust fans over the primary pump motors and a fresh air steam heater unit with a 94-m^3/min capacity intake. There are seven unit heaters in this corridor area. The contaminated air and off-gas system (see Section C.5.2) handles the remaining ventilating air within the building.

C.5.1.5 Fan House

The Fan House (FH) has a gas-fired fresh air heating unit, and the 91-m^3/min makeup is handled by the Reactor Building hot pipe tunnel ventilating air system. There are two gas-fired unit heaters and two wall heaters in this building.

C.5.1.6 Service Equipment Building

The Service Equipment Building (SEB) is a completely uncontaminated area and is isolated from the other buildings by location and operating functions. The SEB has eleven steam unit heaters and a multi-zone air conditioner for heating and cooling the SEB office, control, and laboratory rooms. There are seven roof exhaust fans and two room exhaust fans.

C.5.1.7 Tunnel Areas

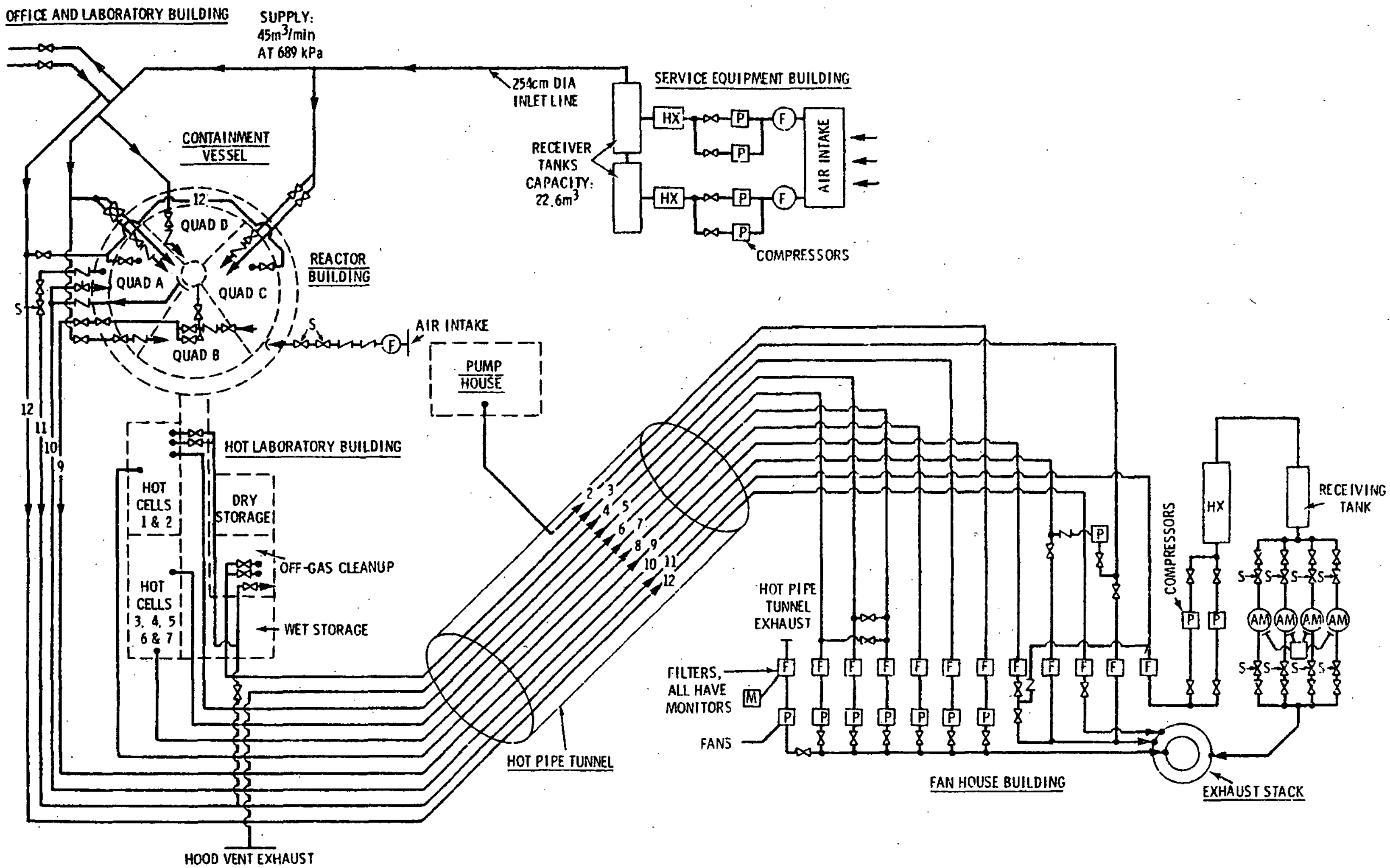
The utility tunnel has a 312-m^3/min-capacity exhaust fan at the cooling tower and the open end from the SEB to the RB is serviced by the RB basement ventilating system. The hot pipe tunnel is in itself a ventilating duct, as described in the contaminated air and off-gas system in Section C.5.2.

C.5.2 Contaminated Air Ventilation Systems

The ventilation systems that service the contaminated areas at the reference test facility are described in this section, and are illustrated in Figure C.5-1. The gaseous waste disposal flow paths are shown in Figure C.5-2.

C.5.2.1 Utility and Service Air Systems

The air supply to contaminated areas in the reference test reactor facilities is provided by the utility air system (UAS). The UAS starts in the SEB (see Figure C.5-1) where the air is filtered, compressed and cooled as a supply to the service air system and to the experimental and test facilities in the RB and the OLB. The used waste air from the experimental and test system is collected, monitored, filtered, and exhausted through the stack at the FH.



SYMBOLS	
⋈ - CONTROL VALVE	—• - SERVICE END
- CHECK VALVE	—) - VENTILATION END
▽ - VENT VALVE	- NORMALLY CLOSED VALVE
[F] - ABSOLUTE AIR FILTER	(F) - STANDARD FILTER
[P] - FAN OR COMPRESSOR	(AM) - AIR MONITOR TANK
[HX] - HEAT EXCHANGER	9 - SYSTEM
S-⧖ - SOLENOID VALVE	[M] - MONITOR

SYSTEMS	
1. COOLING AIR SUPPLY	7. DOOR OPEN, CELLS 3, 4, 5, 6 AND 7
2. PUMP HOUSE VENTILATION	8. VACUUM, CELLS 1 AND 2
3. OFF-GAS	9. REACTOR TANK VENT
4. HOOD VENT EXHAUST	10. PROCESS WASTE AIR
5. VENTILATION, CELLS 1 AND 2	11. CONTAINMENT VESSEL VENTILATION
6. NORMAL VENTILATION, CELLS 3, 4, 5, 6, AND 7	12. WASTE COOLING AIR

FIGURE C.5-1. Schematic Drawing of Ventilation Systems That Service Contaminated Areas

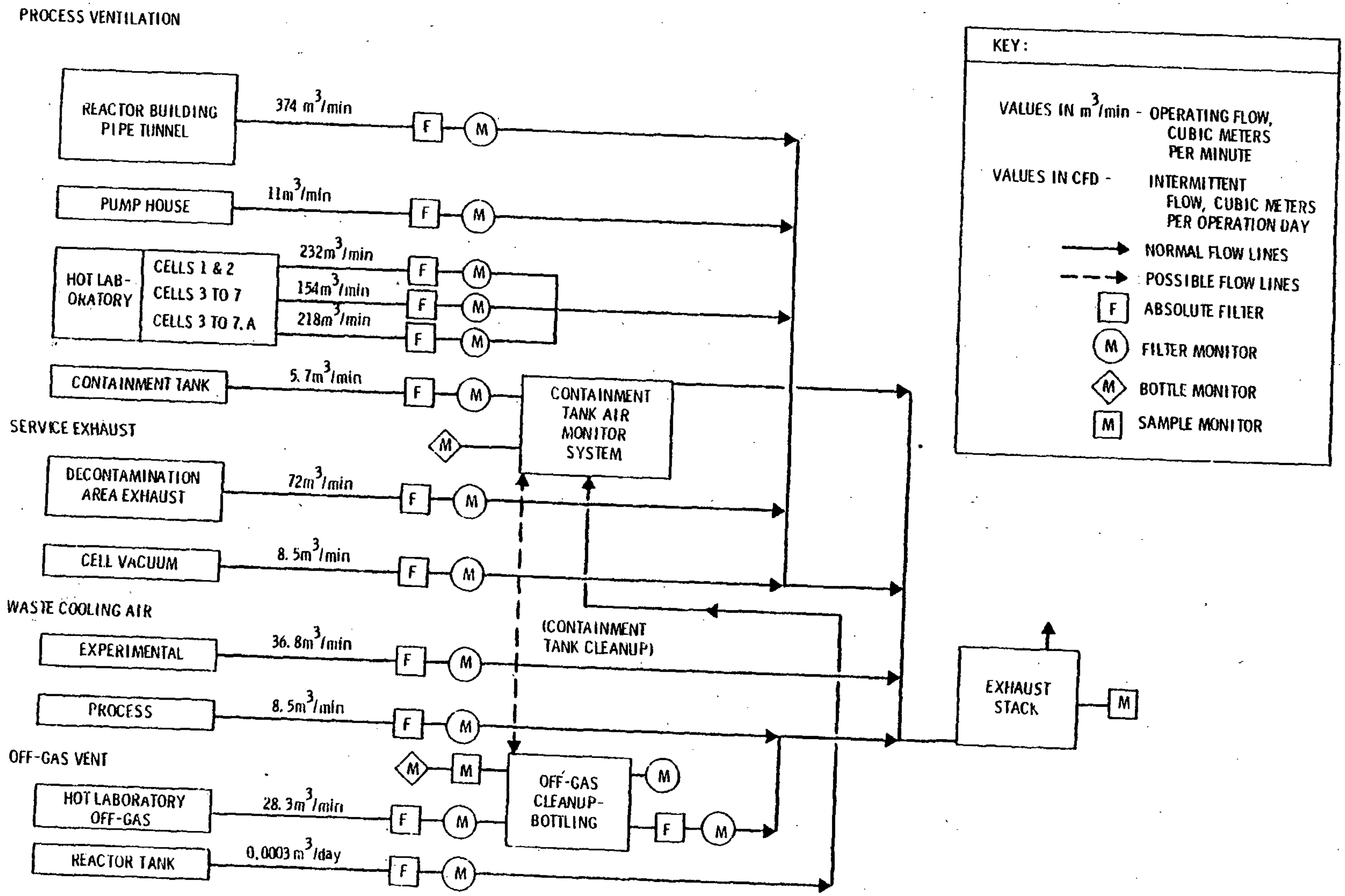


FIGURE C.5-2. Gaseous Waste Disposal Flow Schematic

The intake air for the UAS is drawn through two filters at the SEB by three 77-m^3/min air compressors. The output air, at 689 kPa, passes through two aftercoolers and collects in a double receiver tank with a capacity of 23 m^3. The utility air is supplied through a 255-mm-diameter line at a maximum rate of 45 m^3/min and a pressure of 689 kPa.

The utility air line enters the RB at the northeast corner at the -4.6 m elevation. At this point, a 76-mm-diameter connection is made through an isolation valve and an air filter unit to provide supply air to the service air system.

A 255-mm-diameter remote-operated valve in the supply header isolates the experimental and test cooling air system supply. The 255-mm-diameter supply header circles the containment vessel with supply takeoffs to each of the quadrant areas. Quadrants A and C are both serviced with a 255-mm-diameter experimental cooling air line with two shutoff hand-operated gate valves. Quadrants A, B, C, and D also have 76-mm-diameter test cooling air lines with individual shutoff hand-operated gate valves outside the containment vessel. There is a 203-mm-diameter cooling air supply takeoff valve to the basement of the OLB cold test area which serves design and test facilities.

Waste air from Quadrants A and C experimental units leaves the containment vessel by way of a 406-mm-diameter experimental waste air line, which collects the waste air from the 203-mm-diameter cooling air return from the basement of the OLB, and is piped to the FH. In the basement of the FH, this waste air is monitored for radioactivity, passes through a filter unit composed of a bank of four high-temperature 610-mm by 610-mm by 305-mm absolute filters, and is exhausted by a control valve through a 152-mm-diameter line to the exhaust stack.

Waste air from the test cooling air lines collects in a 102-mm-diameter service ring around the reactor tank and leaves the containment vessel through a 152-mm-diameter process waste air line which exits at the -7.6-m elevation, southwest, in the RB. The process waste air is monitored, passes through a filter unit utilizing a standard 610-mm by 610-mm by 305-mm absolute filter, and exhausts through the 305-mm-diameter waste-vent header to the exhaust stack.

The utility air standby service air system (not shown in Figure C.5-1) supplies the service air systems in the event the main three supply air compressors are not available to supply the system. In that case, service air is supplied by three service air compressors in the SEB. Two of the air compressors have 7.5-hp motors and the third has a 100-hp motor. The combined output is fed to a 1.7-m^3 surge-receiver tank, filtered, and distributed by a 152-mm-diameter line.

C.5.2.2 Contaminated Air and Off-Gas System

The contaminated air and off-gas system (CAOGS) is divided into 14 individual line groups, as shown in Figure C.5-3. Seven of these groups are room ventilation systems for the PH, HLB, WHB, RB, and the Containment Vessel, and the rest are operational and experimental vent lines. Each of the systems which exhausts to the stack passes through a filter bank and is monitored.

The CAOGS includes the ventilating air and operational vent gases exhausted from areas that are postulated to have been contaminated by the presence of radioactive airborne particles, induced and by-product radioactive gases, and radioactive volatile materials. It is assumed that much of this system normally operated in an uncontaminated condition, but since the origin of this ventage is from potentially radioactive areas, the exhaust air and consequently the equipment associated with it are considered contaminated.

Reactor Building--Hot Pipe Tunnel (RB-HPT) Ventilating Air. The RB-HPT ventilating air system starts in the basement areas of the RB from two filtered fresh air intake heater units and passes by general air circulation through a pipe chase on the south wall of the RB to the hot pipe tunnel beneath the HLB, where it is joined by a flow of air from the OLB basement sump room No. 11 through the tunnel which joins the OLB basement and HPT. At this point, the tunnel itself is the passageway for the air flow. There are 20 air changes per hour in the tunnel area, with a flow rate of 283 m^3/min. The tunnel opens into the basement of the FH where additional filtered fresh air heater units add a 91-m^3/min load to the system. After monitoring, the total volume is passed through a 2-m by 2.4-m open filter unit and drawn through a 0.92-m-diameter duct to the fan. The nine filters are standard 610-mm by

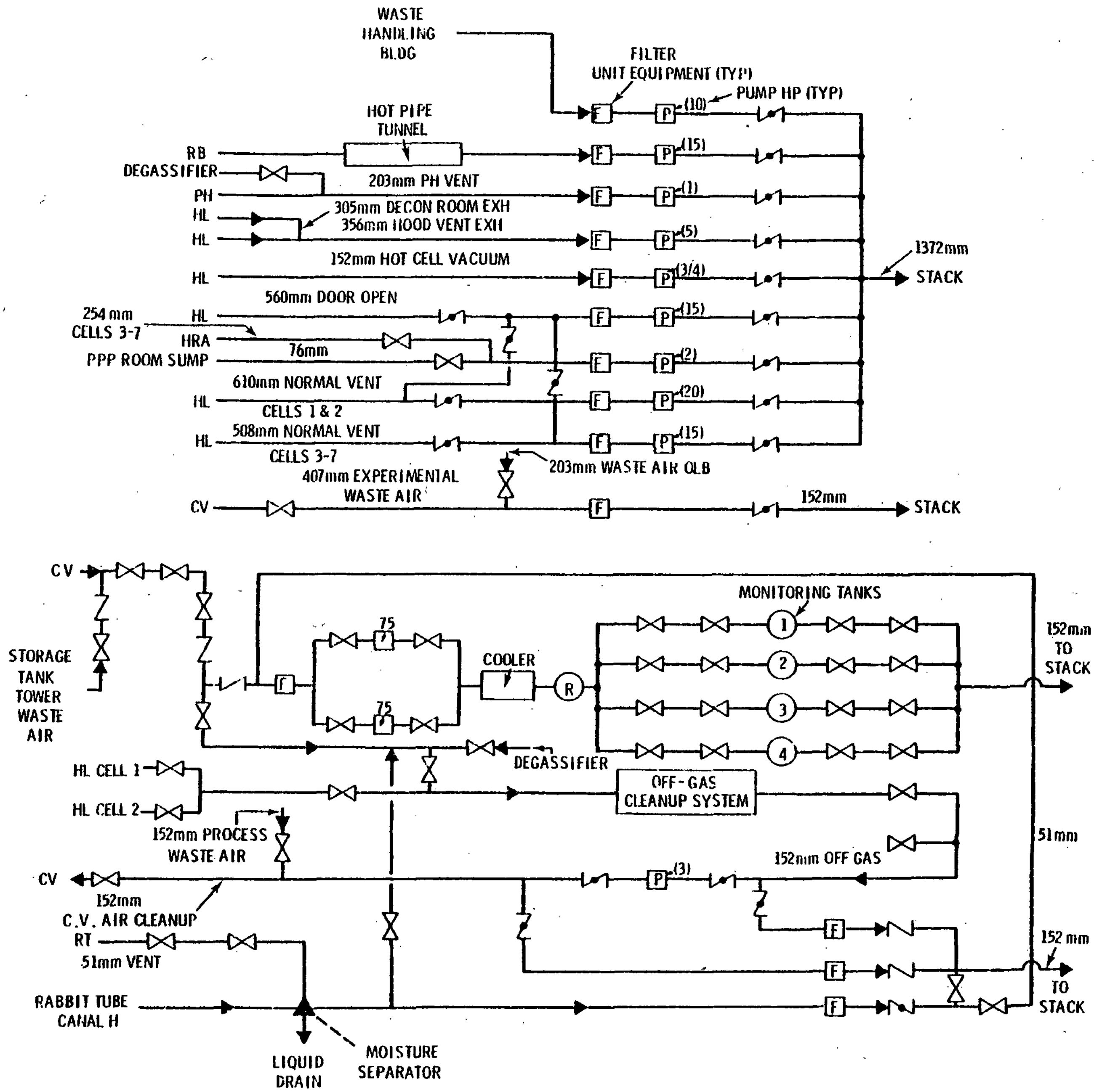


FIGURE C.5-3. Contaminated Air Systems

610-mm by 305-mm absolute filters. A monitoring system in the stack continuously checks the outlet filtered air for radioactivity. The 15-hp fan is rated at 374 m^3/min. This ventilation air is exhausted to the stack by way of a 0.76-m-diameter line and butterfly control valve to the 1.37-m-diameter exhaust header.

<u>Containment Vessel (CV) Ventilating Air</u>. The CV ventilating air system starts as a fresh air intake at the east penthouse where the air is filtered and passes through two balanced check valves, two control valves, and an isolation valve on the 152-mm-diameter inlet line. An air flow of 5.7 m^3/min enters at the top of the containment vessel from this line. Reverse flow through this line is automatically prevented by the balanced check valves. This system is schematically shown in Figure C.5-4.

The ventilating air circulates within the CV and is heated or cooled by the four 1867-m^3/min air conditioners. Ventilation air is purged from the CV at flow rates up to 6.2 m^3/min through a 152-mm-diameter CV ventilating air line on the west side of the CV.

Outside the CV the containment vessel ventilating air passes through a hand-operated valve, two control valves, and a check valve. This 152-mm-diameter line runs from the southwest corner of the RB, through the hot pipe tunnel to the FH basement, through a check valve and through a filter unit. This unit is equipped with a standard 610-mm by 610-mm by 305-mm absolute filter. The filtered air is compressed by two 6.2-m^3/min two-stage air compressors with 75-hp motors. The exhaust pressure is 2070 kPa. There is an aftercooler and a 0.57-m^3 receiver after the compressors.

The compressed air from the receiver passes to a 102-mm-diameter header which supplies the four tanks on the air tank monitoring system (ATMS). Each monitoring tank has a capacity of 1.42 m^3. The system is designed with diaphragm-controlled inlet and exhaust valves so that each tank fills with system air to the full pressure, holds during a radioactivity monitoring period, and exhausts with a cycle time of 20 to 60 seconds. In the event that one of the monitored tanks indicates a high level of radioactivity, the tank is automatically isolated and the system operates on three tanks.

<u>Reactor Tank Gas Vent</u>. The reactor tank gas vent system starts from the float vent valve and the associated gas vent system in the drainage and refill section of the primary cooling water-main loop, as shown in Figure C.5-5.

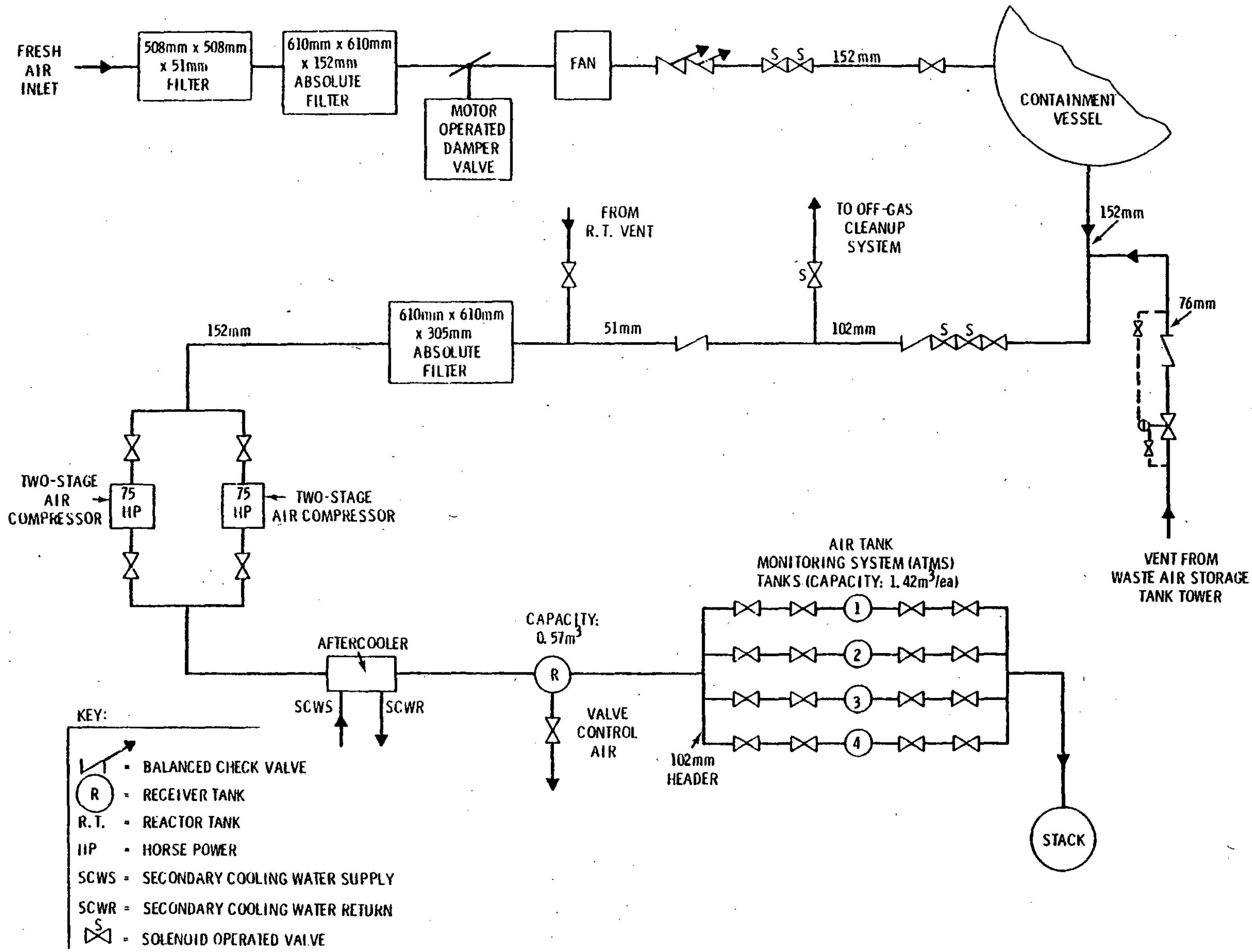
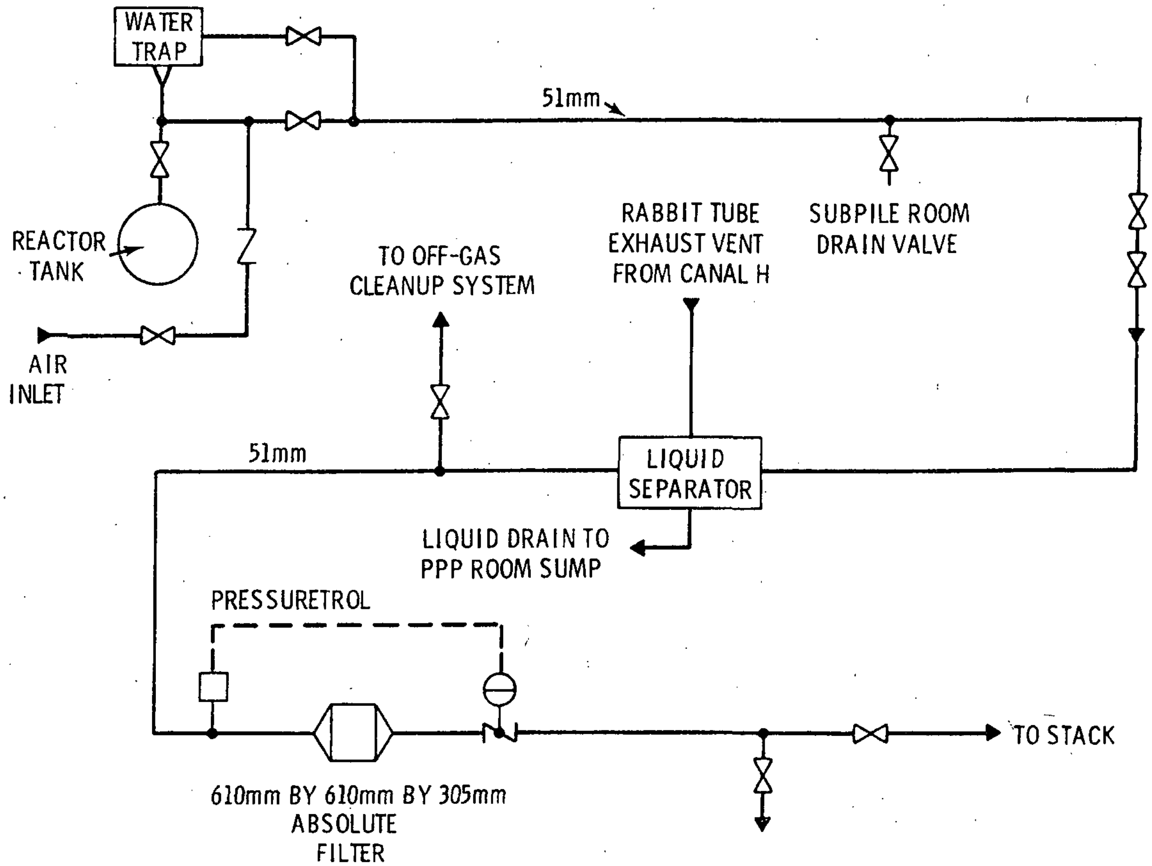


FIGURE C.5-4. Containment Vessel Ventilation System



<u>FIGURE C.5-5</u>. Reactor Tank Vent System

Ventage from the reactor tank can be purged through the 51-mm-diameter reactor tank vent to two valves that serve as isolation control on the reactor containment vessel, located at the -7.62 m elevation in the southwest corner of the RB. One valve is an electrically operated solenoid valve and the other is a hand-operated gate shutoff valve. This tank vent line continues down a shielded pipe trench to the PPP room where it combines with the 51-mm-diameter rabbit tube exhaust vent from Canal H. At this junction, there is a liquid separator which drains any vapor or overflow liquid in these lines to the PPP room sump. The combined gases and waste air continue to the basement of the FH through the 51-mm-diameter reactor tank vent line.

In the event the vented gas activity is too high for discharge to the stack, valves can be operated to allow the RT vent system to discharge to the off-gas cleanup system (OGCUS) for storage and/or cleanup.

At the FH these gases pass through an absolute filter bank which is 610 mm by 610 mm by 152 mm. The radioactivity of the gas at the inlet to the filter unit is continuously checked by the monitoring system, with the differential pressure controlled by a butterfly exhaust valve. Exhaust is to the containment vessel ventilating system, or under special conditions, directly through a 305-mm-diameter waste-vent header to the exhaust stack.

Air Tank Monitoring System (ATMS). The air tank monitoring system is part of the containment vessel ventilation system. Its purpose is:

- to determine the radiation level of the air received from the containment vessel ventilating system and the reactor tank gas vent system
- to provide appropriate signals to the tank (bottle) cycling system for either retaining the air in the bottles or releasing it to the stack
- to provide a printed record of radiation within the tanks.

The basic components of the ATMS are bottles, system electronics, and associated piping and valving. The four bottles, the radiation detection system, and the tank sequential cycling logic system are located in the southeast room in the FH basement.

Pump House (PH) Ventilating Air. The PH ventilating air system, illustrated in Figure C.5-6, originates from eight vent risers in the PH. Each of these vents is a 102-mm-diameter welded steel riser stem with a U-shaped intake elbow located 3.66 m above the floor. The opening is covered with 6.35-mm mesh screen. There is one vent in each of the primary pump rooms and one in the bypass deionizer room. There is an open vent in the PH sump which collects ventage from the resin pit and the sump tank. Also, there is a 152-mm-diameter fresh air inlet vent to this sump which allows a continuous purge through the area. All of the vent risers draw air through the valve stem seals, the roof plugs, interconnecting pipe seals and conduit seals in such a way that air flow is from the uncontaminated side to the contaminated side of the shielding. A single 203-mm-diameter PH ventilating air line connects these vents to the Fan House by way of a buried 508-mm-diameter pipe sheath conduit.

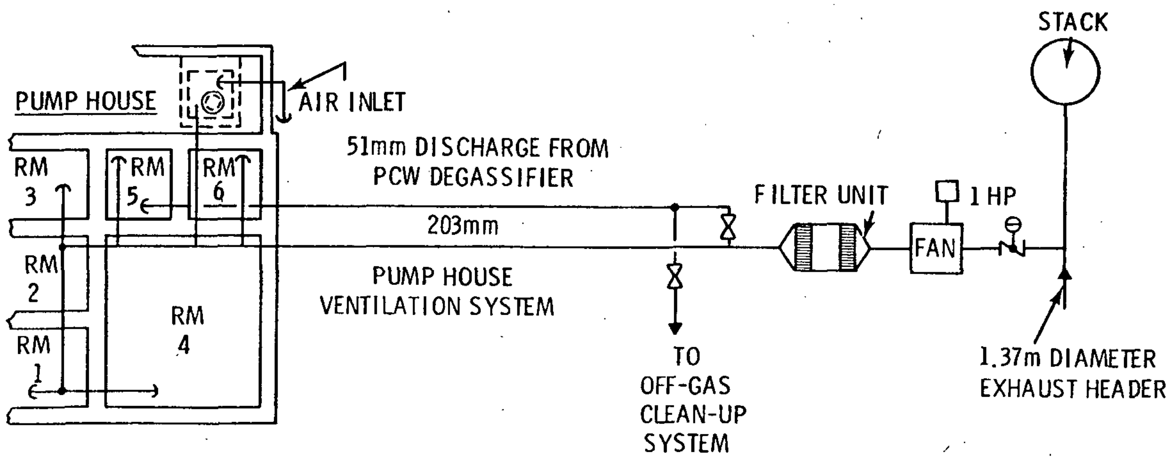


FIGURE C.5-6. Pump House Ventilating Air System

A 51-mm-diameter discharge from the PCW degassifier gas storage tank ties into the 203-mm-diameter PH vent line at the east end of the hot pipe tunnel through a remote operated valve.

In the basement of the FH, the PH ventilating air is drawn through a filter unit, a standard 610-mm by 610-mm by 152-mm absolute filter. A monitoring system continuously checks the air arriving at the filter. Following the filter is a 1-hp fan, rated at 11.3 m^3/min, which exhausts the ventilation air to the exhaust stack by way of a 203-mm-diameter line and a butterfly control valve to the 1.37-m-diameter exhaust header.

Waste Handling Building (WHB) Ventilation. The air for the WHB ventilation system is supplied by a makeup air unit and by infiltration into the building. The air flows to the south basement area through the evaporator area. The air enters the north basement area through roughing filters, a pressure differential controlled damper and fan. The 2-hp fan is rated at 170 m^3/min. The air enters a filter housing through a set of roughing filters and a set of bag-type prefilters. After monitoring, the air flows through a bank of six standard 610-mm by 610-mm by 305-mm absolute filters to a 10-hp fan rated at 170 m^3/min. Exhaust for this ventilation air is to the exhaust stack, as shown in Figure C.5-7.

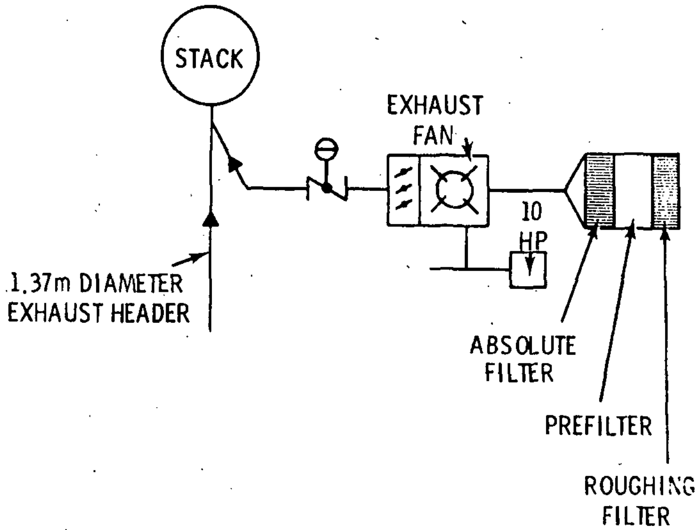


FIGURE C.5-7. Waste Handling Building Ventilating Air System

Hot Laboratory Building (HLB) Cell Ventilating Air. The cell ventilating air system is composed of three subsystems, as shown in Figure C.5-8, which are inter-related by their control circuit and the HLB cell area that they serve (i.e., the seven hot cells). The three subsystems are: 1) normal ventilation, 2) Cells 1 and 2 ventilation, and 3) door open ventilation. In general, ventilation of the cells is accomplished by drawing air into the rear of the cell through a set of hood-type filters and distributing this air through 203-mm- and 254-mm-diameter vent pipes to the top of each cell. The subsystems of the cell ventilating air system are described below.

- Normal Ventilation. When the doors are closed, Cells 3, 4, 5, 6, and 7 are ventilated by the collective normal ventilation header. For this general flow, air is drawn from each cell through another set of hood-type filters at the bottom of the cell and collected by the 508-mm-diameter normal ventilation header which runs under the cells in the hot pipe tunnel. A ventilation air flow of 155 m^3/min, or approximately 60 air changes per hour for each cell, is accomplished. After monitoring, the normal ventilation air is drawn through a filter unit in the basement of the FH. This unit is a bank of six standard 610-mm by 610-mm by 305-mm absolute filters. The air is drawn through a 15-hp fan at 155 m^3/min. Exhaust is to the stack through the 1.37-m-diameter exhaust header.

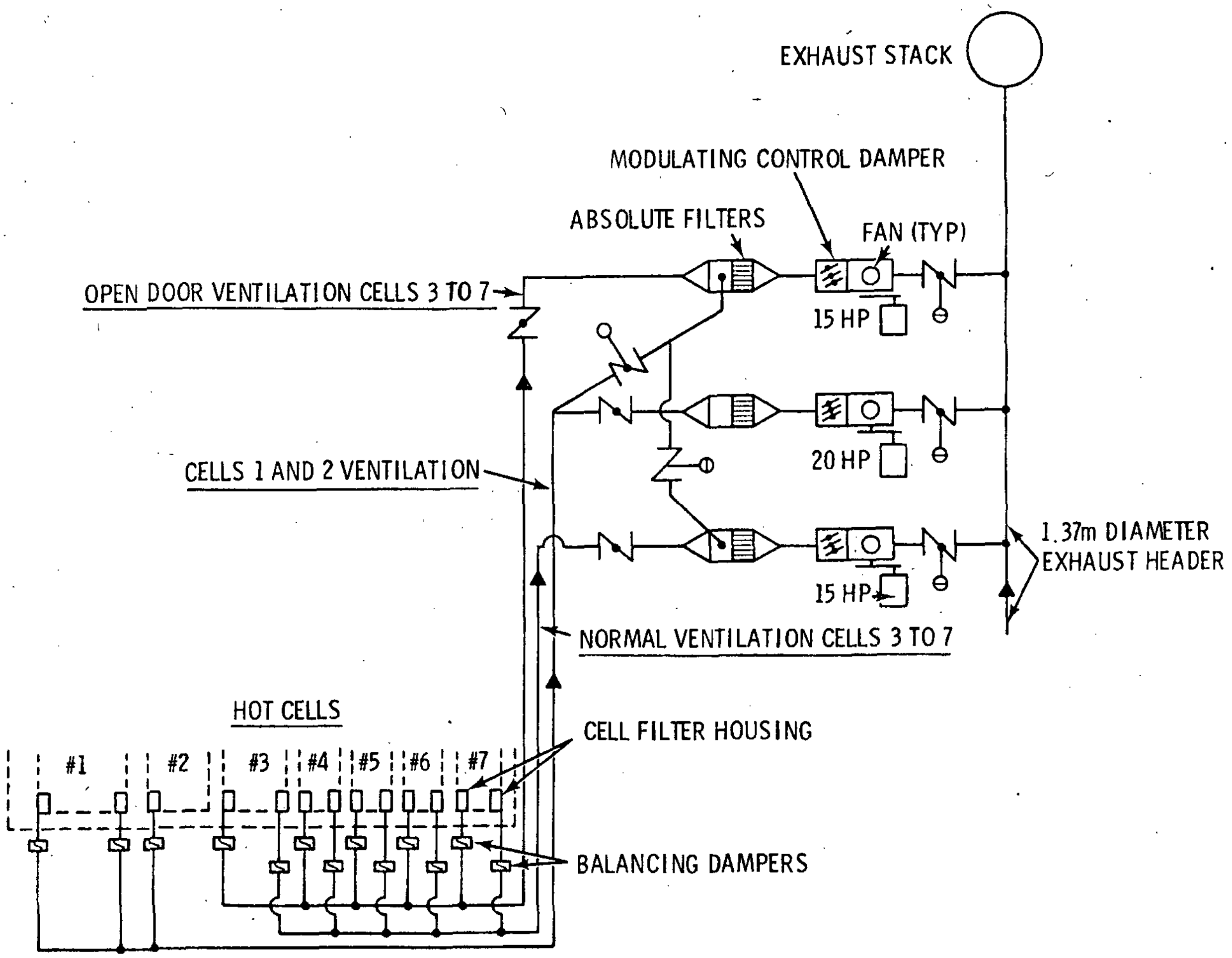


FIGURE C.5-8. Hot Cell Ventilation System

- Cells 1 and 2 Ventilation. Ventilation air flow for Cells 1 and 2 is provided through a 0.61-m-diameter ventilation header. For this general flow, air is drawn from each of the cells through a set of hood-type filters at the bottom of the cell and collected by this header, which runs through the hot pipe tunnel. A ventilation air flow of 233 m^3/min, or approximately 65 air changes per hour for each cell, is accomplished. With the plug-type doors to Cells 1 and 2 open, the flow rate through the open area will be about 3.4 m^3/min. After monitoring, the Cells 1 and 2 ventilation air is drawn through a bank of nine standard absolute filters located in the basement of the FH. The filtered air is drawn through a 20-hp fan at 233 m^3/min and is exhausted through the 1.37-m-diameter exhaust header to he exhaust stack.

• Door Open Ventilation. The door open ventilation system provides an additional ventilation capacity for Cells 3, 4, 5, 6, and 7. The system operates to draw additional air from these cells when the plug-type doors are open, and is electrically controlled to provide a preset dampered exhaust of:

102 m^3/min for 1 door open
127 m^3/min for 2 doors open
165 m^3/min for 3 doors open
219 m^3/min for 4 doors open.

For this flow, the air from the cell with an open door is drawn through a set of hood-type filters at the floor level into a 558-mm-diameter header in the hot pipe tunnel.

Though this system is interconnected between Cells 3, 4, 5, 6, and 7, the cell with an open door, offering less resistance to air flow, will have an air flow velocity of 1.84 m^3/min. A maximum of four open doors at once is the limit of operation. The door open ventilation air is drawn through a bank of nine standard 610-mm by 610-mm by 305-mm absolute filters located in the basement of the FH. After monitoring, the air is drawn through a 15-hp fan at 219 m^3/min. Exhaust is to the exhaust stack through the 1.37-m-diameter exhaust header.

There are two bypass lines between the three Hot Laboratory cell ventilation air lines to facilitate operations and divert air flow between lines when any of the Hot Lab filter units or fans must be shut down. One of these bypasses is a 508-mm-diameter line with a 508-mm-diameter butterfly valve which allows diverting the flow from the Cells 1 and 2 ventilation line to the door open ventilation filter unit and fan. The bypass valve in the other line is similar and diverts the normal ventilation line to the open door ventilation filter unit and fan.

In addition to the three cell ventilating air system subsystems described above, two other HLB contaminated air systems require attention during decommissioning: 1) the hood vent and decon room exhaust system, and 2) the cell vacuum system. These two smaller systems are illustrated in Figure C.5-9 and are described in the following descriptions.

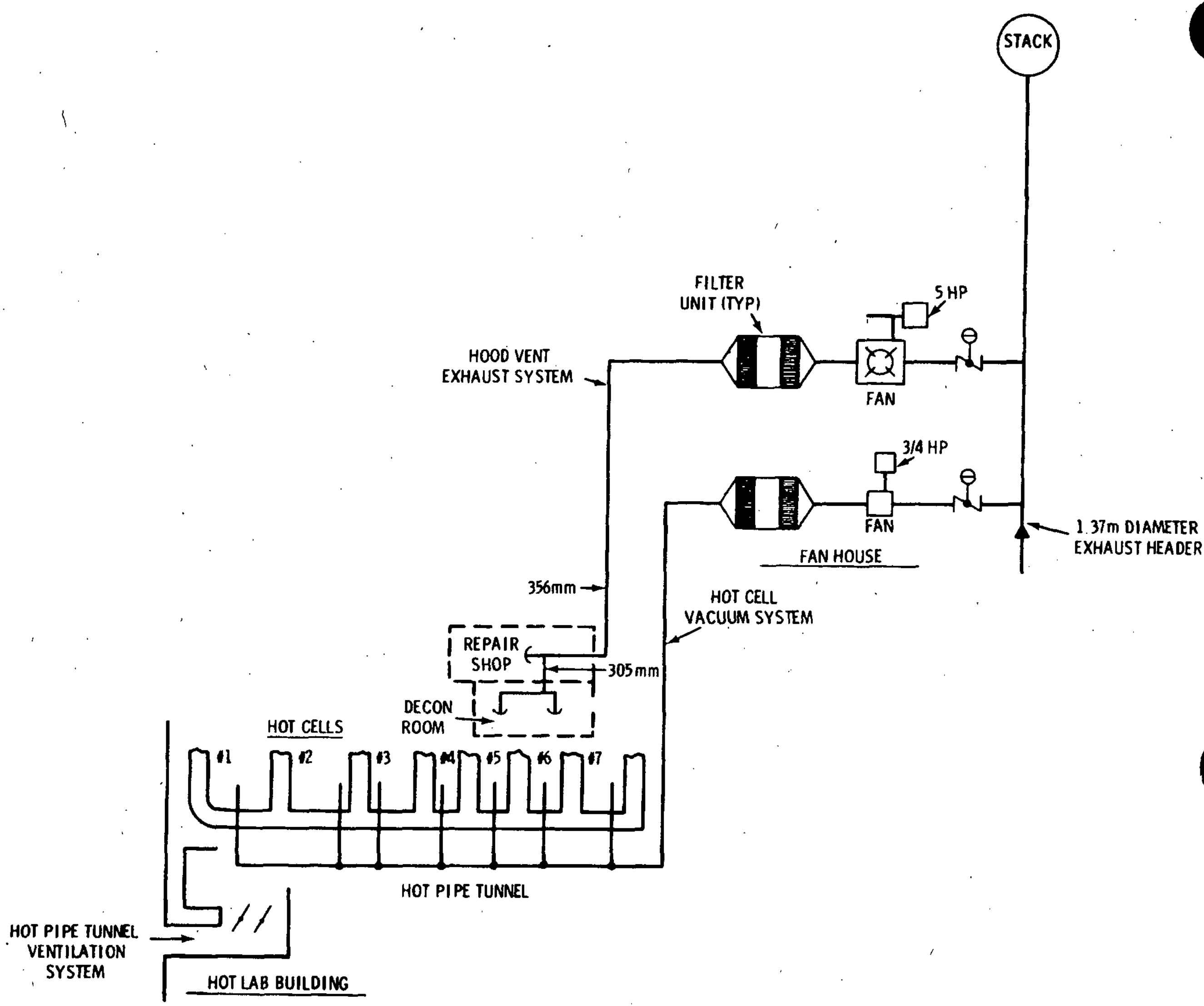


FIGURE C.5-9. Hood Vent and Decon Room and Hot Cell Vacuum Exhaust Systems

- Hot Laboratory Hood Vent and Decon Room Exhaust. The hood vent exhaust is a collective system on the machine operations in the Hot Laboratory machine shop and the decontamination area. In these areas are several collector and assembly hoods where radioactive or contaminated materials and equipment are repaired, disassembled, or cleaned. A 356-mm-diameter line carries the exhaust from these hoods, running from the hot pipe tunnel under the HLB to

the basement of the FH. The exhaust air enters a filter unit, is monitored and flows through a bank of two standard 610-mm by 762-mm by 305-mm absolute filters. The air is drawn through a 5-hp fan at 71 m^3/min. Exhaust is through the 1.37-m-diameter exhaust header to the exhaust stack.

- Hot Laboratory Cell Vacuum. The cell vacuum system serves Cells 1 and 2 with 102-mm-diameter lines and Cells 3 to 7 with 76-mm-diameter lines. This vacuum system exhausts 8.5 m^3/min at -6.5 in. of water pressure for cleanup of dust and solid particles produced in machine grinding, polishing, or filing within the general Hot Laboratory area. The 152-mm-diameter cell vacuum line runs from the HLB to the basement of the FH through the hot pipe tunnel, where the exhaust enters a filter unit, is monitored, and flows through a standard 610-mm by 610-mm by 152-mm absolute filter. The filtered air is drawn through a 3/4-hp fan at 8.5 m^3/min and is exhausted through the 1.37-m-diameter exhaust header to the exhaust stack.

- Off-Gas Cleanup and Storage System. The off-gas cleanup and storage system is shown in Figure C.5-10. It consists of two vacuum pumps, two compressors, and five storage tanks enclosed by a container tank located in the dry storage area of the Hot Lab (see Figure C.2-7). Each vacuum pump has a 1-1/2-hp motor. The exhaust from the vacuum pumps or from an atmospheric pressure feed on the off-gas inlet is collected in a surge tank as supply to the compressors.

One compressor has a 7-1/2-hp motor and the other has a 5-hp motor. The combined output is cooled and by control valving, stored in one of five 2.04-m^3-capacity storage tanks at pressures up to 1725 kPa.

The off-gas cleanup system consists of a liquid-gas scrubber and two carbon bed sorbers. The output from the storage tanks is fed via remotely controlled valving through these units to a 153-mm off-gas exhaust or back to the storage tanks. Sampling of the off-gas at various control points, temperature, pressure, and cycling control is possible through remote operations. Output from the total off-gas system is through valve "A" and is manually operated at the Canal K area.

The off-gas exhaust is a 152-mm-diameter line from the dry storage area of the HLB to the basement of the FH. Discharged gases in this line are monitored and filtered through a filter unit. The gases are then exhausted through the 305-mm-diameter waste-vent header to the exhaust stack.

The Hot Laboratory off-gas vent system shown in Figure C.5-10 starts at Cells 1 and 2. Each cell is serviced with a 102-mm-diameter off-gas vent line which collects vapor and off-gases from the machine operational areas where experimental units are handled. Since the handling operations and equipment size varies with the experiment, this gas collection method is accomplished by hoods, purges, or directly coupled vents. The off-gas vent is remotely controlled through valves located in the Hot Lab contaminated work area.

- Hot Retention Area (HRA) and Process Piping Pump (PPP) Room Sump Vent System. A 76-mm-diameter vent line connected to the top of the PPP Room[a] sump runs through the HPT to the Fan House basement. A 254-mm-diameter vent line stubs into the north end of the HRA pipe tunnel through the Fan House basement wall. As shown in Figure C.5-3, the 76-mm-diameter and the 254-mm-diameter lines are joined to a 305-mm-diameter line at the inlet to a filter housing unit. Isolation valves are provided and exhaust air is monitored for radioactivity. A 2-hp fan, rated at 22 m^3/min, discharges the air to the 1.37-m-diameter header and then to the stack.

C.5.3 Waste Air Systems

There are two systems available to provide for removal of pressurized air or gas that has been used for other than ventilation purposes, mainly cooling processes: 1) a 153-mm-diameter process waste air line, and 2) a 406-mm-diameter experimental waste air line. These systems are described in the following subsections.

C.5.3.1 Process Waste Air Line

The 153-mm-diameter process waste air header originates with the 102-mm-diameter ring header pipe encircling the reactor tank at a centerline elevation of -4.1 meters.

(a) The PPP Room is located in the Reactor Building at the -7.62-m elevation (see Figure C.2-3).

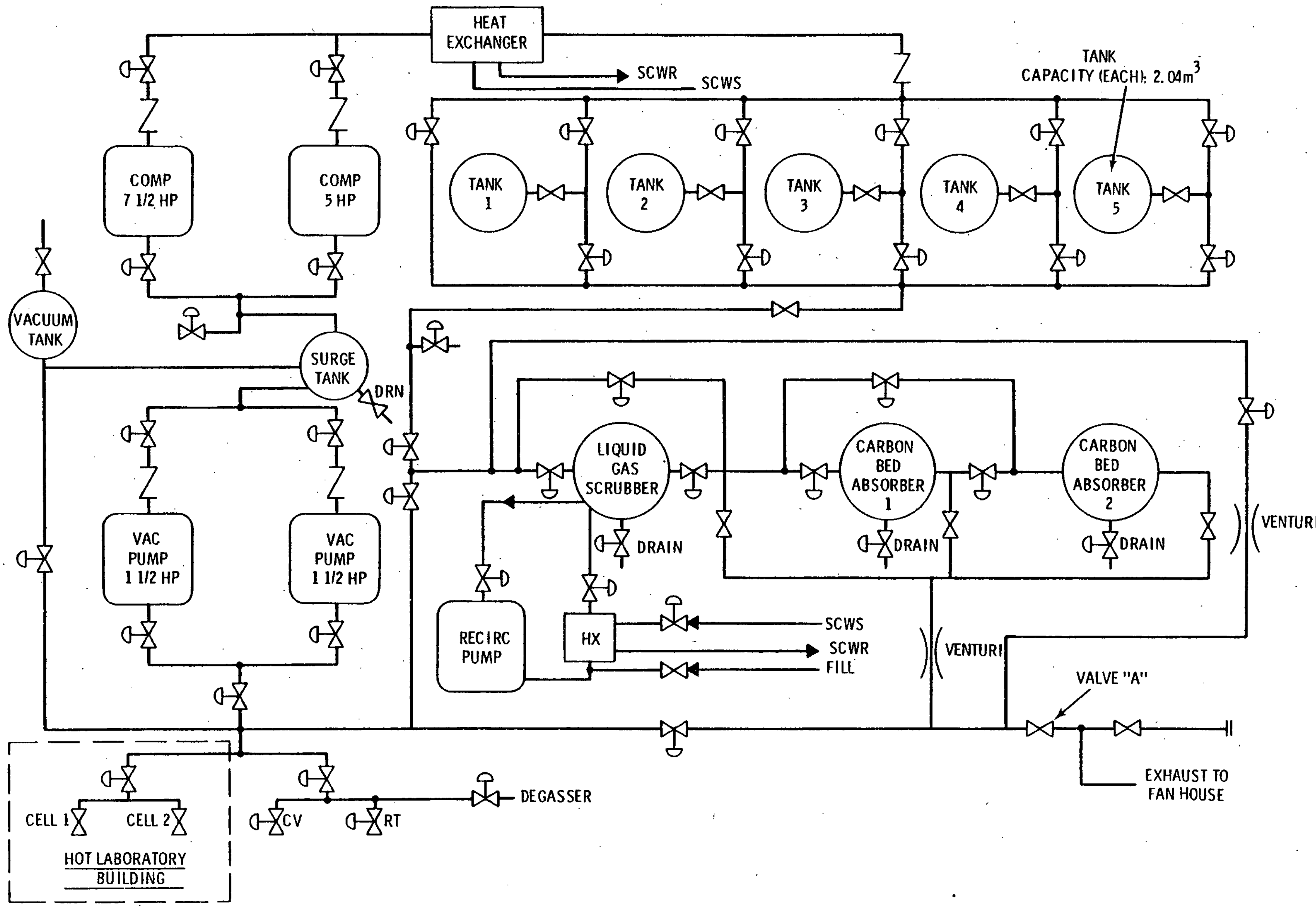


FIGURE C.5-10. Off-Gas Cleanup and Storage System

The waste air ring header is the topmost of the four ring headers. There are connections to the ring header available in each quadrant. The process waste air (see Figure C.5-3) is monitored, filtered, and exhausted to the stack. The 153-mm-diameter process waste air vent system has a capacity of 31 m^3/min with a maximum pressure of 34.5 kPa at the filter housing.

C.5.3.2 Experimental Waste Air Line

The waste air exits from the CV through a 406-mm-diameter line on the -4.6-m level of the Reactor Building at 270°. A 203-mm-diameter cooling air exhaust from the OLB basement joins the 406-mm experimental waste air main in the Reactor Building basement. This system is illustrated in Figures C.5-1, C.5-2, and C.5-3. The 406-mm experimental waste air line runs from the Reactor Building basement through the HPT to the Fan House basement. The waste air flows through a filter housing where it is monitored for radioactivity and filtered. A 152-mm butterfly valve controls the discharge flow to the stack and maintains the system back pressure.

The 406-mm experimental waste air system has a capacity of 125 m^3/min. The maximum operating pressure is 689 kPa.

C.5.4 Exhaust Stack

The exhaust stack is a 30.5-m-high, 1.52-m-diameter vertical steel pipe, with a concrete support stand and a vortex plenum at the base. Bases from the 1.37-m-diameter exhaust header, the 305-mm-diameter waste-vent header, the 152-mm-diameter experimental waste air exhaust, and the 152-mm-diameter containment vessel ventilating air exhaust line enter the plenum and the stack at various points. The output from the exhaust stack is about 1300 m^3/min with all exhaust fans and forced air units in operation. The air flowing through the exhaust stack is monitored by the stack effluent monitoring system (SEMS), which records contamination levels. The flow through the stack is measured and recorded at the FH-contaminated air systems control panel.

C.5.5 Estimate of Contaminated Equipment and Materials in the Air Handling Systems

It is not possible to evaluate quantitatively the amount of radioactivity that is contained in all of the previously described exhaust air handling

systems, since no measurements of these systems are available. Therefore, for conservatism, it is postulated that all of the piping, ductwork, valves, filters, and fans associated with these systems are contaminated. Summaries of the estimated amounts of contaminated ductwork and piping and of contaminated equipment are listed in Tables C.5-1 and C.5-2, respectively.

Waste filters are not included in Table C.5-1. Two types of air filters are used in the air handling systems of the reference test reactor facility: the standard roughing filter, and the absolute filter. There are about 90 intermediate ventilation roughing filters in the contaminated work area of the HLB and 40 absolute filters in the ventilation and off-gas exhaust systems located in the Fan House.

The roughing and absolute filters are postulated to be sealed in drums for offsite disposal. Compressor, vacuum pump, and laboratory hood filters in contaminated areas are handled in a similar manner.

C.6 RADWASTE SYSTEMS

This section describes the systems in the reference test reactor that handle the liquid, solid, and gaseous radwastes. All three of these disposal systems are postulated to be used to some degree during decommissioning, since the disposal of radioactive waste materials from decommissioning involves the same general activities as are encountered during operation--namely, collection, classification, treatment, storage, and disposition.

C.6.1 Liquid Radwaste System

The liquid radwaste systetm is composed of a group of subsystems designed to collect, treat, recycle, and/or discharge different categories of waste water. The flow system for the treatment of liquid radwastes is shown in simplified block diagram form in Figure C.6-1, including estimates of the volumes of waste water from various sources that are treated during operation.

TABLE C.5-1. Estimates of Contaminated Ductwork and Piping in the Air Handling Systems[a,b]

From	Via	To	Line Size (mm)	Length (m)	Volume (m^3)	Weight (kg)	Filter Housing Estimated Volume (m^3)/ Weight (kg)
OLB Waste Air Line	Tunnel	HPT	203	46	1.75	920	---[c]
RB Quadrants A & C Exp. Waste Air Line	HPT	FH	406	100	13.0	1 270	0.45/17
RB Test Cooling Air Line	RT	FH	152 102	100 9	2.3 0.1	1 400 75	0.11/4
RB-HPT	---[c]	Stack Exhaust Header	762	3	1.38	72	2.14/79
CV Ventilation System	HPT	FH	152 102	100 30	2.3 0.33	1 400 252	1.15/43
Reactor Tank Vent	HPT	FH	51	100	0.3	400	0.11/4
PH	Buried Pipe Sheath Conduit	FH	203 51 102	60 50 30	2.28 0.15 0.33	1 200 200 252	0.06/2
WHB	---	Stack	203	20	0.76	400	0.17/6
HLB Cells 3-7 (Normal Vent)	HPT	FH	508	76	15.4	1 246	0.17/6
HLB Cells 1 & 2	HPT	FH	610	76	22.2	1 421	2.14/79
HLB (Door Open Vent)	HPT	FH	558	76	18.62	1 361	2.14/79
HLB (Hood Vent & Decon Room Exh)	HPT	FH	356	76	7.6	851	0.14/5
HLB (Cell Vacuum)	HPT	FH	102 76 152	15 30 76	0.17 0.18 1.75	126 225 1 064	--/-- --/-- 0.06/2
HLB (Off-Gas System Exhaust)	HPT	FH	152 102	76 30	1.75 0.33	1 064 252	1.15/43 --/--
HRA	HPT	FH	254 305	20 2	1.02 0.15	134 16	0.14/5
PPP Room Sump	HPT	FH	76	100	0.6	750	--/--
FH	Stack Exhaust Header	Stack	1372	23	34.04	996	--/--
Totals				1324	128.79	17 347	10.13/374

(a) The number of figures shown is for computational accuracy and does not imply precision to that many significant figures.
(b) During packaging, valves are packaged with the larger-diameter ductwork to provide a reduction in overall total burial volume. In addition, about 50% volume reduction is assumed for the stack exhaust header and the filter housings.
(c) Not applicable.

TABLE C.5-2. Contaminated Equipment in the Air Handling Systems

System	Equipment Piece (quantity)	Estimated Mass (each), kg	Estimated Volume (each), m^3
Contaminent Vessel Ventilation	CVVS Tanks (4)	1 230	1.72
	Air Compressors (2)	2 000	4.5
	Aftercooler	---(a)	---
	Receiver Tank	560	0.57
Off-Gas Cleanup and Storage System	Storage Tanks (5)		
	Vacuum Pumps (2)		
	Compressors (2)		
	Heat Exchanger (2)	35 808 (total)(b)	58 (total)(b)
	Surge Tank		
	Liquid Gas Scrubber		
	Carbon Bed Absorber		
Reactor Tank Vent System	Liquid Separator	---	---
Other	Misc. HVAC Hardware	1 735	13
	Stack	17 455	55

(a) Insufficient data available to make an estimate.
(b) These pieces of equipment are of various shapes and sizes and are estimated to require a total of about 58 m^3 of container volume for transport and subsequent burial. This total volume includes the container tank in which the five storage tanks are enclosed.

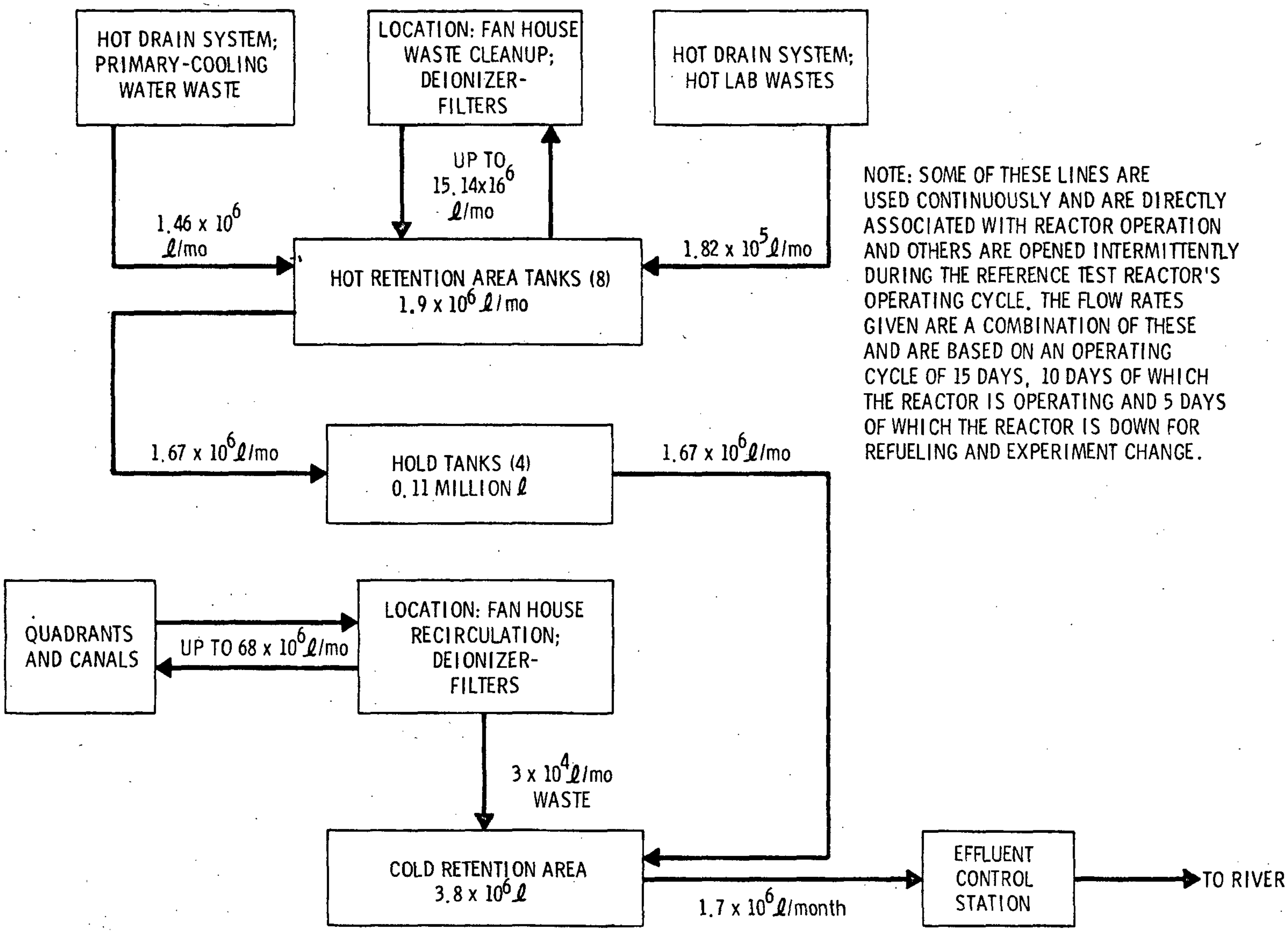


FIGURE C.6-1. Liquid Radwaste Disposal Flow Schematic

All liquid radwaste process streams terminate in a sample or retention tank. Since the liquid radwaste system is operated on a batch basis, each treated batch can be sampled to ensure that the treatment is effective. If the sample indicates that the process liquid is still above acceptable radioactivity limits or is substandard in purity, the batch is recycled, either through the same treatment system or through another subsystem. This process provides a higher degree of treatment. If the sample indicates that the level of radioactivity is within limits for discharge, the processed liquid may be discharged to the environment.

Two types of tank and plumbing installations are used in the liquid waste disposal system to contain liquid wastes. Both types are designed as absolute containment systems for neutralized liquid wastes and both have level indicators, vents, cooling water lines (if required), and plumbing for inlets and outlets. The first type of installation uses tanks and plumbing within a steel outer container. The inner steel tank holds the waste liquid, and the outer steel tank acts as a barrier against the possibility of waste liquids leaking out of the contained area. The open area between the inner and outer steel tanks has a low spot sump, which is monitored for water, either as ground water seeping in through the outer steel surface or as waste liquids leaking from the liquid waste system. When water is detected in the drain sump, a liquid sample is taken to determine which surface is leaking, the tank is emptied, and repairs are made. All buried piping, resin storage pits, and the eight hot retention tanks are of this type.

The second type of installation uses steel tanks and plumbing within a reinforced concrete structure. Between the tank base and the concrete base pad, there is an open annulus with a drainage collection sump. Samples of liquid collection in this sump are checked for radioactivity to determine if the liquid is due to ground-water seepage or individual tank leakage. Wherever possible the concrete surface is waterproofed; otherwise the steel surface is coated for waterproofing. Units of this type are floor drains, hot drain sumps, and connecting pipe in trenches.

The handling of radioactive waste liquids from each major source at the reference test reactor is discussed in the following subsections.

C.6.1.1 Primary Cooling Water Wastes

Contaminated water from the primary cooling water system (PCWS) makes up the greatest volume of high-level radioactive waste liquid. The radioactivity of the primary cooling water is determined by its impurities, both solid and dissolved. Traces of these impurities are found in the deionized water supply; other impurities result from corrosion within the PCWS, and still others are due to recoils from the surface of reactor core materials.

During operation, a small portion of primary cooling water drains or flushes from the system. Where equipment seals or thimble-cooling connections are concerned, these units are pressurized by deionized water from the storage tank at a pressure somewhat higher than the primary cooling water at the seal point. The estimated resultant seal leakage during operations has been about 1,460 m^3 per month.

All drainage from the PCWS is collected via the hot drain system, previously discussed in Section C.3.7.3, into one of three sumps, the subpile room sump, the reactor sump, and the Primary Pump House sump. The pumped discharge from each sump is collected in one of the tanks in the Hot Retention Area.

During reactor shutdown, primary cooling water is drained from the system under one of two procedures: 1) standard shutdown for refueling (which involves draining and flushing about 7.6 m^3 from the PCWS), or 2) decontamination shutdown for system cleanup (which involves draining and flushing about 94.7 m^3 from the PCWS).

The liquid wastes are treated by the combined methods of storage retention for radioactive decay, chemical separation by ion-exchange, and by solidification. The solidification system is discussed further in Section C.6.2.

The capacity of the hot retention tanks to store liquid wastes for radioactive decay is based on a design which allows for 800 hours decay time. The hold tanks allow an additional 800 to 1600 hours. In general, liquid samples from the hot retention tanks and the hold tanks are taken when the tanks are full or as operational procedures require.

Decontamination of the primary cooling water is accomplished by the bypass deionizers, as discussed previously in Section C.3.7 in the connecting circuit with the PCWS. Basically, the deionizer system consists of a mixed bed deionizer preceded by a filter. The design flow of the system is 0.38 m^3/min. The system is located in shielded chambers in the Fan House. All valves are remotely operated. Resin removal and storage are discussed in Section C.6.2.

C.6.1.2 Quadrant and Canal Water Waste

The quadrant and canal (Q&C) areas, described previously in Sections C.1.2.2, C.2.1.2, and C.3.7.2, serve as shielding, transfer, and storage areas for reactor and experimental operations. The water used in these areas is partially deionized water. Each quadrant contains about 681 m^3 of water and each canal about 833 m^3 of water.

No direct generation of radioactivity is expected in the quadrant water because of the concrete shield interposed between the reactor tank and the water. The chief source of contamination occurs through corrosion products from irradiated materials transferred through or stored in the quadrant or canal areas. All surfaces in the quadrant and canal areas are painted with an epoxy resin for waterproofing and to facilitate decontamination.

Normally, the water in the Q&C area is recycled through the recirculating cleanup system, which is a combination of two standard water filters and one mixed bed deionizer. Each quadrant and canal has an overflow type purge line, a float control valve, and a balancing orifice to allow a recycle flow rate of 0.04 to 0.19 m^3/min from each area. The 1.9 m^3/min total flow is split between the filters, and a 0.38 m^3/min portion of the recombined flow passes through the deionizer.

If the quadrant and canal water radioactivity exceeds 10^{-5} Ci/m^3, the recirculating water can be diverted to the Hot Retention Area, through the waste cleanup system and back to the quadrant and canal areas from storage in the Cold Retention Area or the hold tanks.

C.6.1.3 Laboratory and Other Liquid Radwastes

Other liquid wastes with radioactivities greater than 10^{-3} Ci/m^3 will originate from laboratory waste drainage, decontamination liquids, and general drainage from areas of controlled contamination.

Laboratory liquid wastes are collected via the hot drain system, described previously in Section C.3.7.3, into two sumps, the Hot Laboratory Building (HLB) sump and the Office and Laboratory Building (OLB) sump.

The HLB sump handles liquid wastes from hot cell operations, floor drains in the area, and decontamination equipment liquid wastes. The source of the decontamination liquid wastes is the decontamination room in the HLB (see Figure C.2-7, Room 23), which houses the equipment used for surface decontamination of materials. The decontamination equipment includes cleaning sinks and steam and ultrasonic scrubbers. A portion of the volume of these wastes is either acidic or caustic, depending upon cleansing techniques. Therefore, the collection piping is plastic and a neutralizing tank with chemical treatment control precedes liquid waste discharge to the HLB sump.

The OLB sump handles liquid wastes from the low-level laboratory work area. Normally, the radioactivity in the liquid waste to this sump is about 10^{-5} Ci/m^3 on the average, but an occasional batch will increase required hot retention handling. No special chemical treatment is required for these liquid radwastes.

Experiments conducted at the reference test reactor, which had internal waste flow systems that could become contaminated radioactively, required separate cleanup systems, so that the liquid effluent radioactivity to any hot sump was less than the normal radioactivity level for the sump.

The pumped discharge from these sumps is collected in the hot drain system and drains directly into one of the tanks in the Hot Retention Area. For continuous flow waste drainage into the Hot Retention Area, the radioactivity of the liquid was estimated to be less than 1.5×10^{-3} Ci/m^3, at a flow rate of 182 m^3/month during operation. The radioactivity in the intermittent flow waste drainage during decommissioning is anticipated to be about the same order of magnitude.

C.6.1.4 Hot Retention Area Water Wastes

The Hot Retention Area (HRA) is located south of the Fan House, as previously described in Sections C.2.6 and C.3.7.3. Eight storage tanks are located within the HRA rectangular concrete pit to permit for the radioactive decay of liquid wastes from the hot drain system. Each tank has a discharge pump which delivers to the waste cleanup system.

Beneath the eight tanks is a steel plate floor cover with side sections and divider plates sectioning each tank area. These sections serve as individual collection dishes to be monitored for tank leakage indication, and are assumed to be contaminated. It is anticipated that, during decommissioning, a nominal amount of radioactive precipitate in the form of a slurry is removed from the HRA tanks and solidified for disposal.

Immediately north of the concrete pit are four 28.4-m^3 tanks. These tanks are interconnected and have a common discharge pump. They are fed by return waste water from the waste cleanup system and serve as a monitoring storage point in the system. Liquid from these tanks may be transferred to the Cold Retention Area, into the Q&C system, or diluted with uncontaminated waste water prior to disposal.

C.6.1.5 Waste Cleanup System

Decontamination of liquid wastes stored in hot retention tanks is accomplished by the waste cleanup system, which is composed of a mixed bed deionizer preceded by a water filter.

The waste cleanup deionizer is a mixed bed unit with a designed flow rate of 0.38 m^3/min. The deionizer is used for three types of liquid wastes:

- wastes that contain trace quantities of radioactivity in low total concentration wastes
- wastes that contain suspended solids not collected by filtration
- wastes that contain traces of fouling materials.

The last form of waste can either be handled by chemically treating the waste liquid at the point of origin or by normal chemical interaction with the bed itself.

Preceding the deionizer is a standard water filter designed with replaceable cartridges capable of handling 0.95 m^3/min, and removing particles greater than 5 micron in size. Both the filter and the deionizer are located behind shielded cells in the Fan House, and valves on the system are remotely controlled.

It is postulated that during decommissioning activities, the waste cleanup system removes sufficient radioactivity from most of the highly radioactive liquids so that no hot storage problem is encountered. When this system is not adequate, a 0.38-m^3 per hour evaporator is available, or direct solidification methods are employed.

C.6.1.6 Cold Retention Area

Although the Cold Retention Area (CRA) as previously described in Section C.2.7 is primarily a storage point for water pumped from the Q&C water system, it serves doubly as a storage facility for waste water from the hold tanks in the waste cleanup system. At this point the waste water may be transferred for reuse in the quadrants and canals or it may be used for other operational purposes. Cleanup and reuse of contaminated water during decommissioning activities is postulated as a viable technique to limit the total contaminated water inventory.

Waste liquids from the CRA basins containing low levels of radioactivity are postulated to have been pumped through concrete piping to the ditches around the reactor site during reactor operation (see Section C.4). Thus, removal of this concrete piping during decommissioning is anticipated.

C.6.2 Solid Radwaste System

The solid radwaste system collects, monitors, processes, packages, and provides temporary storage facilities for the radioactive wastes prior to offsite disposal. The solid radwaste system is shown in simplified block diagram form in Figure C.6-2.

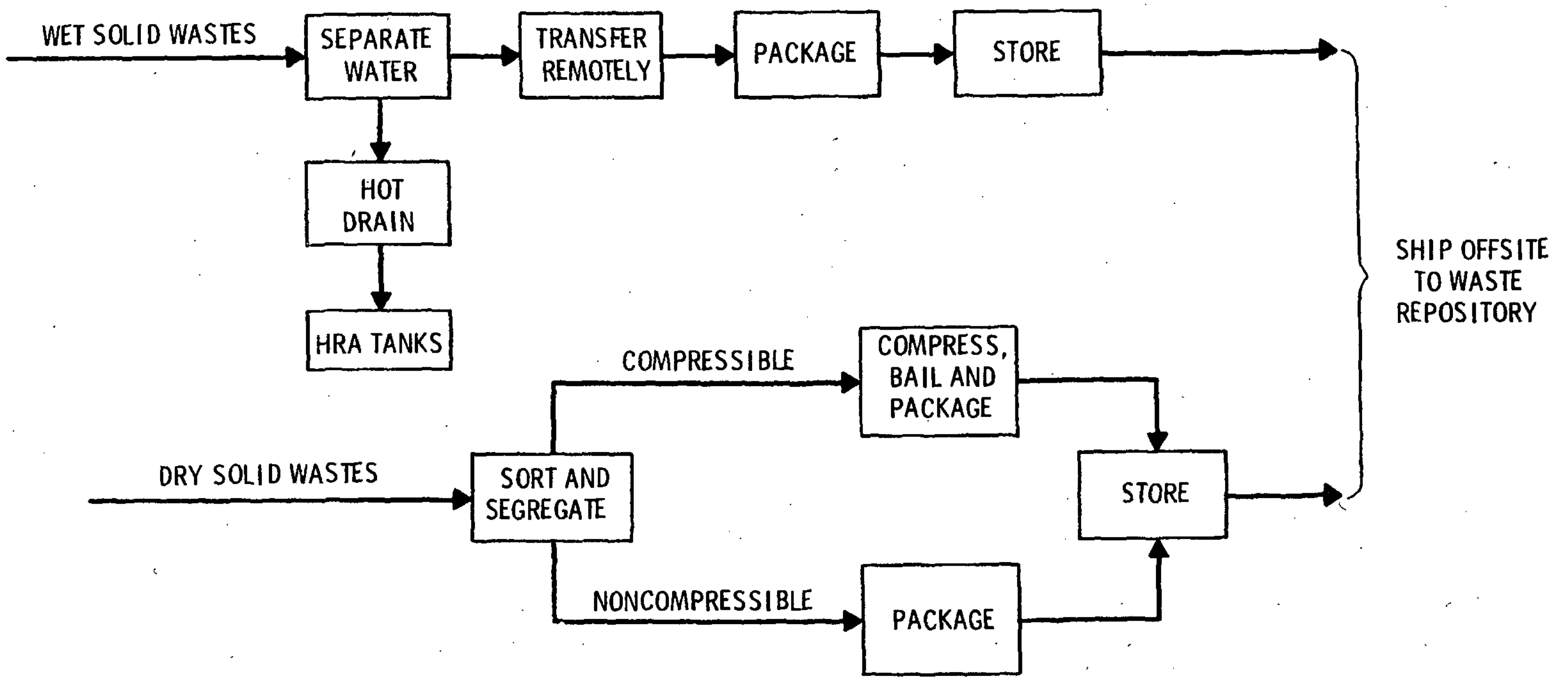


FIGURE C.6-2. Solid Radwaste System Process Diagram

The solid radwaste system processes both wet and dry solid radwastes. Wet wastes include tank slurries, spent resins, and the filters in the liquid radwaste systems. Dry wastes associated with decommissioning include rags, paper, small equipment parts, solid laboratory wastes, air filters, nonrecyclable anti-contamination clothing, etc.

An assortment of waste materials results from operation of the reference test reactor. Some of these materials (such as reactor core components, handling, or control equipment) were subjected to irradiation. The removal and disposal of such materials is postulated to be a logical and integral part of the conceptual decommissioning of the reference test reactor. This category of solid radwastes includes other materials that have been contaminated at the surface (such as machine operational and handling tools, structural components, or operational transfer equipment).

Spent resins from the primary cooling water bypass deionizers and from the waste cleanup deionizer are flushed from the deionizer tank and discharged into underground pits. One resin pit is located outside the Fan House (see Figure C.2-9) and the other is located outside the Primary Pump House (see Figure C.2-10). The flushed material is allowed to settle, and the liquids drain via the hot drain system to the Hot Retention Area, as shown in Figure C.6-2.

The PCWS bypass deionizer resin pits will hold 13.6 m^3 of ion-exchange resin, which contains an estimated 0.3 Ci/m^3 of radioactivity at the time of discharge. When the resin activity has cooled to about 0.01 Ci/m^3, the resin is transferred into shielded or concrete mix shipping containers for transportation to a burial site. The resin pit has a thick concrete cover for shielding, and the inner surface of the pit is lined with stainless steel, equipped with strainers and an overflow drain to insure containment of the resin slurry in the pit.

A similar system of resin transfer and burial is used for spent resins in the waste cleanup deionizer spent resin pit.

The number and kinds of air filters used at the reference test reactor are discussed in Section C.5.5. It is postulated that all contaminated filters will be treated as radioactive solid waste and sealed in drums for offsite disposal.

Cartridge-type water filters are used in the Q&C recirculating water cleanup system and the waste cleanup system. The filters are allowed to drain before being packaged for offsite disposal.

It is postulated that much of the waste packaging is accomplished either in the semi-contaminated work area of the Hot Laboratory Building (see Figure C.2-7, Room 16) or in the Waste Handling Building. The area in the HLB contains packaging and treatment equipment for the disposition of contaminated wastes, including a dustless baler, concrete mixer, and a waste canner.

All of the solid wastes are concentrated in volume and packaged for offsite disposal. Bulk solids are packaged in wooden boxes or 0.189-m^3 drums or other suitable containers to comply with transfer and disposition regulations. Concrete or similar shielding materials are used where required. A solid-waste disposal flow diagram showing handling procedures and associated average radioactivity levels during operation of the reference test facility is presented in Figure C.6-3. It is postulated that wastes generated during decommissioning from the same or similar sources and having similar levels of radioactivity can be handled with confidence.

C.6.3 Gaseous Radwaste System

The gaseous waste disposal system is an integral part of the ventilation and contaminated air handling systems that are described in Section C.5 and illustrated in Figure C.5-2. The ventilating air intake equipment and intermediate ventilation filter and housing equipment serving the reference test reactor facilities are standard HVAC equipment. Air from the ventilation system serving contaminated work areas is filtered and monitored prior to discharge to the exhaust stack. Waste air from experimental cooling, process cooling, and off-gas systems is also passed through filters and monitored in a similar manner. All gaseous discharge at the reference test reactor that may be expected to contain radioactive contamination is passed through roughing filters followed by monitored absolute filters.

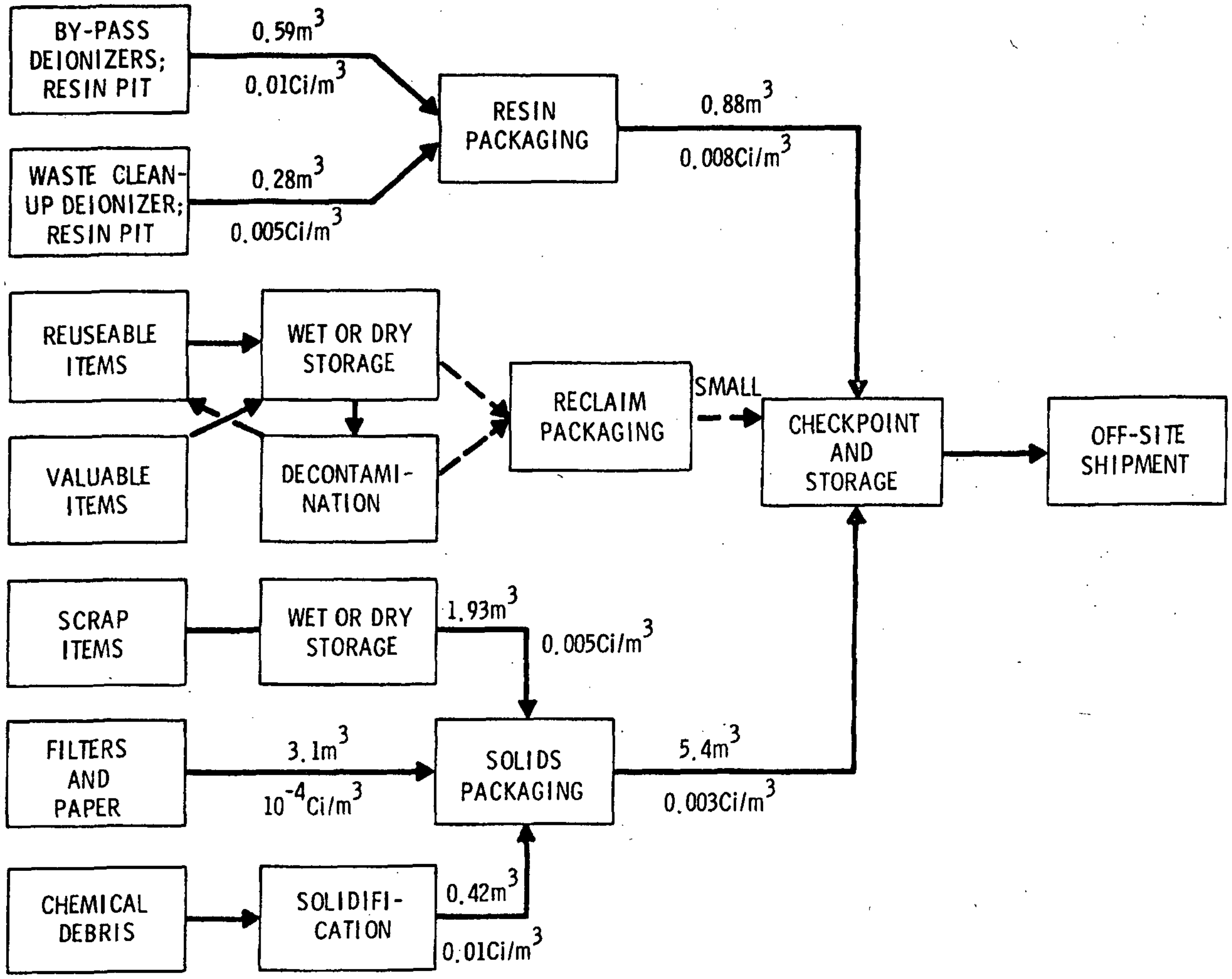


FIGURE C.6-3. Solid Waste Flow Diagram for Solids Disposal on a Monthly Basis During Operation

Major features of the equipment and pertinent design aspects of the gaseous radwaste systems are discussed in the subsequent subsections.

C.6.3.1 Cooling Air Systems

As previously described in Section C.5.2, the cooling air systems originate as a 45 m^3/min (maximum) filtered air supply at a pressure of 689 kPa from the Service Equipment Building. The supply is split for experimental cooling and process service. The experimental cooling air supply is for use with experimental test units. Process air is supplied for pneumatic rabbit

tubes and unfueled test cooling. The waste experimental cooling air flows to the Fan House and is discharged through a high-temperature absolute filter and control line to the exhaust stack. The air is monitored before passing to the stack. This system is designed to handle 478°K exhaust air.

The waste process air flows to the Fan House and is discharged through an absolute filter to the exhaust stack. This air is also monitored before being exhausted. No special high-temperature equipment is needed since this air is not used for high-temperature cooling.

C.6.3.2 Absolute Filters

The pre-exhaust filters on the ventilation air, cooling air, and off-gas systems are 99.97%-efficient absolute filters. Most of the filters used are of a standard variety of glass paper designed to operate at temperatures up to 394°K. However, the experimental cooling air filters are made of ceramic for much higher operating temperatures.

The filter system and fan are shut down and the absolute filters changed when the upstream surface of the filter indicates a residual radioactivity level of 180 mr/hour. The monitoring system indicates this limit for filter change and records the activity buildup rate. The pressure drop across the filters is limited to 4 to 9 in. of water, depending on filter size and materials. The estimated replacement period for the later limitation is 6 to 14 months during normal operation.

C.6.3.3 Off-Gas Process System

During plant shutdown, no radioactive gaseous fission products from the reactor pressure vessel or the hot laboratory (see Figure C.5-2) are anticipated since the sources of these gases are either operation of the reference test reactor or other work involving the fuel elements.

Other process system off-gas flow streams, however, will continue to feed into the main 374-m^3/min venting system and are postulated to be used during decommissioning. The Reactor Building-Hot Pipe Tunnel (RB-HPT), described in Section C.5.2.2, is one of these systems. This vent system operates from the

basement of the Reactor Building through the Hot Pipe Tunnel (also called the utility tunnel) under the Hot Laboratory Building to the Fan House. Vent gases from the reactor sump, the hot laboratory sump, and the treatment tank are collected into this system in the Hot Pipe Tunnel area. The radioactivity of this ventilating system is expected to be negligible in the basement area during shutdown. In any event, the sumps can be emptied and purged when access to the Hot Pipe Tunnel is required.

Vent gases from the Primary Pump House Building, sump, and resin pit are connected to a 11-m^3/min vent system in the Fan House. Under normal operation, the radioactivity of this air is estimated to be negligible. This system will operate at a negative pressure of 4 to 6 inches of water.

C.6.3.4 Off-Gas Experimental System

Since no off gas emanates from the experimental systems during plant shutdown, the off-gas experimental system is not needed during decommissioning. However, since this waste cooling air system is radioactively contaminated, it was described previously in Section C.5.3.2.

C.6.3.5 Off-Gas Cleanup System

The off-gas cleanup system is a high-efficiency unit designed primarily for the removal of gaseous fission products for the off-gas systems. The equipment for this unit was previously described in Section C.5.2.2. It is used in conjunction with the off-gas experimental system described above and as a cleanup purge unit for the reactor containment vessel (see Figure C.5-4). The cleanup unit is located in the dry storage area of the Hot Laboratory Building, Room 19A (see Figure C.2-7).

C.6.3.6 Exhaust Stack

The exhaust stack is described in Section C.5.4. Four exhaust headers from the contaminated ventilating systems, waste cooling air systems, and off-gas systems enter the exhaust stack at various points in the plenum section. The output from the exhaust stack is about 1300 m^3/min with all exhaust fans and forced air units in operation.

The stack effluent gases are monitored by the following systems: 1) a gross beta-gamma radiation detector (G-M tube); 2) a particulate radioactivity monitoring system; 3) a radioiodine monitoring system; and 4) a gaseous radioactivity monitoring system.

The G-M tube, positioned in the stack so that the effluent gas flows past it, is used to detect gross beta-gamma radioactivity. The particulate radioactivity monitoring system consists of a monitored filter. The radioiodine monitoring system consists of a monitored charcoal absorption bed. The gaseous radioactivity monitoring system consists of a monitored spherical volume.

A side stream from the stack is first passed through the particulate monitoring system, then through the radioiodine monitoring system, and finally through the gaseous radioactivity monitoring system. The detectable radioactivity level in each of these systems is 10^{-7} Ci/m^3. This is the lower limit obtainable in the gaseous radioactivity monitoring system; however, in the particulate and radioiodine systems it is possible to lower this limit if desired, since an integrated sample is being monitored in each of these systems.

C.7 ACRONYMS AND ABBREVIATIONS

Table C.7-1 lists the acronyms and abbreviations that are used in Appendix C.

TABLE C.7-1. Acronyms and Abbreviations in Appendix C

AC	Alternating Current
ALARA	As Low as Reasonably Achievable
ANL	Argonne National Laboratory
Anti-C	Anti-contamination
ATMS	Air Tank Monitoring System
Bldg.	Building
CAP	Capacity
CIC	Compensated Ion Chamber
CRA	Cold Retention Area

TABLE C.7-1. (contd)

CRD	Control Rod Drive
CS	Carbon Steel
CV	Containment Vessel
CVVS	Containment Vessel Vent System
DC	Direct Current
DECON	Decontamination
DOT	Department of Transportation
dp	Deep
El or Elev	Elevation
Elec	Electrical
EPA	Environmental Protection Agency
Exp	Experimental
FH	Fan House
FSAR	Final Safety Analysis Report
HB-1	Horizontal Beam Number 1
HEPA	High Efficiency Particulate Air/Absolute (referring to filters)
HLB	Hot Laboratory Building
HPT	Hot Pipe Tunnel
HRA	Hot Retention Area
Ht	Height
HT-1	Horizontal Tube Number 1
HV	High Voltage
HX	Heat Exchanger
HVAC	Heating, Ventilating and Air Conditioning
IX	Ion Exchanger
kV	Kilovolt
kWt	Kilowatt, thermal
LeRC	NASA Lewis Research Center
MCC	Motor Control Center
MG	Motor-Generator
MUR	Mock-Up Reactor

TABLE C.7-1. (contd)

MWT	Megawatt, thermal
NASA	National Aeronautics and Space Administration
NRC	Nuclear Regulatory Commission
OGCUS	Off-Gas Cleanup System
OLB	Office and Laboratory Building
PBRF	Plum Brook Reactor Facility
PCWS	Primary Cooling Water System
PPH	Primary Pump House
PPPR	Process Piping Pump Room
Q&C	Quadrant and Canal
Q&C-Po	Quadrant and Canal Pumpout System
Q&C-Recirc	Quadrant and Canal Recirculation System
RAMS	Remote Area Monitoring System
RB	Reactor Building
RM	Room
ROLB	Reactor Office and Laboratory Building
R&T	Research and Test
RPV	Reactor Pressure Vessel
SCWR	Secondary Cooling Water Return
SCWS	Secondary Cooling Water Supply
SEB	Service Equipment Building
SEMS	Stack Effluent Monitoring System
SPR	Subpile Room
SS	Stainless Steel
SWP	Safe Work Permit
SYS	System
TLD	Thermoluminescent Dosimeter
TYP	Typical
UWBR	Underwater Beam Room
VAFTS	Vertical Adjustable Facility Tubes
VERT	Vertical
WHB	Waste Handling Building

REFERENCES

1. Environmental Report, Plum Brook Reactor Dismantling, National Aeronautics and Space Administration, Lewis Research Center, February 1980.

2. Environmental Report, Plum Brook Mock-Up Reactor Dismantling, National Aeronautics and Space Administration, Lewis Research Center, February 1980.

3. Dismantling Plan, Plum Brook Reactor Dismantling, National Aeronautics and Space Administration, Lewis Research Center, February 1980

4. Dismantling Plan, Plum Brook Mock-Up Reactor Dismantling, National Aeronautics and Space Administration, Lewis Research Center, February 1980.

5. NASA Plum Brook Reactor Facility Final Hazards Summary, Parts I, II, and III Lewis Research Center, Cleveland, Ohio, December 1959.

6. U.S. Atomic Energy Commission, Chemical Processing and Equipment, Superintendent of Documents, GPO, Washington, D.C., 1955.

APPENDIX D

RADIATION DOSE RATES AND CONCRETE SURFACE CONTAMINATION DATA

The radiation dose rate at a specific work site has an important influence on the time needed to complete each decommissioning task. In addition, the degree of concrete contamination determines how much surface requires removal and how much contaminated rubble requires disposal. Established in this appendix are the dose rates and the concrete surface contamination levels that are assumed to be present in the reference research and test (R&T) reactors at final shutdown, except for the dose rates resulting from activated material in and surrounding the reactor vessel. These latter dose rates for the reference R&T reactors are presented in Appendix E.

D.1 ESTIMATED RADIATION DOSE RATES AT SHUTDOWN IN THE REFERENCE RESEARCH REACTOR FACILITY

Radiation dose survey information was obtained for the operational and shutdown levels of the reference research reactor and the 1-MW Texas A&M TRIGA reactor, located at College Station, Texas. A compilation of routine radiation monitoring surveys recently taken at the reference research reactor is summarized in Table D.1-1. The data are in agreement with the Texas A&M reactor information. It is assumed that the reference research reactor has shut down after a 40-year operational lifetime. This assumed lifetime is consistent with that used in previous decommissioning studies in this series.[1,2] No dose rate data are presently available for a licensed research reactor with a 40-year lifespan; therefore, the results of the recent surveys taken at the reference research reactor after 5 years of operation are used to assess the radiation dose rates at shutdown for the locations shown in Table D.1-1.

In general, radiation dose rates were found to be relatively low, due to the low power level of the reactor. Even during operation few areas exist in which restriction of personnel activities are necessary due to radiation

TABLE D.1-1. Annual Summary of Radiation Levels and Contamination Levels Observed During Routine Radiation Surveys for the Year July 1, 1979 Through June 30, 1980, at the Reference Research Reactor[3]

Location	Direct Radiation Levels (mRem/hr) (βγ+neutrons) Average		Direct Radiation Levels Maximum		Contamination Levels (dpm/100 cm²) (βγ)[a] Average		Contamination Levels Maximum	
Reactor Top	<1.00		143.00		≤370		≤370	
Sampling Handling Area	<1.00		143.00		≤370		≤370	
Reactor Room Floor	<1.00		120.00		≤370		≤370	
Beam Port Facilities	<1.00		96.00		≤370		≤370	
Demineralizer Tank	Outside Shield		Inside Shield		Outside Shield		Inside Shield	
	Avg.	Max.	Avg.	Max.	Avg.	Max.	Avg.	Max.
	<1.00	3.00	39.00	150.00	--[b]	--	--	--

(a) No contamination was found at the designated locations during the entire reporting period. The 370 dpm/100 cm² value used in this table is based on the normal beta counting efficiency and a net count rate equal to the normal background counting rate for the portable survey meters routinely used in the field to screen for radioactive contamination (i.e., field measurements would normally have to show a gross counting rate equal to twice the normal background counting rate before contamination would be considered present). However, in addition to normal field screening for contamination by direct surveys and smear samples, those smears suspected of containing removable radioactive contamination are routinely counted in a more sensitive radiation detection system. Based on usual counting times, a normal instrument counting efficiency, and a typical background counting rate, during the current reporting period such as a detection system typically provided a lower limit of detection at 95% confidence of approximately 11-12 dpm for the radionuclides normally expected to be on the smears. Smearing efficiency for radioactivity removal is conservatively assumed to be ∼10%, and positive smear results would usually be multiplied by 10 before final conversion to dpm/100 cm².
(b) No data available.

exposure. Health physics personnel have observed that after a short period following reactor shutdown only few specific components away from the reactor proper have significant radiation levels.

D.1.1 Assumptions and Definitions

This section contains the assumptions and definitions related to the estimated dose rates and surface contamination levels considered to be present in the reference research reactor facility at the time of final shutdown.

The following assumptions and definitions are made concerning dose rate conditions throughout the reference plant:

- The plant is in a shutdown/defueling mode, with no fuel movements in progress.
- "General area" refers to the radiation field not emanating specifically from one discrete source or direction in a room or area, although a specific source may be the sole contributor to the radiation measurement. General area readings are taken at least 1 m from any surface.
- "Contact" means a dose rate at the closest approach to a given surface (a surface dose rate), including the necessary corrections for geometry and source size made in the field by the health physics technician.
- Estimated radiation dose rates are made where specific data is unavailable, but where sufficient information concerning the surrounding areas made estimates meaningful.
- Shutdown time is a time which allows decay of almost all of the short half-life radioisotopes expected to be present in the reference reactor. It does not exceed 30 days in duration.

Study assumptions relating to concrete surface contamination at shutdown provide the bases for concrete rubble volumes for the reference research reactor. These are:

- The final radiation dose rate estimates take into account the ordinary cleanup operations that have taken place during the lifetime of the reactor.
- The percentage of the total area that is assumed contaminated is based upon actual operational area effected or in proximity with known radioactive materials or that portion that can reasonably be expected to be affected.
- The spalling depth for decontamination of concrete surfaces is 0.05 m.
- Contaminated concrete walls are completely broken up and removed as contaminated rubble.

- All concrete associated with contaminated floor storage areas are considered contaminated waste rubble.

It should be recognized that to date no significant surface contamination has been observed at the reference research reactor. Several assumptions and postulations are made throughout this study concerning the eventual use of radioactive material storage and handling facilities that will, by their nature, result in some surface contamination. In addition, certain areas are not accessible for routine inspection by health physics personnel and, in selected situations, these areas are also postulated to contain surface contamination on the concrete.

D.1.2 Presentation of Data

The radiation dose rates for the main reactor room and the first floor of the Reactor Building and adjoining buildings of the reference research reactor site at shutdown are illustrated in Figure D.1-1. It is assumed that dose readings do not exceed 0.0025 R/hr on the remaining floors of the Reactor Building. In most situations, the small amount of general area exposure is due to a particular piece of equipment present within the room or building. All other buildings and structures on the reference site are assessed to have no potential for radiation exposure.

It is clear, from the values shown that, except in a few specific instances, the decontamination activities may proceed with maximum efficiency, with no significant contribution to personnel dose.

Both the lower power level and the construction materials used in the reference research reactor contributed to the lessening of radiation dose at any particular time during operation and shutdown. It is also true that decontamination activities during the plant lifetime will serve to reduce the ultimate radiation dose at shutdown.

Based upon the assumptions made in Section D.1.1, estimates of the contaminated surface area expected throughout the reference research reactor are given in Table D.1-2.

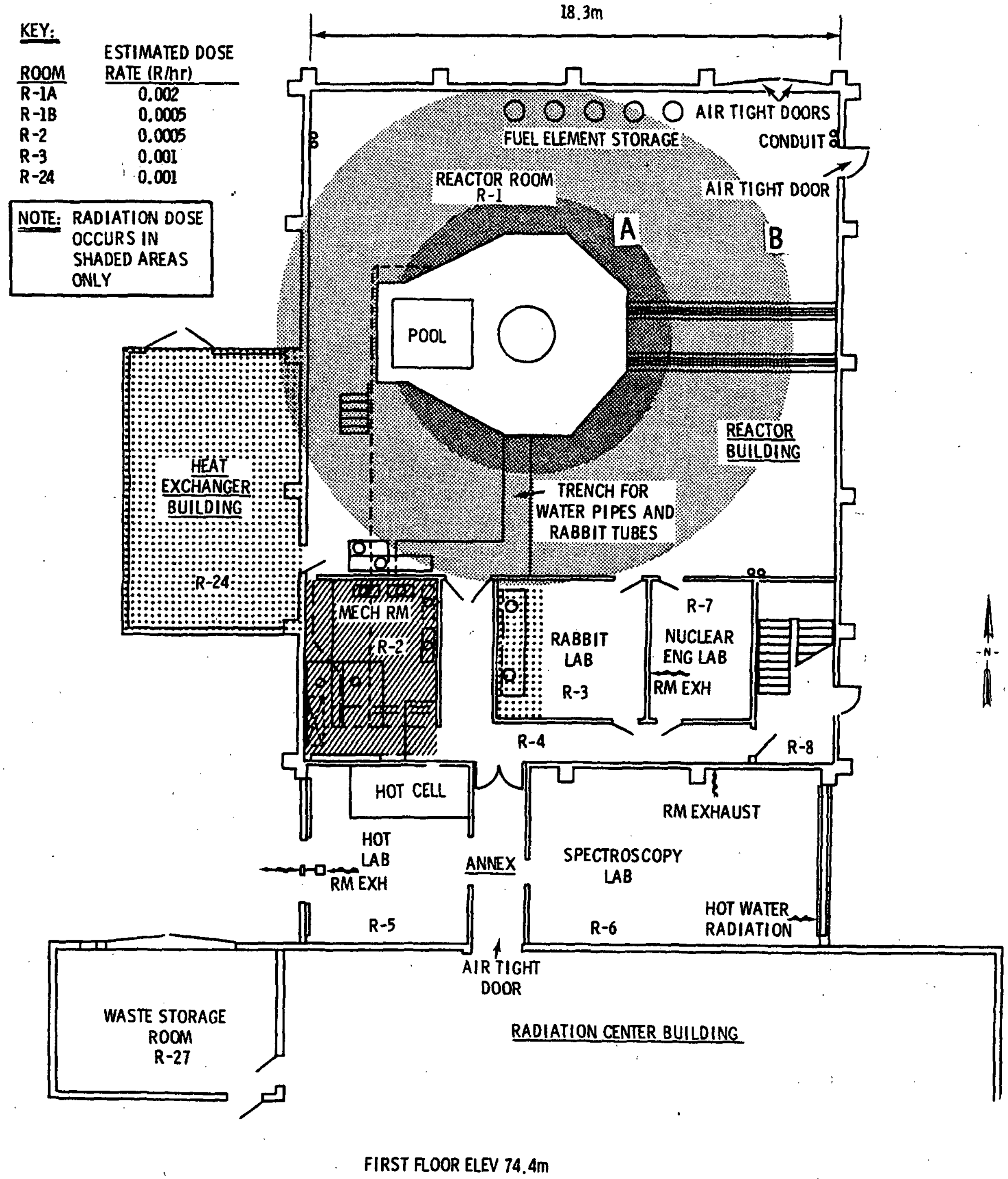


FIGURE D.1-1. Reference Research Reactor Radiation Dose Rates at Time of Decommissioning

TABLE D.1-2. Concrete Surface Contamination at the Reference Research Reactor

Location	Estimated Surface Area (m^2)	Percent of Area Assumed Contaminated	Rubble Volume[a] (m^3)
Reactor Building			
Inner Surface of Reactor Structure	31	100	4.6
Reactor Top	9.4	20	0.1
Storage Pits (3 each)	0.5	100	0.1
Annex			
Hot Cell	13.5	100	8.5[b]
Hot Lab	40	10	0.2
Hot Lab Sump	15	100	0.8
HX Building			
Floor	54	10	0.3
Sump	15	100	0.8
Pump House			
Concrete Floor Pad	60	10	0.4
Sump	15	100	0.8
Waste Storage Room			
Concrete Floor Pad	38	10	0.2
Sump	15	100	0.8
Total Waste Volume			17.6 m^3

(a) Based on a contamination thickness of 0.051 m.
(b) Rubble volume includes entire concrete structure of cell.

D.1.3 Application of Data

The data presented in Section D.1.2 is used to estimate the waste volume generated during the decommissioning tasks of concrete cleanup, as described in Appendices I, J, and K, as well as the manpower requirements for completion of the tasks.

D.2 ESTIMATED RADIATION DOSE RATES AND SURFACE CONTAMINATION LEVELS AT SHUTDOWN IN THE REFERENCE TEST REACTOR FACILITY

A final radiation survey was made when the reference test reactor was placed in safe storage in 1973. The radiation dose rates and contamination levels measured at that time are given in Table D.2-1. Data from more recent surveys made in 1978 are also shown in brackets in the table next to the 1973 values.

A methodology for establishing a useful data base of radiation dose rates and surface contamination levels for conceptually decommissioning the reference test reactor facility as though it had only recently shut down is discussed in the following subsections.

D.2.1 Assumptions and Definitions

To estimate dose rates and surface contamination levels present in the reference test reactor facility at the time of shutdown, the following assumptions and definitions are made:

- For the majority of piping and equipment directly associated with the primary cooling water system (PCWS), the measured 1973, after-cleanup, dose rates should be doubled. This is analgous to assuming a decontamination factor (DF) of 2 for the majority of piping and equipment.
- The mid-1973 contamination levels reported for the hot cells at the reference test reactor facility are increased to account for the post-shutdown cleanup activities (see Appendix E, Section E.2.2.3 for details).
- The radiation dose rate estimates do not include any buildup of large quantities of process material or debris within the test facility because these are assumed to be cleaned up as a part of normal housekeeping procedures during plant operation.
- ^{60}Co is the primary dose contributor in all areas presented in Table D.2-1, except for the hot cells where cutting of fuel plates and other experimental hardware took place (see Appendix E, Section E.2.2.3 for estimated radionuclide inventories).

TABLE D.2-1. Radiation Levels Measured After Placing the Reference Test Reactor in Safe Storage in Mid-1973[(a)]

Location	Direct Radiation (mR/hr) Year: 1973, [1978]	Smearable Contamination (d/m 100 cm²)[(b)] Year: 1973
Quadrants		
General Field	5 to 10 [5]	60 to 10 000
RT Shielding	150 to 350 [150]	90 to 800
Sub-pile		
General Field	10 to 50	90 to 800
Maximum (roof)	250	
Canals E and G		
General	<5 to 10 [5]	120 to 14 500
Maximum (drain)	40	
Canals F and J		
General	10 to 15 [5]	1 100 to 63 000
Maximum (equipment)	65	
Hot Laboratory		
General Behind Cells	2 to 5	15 to 10 000
Cell 1 - General	125 [10 to 25]	--
Cell 2 - General	100 to 150 [20]	20 000 to 400 000
Cell 3 - General	10 [3]	270 to 150 000
Cell 4 - General	<5 [<5]	440 to 10 000
Cell 5 - General	10 [3]	170 to 3 600
Cell 6 - General	25 [2]	2 700 to 200 000
Cell 7 - General	<5 [0.5]	35 to 4 000
Cell 1 - Maximum	-- [3 000]	--
Cell 2 - Maximum	200 to 500 [25]	--
Cell 3 - Maximum	200 [12.5]	--
Cell 4 - Maximum	10 --	--
Cell 5 - Maximum	25 [5]	--
Cell 6 - Maximum	500	--
Cell 7 - Maximum	20 [20]	--
Hot Pipe Tunnel		
General Field	225 to 750 [20 to 300]	3 000 to 20 000
Maximum at Drain Line	7 500 [4 000]	
Primary Pump House (PPH)		
General	5	5 to 50 (minimum range)
Maximum at Valves and and HXs	300 to 5 000	55 to 4 400 (maximum range)
Sumps		
Hot Lab - Maximum	300	
PPH - General	10	
Evaporator - General	60 to 70	
Waste Handling Building		
Evaporator - Maximum	200	
Cold Retention Area		
General	1	
Maximum	<10	1 200 to 9 000
Hot Retention Area Tanks	<1 to 7	46 to 1 800 (minimum range
General (inside)		690 to 23 000 (maximum range

(a) Data from Reference 4.
(b) Disintegrations per minute per 100 square centimeters of surface surveyed.

- "General field" refers to the radiation field not emanating specifically from one discrete source or direction in a room or area, although a specific source may be the sole contributor to the radiation measurement. General field readings are taken at least 1 m from any surface.
- "Contact" means a dose rate at the closest approach to a given surface (a surface dose rate), including the necessary corrections for geometry and source size made in the field by the health physics technician.
- Crud traps that may exist at valves or at sharp bends in piping represent potential high dose rate areas that are not explicitly considered.

Study assumptions relating to concrete surface contamination at shutdown provide the bases for concrete rubble volumes for the reference test reactor. These are:

- The estimated level of surface contamination in a particular area is based on the plant process carried out in, and the accessibility and/or occupancy of, that area during plant operation. Contamination at specific locations within a given area may vary significantly from that in the general area.
- The percentage of the total area that is assumed contaminated is based on the measured data wherever possible and on the plant process carried out in that location.
- The spalling depth for reinforced concrete surfaces is 0.051 m. This depth is assumed to be the minimum depth of concrete cover on reinforcing bar at the reference test reactor and is presumed sufficient to remove all measurable contamination.
- Contaminated concrete block walls or concrete curbs are completely broken up and removed as contaminated rubble.

D.2.2 Presentation of Data

Based on the assumptions presented in Section D.2-1, a new table is generated to provide the bases for conceptually decommissioning the reference

test reactor facility as though it had only recently shut down. These modified estimates are given in Table D.2-2, together with data for additional locations identified at the reference test facility as a result of communications with former operating personnel of the facility.

D.2.3 Application of Data

The entire project required to complete a particular decommissioning alternative is broken down into individual tasks to be performed in each specific area, and the man-hours of radiation exposure are then estimated for each task. In applying the dose rate data presented in this appendix, adjustments are made for ALARA considerations, for decontamination during decommissioning, for special shielding methods, and for other appropriate judgment considerations, as described in Appendices H, I, J, and K. The adjusted dose rate for each specific work site is then multiplied by the estimated man-hours of exposure for each task. After correcting for radioactive decay, the individual task exposures are totaled for the entire decommissioning effort. Radioactive decay of ^{60}Co is the primary basis in this study for calculating the decay factor for each task. Additional information on the makeup of the surface contamination affecting radiation dose rates in the plant is contained in Section E.2.2 of Appendix E.

The measured concrete surface contamination data in the reference test reactor facility are not specifically applied to any analysis in this study, except to help determine the "assumed contamination area" in a specific location and to determine, in those instances where plant process does not dictate, specific areas of concrete surface that will probably need to be decontaminated during decommissioning. In this study, contaminated concrete surfaces are assumed to be decontaminated to acceptable levels only by removing the surface either with a concrete spaller or with other physical-removal means. The volume of concrete rubble or material that requires removal for surface decontamination is calculated as the product of: 1) the total surface area associated with the particular piece of equipment, 2) the "percent of area assumed contaminated," and 3) the thickness of the material to be removed.

With the exception of the contaminated concrete piping discussed in Section C.4 of Appendix C, the estimated concrete areas with surface contamination are presented in Table D.2-3. No concrete surface contamination is assumed in the other buildings and structures at the reference test reactor site.

TABLE D.2-2. Estimated Radiation Dose Rates and Surface Contamination Levels Used for the Conceptual Decommissioning of the Reference Test Reactor Facility

Location	Estimated Dose Rate (R/hr)/ Type of Measurement	Estimated Smearable Contamination $(d/m/100\ cm^2)$[a]
Reactor Building and Containment Vessel		
Reactor Tank with Internals and Water Shield	~0.030 - 0.150 general field[b]	--[c]
Reactor Tank Empty	~0.060/contact[d]	--
Sub-pile Room	0.020 - 0.100/general field	0.2 to 1.6 K[e]
Maximum (roof)	0.250	--
Quadrants	0.010 - 0.020/general field	0.1 - 20 K
RT Shielding	0.300 - 0.700	--
Canals E&G, When Empty	<0.010 - 0.20/general field	0.25 - 29 K
Drains	0.080 - 2.0	--
Canal F, When Empty	0.-20 - 0.030/general field	2.2 - 126 K
Drain	1 - 2	--
Canal H, When Full	<0.001/general field	--
When Empty	<0.010/general field	--
MUR	<0.010 - 0.700/contact[f]	--
Drain	1.0 - 2.0	--
Hot Laboratory Building		
Behind Cells	0.005 - 0.010/general field	1.5 - 1 000 K
Decontamination Room No. 23	<0.005	--
Repair Shop Room No. 22	0.002 - 0.005	--
Sump, Maximum	0.6	--
Cell Drains	1 - 2/contact	--
Cell 1	1 - 1.5/general field	1.0×10^8 K
Cell 2	1 - 1.5/general field	1.0×10^8 K
Cell 3	~0.5/general field	5×10^6 K
Cell 4	~0.5/general field	3×10^6 K
Cell 5	~0.5/general field	1×10^6 K
Cell 6	≥0.5/general field	2×10^7 K
Cell 7	~0.3/general field	4×10^5 K
Cell Manipulators	1 - 2	--
Canal J and K, When Empty	0.020 - 0.030/general field	2.2 - 126
Hot Drain	1 - 2	--

TABLE D.2-2. (contd)

Location	Estimated Dose Rate (R/hr)/ Type of Measurement	Estimated Smearable Contamination (d/m/100 cm^2)(a)
Hot Pipe Tunnel	0.5 - 1.5/general field	6 - 40 K
Maximum at Drain Line	15	
Primary Pump House (PPH)	0.010/general field	0.1 - 9 K
Maximum at Valves and HXs	0.6 - 10	
Pumps	0.050/general field	--
Sumps	0.020/general field	--
Resin Pit (So. side of PPH)		
Tanks Full	2 - 5/contact	--
Tanks Empty	≥0.020/general field	--
Waste Handling Building		
Evaporator Sump	0.120 - 0.140/general field	--
Evaporator, Maximum	0.4	--
Cold Retention Area	0.002 - 0.010	2 - 18 K
Hot Retention Area		
Inside Tank No. 1	∿0.100/general field	0.1 - 3.6 K (minimum range)
Inside Tanks No. 2 through 8	∿0.010 - 0.015/general field	1.4 - 46 K (maximum range)
Fan House		
Pipe Trench and Resin Pit	≤0.010/general field	--
Resin Pits (SW of Fan House), Empty	≤0.010/general field	--

(a) Disintegrations per minute per 100 square centimeters of surface surveyed.
(b) "General field" refers to the radiation field not emanating specifically from one discrete source or direction in a room or area, although a specific source may be the sole contributor to the radiation measurement. General field readings are taken at least 1 m from any surface.
(c) Indicates data not available.
(d) "Contact" means a dose rate at the closest approach to a given surface (a surface dose rate), including the necessary corrections for geometry and source size made in the field by the health physics technician.
(e) 0.2 to 1.6 K stands for 200 to 1,600 d/m/100 cm^2 (typical).
(f) See Section E.2.1.2 for details.

TABLE D.2-3. Summary of Estimated Contaminated Concrete Waste Material Quantities in the Reference Test Reactor Facility[a]

Location	Surface Area (m^2)	Percent of Area Assumed Contaminated	Rubble Volume[b] (m^3)
Reactor Building and Containment Vessel			
1st Floor	2416	100	123.2
Reactor Well Cavity	21	100	1.1
Quadrant "A"	335	100	17.1[c]
Quadrant "B"	353	100	18.0[c]
Quadrant "C"	335	100	17.1[c]
Quadrant "D"	297	100	15.1[c]
Underwater Beam Room	93	100	4.7
Canal "E"	366[d]	100	18.7[c]
Dry Annulus	1022	100	52.1
Sumps (4)	∿50	100	2.6
Experiment Decontamination Room	34	20	0.3
Lily Pad	15	100	0.8
Canal F	170	100	8.7[c]
Canal G	222	100	11.3[c]
Canal H (including MUR)	193	100	9.8
Pump Room Area 22	67	50	1.7
Hot Laboratory Building			
Hot Cells 1-7	581	40	11.9
Hot Dry Storage	340	50	8.7
Canal J	282	100	14.4
Canal K	300	100	15.3
Off-gas Cleanup Room	170	100	8.7
Valve Pit	17	100	0.9
Hot Pipe Tunnel	307	50	7.8
Hot Handling Room 17	56	50	1.4
Hot Work Area Room 16	112	50	2.9

TABLE D.2-3. (contd)

Location	Surface Area (m^2)	Percent of Area Assumed Contaminated	Rubble Volume[b] (m^3)
Decontamination Room 23	12	100	0.6
Repair Shop Room 22	30	100	1.5
Sump	25	100	1.3
Fan House			
Sump Room	25	100	1.3
Resin Pit	13	100	0.7
Pipe Trench	18	50	0.5
Deionizer Room	15	100	0.8
Waste Handling Bldg.			
Decontamination Room 17	38	50	1
Evaporator Room 18	38	100	1.9
Laundry	21	25	0.3
Sumps	25	100	1.3
Equipment Room 8	270	25	3.4
Primary Pump House			
Resin Pit	21	100	1.1
Sump	25	100	1.3
Primary Pump Rooms	48	100	2.4
Degassier Room	14	100	0.7
Deionizer Room	24	100	1.2
HX Room	100	50	2.6
Hot Retention Area			
Floor Area (including sumps)	424	50	11.5
Cold Retention Area			
Floor Area	1252	100	191[e]
Emergency Retention Basin	--[f]	--	--

TABLE D.2-3. (contd)

Location	Surface Area (m^2)	Percent of Area Assumed Contaminated	Rubble Volume(b) (m^3)
Office and Laboratory Bldg.			
Sumps	50	50	1.3
Utility Tunnel	27	50	0.7
Misc.	~25	50	0.6
Total			603.3

(a) Does not include contaminated concrete piping (see Appendix C.4 for details).
(b) Based on a contamination thickness of 0.051; does not include a packing factors.
(c) Includes the drain area.
(d) Does not include that portion of the canal formed by the metal containment vessel wall.
(e) Includes the total basin area for both basins, which is formed by the 0.15-m-thick concrete base slabs.
(f) Included for completeness; negligible amount of contaminated concrete assumed.

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APPENDIX E

RADIONUCLIDE INVENTORIES

The estimated radionuclide inventories at the shutdown, defueled, reference research and test (R&T) reactors are presented in this appendix, together with supporting information on the calculational methods for estimating these inventories. The radionuclide inventories in these facilities at the time of final reactor shutdown (excluding the irradiated spent fuel) are of two types: 1) neutron-activated components in and surrounding the reactor core, and 2) surface contamination from activated material deposited inside certain piping and equipment systems, on some structural surfaces, and on the site.

The radioactivity levels present in the neutron-activated portions of the reference R&T reactors have been calculated to facilitate making estimates of shielding and packaging requirements, disposal costs, and potential personnel radiation exposure rates for the removal and disposal of these materials from the reference R&T reactors. It should be recognized that the data presented in this appendix are calculated estimates specific to the reference R&T reactors defined for this study. Use of these data in an analysis of any other research or test reactor should be made with caution and careful attention to any differences in structural materials, neutron flux levels, and reactor operating histories.

E.1 RADIONUCLIDE INVENTORIES IN THE REFERENCE RESEARCH REACTOR FACILITY

The reference research reactor operates at a power of 1 MW. At this power level the maximum thermal neutron flux in the reactor core is less than 2.5×10^{13} n/cm^2/sec. For the purposes of this study, it is estimated that reactor operation is intermittent over the postulated 40-year operational lifetime, representing little over 5% of the available time. This estimate is based on the 5-year operating history at the reference research reactor. The postulated 40-year lifetime is consistent with previous decommissioning studies in this series. These factors contribute to a low total integrated exposure of the materials near the reactor core.

The inventory of longer half-life radionuclides that remains to be dealt with during decommissioning is very sensitive to the construction materials in the core vicinity. Neutron activation products from stainless steel contribute heavily to the long-term radionuclide inventory, while those of aluminum alloys are much less significant. Aluminum alloys are used extensively in the reactor zones where activation products are produced. In comparison, stainless steel represents only 6.6% by weight of the materials within these zones.

Data on the quantities of neutron activation products of importance to this study and their corresponding radiation dose rates are presented in Section E.1.1. Data on the radionuclides and levels of contamination accrued from plant operation for its postulated 40-year useful lifetime are given in Section E.1.2.

E.1.1 Neutron-Activated Materials in the Reference Research Reactor

The reference research reactor is estimated to accumulate a total of 740 MWd over a 40-year lifetime, with a power level of 1 MW, for a total of 740 effective full-power days (EFPD), or 2.03 effective full-power years (EFPY), by extrapolating the 5-year operating history to a 40-year lifetime.

Using this irradiation time and the 0- to 0.7-m thermal neutron flux profile of the reactor shown in Figure E.1-1, the total integrated exposure for neutron activation is calculated for the location of each component in the core vicinity extrapolated to a maximum distance of 1.8 m. The masses of the components are calculated based on dimensional diagrams and specific data concerning their composition. In some instances portions of larger components are differentially affected by significant changes in the flux zones with distance from the core. In these situations the total neutron exposure for a component is calculated by summing each of the portions exposed to an average neutron flux in a specific zone.

A summary of materials found in the highest neutron flux zones of the reference test reactor is given in Table E.1-1. Essentially the same elemental composition is present in the materials of the research reactor; thus, the types of neutron activation products that are produced in these materials are assumed to be similar for both types of reactors.

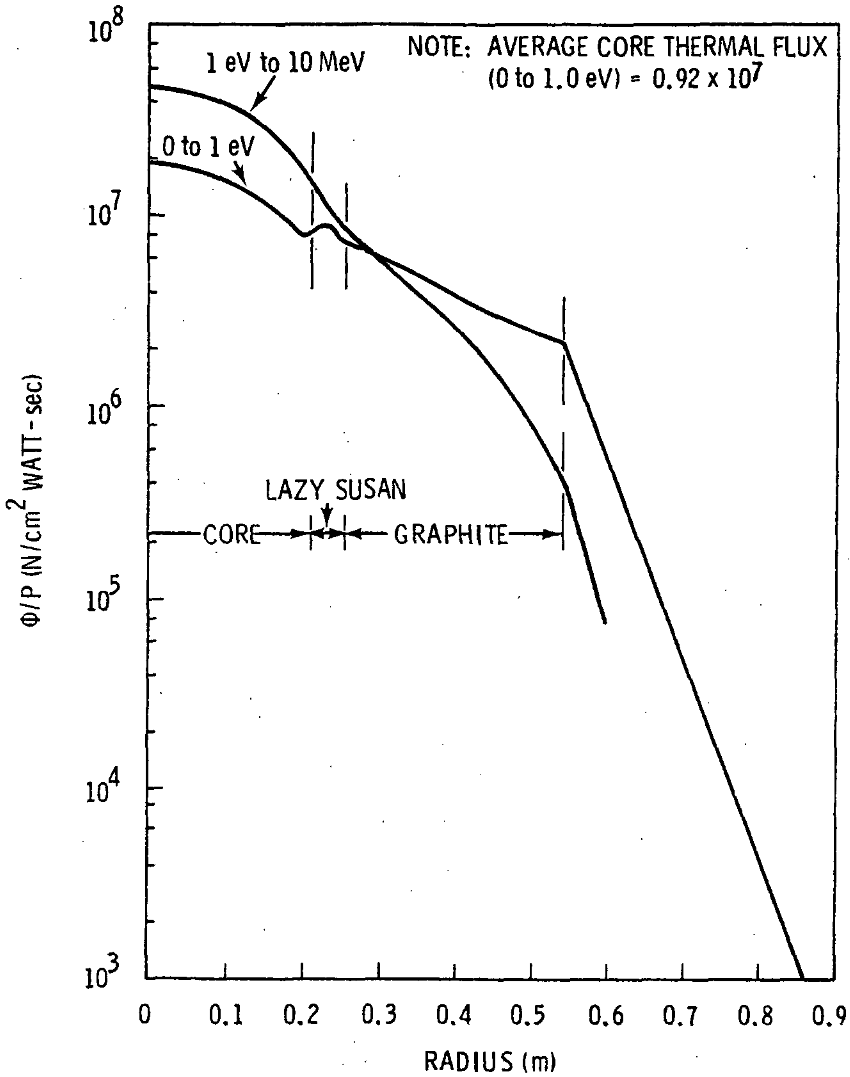

FIGURE E.1-1. Axial Mid-Plane Flux per Watt Versus Radius

Two radionuclides stand out as being particularly important with respect to their impact on the radiological dose to personnel, their disposal requirements, and their potential impact on public safety during decommissioning of the reference research reactor. These are: cobalt in stainless steel, contributing to the production of ^{60}Co; and zinc in aluminum alloys, producing ^{65}Zn. In addition, significant quantities of ^{14}C are produced in the graphite moderator material near the reactor core. While ^{14}C contributes little to the external dose rate since it is a weak beta emitter, the potential for contamination of the larger volumes of materials must be considered in the waste disposal process.

TABLE E.1-1. Elemental Weight Fractions and Neutronic Reactions Calculated in the Reference Test Reactor Materials

	Element Weight Fraction in Material					
Element	304 Stainless Steel	356-T Aluminum	6061-T6 Aluminum	Lockalloy	Beryllium	Neutronic Reactions in Reactor Materials
Aluminum		0.9035	0.9585	0.38	0.0009	$^{27}Al\ (n,\gamma)\ ^{28}Al$
Beryllium				0.5137	0.996	$^{9}Be\ (n,\alpha)\ ^{6}He \xrightarrow{\beta^-} {}^{6}Li\ (n,\alpha)\ ^{3}H$
Cadmium					0.000002	$^{112}Cd\ (n,\gamma)\ ^{113m}Cd$ $^{114}Cd\ (n,\gamma)\ ^{115m}Cd$
Carbon				0.001		
Chromium	0.20		0.0035			
Cobalt	0.001				0.000005	$^{59}Co\ (n,\gamma)\ ^{60m}Co,\ ^{60}Co$
Copper		0.002	0.004	0.0008		$^{63}Cu\ (n,\alpha),\ ^{60}Co,\ ^{63}Cu\ (n,p)\ ^{63}Ni$
Iron	0.649	0.006	0.007	0.002	0.0014	$^{54}Fe\ (n,\gamma)\ ^{55}Fe,\ ^{54}Fe\ (n,p)\ ^{54}Mn$
Magnesium		0.004	0.012		0.0006	
Manganese	0.02	0.003	0.0015			$^{55}Mn\ (n,2n)\ ^{54}Mn$
Nickel	0.12					$^{58}Ni\ (n,\gamma)\ ^{59}Ni,\ ^{58}Ni\ (n,p)\ ^{58m}Co,\ ^{58}Co,$ $^{60}Ni\ (n,p)\ ^{60}Co,\ ^{62}Ni\ (n,\gamma)\ ^{63}Ni$
Phosphorus	0.00045					
Silicon	0.01	0.075	0.008	0.002	0.0008	
Sulfur	0.0003					
Titanium		0.002	0.0015			$^{46}Ti\ (n,p)\ ^{146}Sc$
Zinc		0.003	0.0025			$^{64}Zn\ (n,\gamma)\ ^{65}Zn$
$BeO + Al_2O_3$				0.10		$^{9}Be\ (n,\alpha)\ ^{6}He \xrightarrow{\beta^-} {}^{6}Li\ (n,\alpha)\ ^{3}H$ $^{27}Al\ (n,\gamma)\ ^{28}Al$

From the foregoing information, the quantities of these activation products are calculated for each component. The results of these calculations are summarized in Table E.1-2 for stainless steel components, Table E.1-3 for aluminum alloy components, and Table E.1-4 for carbon components. Because of its long half-life (5730 years), little decrease will take place in the quantity of ^{14}C shown in Table E.1-4 during the decommissioning alternatives considered in this study.

Calculation of activation products at distances comparable to the vicinity of the biological shield indicate that only small quantities of radionuclides are produced in the concrete. For this reason the low-level radioactivity present in the concrete is considered as part of the contaminated materials discussion presented in Section E.1.2. A more complete description of the fractional radionuclide inventory decrease with time (out to a maximum of

TABLE E.1-2. Activated Stainless Steel Inventory and Associated Radiation Dose Rates in the Reference Research Reactor at Shutdown and 10 Years After Shutdown

Stainless Steel Components	Mass (Mg)	Volume (m^3)[a]	Ci (at shutdown)				Ci (10 yrs after shutdown)		Years After Shutdown: 0		10	
			Total Activity[b]	Total Co	^{60}Co	^{58}Co	Total Activity[c]	Total Co	Ci/m^3 (Co)	R/hr (Co)[d]	Ci/m^3 (Co)	R/hr (Co)[d]
Reflector Centering and Leveling Screws	1.0×10^{-3}	1.2×10^{-4}	1.06×10^{1}	9.04×10^{0}	8.41×10^{6}	5.25×10^{-1}	5.32×10^{0}	2.41×10^{0}	7.53×10^{4}	3.39×10^{4}	2.01×10^{4}	9.05×10^{3}
Grid Plate Inserts and Hardware	4.8×10^{-3}	6.0×10^{-4}	2.75×10^{2}	2.35×10^{2}	1.73×10^{2}	6.28×10^{1}	1.38×10^{2}	6.26×10^{1}	3.92×10^{5}	1.76×10^{5}	1.04×10^{5}	4.68×10^{4}
Transient Rod Hardware	3.0×10^{-4}	3.7×10^{-5}	1.93×10^{1}	1.65×10^{1}	1.23×10^{1}	4.23×10^{0}	9.70×10^{0}	4.40×10^{0}	4.46×10^{5}	2.01×10^{5}	1.19×10^{5}	5.36×10^{4}
Control Rods (3 each)	9.0×10^{-3}	1.1×10^{-3}	5.46×10^{2}	4.67×10^{2}	3.46×10^{2}	1.21×10^{2}	2.74×10^{2}	1.24×10^{2}	4.25×10^{5}	1.91×10^{5}	1.12×10^{5}	5.04×10^{4}
Fuel Storage Rack Hardware	6.0×10^{-4}	7.5×10^{-5}	7.47×10^{-4}	6.35×10^{-4}	6.35×10^{-4}	---[e]	3.75×10^{-4}	1.69×10^{-4}	8.47×10^{0}	3.81×10^{0}	2.25×10^{0}	1.01×10^{0}
Rotary Specimen Rack Hardware	1.7×10^{-2}	2.1×10^{-3}	2.19×10^{2}	1.87×10^{2}	1.64×10^{2}	2.30×10^{1}	1.10×10^{2}	4.98×10^{1}	8.90×10^{4}	4.01×10^{4}	2.37×10^{4}	1.07×10^{-2}
Beam Tube Bellows	1.3×10^{-2}	1.6×10^{-3}	1.47×10^{-1}	1.25×10^{-1}	1.25×10^{-1}	---	7.39×10^{-2}	3.33×10^{-2}	7.81×10^{1}	3.51×10^{1}	2.08×10^{1}	9.36×10^{1}
Shadow Shields	3.4×10^{0}	4.2×10^{-1}	4.82×10^{-6}	4.10×10^{-6}	4.10×10^{-6}	---	2.42×10^{-6}	1.09×10^{-6}	9.76×10^{-6}	4.39×10^{-6}	2.59×10^{-6}	1.17×10^{-6}
Outer Beam Tubes	6.5×10^{-2}	8.1×10^{-3}	1.83×10^{-9}	1.56×10^{-9}	1.56×10^{-9}	---	9.24×10^{-10}	4.16×10^{-10}	1.93×10^{-7}	8.67×10^{-8}	5.14×10^{-8}	2.31×10^{-8}
Thermalizing Column Bolt Inserts	3.6×10^{-3}	4.5×10^{-4}	1.02×10^{-13}	8.70×10^{-14}			5.15×10^{-15}	2.32×10^{-14}	1.93×10^{-10}	8.70×10^{-11}	5.16×10^{-11}	2.32×10^{-11}
Totals			1.07×10^{3}				5.37×10^{2}					
Adjusted Totals[f]			1.42×10^{3}				7.77×10^{2}					

(a) Inferred from mass, assuming density of stainless steel is 8.04 Mg/m^3.
(b) Calculated from ^{60}Co activity and 1.17 ratio of total activity/^{60}Co in test reactor data (see Section E.2).
(c) Calculated from ^{60}Co activity and 2.22 ratio of total activity/^{60}Co in test reactor data (see Section E.2).
(d) Activity to dose conversion factor used is $\sim$0.45 R/hr/Ci/m^3, derived from calculated activities reported in Reference 1. Dose rates are based on volume-averaged specific radioactivities and represent average rather than maximum values.
(e) Indicates insignificant quantities.
(f) Adjusted for inclusion of radionuclides not directly calculated.

TABLE E.1-3. Activated Aluminum Inventory and Associated Radiation Dose Rates in the Reference Research Reactor at Shutdown and 10 Years After Shutdown

Aluminum Components	Mass (Mg)	Volume (m^3)(a)	Ci (at shutdown) Total Activity(b)	Ci (at shutdown) ^{65}Zn	Ci (10 yrs after shutdown) Total Activity(c)	Ci (10 yrs after shutdown) ^{65}Zn	Years After Shutdown 0: Ci/m^3 (^{65}Zn)	Years After Shutdown 0: R/hr (^{65}Zn)(d)	Years After Shutdown 10: Ci/m^3 (^{65}Zn)	Years After Shutdown 10: R/hr (^{65}Zn)(d)
Reactor Vessel (core zone)	1.3×10^{-1}	1.2×10^{-1}	6.16×10^{-5}	2.75×10^{-5}	2.19×10^{-6}	8.29×10^{-10}	2.29×10^{-4}	4.35×10^{-5}	6.90×10^{-9}	1.3×10^{-9}
Reactor Vessel (above core)	6.2×10^{-2}		1.89×10^{-9}	8.43×10^{-10}	---(e)	---				
Reflector Platform	1.2×10^{-1}	4.6×10^{-2}	1.0×10^{0}	4.53×10^{-1}	3.62×10^{2}	1.37×10^{-5}	9.85×10^{0}	1.87×10^{0}	2.98×10^{-4}	5.66×10^{-5}
Reflector and Shroud	3.4×10^{-1} 1.0×10^{-1}	1.3×10^{-1} 3.7×10^{-2}	1.73×10^{1} 1.18×10^{-1}	7.71×10^{0} 5.29×10^{-2}	6.12×10^{-1} 4.22×10^{-3}	2.32×10^{-4} 1.60×10^{-6}	5.93×10^{1} 1.43×10^{0}	1.13×10^{1} 2.72×10^{-1}	1.78×10^{-3} 4.32×10^{-5}	3.39×10^{-4} 8.22×10^{-6}
Grid Plates	2.6×10^{-2}	9.4×10^{-3}	4.91×10^{0}	2.64×10^{0}	2.10×10^{-1}	7.96×10^{-5}	2.81×10^{2}	5.34×10^{1}	8.47×10^{1}	1.61×10^{0}
Safety and Grid Adapter Plates	1.2×10^{-2}	4.4×10^{-3}	2.84×10^{0}	1.27×10^{0}	1.01×10^{-1}	3.83×10^{-5}	2.89×10^{2}	5.48×10^{1}	2.29×10^{1}	4.36×10^{0}
Neutron Source Holder	1.8×10^{-3}	6.5×10^{-4}	2.69×10^{-1}	1.20×10^{-1}	9.56×10^{-3}	3.62×10^{-6}	1.85×10^{2}	3.51×10^{1}	5.57×10^{-3}	1.06×10^{-3}
Dummy Fuel Elements	8.5×10^{-3}	3.1×10^{-3}	1.27×10^{0}	5.65×10^{-1}	4.49×10^{-2}	1.70×10^{5}	1.82×10^{2}	3.46×10^{1}	5.48×10^{-3}	1.04×10^{-3}
Transient Rod and Guide Tubes	1.3×10^{-3}	4.8×10^{-4}	3.74×10^{-1}	1.67×10^{-1}	1.33×10^{-2}	5.04×10^{-6}	3.48×10^{2}	6.61×10^{1}	1.05×10^{-2}	2.00×10^{-3}
Fuel Storage Racks (3 each)	3.3×10^{-2}	1.2×10^{-2}	2.23×10^{-4}	9.97×10^{-5}	7.95×10^{-6}	3.01×10^{-9}	8.31×10^{-3}	1.58×10^{-3}	2.5×10^{-7}	5.00×10^{-6}
Ion Chambers and Holders (4 each)	3.3×10^{-2}	1.2×10^{-2}	7.25×10^{-2}	3.24×10^{-2}	2.58×10^{-3}	9.77×10^{-7}	2.7×10^{0}	5.13×10^{-1}	8.14×10^{-5}	1.55×10^{-5}
Rotary Specimen Rack	5.8×10^{-2}	2.2×10^{-2}	4.98×10^{0}	2.63×10^{0}	2.09×10^{-1}	7.93×10^{-5}	1.2×10^{2}	2.27×10^{1}	3.60×10^{-3}	6.85×10^{-4}
Pneumatic Transfer System (in core)	2.0×10^{-3}	7.4×10^{-4}	2.98×10^{-1}	1.33×10^{-1}	1.06×10^{-2}	4.01×10^{-6}	1.80×10^{2}	3.41×10^{1}	5.42×10^{-3}	1.03×10^{-3}
Pneumatic Transfer System (piping)	1.6×10^{-3}	6.0×10^{-4}	7.86×10^{-7}	3.51×10^{-7}						
Central Thimble (in core)	3.6×10^{-3}	1.3×10^{-3}	1.16×10^{0}	5.17×10^{-1}	4.12×10^{-2}	1.56×10^{-5}	3.98×10^{2}	7.56×10^{1}	1.20×10^{-2}	2.29×10^{-3}
Central Thimble (piping)	7.0×10^{-4}	2.6×10^{-4}								
Inner Beam Tubes (in core)	3.3×10^{-2}	1.2×10^{-2}	2.40×10^{-3}	1.07×10^{-3}	8.53×10^{-5}	3.23×10^{-8}	8.9×10^{-2}	1.69×10^{-2}	2.69×10^{-6}	5.1×10^{-7}
Beam Tubes (99 to 180 cm)	4.5×10^{-2}	1.7×10^{-2}	4.61×10^{-9}	2.06×10^{-9}						
Thermal Column (in vessel)	2.1×10^{-1}	7.8×10^{-2}	1.53×10^{-2}	6.92×10^{-3}	5.44×10^{-4}	2.06×10^{-7}	8.74×10^{-2}	1.66×10^{-2}	2.64×10^{-6}	5.0×10^{-7}
Thermal Column (outer)	4.1×10^{-1}	1.5×10^{-1}	6.94×10^{-8}	3.10×10^{-8}						
Thermalizing Column (in vessel)	4.5×10^{-2}	1.7×10^{-2}	3.49×10^{-3}	1.56×10^{-3}	1.24×10^{-4}	4.79×10^{-8}	9.17×10^{-2}	1.74×10^{-2}	2.76×10^{-6}	5.3×10^{-7}
Thermalizing Column (outer)	5.6×10^{-2}	2.1×10^{-2}	4.75×10^{-10}	2.12×10^{-10}						
Totals			3.65×10^{1}		1.3×10^{0}					

(a) Inferred from mass, assuming density of aluminum is 2.7 Mg/m^3.
(b) Calculated from ^{60}Co activity and 2.24 ratio of total activity/^{60}Co in test reactor data (see Section E.2).
(c) Calculated from ^{60}Co activity and 2640 ratio of total activity/^{60}Co in test reactor data (see Section E.2).
(d) Activity to dose conversion factor is $\sim$0.19 R/hr/Ci/m^3, based on ^{65}Zn activity. Dose rates are based on volume-averaged specific radioactivities and represent average rather than maximum values.
(e) Indicates a value less than 1.0×10^{-10}.

TABLE E.1-4. Activated Carbon Inventory Present During Decommissioning Tasks in the Reference Research Reactor

Carbon Component	Mass (Mg)	Volume (m^3)	Total Activity[a] ^{14}C (Ci)
Reflector	5.90×10^{-1}	3.6×10^{-1}	1.02×10^{0}
Dummy fuel Elements	1.96×10^{-2}	1.21×10^{-2}	6.67×10^{-2}
Thermal Column (within vessel)	1.1×10^{0}	6.8×10^{-1}	1.89×10^{-3}
Thermal Column (outside vessel)	2.71×10^{0}	1.69×10^{0}	2.69×10^{-8}
Thermalizing Column (within vessel)	1.39×10^{-1}	8.57×10^{-2}	2.69×10^{-3}
Thermalizing Column (outside vessel)	3.67×10^{-1}	2.27×10^{-1}	2.52×10^{-11}
Total			1.09×10^{0}

(a) Calculated from the neutron flux exposure.

100 years after shutdown) is presented in Table E.1-5 for stainless steel (reference radionuclide inventory 1) and for aluminum (reference radionuclide inventory 2) in Table E.1-6 for selected reactor components. Trace quantities of other isotopes known to be produced in the research reactor and not considered in Tables E.1-2 and E.1-3 are also included in these tables.

The cobalt isotopes given in Table E.1-5 are calculated as previously described, and the other radionuclides given in the table are inferred from their ratio to cobalt as developed in the test reactor analysis (see Table E.2-1). In a similar manner Table E.1-6 is completed by reference to the calculated quantities of ^{65}Zn and its ratio to other radionuclides seen in the test reactor (see Table E.2-2).

Estimates of the radionuclide composition and quantities in the activated portions of the concrete biological shield at the reference research reactor are made by analogy to data presented in the pressurized water reactor (PWR) decommissioning study.[1] In the PWR study, the composition of activated concrete was identified using the ORIGEN[2] computer code and 30 EFPY of

TABLE E.1-5. Reference Radionuclide Inventory 1, Neutron-Activated Stainless Steel[(a)] in the Reference Research Reactor

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{14}C	9.22×10^{0} (b)	1.75×10^{-5}	1.75×10^{-5}	1.74×10^{-5}	1.74×10^{-5}	1.73×10^{-5}
^{51}Cr	1.27×10^{5} (b)	2.41×10^{-1}	---(c)	---	---	---
^{54}Mn	1.61×10^{4} (d)	2.09×10^{-2}	3.49×10^{-6}	---	---	---
^{55}Fe	2.52×10^{4} (d)	4.76×10^{-2}	3.63×10^{-2}	2.15×10^{-5}	1.27×10^{-7}	---
^{59}Fe	2.41×10^{3} (b)	4.56×10^{-3}	---	---	---	---
^{58}Co	1.03×10^{5} (e)	1.28×10^{-1}	---	---	---	---
^{60}Co	2.88×10^{5} (e)	5.45×10^{-1}	6.33×10^{-2}	4.56×10^{-3}	3.28×10^{-4}	4.58×10^{-7}
^{59}Ni	5.59×10^{1} (d)	1.06×10^{-4}	1.06×10^{-4}	1.06×10^{-4}	1.06×10^{-4}	1.06×10^{-4}
^{63}Ni	6.40×10^{3} (d)	1.21×10^{-2}	1.14×10^{-2}	9.82×10^{-3}	9.07×10^{-3}	6.41×10^{-3}
^{93m}Nb	1.02×10^{-2} (b)	2.25×10^{-8}	1.35×10^{-8}	4.88×10^{-9}	1.76×10^{-7}	1.38×10^{-10}
^{94}Nb	1.32×10^{-1} (b)	2.50×10^{-7}	2.50×10^{-7}	2.50×10^{-7}	2.50×10^{-7}	2.49×10^{-7}
^{95}Nb	1.06×10^{1} (b)	2.00×10^{-5}	---	---	---	---
Totals	5.61×10^{5}	1.00	1.11×10^{-1}	1.45×10^{-2}	9.52×10^{-3}	6.53×10^{-3}

(a) Grid plate inserts and hardware.
(b) Not calculated, inferred by analogy with ^{59}Ni activity as calculated in Reference 3.
(c) Indicates a value of less than 1×10^{-10}.
(d) Based upon ratio of radionuclide to ^{60}Co observed in the reference test reactor calculation (see Section E.2).
(e) Calculated from neutron exposure.

TABLE E.1-6. Reference Radionuclide Inventory 2, Neutron-Activated Aluminum[(a)] in the Reference Research Reactor

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{46}Sc	9.80×10^{-2}[(b)]	1.74×10^{-4}	---[(c)]	---	---	---
^{54}Mn	3.90×10^{0}[(b)]	6.93×10^{-3}	1.15×10^{-6}	---	---	---
^{55}Fe	2.77×10^{2}[(b)]	4.93×10^{-1}	3.74×10^{-2}	2.22×10^{-4}	1.31×10^{-6}	---
^{60}Co	1.36×10^{-1}[(b)]	2.42×10^{-4}	6.48×10^{-6}	4.68×10^{-6}	3.37×10^{-7}	4.70×10^{-10}
^{63}Ni	3.37×10^{-2}[(b)]	6.00×10^{-5}	5.67×10^{-5}	4.94×10^{-5}	4.30×10^{-5}	3.04×10^{-5}
^{65}Zn	2.81×10^{2}[(d)]	5.00×10^{-1}	1.62×10^{-5}	---	---	---
Totals	5.62×10^{2}	1.00	3.75×10^{-2}	2.76×10^{-4}	4.46×10^{-5}	3.04×10^{-5}

(a) Averaged over grid plates for 2.03 EFPY of operation, from Reference 4.
(b) Based upon ratio of radionuclide to ^{65}Zn observed in the reference test reactor calculation (see Section E.2).
(c) Indicates a value of less than 1.0×10^{-10}.
(d) Calculated from neutron exposure.

reactor operation. The fractional activity of the activated concrete present at the PWR is shown in Table E.1-7 at reactor shutdown and for decay times of 10, 30, 50 and 100 years. This mixture of radionuclides is assumed to be present in the activated concrete surrounding the reference research reactor core. The total activity present is estimated by correcting for the different neutron flux levels at the reference research reactor (shown in Figure E.1-1) and by correcting for the difference in reactor lifetimes between the PWR in Reference 1 and the reference research reactor. The average specific activity of the concrete is estimated to be about 0.49 Ci/m^3, as shown in Table E.1-7.

Only those radionuclides whose half-lives and/or concentration at shutdown result in a significant contribution to the total activity after 1 year and/or after 100 years of decay are shown in Table E.1-7. No rare earth elements (e.g., europium) are included in the radionuclide mixture since definitive data on their probable initial concentrations are not available. During an actual decommissioning operation, it will be necessary to either analyze a sample for trace elements and calculate the probable activation levels, or to obtain and analyze core samples from the concrete biological shield after reactor shutdown to give a definitive estimate of the radionuclide inventory. The potentially large impact of a minor trace constituent (such as europium) in the inventory may require a careful analysis.

E.1.2 Surface Contamination in the Reference Research Reactor Facility

Surface contamination can be expected to be found on equipment and in work areas designated for handling radioactive materials, such as the hot cell, the terminus of the rabbit facility and the fuel storage facilities. In addition, materials in contact with the reactor water may contain deposited radionuclides carried through the recirculating system. Little information is available about the accumulation of surface contamination at the reference research reactor; however, radiation surveys conducted by OSTR personnel indicate that the exposure from such accumulations is typically 1 mR/hr or less. This confirms the expectation that in the absence of fuel failures, surface contamination is not a significant factor in a research reactor that operates at a relatively low power level ($\sim$1 MW). This section presents the known data, judgements,

TABLE E.1-7. Reference Radionuclide Inventory 3, Activated Biological Shield Concrete at the Reference Research Reactor[a]

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at the Reference PWR Decay Time of:[b]				
		Shutdown	10 Years	30 Years	50 Years	100 Years
^{39}Ar	5.4×10^{-4}	1.1×10^{-3}	1.1×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	8.8×10^{-4}
^{41}Ca	9.8×10^{-3}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}
^{45}Ca	4.9×10^{-2}	1.0×10^{-1}	2.3×10^{-8}	---[c]	---	---
^{54}Mn	2.4×10^{-3}	4.8×10^{-3}	1.0×10^{-6}	---	---	---
^{55}Fe	4.2×10^{-1}	8.6×10^{-1}	6.6×10^{-2}	3.9×10^{-4}	2.3×10^{-6}	---
^{60}Co	9.3×10^{-3}	1.9×10^{-2}	5.2×10^{-3}	3.7×10^{-4}	2.7×10^{-5}	3.7×10^{-8}
^{59}Ni	1.7×10^{-5}	3.4×10^{-5}	3.4×10^{-5}	3.4×10^{-5}	3.4×10^{-5}	3.4×10^{-5}
^{63}Ni	2.0×10^{-3}	4.0×10^{-3}	3.8×10^{-3}	3.2×10^{-3}	3.2×10^{-3}	2.0×10^{-3}
Totals	4.9×10^{-1}	1.0	7.7×10^{-2}	5.3×10^{-3}	4.1×10^{-3}	3.1×10^{-3}

(a) The radionuclides listed include only those whose half-life and/or initial concentration result in a significant contribution after one year of decay and/or one hundred years of decay.
(b) Based on data from Table 7.3-5 of Reference 1.
(c) Dashes mean the fractional level of radioactivity was less than 10^{-10}.

and quantitative calculations used to estimate the contamination levels in piping, equipment, and other reactor areas.

E.1.2.1 Radionuclide Inventories in Internally Contaminated Piping and Equipment

A thin surface layer of material is deposited from the ionic species of the reactor water onto internal components of the reactor primary water cooling and purification system. This system is shown in Figure B.3-9 and is described in Section B.3.3.1 of Appendix B. In addition, neutron-activated particulate corrosion products add to this surface layer by deposition and absorption. The composition and amount of radioactivity found on these internal surfaces at the time of facility shutdown are dependent on several reactor design parameters. The reference research reactor is operated only intermittently at low power levels and, therefore, only small amounts of activation products are produced. Since aluminum is the structural material used in the core region, primary production of radionuclides with long half-lives is limited (see Table E.1-3).

Fission products may enter the water recirculation system on infrequent occasions via leaks from fuel elements. The high structural integrity of the fuel elements has resulted in only one fuel element leak to date at the reference research reactor.[a] Based on this historical experience, it is postulated that a fuel element leak occurs about once every 5 years during an estimated 40-year operating lifetime for the reference research reactor. By carefully monitoring the radioactivity levels at the demineralizer, prompt corrective action is assumed to prevent any long-term introduction of fission products into the reactor water. From the postulated frequency of this occurrence and its short duration in comparison with the normally monitored corrosion products (each fuel failure resulting in doubling the exposure rate for about 1 week), it is estimated that the radiation dose from fission product surface contamination in the primary water recirculating system contributes less than 2% of the dose that results from activated corrosion product adsorption and deposition.

(a) Based on information supplied by Mr. T. V. Anderson, Oregon State University, Corvallis, Oregon.

Historically, radiation surveys were made of the equipment and piping at the reference research reactor during periods of shutdown. These surveys confirm that within 2 weeks after shutdown, the exposure levels at 0.3 m distance from most equipment and piping were 1 mR/hr or less. The major contributor to γ-dose from neutron activation products is ^{60}Co (see Tables E.1-2, E.1-3, and E.1-4). The relationship between exposure rate and quantity of ^{60}Co on a surface has been calculated from exposure rate versus distance data for 1.25 MeV photons and is presented in Figure E.1-2 for three distances from the surface. This information is used to directly estimate the ^{60}Co accumulation on a surface. More often, radiation from the internally deposited radionuclides is attenuated by the walls of the equipment or pipes. The expected ^{60}Co γ-ray attenuation versus material thickness is calculated from the mass absorption coefficients for the structural materials of interest. The results of these calculations are illustrated in Figure E.1-3.

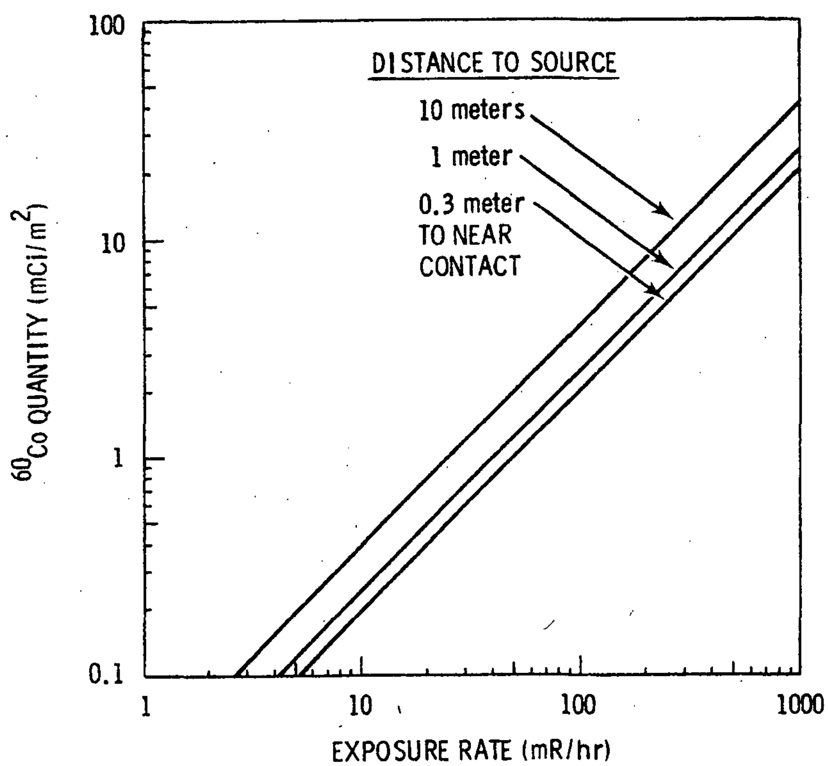

FIGURE E.1-2. Exposure Rate Above an Infinite Plane as a Function of Distance and Quantity of ^{60}Co Surface Contamination.

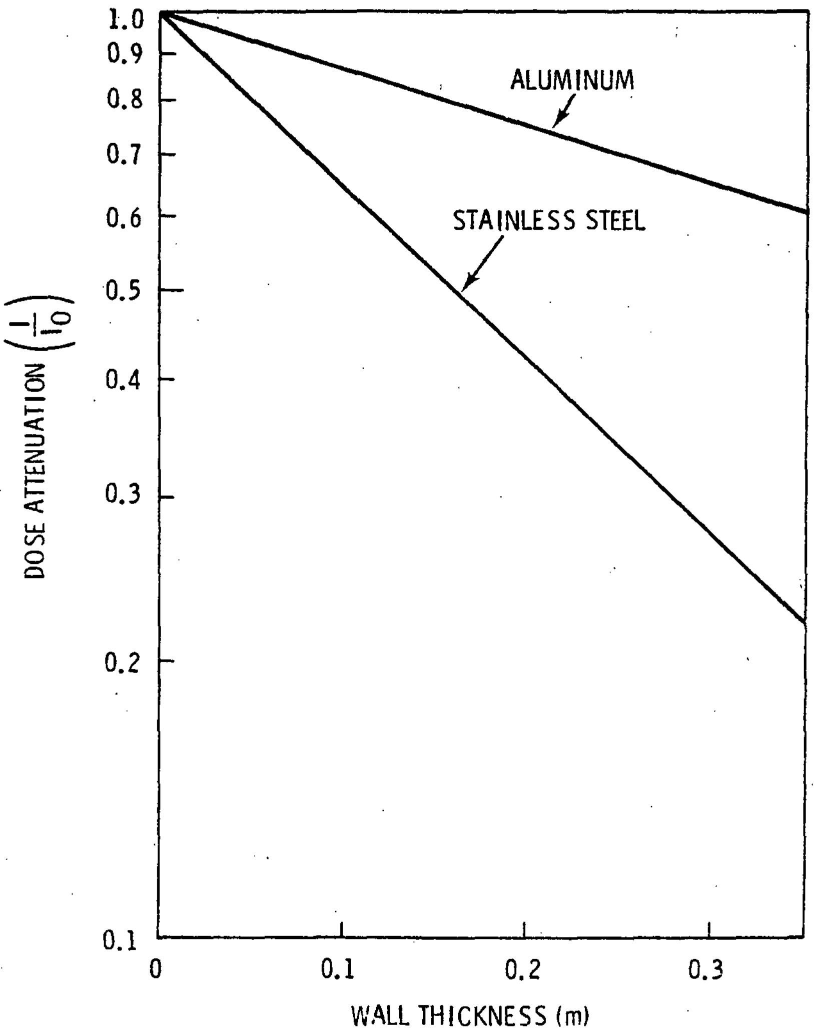

FIGURE E.1-3. Wall Attenuation of Radiation Dose from Internally Deposited ^{60}Co as a Function of Wall Thickness

The levels of surface contamination inside equipment and piping are estimated based on dose rates measured at the reference research reactor. The estimates are doubled to estimate the projected radionuclide accumulation to the end of the reactor operational life. The 5-year operational data to date suggest a very slow increase in exposure rate to less than, but near, 1 mR/hr. Based on this accumulation rate, the dose rate at shutdown is estimated by extrapolation to be about 2 mR/hr. For dose rates in the 1 mR/hr range from stainless steel equipment or piping whose wall thickness is conservatively estimated to be no greater than 12.5 mm (the maximum thickness anticipated for

any equipment or piping at the reference research reactor), the ^{60}Co internal surface contamination levels is estimated to be 7×10^{-2} mCi/m^2.

E.1.2.2 Radionuclide Inventories on Contaminated Surfaces of the Hot Cell, the Storage Pits, and the Rabbit Facility Terminus

Several pieces of equipment at the reference research reactor are designed to handle high levels of radionuclides. These are the hot cell facility, the fuel storage pits, and the hoods at the rabbit facility terminus.

Hot Cell Facility. The hot cell contains equipment for remote maintenance and decontamination and is used to prepare failed fuel for transfer to the onsite fuel storage facilities. It is equipped with shielding windows and slave manipulators. The hot cell is presumed to have had light-use compared to typical hot cells in the nuclear industry, since no actual use has been made of this hot cell facility after 5 years of reactor operation. For the purposes of this study, a "light-use" hot cell is postulated to have been in use at the reference research reactor during its operating lifetime. Such a cell is described as one of several used in a reference nuclear fuel reprocessing plant.(5) Although small amounts of many fission products are expected in the residual surface contamination of this hot cell, only the major items of concern are estimated.(5) These estimates are given in Table E.1-8. The ^{137}Cs value is in good agreement with the lower limit of contamination estimated for hot cell work when handling this isotope.(6) Contamination can be expected to be minimized in a low-service facility.

TABLE E.1-8. Estimated Inventory of Major Radionuclides in the Hot Cell at the Reference Research Reactor

Radionuclide	Radionuclide Inventory(Ci)
^{90}Sr, ^{90}Y	4
^{134}Cs	3
^{137}Cs	3
Total Actinides	≤ 1
Total	10

Storage Pits. It is postulated that three of the five radioactive material storage pits will see service during the operating lifetime of the reference research reactor. Although used for fuel elements and high-level radioactive sample storage, no manipulative activity (such as that carried out in the hot cell) is associated with these storage pits. No actual data are available on the radionuclide inventory within these areas; however, it is postulated to contain approximately 10% of the inventory estimated for the hot cell.

Radioactive Contamination in Hoods. There are two fume hoods located in Room R-3 (see Figure B.2-1a in Appendix B) that contain surface contamination. They are used as the receptor point for the pneumatic transfer system facility, which produces radionuclides by moving materials from the hood to the reactor core. As such, it is a radionuclide manufacturing facility and can be considered to contain materials and quantities similar to those described in the non-fuel-cycle reference facility.[6] The total radionuclide inventory for both hoods is presented in Table E.1-9. Each hood found in the reference research reactor has a surface area of approximately 5 m^2.

TABLE E.1-9. Estimated Radionuclide Inventory in the Contaminated Hoods at the Rabbit Facility Terminus

Nuclide	Radionuclide Inventory(Ci)
^{3}H	4.5×10^{-1}
^{14}C	4.5×10^{-5} to 4.5×10^{-4}
^{125}I	4.5×10^{-8} to 4.5×10^{-7}
^{137}Cs	4.5×10^{-4} to 4.5×10^{-3}
Transuranics	4.5×10^{-8} to 4.5×10^{-7}

E.1.2.3 Radionuclide Inventory on the Reference Research Reactor Site

It is postulated that no radioactivity is distributed around the reference research reactor site. Normal operation of this reactor does not result in deposition of radionuclides in the immediate site vicinity. Should internal monitoring information indicate the accidental release of fission products,

the reactor ventilation system is designed to shut down, seal, and isolate the reactor room until cleanup is complete. In any event, immediate external cleanup would take place due to the very close proximity of public lands. Therefore, it is assumed that this cleanup would preclude the presence of site contamination at the time of decommissioning.

E.2 RADIONUCLIDE INVENTORIES IN THE REFERENCE TEST REACTOR FACILITY

The quantities of radioactivity and the radiation dose rates are significantly greater in the reference test reactor facility than in the reference research reactor facility because of the generally higher neutron flux levels and longer integrated exposure of materials to neutrons. Available data on the quantities and levels of radioactivity in neutron-activated materials for the reference test reactor are presented in Section E.2.1; and for the Mock-Up Reactor (MUR) in Section E.2.1.2. A limited amount of information is available on the radionuclides and contamination levels throughout the reference test reactor facility. This information is presented in Section E.2.2.

E.2.1 Neutron-Activated Materials in the Reference Test Reactor Facility

Radioactive material is produced in two locations in the reference test reactor facility. The principal source is the test reactor, and the inventory produced during the operating lifetime of the test reactor is described in Section E.2.1.1. The second and much lesser source of radioactive material is the MUR. Its inventory is described in Section E.2.1.2. The characteristics of the principal radionuclides produced by neutron activation in the reactor components are described in Section E.2.1.3.

E.2.1.1 Neutron-Activated Materials in the Reference Test Reactor

The reference test reactor operated for a total of 98,000 MWd, with a nominal level of 60 MW, for a total of 1633 EFPD, or 4.47 EFPY.[4]

Based on this operating history and detailed neutron flux information, NASA consultants calculated the types and quantities of radionuclides that should be present in the neutron-activated reactor materials at the end of operating life, using the methodology described in Appendix A of Reference 4.

These calculations are straightforward production-removal calculations performed over the cyclic power history of the reference test reactor for the principal constituents of the reactor core structure. Burnup of both target and product atoms is neglected in the calculations, resulting in conservative (larger than probable) estimates of the concentrations of the various radionuclides in the reactor components.

Those elements for which levels of radioactivity are explicity calculated are listed in Table E.1-1 (see Section E.1-1) for each of the principal materials exposed to the neutron flux. The neutron reactions that produce radioactivity in these materials are also listed.

The results of these calculations are summarized in Tables E.2-1 through E.2-4 for selected components made of 304 stainless steel, aluminum, concrete, and beryllium, respectively. In the case of stainless steel, several radionuclides likely to be present were not calculated directly but were inferred from other calculations made for stainless steel in a previous decommissioning study.[3] Three of these additional radionuclides, ^{51}Cr, ^{59}Fe, and ^{95}Nb, are of significance only immediately following reactor shutdown. However, ^{14}C, ^{93m}Nb, and ^{94}Nb are long-lived and are important even after extended safe storage periods.

Estimates of the radionuclide composition and quantities in the activated portions of the biological shield at the reference test rector are made by analogy to data presented in the pressurized water reactor (PWR) decommissioning study.[1] In the PWR study, the composition of activated concrete was identified using the ORIGEN[2] computer code and 30 EFPY of reactor operation. The fractional activity of the activated concrete present in the PWR is shown as reference radionuclide inventory 3 in Table E.2-3 at reactor shutdown and for decay time of 10, 30, 50 and 100 years. This mixture of radionuclides is assumed to be present in the activated concrete surrounding the reference test reactor core. The total activity present is estimated by correcting for the different neutron flux levels at the test reactor (assumed to equal those at the reference research reactor shown in Figure E.1-1), and by correcting for the different reactor lifetimes between the PWR in Reference 1 and the reference test reactor. The average specific activity of the concrete is estimated to be about 1.1 Ci/m^3, as shown in Table E.2-3.

TABLE E.2-1. Reference Radionuclide Inventory 1, Neutron-Activated Stainless Steel[a] in the Reference Test Reactor

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{14}C[b]	2.97×10^{0}	1.75×10^{-5}	1.75×10^{-5}	1.74×10^{-5}	1.74×10^{-5}	1.73×10^{-5}
^{51}Cr[b]	4.10×10^{4}	2.41×10^{-1}	---[c]	---	---	---
^{54}Mn	3.56×10^{3}	2.09×10^{-2}	3.49×10^{-6}	---	---	---
^{55}Fe	8.10×10^{3}	4.76×10^{-2}	3.63×10^{-2}	2.15×10^{-5}	1.27×10^{-7}	---
^{59}Fe[b]	7.75×10^{2}	4.56×10^{-3}	---	---	---	---
^{58}Co	2.18×10^{4}	1.28×10^{-1}	---	---	---	---
^{60}Co	9.27×10^{4}	5.45×10^{-1}	6.33×10^{-2}	4.56×10^{-3}	3.28×10^{-4}	4.58×10^{-7}
^{59}Ni	1.80×10^{1}	1.06×10^{-4}	1.06×10^{-4}	1.06×10^{-4}	1.06×10^{-4}	1.06×10^{-4}
^{63}Ni	2.06×10^{3}	1.21×10^{-2}	1.14×10^{-2}	9.82×10^{-3}	9.07×10^{-3}	6.41×10^{-3}
^{93m}Nb[b]	3.82×10^{-3}	2.25×10^{-8}	1.35×10^{-8}	4.88×10^{-9}	1.76×10^{-9}	1.38×10^{-10}
^{94}Nb[b]	4.25×10^{-2}	2.50×10^{-7}	2.50×10^{-7}	2.50×10^{-7}	2.50×10^{-7}	2.50×10^{-7}
^{95}Nb[b]	3.40×10^{0}	2.0×10^{-5}	---	---	---	---
Totals	1.70×10^{5}	1.00	1.11×10^{-1}	1.45×10^{-2}	9.52×10^{-3}	6.53×10^{-3}

(a) Averaged over the upper flow guide, for 4.47 EFPY of operation, from Reference 4.
(b) Not calculated, inferred by analogy with ^{59}Ni activity as calculated in Reference 3.
(c) Indicates a value of less than 1×10^{-10}.

TABLE E.2-2. Reference Radionuclide Inventory 2, Neutron-Activated Aluminum[a] in the Reference Test Reactor

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{46}Sc	1.96×10^{1}	1.74×10^{-4}	---[b]	---	---	---
^{54}Mn	7.78×10^{2}	6.93×10^{-3}	1.15×10^{-6}	---	---	---
^{55}Fe	5.53×10^{4}	4.93×10^{-1}	3.74×10^{-2}	2.22×10^{-4}	1.31×10^{-6}	---
^{60}Co	2.72×10^{1}	2.42×10^{-4}	6.48×10^{-5}	4.68×10^{-6}	3.37×10^{-7}	4.70×10^{-10}
^{63}Ni	6.74×10^{0}	6.00×10^{-5}	5.67×10^{-5}	4.94×10^{-5}	4.30×10^{-5}	3.04×10^{-5}
^{65}Zn	5.61×10^{4}	5.00×10^{-1}	1.62×10^{-5}	---	---	---
Totals	1.12×10^{5}	1.00	3.75×10^{-2}	2.76×10^{-4}	4.46×10^{-5}	3.04×10^{-5}

(a) Averaged over the Upper Grid, for 4.47 EFPY of operation, from Reference 4.
(b) Indicates a value of less than 1.0×10^{-10}.

TABLE E.2-3. Reference Radionuclide Inventory 3, Activated Concrete at the Reference Test Reactor[a]

Radionuclide	Radioactivity[b] Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at the Reference PWR Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{39}Ar	1.2×10^{-3}	1.1×10^{-3}	1.1×10^{-3}	1.0×10^{-3}	1.0×10^{-3}	8.8×10^{-4}
^{41}Ca	2.2×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}
^{45}Ca	1.1×10^{-1}	1.0×10^{-1}	2.3×10^{-8}	---[c]	---	---
^{54}Mn	5.3×10^{-3}	4.8×10^{-3}	1.0×10^{-6}	---	---	---
^{55}Fe	9.5×10^{-1}	8.6×10^{-1}	6.6×10^{-2}	3.9×10^{-4}	2.3×10^{-6}	---
^{60}Co	2.1×10^{-2}	1.9×10^{-2}	5.2×10^{-3}	3.7×10^{-4}	2.7×10^{-5}	3.7×10^{-8}
^{59}Ni	3.7×10^{-5}	3.4×10^{-5}	3.4×10^{-5}	3.4×10^{-5}	3.4×10^{-5}	3.4×10^{-5}
^{63}Ni	4.4×10^{-3}	4.0×10^{-3}	3.8×10^{-3}	3.2×10^{-3}	2.8×10^{-3}	2.0×10^{-3}
Totals	1.1	1.0	7.7×10^{-2}	5.3×10^{-3}	4.1×10^{-3}	3.1×10^{-3}

(a) The radionuclides listed include only those whose half-life and/or initial concentration result in a significant contribution after one year of decay and/or one hundred years of decay.
(b) Based on data from Table 7.3-5 of Reference 1.
(c) Dashes mean the fractional level of radioactivity was less than 10^{-10}.

TABLE E.2-4. Reference Radionuclide Inventory 4, Neutron-Activated Beryllium[a] in the Reference Test Reactor

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{3}H	8.28×10^{5}	9.01×10^{-1}	5.22×10^{-1}	1.70×10^{-1}	5.51×10^{-2}	3.31×10^{-3}
^{54}Mn	8.85×10^{1}	9.63×10^{-5}	1.61×10^{-8}	---[b]	---	---
^{55}Fe	2.52×10^{4}	2.74×10^{-2}	2.08×10^{-3}	1.23×10^{-5}	7.22×10^{-8}	---
^{60}Co	6.55×10^{4}	7.13×10^{-2}	8.27×10^{-3}	5.96×10^{-4}	4.29×10^{-5}	5.98×10^{-8}
^{113m}Cd	1.28×10^{0}	1.39×10^{-6}	8.53×10^{-7}	3.30×10^{-7}	1.10×10^{-7}	1.02×10^{-8}
^{115m}Cd	9.39×10^{-3}	1.02×10^{-8}	---	---	---	---
Totals	9.19×10^{5}	1.00	5.32×10^{-1}	1.71×10^{-1}	5.52×10^{-2}	3.31×10^{-3}

(a) Averaged over 8 each RA blocks, for 4.47 EFPY of operation, from Reference 4.
(b) Indicates value less than 1.0×10^{-10}.

Only those radionuclides whose half-lives and/or concentrations at shutdown result in a significant contribution to the total activity after 1 year and/or after 100 years of decay are shown in Table E.2-3. No rare earth elements (e.g., europium) are included in the radionuclide mixture since definitive data on their probable initial concentrations are not available. During an actual decommissioning operation, it will be necessary to either analyze a sample for trace elements and calculate the probable activation levels, or to obtain and analyze core samples from the activated concrete after reactor shutdown to give a definitive estimate of the radionuclide inventory. The potentially large impact of a minor trace constituent, such as europium, in the inventory may require a careful analysis.

The reference radionuclide inventories presented in Tables E.2-1 through E.2-4 are used for several purposes in this study. In this section they are used to calculate the total quantities of radioactivity in the structural components in the reactor vessel. This information is used in determining disposal requirements and costs whenever removal of any or all of these components is required by a particular decommissioning alternative. In Appendix N these inventories are used in assessing public safety of decommissioning activities.

The radioactivity levels in all of the activated reactor components are listed in Tables E.2-5 through E.2-7 for stainless steel, aluminum alloys, and beryllium, respectively. For each component, the following items are listed: The mass (if known), the inferred volume, the total radioactivity in curies at reactor shutdown and after 10 years' decay, the specific activity (Ci/m^3) of the radionuclide that is the principal source of external radiation dose at reactor shutdown and after 10 years' decay, and the calculated external radiation dose rate (R/hr) at reactor shutdown and after 10 years' decay. If the mass of a component is not known, only the total radioactivity is listed.

The specific activities are converted to external radiation dose rate using activity-to-dose conversion factors derived for stainless steel from calculations presented in a previous decommissioning study.[3] For aluminum and beryllium these conversions are made using the method outlined in Reference 7. The dose conversion factor for stainless steel calculated using the method given in Reference 7 agreed with the factor derived from the more

TABLE E.2.5. Activated Stainless Steel Inventory and Associated Radiation Doses in the Referenced Test Reactor at Shutdown and 10 Years After Shutdown

			Ci (at shutdown)[a]		Ci (10 yrs after shutdown)[a]		Years After Shutdown			
							0		10	
Stainless Steel Components	Mass (Mg)[a]	Volume (m^3)[b]	Total Activity	Co Only	Total Activity	Co Only	Ci/m^3 (Co)	R/hr (Co)[c]	Ci/m^3 (Co)	R/hr (Co)[c]
Thermal Shield, Outer Upper	5.65×10^{0}	7.03×10^{-1}	2.03×10^{-3}	1.75×10^{-3}	5.25×10^{-5}	2.58×10^{-5}	2.49×10^{-3}	1.12×10^{-3}	3.67×10^{-3}	1.64×10^{-5}
Outer Lower	4.10×10^{0}	5.10×10^{-1}	6.99×10^{-4}	5.96×10^{-4}	1.46×10^{-5}	5.16×10^{-6}	1.17×10^{-3}	5.27×10^{-4}	1.01×10^{-5}	4.55×10^{-6}
Middle Upper	4.70×10^{0}	5.85×10^{-1}	3.30×10^{-3}	2.83×10^{-3}	7.84×10^{-5}	3.39×10^{-5}	4.84×10^{-3}	2.18×10^{-3}	5.79×10^{-5}	2.62×10^{-5}
Middle Lower	3.80×10^{0}	4.73×10^{-1}	1.85×10^{-3}	1.35×10^{-3}	9.50×10^{-5}	3.81×10^{-5}	2.85×10^{-3}	1.28×10^{-3}	8.06×10^{-5}	2.85×10^{-5}
Upper Inner	3.76×10^{0}	4.68×10^{-1}	4.61×10^{-3}	3.65×10^{-3}	1.70×10^{-4}	4.50×10^{-5}	7.80×10^{-3}	3.51×10^{-3}	9.62×10^{-5}	4.33×10^{-5}
Pressure Vessel, Wall	9.23×10^{0}	1.15×10^{0}	1.24×10^{-3}	1.05×10^{-3}	4.37×10^{-5}	1.29×10^{-5}	9.13×10^{-3}	4.11×10^{-3}	1.12×10^{-5}	5.05×10^{-6}
Bottom	2.30×10^{0}	2.86×10^{-1}	2.88×10^{-7}	2.69×10^{-7}	2.71×10^{-8}	2.40×10^{-8}	9.41×10^{-7}	4.23×10^{-4}	8.39×10^{-6}	3.78×10^{-8}
Flow Guide, Upper	1.05×10^{0}	1.31×10^{-1}	1.96×10^{3}	1.75×10^{3}	2.89×10^{2}	1.65×10^{2}	1.34×10^{4}	6.03×10^{3}	1.26×10^{3}	5.67×10^{2}
Lower	5.14×10^{-1}	6.39×10^{-2}	1.27×10^{-1}	1.06×10^{-1}	1.21×10^{-2}	8.49×10^{-3}	1.66×10^{0}	7.47×10^{-1}	1.33×10^{-1}	5.98×10^{-2}
Support	1.90×10^{0}	2.36×10^{-1}	2.60×10^{-5}	2.46×10^{-5}	2.45×10^{-6}	2.19×10^{-6}	1.04×10^{-4}	4.69×10^{-5}	9.28×10^{-6}	4.18×10^{-6}
Metering Plate	6.92×10^{-1}	7.36×10^{-2}	3.93×10^{2}	3.53×10^{2}	3.53×10^{1}	1.73×10^{1}	2.24×10^{3}	1.01×10^{3}	2.35×10^{2}	1.06×10^{2}
Control Rod, Upper Rollers	1.25×10^{-2}	1.55×10^{-3}	1.78×10^{3}	1.05×10^{3}	1.25×10^{2}	6.49×10^{1}	6.77×10^{6}	3.05×10^{5}	4.19×10^{4}	1.88×10^{4}
Drive Box	7.04×10^{-1}	8.76×10^{-2}	8.63×10^{-6}	8.17×10^{-6}	8.13×10^{-7}	7.27×10^{-7}	9.33×10^{-5}	4.20×10^{-5}	8.30×10^{-6}	3.73×10^{-6}
Miscellaneous Bolts	1.94×10^{-3}	2.41×10^{-4}	3.35×10^{2}	1.78×10^{2}	2.72×10^{1}	1.45×10^{1}	7.39×10^{6}	3.32×10^{5}	6.02×10^{-4}	2.71×10^{4}
Instrumentation Thimbles	(d)		2.53×10^{4}		6.63×10^{3}					
Fast Flux Facility	(d)		7.28×10^{2}		1.23×10^{2}					
Shim Rod Section	(d)		2.20×10^{3}		5.79×10^{2}					
Totals			3.38×10^{4}		7.81×10^{3}					
Adjusted Totals[e]			4.49×10^{4}		1.13×10^{4}					

(a) Data from Appendix A of Reference 4.
(b) Inferred from the mass, assuming density of stainless steel is 8.04 Mg/m^3.
(c) Activity to dose conversion factor used is 0.45 $R/hr/Ci/m^3$, derived from calculated activities reported in Reference 1. Dose rates are based on volume-averaged specific ratio activities and represent average rather than maximum values.
(d) Data not available.
(e) Adjusted for inclusion of radionuclides not directly calculated, from Table E.2-1. For upper flow guide, Adjusted/Base (0 years) = 1.328, Adjusted/Base (10 years) = 1.448.

FIGURE E.2-6. Activated Aluminum and Cadmium Inventory and Associated Radiation Dose Rates in the Reference Test Reactor at Shutdown and 10 Years After Shutdown

Aluminum Components	Mass (Mg)[a]	Volume (m^3)[b]	Ci (at shutdown)[a] Total Activity	Ci (at shutdown)[a] ^{65}Zn Only	Ci (10 yrs after shutdown)[a] Total Activity	Ci (10 yrs after shutdown)[a] ^{65}Zn Only	Years After Shutdown 0: Ci/m^3 (^{65}Zn)	Years After Shutdown 0: R/hr (^{65}Zn)[c]	Years After Shutdown 10: Ci/m^3 (^{65}Zn)	Years After Shutdown 10: R/hr (^{65}Zn)[c]
Upper Grid, in Core	9.37×10^{-2}	3.47×10^{-2}	3.89×10^{3}	1.95×10^{3}	1.46×10^{2}	6.32×10^{-2}	5.62×10^{4}	1.07×10^{4}	1.82×10^{0}	3.46×10^{-1}
in Hot Storage	9.37×10^{-2}	3.47×10^{-2}	1.16×10^{2}	2.38×10^{0}	3.77×10^{-1}	1.63×10^{-4}	6.86×10^{1}	1.30×10^{1}	4.70×10^{-3}	8.36×10^{-4}
Beam Tubes, V-2	2.89×10^{-2}	1.07×10^{-2}	3.69×10^{2}	1.64×10^{2}	1.56×10^{1}	5.32×10^{-3}	1.53×10^{4}	2.90×10^{3}	4.97×10^{-1}	9.44×10^{-2}
HB-4	1.72×10^{-2}	6.37×10^{-3}	1.29×10^{-1}	5.85×10^{-2}	5.35×10^{-3}	1.90×10^{-6}	9.18×10^{0}	1.74×10^{0}	2.98×10^{-4}	5.66×10^{-5}
HB-5,6	1.69×10^{-2}	2.51×10^{-2}	2.62×10^{3}	1.12×10^{2}	3.74×10^{3}	1.87×10^{-6}	9.22×10^{0}	1.75×10^{0}	2.99×10^{4}	5.68×10^{5}
HT-1	6.79×10^{-2}	2.51×10^{-2}	2.62×10^{3}	1.15×10^{3}	1.12×10^{2}	3.74×10^{-2}	4.57×10^{4}	8.68×10^{3}	1.49×10^{0}	2.83×10^{1}
HT-2	8.55×10^{-2}	3.17×10^{-2}	7.64×10^{2}	3.28×10^{2}	3.32×10^{1}	1.07×10^{-2}	1.04×10^{4}	1.98×10^{3}	3.38×10^{-1}	6.42×10^{-2}
HB-1,3	2.20×10^{-2}	8.15×10^{-3}	8.36×10^{-1}	3.68×10^{-1}	3.55×10^{-2}	1.19×10^{-5}	4.53×10^{1}	8.61×10^{0}	6.88×10^{4}	1.31×10^{4}
Far South Box Plate	6.20×10^{-2}	2.30×10^{-2}	1.47×10^{2}	6.21×10^{1}	6.43×10^{0}	2.02×10^{-3}	2.70×10^{3}	5.13×10^{2}	8.80×10^{-2}	1.67×10^{-2}
Side Plate (2 each)	1.00×10^{-1}	3.70×10^{-2}	7.43×10^{3}	3.91×10^{3}	2.67×10^{2}	1.27×10^{-1}	1.06×10^{5}	2.01×10^{4}	3.43×10^{0}	6.52×10^{-1}
Lower Grid	1.66×10^{-1}	6.15×10^{-2}	2.26×10^{4}	1.27×10^{4}	7.49×10^{2}	4.11×10^{-1}	2.07×10^{5}	3.93×10^{4}	6.68×10^{0}	1.27×10^{0}
VAFT Lower Section (3 each)	(d)		2.28×10^{3}	4.67×10^{1}	4.46×10^{2}	1.93×10^{-1}				
VAFT, 68-03	(d)		2.80×10^{-1}	5.74×10^{-3}	1.20×10^{-25}	---(e)				
69-01	(d)		2.00×10^{-3}	4.10×10^{-5}	3.55×10^{-25}	---				
Instrument Thimbles	(d)		2.00×10^{-3}	4.10×10^{-6}	1.20×10^{-15}	---				
Cadmium Control Rods										
Test Reactor (6 each)	(d)		9.17×10^{1}	(d)	2.46×10^{1}	(d)				
Totals			4.03×10^{4}		1.80×10^{3}					

(a) Data from Appendix A of Reference 4.
(b) Inferred from the mass, assuming density of aluminum is 2.7 Mg/m^3.
(c) Activity to dose conversion factor is ~0.19 R/hr/Ci/m^3, based on ^{65}Zn activity. Dose rates are based on volume-averaged specific radioactivities and represent average rather than maximum values.
(d) Data not available.
(e) Indicates a value less than 1.0×10^{-10}.

TABLE E.2-7. Activated Beryllium Inventory and Associated Radiation Dose Rates in the Reference Test Reactor at Shutdown and 10 Years After Shutdown

Beryllium Components	Mass (Mg)[a]	Volume (m^3)[b]	Ci (at shutdown)[a] Including ^3H	Ci (at shutdown)[a] ^{60}Co Only	Ci (10 yrs after shutdown)[a] Including ^3H	Ci (10 yrs after shutdown)[a] ^{60}Co Only	Years After Shutdown 0: Ci/m^3 (^{60}Co)	Years After Shutdown 0: R/hr (^{60}Co)[c]	Years After Shutdown 10: Ci/m^3 (^{60}Co)	Years After Shutdown 10: R/hr (^{60}Co)[c]
North Core Box Plate, in Core	4.20×10^{-2}	2.27×10^{-2}	5.93×10^{3}	2.99×10^{2}	3.20×10^{3}	3.47×10^{1}	1.32×10^{4}	8.98×10^{3}	1.52×10^{3}	1.04×10^{3}
in Hot Storage	4.20×10^{-2}	2.27×10^{-2}	2.96×10^{3}	$\sim 4.67 \times 10^{1}$	1.71×10^{3}	$\sim 1.85 \times 10^{1}$	2.06×10^{3}	1.40×10^{3}	8.15×10^{2}	5.56×10^{2}
RD Blocks (8 each)	1.63×10^{-1}	8.81×10^{-2}	4.40×10^{3}	1.28×10^{2}	1.49×10^{2}	1.48×10^{1}	1.45×10^{3}	9.91×10^{2}	1.68×10^{2}	1.15×10^{2}
RC Blocks (8 each)	1.63×10^{-1}	8.81×10^{-2}	2.22×10^{3}	4.50×10^{2}	9.20×10^{2}	5.22×10^{1}	6.11×10^{3}	3.48×10^{3}	5.92×10^{2}	4.04×10^{2}
RB Blocks (8 each)	1.13×10^{-1}	6.11×10^{-2}	1.23×10^{4}	1.35×10^{3}	6.08×10^{3}	1.57×10^{2}	2.21×10^{4}	1.51×10^{4}	2.57×10^{3}	1.75×10^{3}
RA Blocks (8 each)	1.13×10^{-1}	6.11×10^{-2}	5.61×10^{4}	4.00×10^{3}	2.99×10^{4}	4.64×10^{2}	6.55×10^{4}	4.47×10^{4}	7.60×10^{3}	5.18×10^{3}
LI, II Blocks, (8 each)	5.44×10^{-2}	2.94×10^{-2}	4.00×10^{4}	9.19×10^{2}	2.77×10^{4}	1.07×10^{2}	3.13×10^{4}	2.13×10^{4}	3.64×10^{3}	2.48×10^{3}
LA Blocks (19 each)	1.29×10^{-1}	6.98×10^{-2}	1.09×10^{5}	1.59×10^{3}	5.92×10^{4}	1.85×10^{2}	2.28×10^{4}	1.55×10^{4}	2.65×10^{3}	1.81×10^{3}
R and L Block Plugs (11R, 5L)	(d)		1.66×10^{4}	$\sim 3.73 \times 10^{2}$	8.37×10^{3}	$\sim 1.11 \times 10^{2}$				
Flow Divider Plate	4.09×10^{-2}	2.21×10^{-2}	1.76×10^{4}	4.77×10^{-1}	1.00×10^{4}	1.28×10^{-1}	2.16×10^{1}	1.47×10^{1}	5.79×10^{0}	3.95×10^{0}
Be Control Rod (5 each)	$\sim 3.3 \times 10^{-1}$	1.78×10^{-2}	6.02×10^{4}	$\sim 6.50 \times 10^{4}$	1.06×10^{2}	$\sim 4.10 \times 10^{-1}$	2.64×10^{4}	2.49×10^{4}	2.30×10^{3}	1.57×10^{3}
Totals			3.23×10^{5}		1.53×10^{5}					

(a) Data from Appendix A of Reference 4.
(b) Inferred from the mass, assuming density of beryllium is 1.85 Mg/m^3.
(c) Activity to dose conversion factor is $\sim$0.68 R/hr/Ci/m^3 (^{60}Co), based on ^{60}Co activity. Dose rates are based on volume-averaged specific radioactivities and represent average rather than maximum values.
(d) Data not available.

rigorous calculations in Reference 3 to within a factor of 2. Thus, the dose rate calculations for aluminum and beryllium are considered to be sufficiently reliable for use in planning purposes.

The data presented in Tables E.2-1 through E.2-7 are illustrated in Figures E.2-1 and E.2-2. The decay of the total radioactivity in the reference test reactor components is shown in Figure E.2-1 as a function of time after reactor shutdown, to about 120 years. The largest single source of radioactivity is the tritium produced in the beryllium components. The radiation

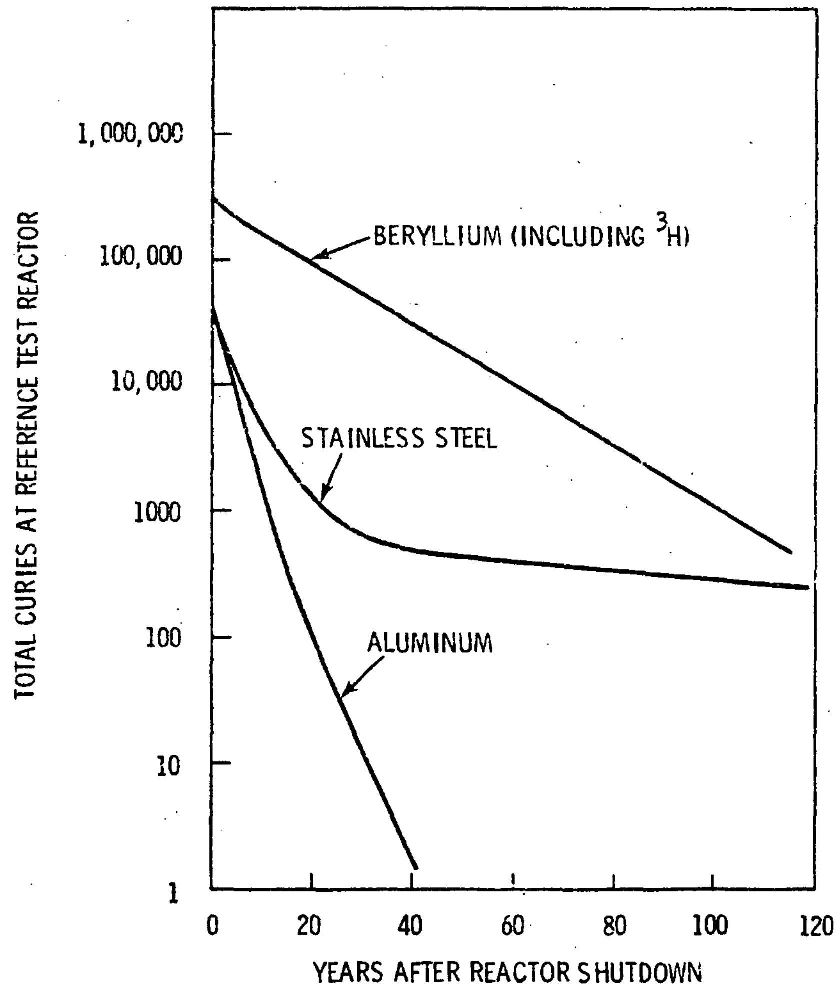

FIGURE E.2-1. Time Dependence of Total Radioactivity in Neutron-Activated Reactor Components Following Final Reactor Shutdown

dose rates from selected reactor components are shown in Figure E.2-2 as a function of time after reactor shutdown, to about 120 years. The principal source of external radiation dose is ^{60}Co for the first 100 years after reactor shutdown. Beyond 100 years, ^{94}Nb in the stainless steel becomes the dominant dose source.

E.2.1.2 Neutron-Activated Materials in the Mock-Up Reactor

The MUR operated for a total of 0.198 MWd, with a maximum power level of 100 kW, for a total of 1.98 EFPD. MUR operation was generally at power levels

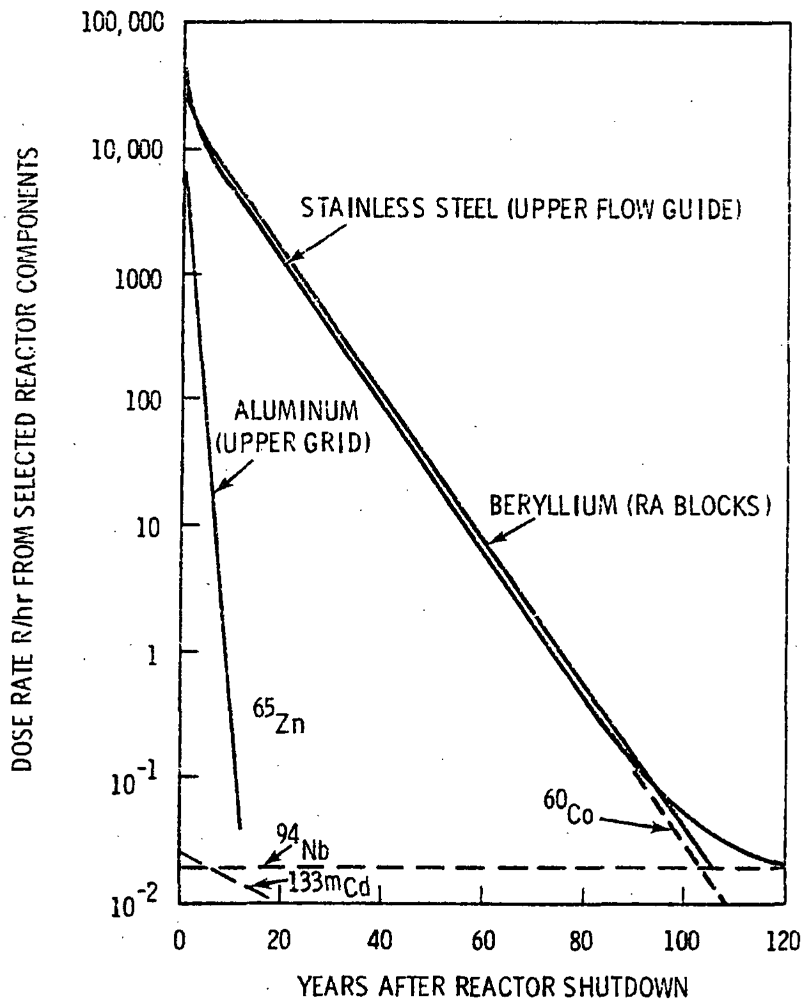

FIGURE E.2-2. Time Dependence of Radiation Dose Rate from Selected Reactor Components Following Final Reactor Shutdown

considerably less than full power and at intermittent intervals, over about 2 years less time than the reference test reactor. However, for purposes of this study, both reactors are postulated to have had similar operating time frames.

The inventory of neutron-activated materials is estimated from the inventory calculated for the reference test reactor by multiplying those values by the ratio of integrated power production (0.198 MWd/98,000 MWd) in the MUR and the test reactor. This approximation accounts in a reasonable (conservative) way for the intermittent operation of the MUR, the buildup and decay of activated radionuclides present, and the maximum values of radiation dose rate likely to be encountered using the data from Tables E.2-5, E.2-6, and E.2-7. It also accounts for the ratio of integrated power production as given in the previous paragraph. These estimates are listed in Table E.2-8.

The total inventory of radioactivity in the MUR is estimated to be quite small (<1-1/2 Ci) and the estimated maximum dose rates are also quite small (<700 mR/hr) as shown in Table E.2-8. Thus, the neutron-activated materials in the MUR do not present any significant problems for packaging or disposal during decommissioning.

E.2.1.3 Radioactivity Decay Characteristics of Selected Radionuclides

Many of the radionuclides present in the neutron-activated structures at shutdown in the reference test reactor have specific radioactivity levels but, because of their particular decay half-lives or decay processes or both, do not contribute significantly to the radiation dose rate during decommissioning. Other radionuclides have lower specific radioactivity levels but produce large contributions to an external dose rate because they emit high-energy gamma rays.

The following paragraphs present the characteristics (i.e., production process, radioactive decay mode, and radioactive emissions) of selected radionuclides of concern during decommissioning.

Hydrogen-3 ($^{3}_{1}H$). Tritium is produced in beryllium by a two-step neutron capture radioactive decay process:

TABLE E.2-8. Estimated Total Radioactivity and Maximum Radiation Dose Rates in the MUR at Reactor Shutdown, Based on Reference Test Data

	Total Radioactivity in Reactor Structural Materials (Ci)				Radiation Dose Rate in Components Having Greatest Activation (R/hr)		
	Stainless Steel	Beryllium	Aluminum	Total	Misc. Bolts (SS)	RA Blocks (Be)	Lower Grid (Al)
Test Reactor[a]	5.41×10^{3}[b]	3.23×10^{5}	4.03×10^{4}	3.69×10^{5}	3.32×10^{5}	4.47×10^{4}	3.93×10^{4}
Mock-Up Reactor[c]	1.09×10^{-2}	6.53×10^{-1}	8.14×10^{-1}	1.48×10^{0}	6.71×10^{-1}	9.03×10^{-2}	7.94×10^{-2}

(a) Data from Tables E.2-5, E.2-6, and E.2-7.
(b) This total activity represents only those reactor components present in both reactor facilities times the ratio: Adjusted Total Activity, Ci/Total, Ci, shown in Table E.2-5.
(c) Postulated ratio of integrated power production (0.198/98,000) = 2.02×10^{-6}.

$$^{9}Be\ (n,\alpha)\ ^{6}He \xrightarrow{\beta^-} {}^{6}Li\ (n,\alpha)\ ^{3}H$$

Tritium has a radioactive decay half-life of about 12.4 years and decays by beta emission, with a maximum energy of 18.6 keV. The contribution of ^{3}H to external occupational radiation exposure is insignificant since the radioactivity is tightly bound in the metallic beryllium. The principal concern is to prevent its release into the environment where it could enter the biological pathways for internal exposure.

Carbon-14 ($^{14}_{6}C$). Carbon-14 is produced by two processes: 1) by multiple neutron capture in ^{12}C and ^{13}C, and 2) by neutron capture-proton emission in ^{14}N. The second reaction is the predominant one. Both carbon and nitrogen are present in trace quantities in stainless and carbon steels. Carbon-14 has a radioactive decay half-life of about 5,750 years and decays by beta emission, with a maximum energy of 156 keV. The contribution of ^{14}C to external occupational radiation exposure is insignificant since the radioactivity is bound tightly in the metal alloys. The principal concern is to prevent its release into the environment where it could enter biological internal-exposure pathways.

Iron-55 ($^{55}_{26}Fe$). Iron-55 is produced principally by thermal neutron capture in 5.8%-abundant ^{54}Fe in iron, a constituent in stainless steel and carbon steel. Iron-55 has a decay half-life of 2.7 years and decays by orbital electron capture (EC) by the nucleus to form $^{55}_{25}Mn$.

Each EC event is accompanied by the emission of either x-rays or Auger electrons associated with ^{55}Mn stabilization to the ground energy state, with energies less than 7 keV. As is typical with such emissions from nuclides in this atomic number range, most ($\sim$74% in this case) of the x-rays are not emitted, and Auger electrons of the same energy appear instead. These emissions do not contain the energy (i.e., >68 keV) required to penetrate the outer dead layers of skin and, thus, do not contribute to a skin dose.

Each EC decay event is accompanied by a continuous spectrum of gamma rays from the associated inner bremsstrahlung (IB) process. The maximum energy of this spectrum is 220 keV. For computing radiation dose in this study, the IB

spectrum of ^{55}Fe decay is divided into four energy ranges, with a specific yield fraction per decay event for each energy range as follows:

- 0.01 to 0.07 MeV, 1.37×10^{-5}
- 0.07 to 0.12 MeV, 8.37×10^{-6}
- 0.12 to 0.21 MeV, 4.77×10^{-6}
- 0.21 to 0.04 MeV, 8.5×10^{-9}.

<u>Nickel-59 ($^{59}_{28}Ni$)</u>. Nickel-59 is produced primarily by thermal neutron capture in 68.3%-abundant ^{58}Ni in nickel, a constituent in stainless steel. Nickel-59 has a decay half-life of approximately 80,000 years and decays by the EC process to $^{59}_{27}Co$.

Again, x-rays or Auger electrons are emitted during stabilization to the ground energy state. In the case of ^{59}Co, the energies are less than 8 keV, and approximately 33% of the emissions are x-rays and the rest are Auger electrons. These emissions do not contribute to the external radiation dose rate.

The contribution of the ^{59}Ni decay process to external dose rate comes from the EC-associated IB process. A continuous spectrum of gamma rays is emitted, with the maximum energy being 1.07 MeV. For computing radiation dose in this study, the IB spectrum of ^{59}Ni decay is divided into seven energy ranges, with a specific yield fraction per decay event for each energy range, as follows:

- 0.01 to 0.07 MeV, 2.23×10^{-4}
- 0.07 to 0.12 MeV, 6.25×10^{-5}
- 0.12 to 0.21 MeV, 1.32×10^{-4}
- 0.21 to 0.4 MeV, 3.21×10^{-4}
- 0.4 to 0.6 MeV, 3.31×10^{-4}
- 0.6 to 0.9 MeV, 3.24×10^{-4}
- 0.9 to 1.2 MeV, 3.01×10^{-5}.

<u>Nickel-63 ($^{63}_{28}Ni$)</u>. Nickel-63 is produced by thermal neutron capture in 3.6%-abundant ^{62}Ni in nickel, a constituent in stainless steel. Nickel-63 has a decay half-life of about 100 years and decays by beta emission to $^{63}_{29}Cu$. The beta emissions have maximum energies of about 66 keV, which is insufficient to penetrate the outer surface of the skin. No gamma rays are emitted. Therefore, ^{63}Ni does not contribute to the external dose rate.

Cobalt-60 ($^{60}_{27}Co$). Cobalt-60 is produced by thermal neutron capture in 100%-abundant ^{59}Co, a trace constituent in all three structural materials. Cobalt-60 has a decay half-life of 5.27 years and decays by beta emission to excited energy states of $^{60}_{28}Ni$. Over 99% of the decay events are to an excited energy state at 2.505 MeV, which goes to the ground energy state, with a sequential emission of two gamma rays of 1.173 MeV and 1.332 MeV. These are the major contributors to the dose rate from ^{60}Co. The beta spectrum (maximum-energy 0.32 MeV) contributes a minor fraction.

Zinc-65 ($^{65}_{30}Zn$). Zinc-65 is formed by neutron capture in 48.6%-abundant ^{64}Zn, an element found in aluminum alloys. Neutron capture leads to $^{65}_{30}Zn$, which decays by beta emissions to the 1.115 MeV level of $^{65}_{29}Cu$ (49%) or by EC (49%) or beta emission (1.7%) to the ground state of $^{65}_{29}Cu$. A 1.115-MeV gamma ray is emitted when the excited level drops to the ground state. The maximum beta energy is about 0.3 MeV. The half-life for radioactive decay of $^{65}_{29}Zn$ is about 244 days.

Cadmium-113m ($^{113m}_{48}Cd$). Cadmium-113m is formed by neutron capture in 24.1%-abundant ^{112}Cd, a trace element found in beryllium. The excited state, $^{113m}_{48}Cd$, decays to the ground state by emitting an 0.265-MeV gamma ray, ($\sim$0.1% of transitions) or decays to the ground state of $^{113}_{49}In$ by emitting a 0.58-MeV beta (99 + % of transitions). The radioactive decay half-life is about 14.6 years.

Niobium-94 ($^{94}_{41}Nb$). Niobium-94 is formed by thermal neutron capture in 100%-abundant ^{93}Nb, a trace constituent in stainless steel. Neutron capture leads either to the ground state of ^{94}Nb or to an isomeric state that is 41 keV above the ground state. The isomeric state decays with a half-life of 6.26 minutes. Most (99.5%) of these decays go to the ground energy state. Thus, for long-time radiation dose rate considerations, all neutron captures lead to the ground energy state. Niobium-94 has a decay half-life of 20,300 years and decays by beta emission, with a maximum beta energy of 471 keV. All of the decays go to a single excited energy state of ^{94}Mo at 1.574 MeV excitation. All of the decays from this state go through an excited

energy state at 871 keV. Thus, each ^{94}Nb decay yields two gamma rays with energies of 703 and 871 keV, respectively; these dominate the dose rate from ^{94}Nb.

E.2.2 Surface Contamination in the Reference Test Reactor Facility

While activated corrosion products from structural materials in contact with the reactor water and fission products from leaking fuel can both contribute to radionuclide mixtures and levels of surface contamination, based on historical data no fuel failures are assumed to have occurred at the reference test reactor.[4] Therefore, fission products from leaking fuel are neglected as a general contributor to surface contamination levels. It is assumed, however, that the cutting of fuel did occur in certain hot cells in the Hot Laboratory Building. These activites were conducted under rigidly controlled conditions within local confinement envelopes and using appropriate bag-out procedures to limit surface contamination from this source to specific areas of the cell itself.

The limited amount of information on radionuclide mixtures and/or inventories present at shutdown at the reference test facility is presented in the following subsections. In those areas where actual data are unavailable, estimates are made using past experience and engineering judgement.

E.2.2.1 Radionuclide Inventories in Internally Contaminated Piping and Equipment

A thin layer of radioactive contamination is deposited on the internal surfaces of piping and equipment in the reference test reactor during its normal operating lifetime. The piping and equipment systems involved are described in Appendix C. The composition and amount of radioactivity found on internal surfaces at plant shutdown are dependent on such reactor parameters as: 1) structural material composition, 2) reactor size, design, and operating history, and 3) reactor fuel conditions. In general, the internal surface contamination is characterized by the mixture of activated corrosion products and fission products (if any) found in the reactor water.

It is estimated that after draining and flushing tasks are completed, the presence of radioactive materials elsewhere in the reference plant is minimal,

mostly as trace internal and surface contamination.[4] Three exceptions are: the interior of the PCWS, the interior of certain hot cells, and the hot cell drain pipe in the Hot Pipe Tunnel. These areas are estimated to contain quantities of radioactivity ranging from a few millicuries to a few curies. These estimates are based on the actual monitoring of accessible system components performed during SAFSTOR activities at the reference test reactor in early-1973 and subsequent surveys, as reported in Reference 4. The primary system contamination is assumed to be concentrated at the inlet end of the heat exchangers. Contamination in the hot cells is most prevalent on equipment located in Cells 1 and 2.

Although the exact quantities of the individual constituents of the radionuclide mixtures contributing to the various surface contamination levels are unavailable, their general composition and characteristics are known and are given in Table E.2-9.

Production of the radionuclides given in Table E.2-9 is described in Section E.2.1.3. Although significant inventories of ^{3}H are present, it is contained and confined within the metal matrix of beryllium pieces. During the operating years of the reference test reactor, operational sampling and experiments confirmed that no tritium was released even during underwater cutting and burning of beryllium components during replacement of bowed and fractured beryllium reactor core side plates. This lends support to the belief that tritium is well-contained within the metal matrix. Of all the

TABLE E.2-9. Radionuclide Composition and Characteristics of Surface Contamination

Nuclide	Half-Life (years)	Emission
^{3}H	12.3	β only
^{60}Co	5.2	β and γ
^{55}Fe	2.4	X-Ray and γ
^{63}Ni	92	β only γ
^{59}Ni	8×10^{4}	X-Ray and
^{65}Zn	0.7	β and γ
^{26}Al	7.4×10^{5}	β and γ

radionuclides, ^{60}Co is of prime concern as a surface contaminant since this isotope heavily influences the degree of shielding and remote operations necessary to control external dose rates.

Based upon actual monitoring data obtained at the reference test reactor (see Appendix D) and making conservative upward adjustments in radiation dose rate to account for original systems shutdown conditions, the internal surface contamination of PCWS piping and equipment is estimated. For example, using Figures E.1-1 and E.1-2, a 12.5-mm stainless steel pipe in the PCWS with a contact dose rate in the 10 mR/hr range will have a ^{60}Co internal surface contamination of about 0.3 mCi/m^2 at final reactor shutdown.

E.2.2.2 Radionuclide Inventories on Externally Contaminated Structural Surfaces

In general, the radionuclide mixture found on most externally contaminated structural surfaces in the reference test reactor, with the exception of the hot cells, is assumed to reflect the mixture of radionuclides found in the reactor water (as previously discussed in Section E.2.2.1). Leaks occurring in normally accessible areas are assumed to be repaired and cleaned up according to standard operating procedures. Leaks occurring in areas not normally accessible are assumed to accumulate and build up over a 12-year plant operating lifetime. A 12-year plant operating lifetime is considered conservative based on the operating lifetimes of the eight NRC-licensed test reactors in existence. Currently (see Section 3, Volume 1), seven of the eight test reactors are shut down. The average operating lifetime of these seven test reactors was about 8.4 years, with the reference test reactor above average at about 12 years. The radionuclide inventories in the hot cells are discussed in the following subsection.

E.2.2.3 Radionuclide Inventories on Contaminated Hot Cell Surfaces

An estimate of the amounts of radioactive contaminants in the hot cells after shutdown is presented in Table E.2-10. The inventory is based on: 1) actual post-shutdown measurements of total radioactivity at the reference test facility, taken in 1973 after considerable cleanup activities had been completed,[8] and 2) the distribution of major isotopes of radionuclide

TABLE E.2-10. Estimated Inventory of Major Radionuclides in Hot Cells of the Hot Laboratory Building at Shutdown[a]

Fission Products	Estimated Radioactivity (Ci) Hot Cell No. 1	Hot Cell No. 2	Hot Cell No. 3	Hot Cell No. 4	Hot Cell No. 5	Hot Cell No. 6	Hot Cell No. 7	Total Inventory by Isotope, Ci
^{90}Sr, ^{90}Y[b]	3.3×10^{0}	1.7×10^{0}	8.1×10^{-2}	3.2×10^{-2}	1.2×10^{-2}	2.1×10^{-2}	4×10^{-3}	5.2
^{106}Ru, ^{106}Rh	6.6×10^{-1}	3.4×10^{-1}	1.6×10^{-2}	6.5×10^{-3}	2.4×10^{-3}	4.3×10^{-3}	8×10^{-4}	1.0
^{134}Cs	3.3×10^{-1}	1.7×10^{-1}	8×10^{-3}	3.2×10^{-3}	1.2×10^{-3}	2.1×10^{-3}	4×10^{-4}	0.5
^{137}Cs	2.6×10^{0}	1.4×10^{0}	6.5×10^{-2}	2.6×10^{-2}	9.7×10^{3}	1.7×10^{-2}	3.2×10^{-3}	4.1
^{144}Ce, ^{144}Pr	6.6×10^{-1}	3.4×10^{-1}	1.6×10^{-2}	6.5×10^{-3}	2.4×10^{-3}	4.3×10^{-3}	8×10^{-4}	1.0
^{147}Pm	3.3×10^{-1}	1.7×10^{-1}	8×10^{-3}	3.1×10^{-3}	1.2×10^{-3}	2.1×10^{-3}	4×10^{-4}	0.5
^{151}Sm	1.3×10^{-1}	6.8×10^{-2}	3×10^{-3}	1.3×10^{-3}	5×10^{-4}	8.6×10^{-4}	2×10^{-4}	0.2
^{154}Eu	1.3×10^{-1}	6.8×10^{-2}	3×10^{-3}	1.3×10^{-3}	5×10^{-4}	8.6×10^{-4}	2×10^{-4}	0.2
Actinides								
U (all isotopes)	5.7×10^{-3}	3×10^{-3}	1.4×10^{-4}	6×10^{-5}	2.1×10^{-5}	4×10^{-5}	7×10^{-6}	<0.1
^{244}Cm	3×1^{-1}	1.6×10^{-1}	7×10^{-3}	3×10^{-3}	1.1×10^{-3}	2×10^{-3}	4×10^{-4}	0.5
Total Radioactivity	8.5×10^{0}	4.4×10^{0}	2.1×10^{-1}	8.3×10^{-2}	3.1×10^{-2}	5.5×10^{-2}	1×10^{-2}	~13.3

(a) 60% of all inventory assumed to be on stainless steel walls and 40% on concrete.
(b) Where isotopes are grouped, radioactivities are total for the groups.

inventory that was estimated previously for an intermediate-level cell located at a reference nuclear fuel reprocessing plant (FRP).[5] Except for size, the FRP cell is considered to be similar to the reference test facility's hot cells in scope of operations and design. For each cell about 60% of this contamination is assumed to be on stainless steel and 40% on concrete. In addition, this estimate does not include any buildup of large quantities of process material or debris in the cells, because these are assumed to be cleaned up as a part of normal housecleaning procedures during plant operation.

The estimated inventory appears reasonable and consistent with the stated bases and assumptions. However, these estimates are highly dependent on the operating philosophy at the plant, and the values presented in Table E.2-10 represent what is expected to be a typical case for the reference hot cells for the assumptions used. Actual hot cell operations could result in values different from those given.

E.2.2.4 Radionuclide Inventory on the Reference Test Reactor Site

This subsection contains a discussion of the radionuclide mixture and contamination level present on the reference site resulting from normal test reactor operation. Releases of radionuclides resulting from accidents are not expected to significantly increase the radioactivity on the reference site and, therefore, are not considered in this analysis. Information about the level and nature of the radioactive contamination present at the time of decommissioning is needed to determine the alternative future uses of the site.

For this study, deposition of airborne radionuclides during 12 years of normal test reactor operation is considered to be insignificant because of the relatively small plant size, the absence of any fuel failures, and the extensive use of gaseous radwaste treatment systems. Naturally occurring radionuclides and those resulting from nuclear weapons-testing fallout are present on the site, but deposition of these latter radionuclides is not quantified in this study. However, low levels of radioactive contamination are anticipated to be present in three areas on the reference site as a result of deposition of waterborne radionuclides. The three areas are: 1) the contaminated drainage ditches, 2) the Emergency Retention Basin, and 3) the soil beneath the two Cold Retention Basins. Descriptions of these areas are given in Section C.4,

Appendix C and are not repeated here. The results of a recent (1981) soil surface sample taken from the Emergency Retention Basin at the point of highest concentration are given in Table E.2-11. The calculated deposited radioactivity values at various times after shutdown are shown to account for decommissioning of the aforementioned areas after specific periods of radioactive decay. A subsurface sample taken at a depth of 0.3 m directly below the surface sample indicated decreasing values for all radionuclides by factors ranging from about 5 for ^{90}Sr to 466 for ^{60}Co. The maximum surface level given in Table E.2-11 is assumed to be the same for all three of the aforementioned areas that contain contaminated soil and is used in Appendix F to determine the maximum annual dose to the maximum-exposed individual living on the decommissioned reference site.

TABLE E.2-11. Reference Radionuclide Inventory, Soil Contamination on the Reference Test Reactor Site[(a)]

Radionuclide	Deposited Radioactivity (pCi/g) at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{60}Co	1.73×10^{2}	4.7×10^{1}	3.4×10^{0}	2.4×10^{-1}	3.4×10^{-4}
^{63}Ni	1.37×10^{1}	1.28×10^{1}	1.1×10^{1}	9.7×10^{0}	6.9×10^{0}
^{90}Sr	8.23×10^{0}	6.48×10^{0}	4×10^{0}	2.5×10^{0}	7.5×10^{-1}
^{134}Cs	1.59×10^{1}	5.5×10^{-1}	6.6×10^{-4}	7.8×10^{-7}	3.9×10^{-14}
^{137}Cs	6.59×10^{1}	5.23×10^{1}	3.3×10^{1}	2.1×10^{1}	6.6×10^{0}
^{239}Pu	3×10^{-2}	3×10^{-2}	3×10^{-2}	3×10^{-2}	3×10^{-2}

(a) Based on information supplied by NASA Lewis Research Center; early-1981 sample results.

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APPENDIX F

PUBLIC RADIATION DOSE MODELS AND CALCULATED MAXIMUM ANNUAL DOSES

This appendix contains information to support the discussions in Sections 9 and 12. Terminology, models, equations and parameters used in radiation dose calculations, as well as tabulated maximum annual doses to the maximum-exposed individual, are presented. The tabulated maximum annual doses are based on the rerference radionuclide inventories detailed in Appendix E. These doses are used to determine the acceptable contamination levels for public release of the decommissioned reference research and test reactors. It should be noted that the occupational radiation doses during decommissioning operations are not calculated using the dose models presented in this appendix.

F.1 RADIATION DOSE TERMINOLOGY AND DEFINITIONS

The following terminology and definitions apply for the radiation dose calculations made for this study:

- Routine Release. An atmospheric release of radionuclides from routine decommissioning tasks that is assumed to occur at a constant rate for one year.
- Acute Release. A short-duration (usually less than 8 hours) release of radionuclides. The accident analysis, discussed in Section 12 and Appendix N, is based on acute releases.
- Organs of Reference. The organs of the human body for which radiation doses are calculated. For this study, the organs of reference are the total body, lungs, bone, and lower large intestine (LLI) of the GI-tract. The total body is the head and trunk of the human body and includes active blood-forming organs, eye lenses, and gonads.
- Exposure Pathways. The potential routes by which people may be exposed to radionuclides or radiation. Radiation exposure pathways in the environment that are considered in this study are: external exposure to contamination deposited on the ground, ingestion of food

products containing radionuclides, and inhalation of airborne radionuclides. Radiation exposure pathways inside the research and test reactors are: external exposure from contaminated room surfaces and inhalation of airborne radionuclides. External exposure from airborne radionuclides (air submersion) is not considered, since previous decommissioning studies in this series have shown this exposure pathway to be insignificant compared to the others.(1-3)

- Maximum-Exposed Individual. The individual who receives the maximum radiation dose to an organ of reference. The maximum-exposed individual is assumed to reside at the location of the highest airborne radionuclide concentration. Maximized exposure pathway parameters are used.
- Radiation Dose Equivalent. A quantity used to express the absorbed dose from all radiations on a common scale in units of rem. It is defined as the product of the absorbed dose in rads, a quality factor intended to allow for the effect of the microscopic distribution of absorbed energy, and other modifying factors as defined by the International Commission on Radiological Protection (ICRP).
- Committed Radiation Dose Equivalent. The dose equivalent accumulated over a specific number of years from radionuclides deposited in the body from the various radiation pathways during a given year.
- Collective Dose. The summation of radiation dose equivalents received by all individuals in the population of concern. It is calculated by multiplying the dose to the average individual by the population distribution for the site listed in Appendix A. Average individual exposure parameters values are used for the exposure pathways.
- First-Year Dose. The radiation dose equivalent accumulated during the year of the release period.
- Fifty-Year Committed Dose Equivalent. The committed radiation dose equivalent for 50 years following a 1-year exposure to radionuclides.

- Annual Dose. The radiation dose equivalent calculated during any year following the start of continuous exposure. It is the sum of the dose received by an organ of reference during the year of interest from all exposure pathways and the dose received that year from radionuclides deposited in the organ of reference during previous years of exposure.
- Maximum Annual Dose. The largest of the 50 annual doses calculated to occur during the 50 years following the start of continuous exposure.
- Biological Half-Life. The time required for the body to eliminate half of any substance by regular body elimination processes.
- Solubility Classes D, W, and Y. Radionuclides are classified according to the rate at which they are dissolved into body fluids in the lung after inhalation. The three classifications used in this study are: Class D materials, with a maximum biological half-life of less than one day; Class W materials, with a maximum biological half-life ranging from a few days to a few months; and Class Y materials, with a maximum biological half-life of from six months to a few years.[4] Table F.1-1 contains a list of the solubility classes of the elements considered in this study.

F.2 GENERAL MODELS FOR ESTIMATING RADIATION DOSE

The methodology and parameters necessary to calculate the first-year dose or the fifty-year committed dose equivalent are presented in this section. The fundamental relationship for calculating radiation doses to people from any radionuclide or exposure pathway is given in Equation F.1.[6]

$$R_{ipr} = C_{ip} U_p D_{ipr} \tag{F.1}$$

where:

R_{ipr} • first-year dose of fifty-year committed dose equivalent from radionuclide i via exposure pathway p to organ r

TABLE F.1-1. Solubility Classes of the Elements Present in the Radionuclide Inventories at Final Shutdown for the Reference Research and Test Reactors

Element	Solubility Class[a] (all other organs/lungs)	Reference[b]
Hydrogen	D/D	4
Carbon	D/D	4
Phosphorus	D/W	4
Chromium	W/Y	4
Manganese	D/W	5
Iron	W/Y	5
Cobalt	W/Y	5
Nickel	W/W	4
Zinc	W/Y	4
Strontium	D/Y	5
Yitrium	D/Y	5
Zirconium	W/Y	5
Niobium	W/Y	5
Technetium	D/W	5
Ruthenium	Y/Y	5
Iodine	D/D	5
Cesium	D/D	5
Barium	D/W	4
Lanthanum	W/Y	4
Cerium	W/Y	4

(a) Solubility classes are for use in the ICRP Task Group Lung Model.
(b) Reference numbers shown at the end of this appendix.

D_{ipr} • Radiation dose equivalent factor of fifty-year committed radiation dose equivalent factor (for a given radionuclide i, exposure pathway p, and organ r) that converts the concentration and usage parameters to the first-year dose or fifty-year committed dose equivalent.

Each of the terms in the general dose equation (Equation F.1) is discussed in detail below:

- $\underline{C}_{ip}$. An analysis of the radiation doses from separate exposure pathways requires a calculation of the radionuclide concentrations associated with each pathway. An external exposure analysis requires a soil concentration or surface contamination level. The radionuclide concentration in food crops grown in contaminated soil is directly related to the radionuclide concentration in the soil plow layer (a depth of 0.15 m). Exposure pathways that require the concentration of radionuclides in air from airborne radionuclide releases use dilution factors discussed in the site description in Appendix A. They are: 7.2×10^{-8} sec/m^3 for the maximum-exposed individual located 1 km from the site and 2.3×10^{-11} sec/m^3 for the average member of the population residing within 80 km of the site.
- $\underline{U}_p$. Each radiation exposure pathway has an associated exposure rate or intake rate. Parameters used to calculate radiation doses from the consumption of food products grown in contaminated soil are listed in Table F.2-1. For the inhalation pathway, the intake rate (or ventilation rate) is assumed to be 2.7×10^{-4} m^3/sec for members of the public and 3.3×10^{-4} m^3/sec for workers with a 40-hr work week.
- $\underline{D}_{ipr}$. The radiation dose equivalent factors and the fifty-year committed radiation dose equivalent factors that are used in this study for internal exposure pathways are obtained from several sources. Those for ingestion are obtained primarily from References 6, 7, 8, 9, and 10, which are derived from information given

TABLE F.2-1. Parameters Used for Calculation of Radiation Doses from Consumption of Foods[a]

Food	Growing Period (days)	Yield (kg/m^2)	Holdup (days)[b]	Up, Maximum Individual Consumption (kg/year)[c]	Up, Average Individual Consumption (kg/yr)
Leafy Vegetables	90	1.5	1	30	15
Other Above-Ground Vegetables	60	0.7	1	30	15
Potatoes	90	4.0	10	110	71
Other Root Vegetables	90	5.0	1	72	47
Berries	60	2.7	1	30	6
Melons	90	0.8	1	40	8
Orchard Fruit	90	1.7	10	265	50
Wheat	90	0.72	10	80	72
Other Grain	90	1.7	1	8	8
Eggs	90	0.84[d]	2	30	20
Milk	30	1.3[d]	2	274[e]	230[e]
Beef	90	0.84[d]	15	40	40
Pork	90	0.84[d]	15	40	30
Poultry	90	0.84[d]	2	18	8

(a) Based on information in Reference 11.
(b) Time between harvest and consumption.
(c) Only that fraction of the diet grown locally (and, therefore, potentially contaminated) is listed.
(d) Yield of animal feeds (i.e., grain or pasture grass).
(e) Units of ℓ/year.

by the ICRP for the body burden and the maximum permissible concentration of each radionuclide.[11] Ingestion factors not obtainable from the references are calculated by the method given in Reference 10. In this method, effective decay energies for the radionuclides are calculated from the ICRP model, which assumes that all of a given radionucide is in the center of a spherical organ with an appropriate effective radius. Dose factors for inhalation are calculated using the computer code DACRIN.[12] This code incorporates the ICRP Task Group Lung Model to calculate the dose to the lung and other organs or reference. A particle size of 1 micrometer in diameter is assumed. Radiation dose equivalent factors for internal pathways have units of mrem per year per pCi taken into the body during that same year. Committed radiation dose equivalent factors have units of mrem/50-yr per pCi taken into the body in the first year. External exposure can arise from either contaminated land or contaminated building surfaces. In this study, external dose equivalent factors are calculated for these sources using the ISOSHLD computer code.[13,14] Ground contamination is modeled by calculating the dose at a point 1 m in air above a semi-infinite slab of radionuclides evenly distributed in a 0.15-m layer of soil. Total body dose rates are obtained by using a 50-mm tissue equivalent thickness around the dose point.

Discussion of the various radiation exposure pathways considered in this study, together with appropriate modifications to the general dose equation and its parameters, are given in the following sections.

F.2.1 External Exposure

Calculation of the radiation dose resulting from external total body exposure to surface contamination is done with no modification to Equation F.1. Soil concentrations are calculated using a simple cloud-depletion model, as is described for the food ingestion pathway analysis in the next section. For routine releases from decommissioning operations, the maximum-exposed individual is assumed to be exposed to contaminated soil for 12 hr/day for a full year. The average individual is assumed to be exposed for 6 hr/day for a full

year. The 50-yr committed dose equivalent for external exposure to surface contamination is equal to the first-year dose, since there is no internal radionuclide deposition.

F.2.2 Ingestion

The radiation dose equivalent or committed radiation dose equivalent to a person consuming vegetation grown on the contaminated site is given in Equation F.2.

$$R_{vr} = \sum_{i=1}^{n} C_{iv} U_v D_{ir} \tag{F.2}$$

Similarly, the radiation dose equivalent or committed radiation dose equivalent to a person consuming a contaminated animal product is given in Equation F.3.

$$R_{ar} = \sum_{i=1}^{n} C_{ia} U_a D_{ir} \tag{F.3}$$

where:

R_{vr}, R_{ar} • radiation dose equivalent or committed radiation dose equivalent from vegetation (v) or animal (a) food product pathways, mrem or mrem/50 yr

C_{iv}, C_{ia} • concentration of radionuclide i in vegetation (v) or animal product (a), pCi/kg or pCi/ℓ

U_v, U_a • annual consumption of contaminated vegetable (v) or animal products (a), kg or ℓ

D_{ir} • radiation dose equivalent factor or committed radiation dose equivalent factor, mrem/pCi or mrem/50 yr/pCi.

Specific values of U_v and U_a are shown in Table F.2-1.

F.2.2.1 Deposition of Food Products or Ground Surfaces

Equation F.4 is used to describe the deposition of airborne particulate radionuclides directly on food products or ground surfaces.

$$d_i = 86{,}400\ \bar{\chi}_i V_{di} \tag{F.4}$$

where

d_i • deposition rate of flux of radionuclide i, pCi/(m^2-day)

86,400 • dimensional conversion factor, seconds/day

V_{di} • deposition velocity of radionuclide i, m/second

$\bar{\chi}_i$ • annual average air concentration from airborne releases of radionuclide i, pCi/m^3.

Specific values of the deposition velocity, V_{di}, are obtained from Reference 15.

F.2.2.2 Concentration in Vegetation

The concentration of radioactive material in vegetation resulting from direct deposition onto plant foliage and uptake of prior depositions from the soil is given in Equation F.5.

$$C_{iv} = d_i \left[\frac{r\ T_v(1-e^{-\lambda_{Ei}\ t_e})}{Y_v \lambda_{Ei}} + \frac{B_{vi}\ (1-e^{-\lambda_i\ t_b})}{P\ \lambda_i} \right] e^{-\lambda_i\ t_h} \tag{F.5}$$

where:

C_{iv} • concentration of radionuclide i in the edible portion of the vegetation (v), pCi/kg

d_i • deposition rate or flux or radionuclide i, pCi/(m^2-day)

r • fraction of deposition retained on the vegetation (dimensionless), assumed to be 0.25 in this study

T_v • factor for translocation of externally deposited radionuclides to the edible parts of the vegetation (dimensionless). For simplicity it is assumed in this study to be independent of the radionuclide

and is 1.0 for leafy vegetables and fresh forage, and 0.1 for all other produce, including grain. (Reference 8 lists values of this parameter for various radionuclides.)

λ_i • radiological decay constant for radionuclide i, day^{-1}

λ_{Ei} • effective removal constant for radionuclide i, day^{-1}; $\lambda_{Ei} = \lambda_i + \lambda_w$, where λ_i is as defined above and λ_w is the weathering removal constant for vegetation, day^{-1}, assumed to be $0.693/14 = 5 \times 10^{-2}$ day^{-1}

Y_v • yield of vegetation (v), kg (wet weight)/m^2

B_{vi} • concentration factor for uptake of radionuclide i from the soil in vegetation (v), pCi/kg (net weight) per pCi/kg (dry soil)

t_b • time for buildup of radionuclides in the soil, days; assumed in this study to be 40 yr of routine research and test reactor operation

t_e • time of exposure of above-ground vegetation to contamination during growing season, days

t_h • holdup time between harvest and food consumption, days

P • soil "surface density," kg (dry soil)/m^2; a value of 224 kg/m^2 is used assuming the contaminated ground is plowed to a depth of 0.15 m after decommissioning.

The first term inside the brackets of Equation F.5 relates to the concentration resulting from direct deposition on foliage during the growing season. The second term relates to the uptake from the soil and reflects the deposition throughout the research and test reactor operating lives. Specific values for the parameters in Equation F.5 are obtained from References 7, 15, and 16.

F.2.2.3 Concentration in Animal Products

The radionuclide concentration in animal products such as meat, milk, or eggs is dependent on the amount of contaminated forage of feed eaten by the animal. This concentration is described by Equation F.6.

$$C_{ia} = S_{ia} \, C_{iF} \, Q_F \tag{F.6}$$

where:

C_{ia} • concentration of radionuclide i in animal products (a), pCi/kg or pci/ℓ

S_{ia} • transfer coefficient of radionuclide i to the edible portion of animal product (a) from daily intake by the animal, pCi/ℓ (milk) per pCi/day or pCi/kg (animal product) per pCi/day.

C_{iF} • concentration of radionuclide i in feed or forage F, pCi/kg; calculated from Equation F.5

Q_F • animal consumption rate of contaminated feed or forage F, kg/day.

Specific values for the parameters used in Equation F.6 are obtained from References 7, 15, and 16. Where data are lacking, comparisons are made with biological data from chemically similar elements.

F.2.3 Inhalation

The radiation dose equivalent or the committed radiation dose equivalent via inhalation is calculated using Equation F.7.

$$R_{ir} = \sum_{i=1}^{n} \bar{X}_i \, B \, T \, D_{ir} \tag{F.7}$$

where:

R_{ir} • radiation dose equivalent or committed radiation dose equivalent to organ r from radionuclide i, mrem or mrem/50 yr

$\bar{X}_i$ • annual average airborne concentration of radionuclide i, pCi/m^3

B • ventilation rate of exposed individual:
for a chronic release, $B = 2.7 \times 10^{-4}$ m^3/sec;
for an acute release, $B = 3.3 \times 10^{-4}$ m^3/sec[11]

T • time of exposure to the airborne radionuclide concentration, sec

D_{ir} = radiation dose equivalent factor or committed radiation dose equivalent factor, mrem/pCi or mrem/50 yr/pCi.

The inhalation dose model presented in Equation F.7 is consistent with the ICRP Task Group Lung Model.

F.3 MODEL FOR ESTIMATING ANNUAL DOSE

The determination of acceptable residual contamination levels at the decommissioned reference research and test reactors (discussed in Section 9) is made by direct comparisons of calculated annual doses with an assumed annual dose limit. The annual dose to internal body organs from internally deposited radionuclides tends to increase for a time (after the start of continuous exposure to a radioactively decaying source) until a maximum is reached. The annual dose then tends to decrease with time because of radioactive decay, decrease in the radionuclide concentrations of specific exposure pathways, and biological elimination of radionuclides deposited in the organ. For continuous exposure to a radioactively decaying source, the year in which the annual dose is a maximum depends on the chemical and physical characteristics of the radionuclides in the source, the organ of reference, and the exposure pathway.

The calculated first-year dose from ingested or inhaled radionuclides will most likely underestimate the maximum annual dose during continuous exposure, and therefore does not provide the best comparison to annual dose limit. Nor is it appropriate to compare a fifty-year dose (accounting for 50 years of committed radiation dose equivalent) to an annual dose limit. Therefore, an extension of the general dose relationship given in Equation F.1 is required to calculate annual doses for each year of a continuous exposure period. The maximum annual dose occurring in the continuous exposure period is then compared to the annual dose limit to determine acceptable residual contamination levels.

Calculation of the annual dose to an organ of reference requires the dose equivalent from exposure during the year of interest. It also requires a detailed accounting of the committed dose equivalents to that organ from radionuclides deposited during previous years of intake. The general expression for

annual dose calculations is deduced by inspecting the annual dose equations for the first 3 years of continuous exposure. The annual dose for the first year to an organ of reference is simply the summation of the radiation dose equivalents from all internal and external exposure pathways. For the second year, the annual dose is calculated by the following expression:

$$A_2 = R_2^* + (R_{1,2} - R_{1,1}) \tag{F.8a}$$

where:

A_2 • the annual dose during the second year from all exposure pathways to the organ of reference, mrem

R_2^* • the radiation dose equivalent in the second year to the organ of reference from all internal and external exposure pathways from intake and exposure in the second year of continuous exposure, mrem

$R_{1,2}$ • the committed dose equivalent to the organ of reference for the first two years from radionuclides internally deposited during intake from exposure pathways in the first year, mrem

$R_{1,1}$ • the radiation dose equivalent to the organ of reference for the first year from radionuclides internally deposited during intake from exposure pathways in the first year (no external component to the dose equivalent), mrem.

The second-year annual dose to an organ of reference (A_2 in Equation F.8a) is the summation of the radiation dose equivalents from all exposure pathways during the second year and the dose equivalent resulting from the radionuclides internally deposited in that organ during the first year. The term in parentheses in Equation F.8a is the expression for the dose equivalent to the organ of reference from radionuclides deposited in that organ in the first year. It is found by subtracting the first-year dose equivalent from internally deposited radionuclides from the two-year committed dose equivalent.

The mathematical expression for the annual dose to an organ of reference in the third year of continuous exposure is shown in Equation F.8b.

$$A_3 = R_3^* + (R_{1,3} - R_{1,2}) + (R_{2,2} - R_{2,1}) \tag{F.8b}$$

where:

A_3 • the annual dose during the third year from all exposure pathways to the organ of reference, mrem

R_3^* • the radiation dose equivalent in the third year to the organ of reference from all internal and external exposure pathways from intake and exposure in the third year of continuous exposure, mrem.

The terms $R_{1,3}$, $R_{1,2}$, $R_{2,2}$, and $R_{2,1}$ are of similar form, each having two subscripts. The first subscript defines the year of intake or exposure after the start of continuous exposure, and the second defines the number of years used in calculating the committed dose equivalent.

The quantity in the first parentheses in Equation F.8b is the dose equivalent to the organ of reference in the third year from radionuclides deposited in the first year of continuous exposure (i.e., the difference between the 3-yr committed dose equivalent and the 2-yr committed dose equivalent). The quantity in the second parentheses is the dose equivalent in the third year to the organ of reference from radionuclides deposited in the second year of continuous exposure (i.e., the difference between the 2-yr committed dose equivalent and the first-year committed dose equivalent).

The general expression for calculating the annual dose to an organ of reference during any year after the start of continuous exposure can be expressed as:

$$A_t = R_t^* + \sum_{i=1}^{t-1} \left[R_{i,(t-i+1)} - R_{i,(t-i)}\right] \tag{F.9}$$

where:

A_t • the annual dose during the year t from all exposure pathways to the organ of reference, mrem

R_t^* = the radiation dose equivalent in year t to the organ of reference from all internal and external exposure pathways from intake and exposure in the year t, mrem.

The summation term represents the dose equivalent delivered to the organ of reference in year t from radionuclides deposited in the organ from intake in all previous years since the start of continuous exposure.

The summation term in Equation F.9 is valid only for integer values of t greater than 1. For values of t equal to or less than 1, the subscripts define a non-real case, and the summation term is set equal to zero.

The annual dose, A_t, to the organ of reference is calculated for each value of t from 1 to 50, and the maximum annual dose is determined by inspection. The radionuclide inventories are adjusted for radionuclide decay and daughter-product buildup during the calculation.

To determine acceptable residual contamination levels for the research and test reactors a model of a reference room is required. Modifications to the models for the airborne exposure pathway are also required since no further airborne release occurs after decommissioning. The following sections contain discussions of these topics.

F.3.1 Model of a Reference Room at a Research or Test Reactor

To determine the acceptable contamination levels in a decommissioned research or test reactor, the dimensions of a reference room must be defined. The results of a sensitivity study of the relationship between room volume, surface radionuclide levels, and calculated dose rate are shown in Figure F.3-1. Dose rates are calculated at a point in air (in a tissue equivalent 50 mm thick) 1 m above the floor of a room with 1 Ci/m^2 of ^{60}Co uniformly distributed on the room surfaces. The curves in Figure F.3-1 graphically illustrate the difference between an evenly contaminated room and one whose walls and ceiling contain 50% and 10%, respectively, of the floor contamination level. The straight line represents the dose rate from an infinite flat plane of evenly distributed ^{60}Co contamination with a concentration of 1 Ci/m^2. It can be seen that the dose rate for a large, non-uniformly contaminated room may exceed that of a uniformly contaminated infinite plane.

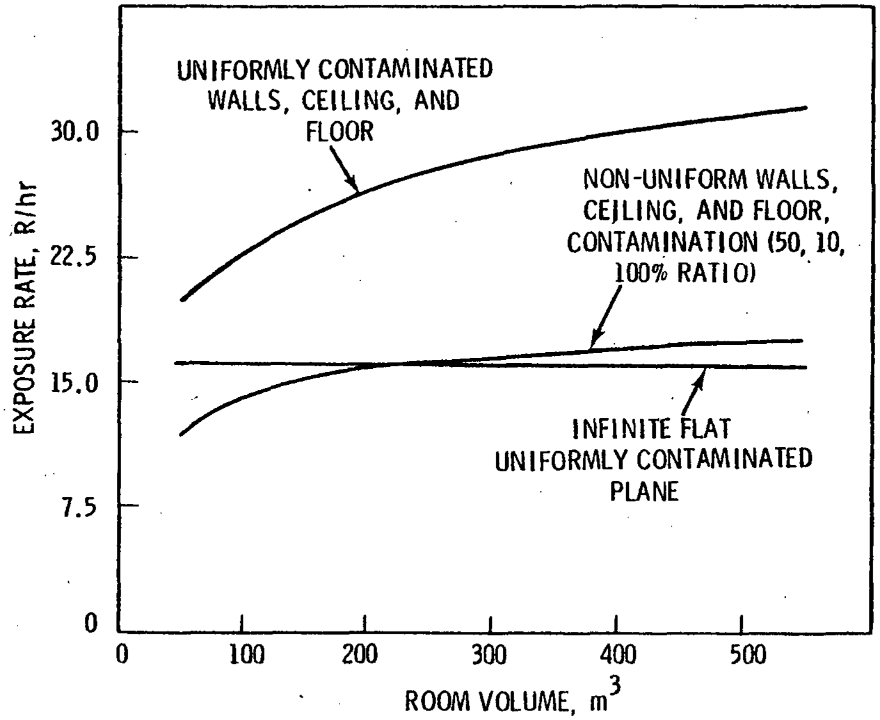

FIGURE F.3-1. Dose Rate as a Function of Room Volume for a ^{60}Co Deposition of 1 Ci/m^2

To be conservative, a room with uniformly contaminated walls, floor, and ceiling is used in this study. It has a floor area of 154 m^2 and 3-m-high walls, for a total volume of about 460 m^3. External radiation dose equivalent factors for this reference room are calculated using the ISOSHLD computer code.(13,14) Annual doses are determined for a person working in this reference room at a decommissioned research or test reactor for 40 hr/week for 50 years.

F.3.2 Airborne Radioactivity Concentrations for Annual Dose Calculations

Resuspended surface or ground contamination is assumed to be the only source of airborne radionuclides following decommissioning of the reference research and test reactors. Airborne radionuclide concentrations after decommissioning of research and test reactors are calculated using a resuspension analysis. The average airborne concentration is the product of a resuspension factor and the surface contamination level as shown in Equation F.10.

$$\bar{\chi} = S_f \, S_A \quad \text{(F.10)}$$

where:

$\bar{\chi}$ • average airborne concentration, Ci/m^3

S_f • resuspension factor, m^{-1}

S_A • surface radioactivity deposition, Ci/m^2.

Inside the decommissioned research or test reactors, airborne concentrations are calculated using a constant resuspension factor of $5 \times 10^{-6}\ m^{-1}$. This value is one-tenth of the value suggested by the IAEA for surfaces with removable contamination.(17) This lower value is assumed since unrestricted release of nuclear facilities should not be contemplated with readily removable contamination remaining on accessible surfaces. Actual facility surface resuspension factors may be considerably lower than this value.

Airborne radionuclide concentrations on the reference site are calculated using a time dependent suspension factor to account for the environmental "aging" of radionuclides. The relationship for this factor is given in Equation F.11.(18)

$$S_f = \left(10^{-4}\, e^{-\lambda\sqrt{t}}\right) + 10^{-9} \quad \text{(F.11)}$$

where:

S_f • resuspension factor, m^{-1}

10^{-4} • resuspension factor at time t = 0, m^{-1}

$e^{-\lambda}$ • effective decay constant controlling the availability of material for resuspension, 0.15 $day^{-1/2}$

t • time after deposition, days

10^{-9} • resuspension factor after 17 years, m^{-1}.

The second term in Equation F.11 (10^{-9}) is added based on the assumption that there is no further measurable decrease in the resuspension factor process after 17 years, which is the longest period for which data are available. This

is deemed appropriate since this model is derived empirically to simulate experimental measurements and contains no fundamental understanding of the resuspension process.[18] The time dependence of the site resuspension factor is illustrated in Figure F.3-2.

After decommissioning, unrestricted use of the site will allow activities such as unlimited farming. Therefore, the amount of surface contamination available for resuspension must be defined. For the calculation of the maximum annual doses, it is assumed that the site surface is plowed after decommissioning to a depth of 0.15 m, and the original surface contamination is assumed to be uniformly mixed into that volume of soil. This reduces the amount of material available for resuspension.

F.4 CALCULATED MAXIMUM ANNUAL DOSES AT THE DECOMMISSIONED R&T REACTORS

This section contains the assumptions, specifies the radionuclide inventories, and presents listings of the resultant calculated maximum annual doses.

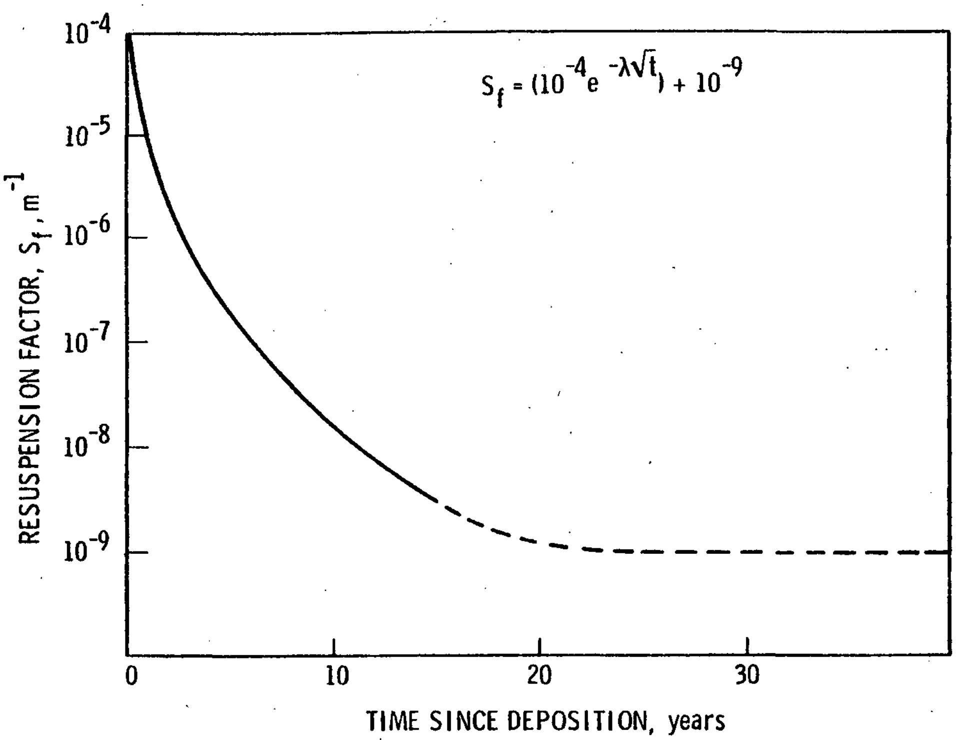

FIGURE F.3-2. Time-Dependent Surface Resuspension Factor for the Site

These maximum annual doses are used in determining the acceptable residual contamination levels for the decommissioned reference research and test reactors that are presented in Section 9. Doses are calculated and reported for the maximum-exposed individual working in the research or test reactor, or residing on the sites after decommissioning.

F.4.1 Assumptions for Calculating Maximum Annual Dose

The following assumptions are used in calculating the maximum annual doses at the decommissioned research or test reactor.

- The radionuclide inventories are undisturbed, except for radioactive decay after plant shutdown. Doses are calculated for the mixtures at shutdown, and for decay periods of 10, 30, 50, and 100 years.
- The maximum-exposed individual is considered to have realistically maximized exposure pathway parameters.
- the maximum-exposed individual working in the facility receives continuous exposure for 40 hr/week, 50 weeks/year.
- The maximum-exposed individual residing on the site is continuously exposed for 168 hr/week, 52 weeks/year.
- The airborne radionuclide concentrations are calculated using a resuspension factor of $5 \times 10^{-6}\ m^{-1}$ inside the facility, and the time dependent factor from Equation F.11 on the site.

F.4.2 Radionuclide Inventories for Calculating Maximum Annual Doses

The radioactive contamination present at the reference research and test reactors is discussed in Appendix E. To demonstrate the acceptable contamination level analysis for example mixtures of radionuclides at the research and test reactors, maximum annual doses are first calculated. For the research reactor, these dose calculations are made using the fractional activity found in neutron-activated stainless steel, normalized to a surface contamination level of 1 $\mu Ci/m^2$. The inventory for neutron-activated stainless steel is given in Table E.1-5 of Appendix E. No radiation dose calculations are made for the research reactor site since planned releases during routine operations are very small and no site contamination results. For the test reactor, the

fractional surface contamination present in the hot cells, normalized to 1 $\mu Ci/m^2$, is used in the maximum annual dose calculations. The radionuclide inventory in the hot cells is given in Table E.2-10. Since limited areas of the test reactor site are contaminated, maximum annual doses are calculated for the site using the inventory shown in Table E.2-11.

F.4.3 Maximum Annual Dose Listings

Maximum annual doses to the maximum-exposed individual using the decommissioned research or test reactor facilities, or their sites, are listed in this section. These doses are calculated using the models, parameters, and assumptions presented earlier in this appendix. Dose pathways of concern in the decommissioned research or test reactor facilities are inhalation of resuspended surface contamination and external exposure from surface contamination. The calculated maximum annual doses to an individual working 40 hr/week, 50 week/year for 50 years, are listed in Tables F.4-1 and F.4-2 for the research and test reactors, respectively. The doses are shown for various times after plant shutdown to account for decommissioning of the facilities after specific periods of radioactive decay.

Dose pathways of concern on the decommissioned reference test reactor site are inhalation of resuspended ground contamination, external exposure to ground contamination, and ingestion of food products raised on the site. The calculated maximum annual doses to an individual residing on the reference site 24 hr/day, 52 weeks/year, are listed in Table F.4-3. Again, these doses are shown for various times after plant shutdown to account for decommissioning the reactors after specific periods of radioactive decay. No calculations are made for the research reactor since the site is assumed to remain free of contamination during routine operation of the reactor.

The doses shown in Tables F.4-1 through F.4-3 are used to determine the acceptable residual contamination levels for unrestricted release of the decommissioned research and test reactors. A more detailed discussion of the method used to determine the acceptable residual contamination levels is presented in Section 9.

TABLE F.4-1. Estimated Maximum Annual Dose to the Maximum-Exposed Individual Working in the Decommissioned Research Reactor at Various Times After Reactor Shutdown(a)

Continuous Exposure in the Facility at Shutdown

Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
Total Body/Year 1	^{54}Mn	9.5×10^{-7}	2.0×10^{-3}	2.0×10^{-3}
	^{58}Co	1.8×10^{-6}	1.5×10^{-2}	1.5×10^{-2}
	^{60}Co	2.2×10^{-5}	1.3×10^{-1}	1.3×10^{-1}
	Totals	2.6×10^{-5}	1.5×10^{-1}	1.5×10^{-1}
Bone/Year 1	^{54}Mn	---(d)	2.0×10^{-3}	2.0×10^{-3}
	^{58}Co	---	1.5×10^{-2}	1.5×10^{-2}
	^{60}Co	---	1.3×10^{-1}	1.3×10^{-1}
	Totals	---	1.5×10^{-1}	1.5×10^{-1}
Lungs/Year 1	^{54}Mn	2.2×10^{-5}	2.0×10^{-3}	2.8×10^{-3}
	^{58}Co	2.2×10^{-4}	1.5×10^{-4}	3.7×10^{-4}
	^{60}Co	5.0×10^{-3}	1.3×10^{-1}	1.3×10^{-1}
	Totals	5.3×10^{-3}	1.5×10^{-1}	1.5×10^{-1}
Thyroid/Year 1	^{51}Cr	1.6×10^{-7}	1.1×10^{-3}	1.1×10^{-3}
	^{54}Mn	---	2.0×10^{-3}	2.0×10^{-3}
	^{58}Co	---	1.5×10^{-2}	1.5×10^{-2}
	^{60}Co	---	1.3×10^{-1}	1.3×10^{-1}
	Totals	1.6×10^{-7}	1.5×10^{-1}	1.5×10^{-1}

Continuous Exposure in the Facility 10 Years After Shutdown

Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
Total Body/Year 11	^{54}Mn	2.9×10^{-10}	6.1×10^{-7}	6.1×10^{-7}
	^{60}Co	6.0×10^{-6}	3.4×10^{-2}	3.4×10^{-2}
	Totals	6.0×10^{-6}	3.4×10^{-2}	3.4×10^{-2}
Bone/Year 11	^{54}Mn	---(d)	6.1×10^{-7}	6.1×10^{-7}
	^{60}Co	---	3.4×10^{-2}	3.4×10^{-2}
	Totals	---	3.4×10^{-2}	3.4×10^{-2}
Lungs/Year 11	^{54}Mn	6.5×10^{-9}	6.1×10^{-7}	6.1×10^{-7}
	^{58}Co	3.0×10^{-7}	---	3.0×10^{-7}
	^{60}Co	1.3×10^{-3}	3.4×10^{-2}	3.6×10^{-2}
	Totals	1.3×10^{-3}	3.4×10^{-2}	3.6×10^{-2}
Thyroid/Year 11	^{54}Mn	---	6.1×10^{-7}	6.1×10^{-7}
	^{60}Co	---	3.4×10^{-2}	3.4×10^{-2}
	Totals	---	3.4×10^{-2}	3.4×10^{-2}

Continuous Exposure in the Facility 30 Years After Shutdown

Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
Total Body/Year 31	^{60}Co	4.3×10^{-7}	2.5×10^{-3}	2.5×10^{-3}
	Totals	4.3×10^{-7}	2.5×10^{-3}	2.5×10^{-3}
Bone/Year 31	^{55}Fe	2.9×10^{-10}	---(d)	2.9×10^{-10}
	^{60}Co	---	2.5×10^{-3}	2.5×10^{-3}
	Totals	2.9×10^{-10}	2.5×10^{-3}	2.5×10^{-3}
Lungs/Year 31	^{55}Fe	1.8×10^{-9}	---	1.8×10^{-9}
	^{60}Co	9.6×10^{-5}	2.5×10^{-3}	2.5×10^{-3}
	Totals	9.6×10^{-5}	2.5×10^{-3}	2.6×10^{-3}
Thyroid/Year 31	^{60}Co	---	2.5×10^{-3}	2.5×10^{-3}
	Totals	---	2.5×10^{-3}	2.5×10^{-3}

Continuous Exposure in the Facility 50 Years After Shutdown

Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
Total Body/Year 51	^{60}Co	3.1×10^{-8}	1.7×10^{-4}	1.7×10^{-4}
	Totals	3.1×10^{-8}	1.7×10^{-4}	1.7×10^{-4}
Bone/Year 51	^{55}Fe	1.7×10^{-12}	---(d)	1.7×10^{-12}
	^{60}Co	---	1.7×10^{-4}	1.7×10^{-4}
	Totals	1.7×10^{-12}	1.7×10^{-4}	1.7×10^{-4}
Lungs/Year 51	^{55}Fe	1.0×10^{-10}	---	1.0×10^{-10}
	^{60}Co	6.9×10^{-6}	1.7×10^{-4}	1.8×10^{-4}
	Totals	6.9×10^{-6}	1.7×10^{-4}	1.8×10^{-4}
Thyroid/Year 31	^{60}Co	---	1.7×10^{-4}	1.7×10^{-4}
	Totals	---	1.7×10^{-4}	1.7×10^{-4}

Continuous Exposure in the Facility 100 Years After Shutdown

Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
Total Body/Year 101	^{60}Co	4.2×10^{-10}	2.4×10^{-7}	2.4×10^{-7}
	Totals	4.2×10^{-10}	2.4×10^{-7}	2.4×10^{-7}
Bone/Year 101	^{60}Co	---(d)	2.4×10^{-7}	2.4×10^{-7}
	Totals	---	2.4×10^{-7}	2.4×10^{-7}
Lungs/Year 101	^{60}Co	9.5×10^{-9}	2.4×10^{-7}	2.5×10^{-7}
	Totals	9.5×10^{-9}	2.4×10^{-7}	2.5×10^{-7}
Thyroid/Year 31	^{60}Co	---	2.4×10^{-7}	2.4×10^{-7}
	Totals	---	2.4×10^{-7}	2.4×10^{-7}

(a) Based on the radionuclide inventory for neutron-activated stainless steel at the reference test reactor, shown in Table E.1-5, at an assumed contamination level of 1 μCi/m².
(b) The year in which the maximum annual dose occurs after the start of continuous exposure.
(c) The dose from all radionuclides in the inventory is calculated, but only select contributors to the total dose to any exposure pathway are reported in this table.
(d) Indicates a contribution to the total dose of less than 1%.

TABLE F.4-1. Estimated Maximum Annual Dose to the Maximum-Exposed Individual Working in the Decommissioned Research Reactor at Various Times After Reactor Shutdown

TABLE F.4-2. Estimated Maximum Annual Dose to the Maximum-Exposed Individual Working in the Decommissioned Test Reactor at Various Times After Reactor Shutdown[a]

Continuous Exposure in the Facility at Shutdown

Organ Reference/ Maximum Year[b]	Radionuclide[c]	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
l Body/Year 30	^{90}Sr+D[d]	4.9×10^{-2}	1.2×10^{-3}	5.0×10^{-2}
	^{134}Cs	1.8×10^{-9}	6.2×10^{-7}	6.2×10^{-7}
	^{147}Pm	1.6×10^{-8}	---[e]	1.5×10^{-8}
	^{234}Th	2.0×10^{-7}	---	2.0×10^{-7}
	^{244}Cm	8.8×10^{-3}	---	8.8×10^{-3}
	^{240}Pu	2.9×10^{-5}	---	2.9×10^{-5}
	Totals	5.8×10^{-2}	1.3×10^{-3}	6.0×10^{-2}
/Year 28	^{90}Sr+D	2.0×10^{-1}	1.3×10^{-3}	2.0×10^{-1}
	^{134}Cs	2.6×10^{-9}	1.2×10^{-6}	1.2×10^{-6}
	^{147}Pm	1.1×10^{-9}	1.2×10^{-6}	1.2×10^{-6}
	^{234}Th	6.9×10^{-6}	---	---
	^{244}Cm	3.2×10^{-1}	---	3.2×10^{-1}
	^{240}Pu	4.8×10^{-4}	---	4.8×10^{-4}
	Totals	5.2×10^{-1}	1.3×10^{-3}	5.2×10^{-1}
s/Year 6	^{154}Eu	2.8×10^{-2}	---	2.8×10^{-2}
	^{90}Sr+D	8.0×10^{-4}	2.3×10^{-6}	8.9×10^{-4}
	^{134}Cs	1.9×10^{-6}	2.0×10^{-3}	2.9×10^{-3}
	^{144}Ce	4.9×10^{-4}	1.3×10^{-5}	5.0×10^{-4}
	^{147}Pm	5.4×10^{-4}	---	5.4×10^{-4}
	^{234}Th	2.3×10^{-5}	---	2.3×10^{-5}
	^{244}Cm	2.4×10^{-7}	---	2.4×10^{-7}
	^{240}Pu	8.9×10^{-5}	---	8.9×10^{-5}
	Totals	2.7×10^{-1}	4.3×10^{-6}	2.7×10^{-1}
oid/Year 2	^{90}Sr+D	---	2.5×10^{-3}	2.5×10^{-3}
	^{134}Cs	---	7.6×10^{-3}	7.6×10^{-3}
	^{144}Ce	---	4.8×10^{-4}	4.8×10^{-4}
	Totals	---	1.1×10^{-2}	1.1×10^{-2}

Continuous Exposure in the Facility 10 Years After Shutdown

Organ of Reference/ Maximum Year[b]	Radionuclide[c]	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
Total Body/Year 39	^{90}Sr+D[d]	4.0×10^{-2}	1.0×10^{-3}	4.1×10^{-2}
	^{134}Cs	8.9×10^{-11}	3.0×10^{-8}	3.0×10^{-8}
	^{234}Th	2.0×10^{-7}	---[e]	2.0×10^{-7}
	^{244}Cm	6.3×10^{-3}	---	6.3×10^{-2}
	^{240}Pu	4.1×10^{-5}	---	4.1×10^{-5}
	Totals	4.6×10^{-2}	1.0×10^{-3}	4.7×10^{-2}
Bone/Year 27	^{90}Sr+D	1.6×10^{-1}	1.0×10^{-3}	1.6×10^{-1}
	^{134}Cs	1.2×10^{-10}	5.9×10^{-8}	5.9×10^{-8}
	^{147}Pm	1.0×10^{-7}	---	1.0×10^{-7}
	^{234}Th	6.9×10^{-6}	---	6.9×10^{-6}
	^{244}Cm	2.2×10^{-1}	---	2.2×10^{-1}
	^{240}Pu	8.3×10^{-4}	---	8.3×10^{-4}
	Totals	3.8×10^{-1}	1.1×10^{-3}	3.9×10^{-1}
Lungs/Year 16	^{154}Eu	1.3×10^{-2}	---	1.3×10^{-2}
	^{90}Sr+D	6.3×10^{-4}	1.7×10^{-6}	2.3×10^{-4}
	^{106}Ru	1.5×10^{-6}	---	1.5×10^{-6}
	^{134}Cs	6.4×10^{-8}	6.4×10^{-5}	4.0×10^{-5}
	^{147}Pm	4.0×10^{-5}	---	2.3×10^{-5}
	^{234}Th	2.3×10^{-5}	---	1.7×10^{-1}
	^{244}Cm	1.7×10^{-1}	---	1.7×10^{-1}
	^{240}Pu	2.9×10^{-4}	---	2.9×10^{-4}
	Totals	1.8×10^{-1}	1.9×10^{-3}	1.8×10^{-1}
Thyroid/Year 11	^{90}Sr+D	---	2.0×10^{-3}	2.0×10^{-3}
	^{134}Cs	---	3.7×10^{-4}	3.7×10^{-4}
	^{144}Ce	---	2.4×10^{-7}	1.6×10^{-7}
	Totals	---	2.4×10^{-3}	2.4×10^{-3}

Continuous Exposure in the Facility 30 Years After Shutdown

Organ of Reference/ Maximum Year[b]	Radionuclide[c]	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
Total Body/Year 59	^{90}Sr+D[d]	2.4×10^{-2}	6.3×10^{-4}	2.4×10^{-2}
	^{244}Cm	2.9×10^{-3}	---[e]	3.0×10^{-8}
	^{240}Pu	5.7×10^{-5}	---	5.7×10^{-5}
	Totals	2.7×10^{-2}	6.3×10^{-4}	2.8×10^{-2}
Bone/Year 58	^{90}Sr+D	9.8×10^{-2}	6.4×10^{-4}	9.8×10^{-2}
	^{234}Th	6.9×10^{-6}	---	6.9×10^{-6}
	^{244}Cm	1.0×10^{-1}	---	1.0×10^{-1}
	^{240}Pu	1.2×10^{-3}	---	1.2×10^{-3}
	Totals	2.0×10^{-1}	6.4×10^{-1}	2.0×10^{-1}
Lungs/Year 36	^{154}Eu	2.6×10^{-3}	---	2.6×10^{-3}
	^{90}Sr+D	3.9×10^{-4}	1.1×10^{-3}	1.5×10^{-3}
	^{147}Pm	2.1×10^{-7}	---	2.1×10^{-7}
	^{234}Th	2.3×10^{-5}	---	2.3×10^{-5}
	^{244}Cm	7.9×10^{-2}	---	7.9×10^{-2}
	^{240}Pu	5.3×10^{-4}	---	5.3×10^{-4}
	Totals	1.8×10^{-1}	1.9×10^{-3}	1.8×10^{-1}
Thyroid/Year 31	^{90}Sr+D	---	1.2×10^{-3}	1.2×10^{-3}
	^{134}Cs	---	4.5×10^{-7}	4.5×10^{-7}
	Totals	---	1.2×10^{-3}	1.2×10^{-3}

Continuous Exposure in the Facility 50 Years After Shutdown

Organ of Reference/ Maximum Year[b]	Radionuclide[c]	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
Total Body/Year 60	^{90}Sr+D[d]	1.5×10^{-2}	3.8×10^{-4}	1.5×10^{-2}
	^{234}Th	2.0×10^{-7}	---[e]	2.0×10^{-7}
	^{244}Cm	1.3×10^{-3}	---	1.3×10^{-3}
	Totals	1.6×10^{-2}	3.8×10^{-4}	1.7×10^{-2}
Bone/Year 79	^{90}Sr+D	6.0×10^{-2}	3.8×10^{-4}	6.0×10^{-2}
	^{234}Th	6.9×10^{-6}	---	6.9×10^{-6}
	^{244}Cm	4.8×10^{-2}	---	4.8×10^{-2}
	^{240}Pu	1.4×10^{-3}	---	1.4×10^{-3}
	Totals	1.1×10^{-1}	3.8×10^{-4}	1.1×10^{-1}
Lungs/Year 56	^{154}Eu	5.2×10^{-4}	---	5.2×10^{-4}
	^{90}Sr+D	2.4×10^{-4}	6.7×10^{-6}	9.1×10^{-4}
	^{234}Th	2.3×10^{-5}	---	2.3×10^{-5}
	^{244}Cm	3.7×10^{-2}	---	3.7×10^{-2}
	^{240}Pu	6.4×10^{-4}	---	6.4×10^{-4}
	Totals	3.8×10^{-2}	6.7×10^{-4}	3.9×10^{-2}
Thyroid/Year 51	^{90}Sr+D	---	7.6×10^{-4}	7.6×10^{-4}
	^{134}Cs	---	5.4×10^{-10}	5.4×10^{-10}
	Totals	---	7.6×10^{-4}	7.6×10^{-4}

Continuous Exposure in the Facility 100 Years After Shutdown

Organ of Reference/ Maximum Year[b]	Radionuclide[c]	Maximum Annual Dose (rem) via: Inhalation	External	All Pathways
Total Body/Year 130	^{90}Sr+D[d]	4.4×10^{-3}	1.1×10^{-4}	4.5×10^{-3}
	^{234}Th	2.0×10^{-7}	---[e]	2.0×10^{-7}
	^{244}Cm	2.0×10^{-4}	---	2.0×10^{-4}
	^{240}Pu	7.2×10^{-5}	---	7.2×10^{-5}
	Totals	4.7×10^{-3}	1.1×10^{-4}	4.8×10^{-3}
Bone/Year 28	^{90}Sr+D	1.8×10^{-2}	1.0×10^{-4}	1.8×10^{-2}
	^{234}Th	6.9×10^{-6}	---	6.9×10^{-6}
	^{244}Cm	6.9×10^{-3}	---	6.9×10^{-3}
	^{240}Pu	1.7×10^{-3}	---	1.7×10^{-3}
	Totals	2.7×10^{-2}	1.1×10^{-4}	2.7×10^{-2}
Lungs/Year 6	^{154}Eu	9.2×10^{-6}	---	9.2×10^{-6}
	^{90}Sr+D	7.0×10^{-5}	2.0×10^{-4}	2.7×10^{-4}
	^{234}Th	2.3×10^{-5}	---	2.3×10^{-5}
	^{244}Cm	5.4×10^{-3}	---	5.4×10^{-3}
	^{240}Pu	7.1×10^{-4}	---	7.1×10^{-4}
	Totals	6.2×10^{-3}	2.0×10^{-4}	6.4×10^{-3}
Thyroid/Year 2	^{90}Sr+D	---	2.2×10^{-4}	2.2×10^{-4}
	Totals	---	2.2×10^{-4}	2.2×10^{-4}

Based on the radionuclide inventory for surface contamination in a hot cell at the reference test reactor, shown in Table E.2-10, at an assumed contamination level of 1 μCi/m².
The year in which the maximum annual dose occurs after the start of continuous exposure.
The dose from all radionuclides in the inventory is calculated, but only select contributors to the total dose to any exposure pathway are reported in this table.
+D means plus daughter product radionclides.
Indicates a contribution to the total dose of less than 1%.

TABLE F.4-2. Estimated Maximum Annual Dose to the Maximum-Exposed Individual Working in the Decommissione Test Reactor at Various Times After Reactor Shutdown

TABLE F.4-3. Estimated Maximum Annual Dose to the Maximum-Exposed Individual Residing on the Decommissioned Test Reactor Site at Various Times After Shutdown[(a)]

Continuous Exposure on the Site at Shutdown

Organ of Reference/ Maximum Year[(b)]	Radionuclide[(c)]	Maximum Annual Dose (rem) via: Ingestion	Inhalation	External	All Pathways
Total Body/Year 1	^{60}Co	5.0×10^{-3}	9.6×10^{-7}	1.9×10^{0}	1.9×10^{0}
	$^{90}Sr+D$[(d)]	5.0×10^{-2}	9.3×10^{-7}	1.9×10^{-4}	5.0×10^{-2}
	^{134}Cs	2.2×10^{-3}	8.6×10^{-7}	1.2×10^{-1}	1.2×10^{-1}
	$^{137}Cs+D$	5.0×10^{-3}	2.0×10^{-6}	1.9×10^{-1}	2.0×10^{-1}
	Totals	6.2×10^{-2}	4.8×10^{-6}	2.1×10^{0}	2.2×10^{0}
Bone/Year 31	^{60}Co	---[(e)]	---	3.5×10^{-2}	3.5×10^{-2}
	$^{90}Sr+D$	4.5×10^{0}	1.3×10^{-5}	9.2×10^{-5}	4.5×10^{0}
	$^{137}Cs+D$	4.8×10^{-3}	2.1×10^{-7}	9.7×10^{-2}	1.0×10^{-1}
	Totals	4.5×10^{0}	1.3×10^{-5}	1.3×10^{-1}	4.7×10^{0}
Lungs/Year 1	^{60}Co	---	2.2×10^{-4}	1.8×10^{0}	1.8×10^{0}
	^{134}Cs	2.5×10^{-4}	2.2×10^{-4}	1.8×10^{0}	1.8×10^{0}
	^{137}Cs	8.0×10^{-4}	2.6×10^{-7}	1.9×10^{-1}	1.2×10^{-1}
	Totals	1.0×10^{-3}	2.2×10^{-4}	2.2×10^{0}	2.2×10^{0}
Thyroid/Year 1	^{60}Co	---	5.6×10^{-6}	1.8×10^{0}	1.8×10^{0}
	^{134}Cs	---	1.0×10^{-8}	1.2×10^{-1}	1.2×10^{-1}
	$^{137}Cs+D$	---	3.8×10^{-8}	1.9×10^{-1}	1.9×10^{-1}
	Totals	---	5.9×10^{-6}	2.2×10^{0}	2.2×10^{0}

(a) Based on the radionuclide inventory for contaminated soil at the reference test reactor, shown in Table E.2-11.
(b) The year in which the maximum annual dose occurs after the start of continuous exposure.
(c) The dose from all radionuclides in the inventory is calculated, but only select contributers to the total dose to any exposure pathway are reported in this table.
(d) +D means plus daughter product radionclides.
(e) Indicates a contribution to the total dose of less than 1%.

Continuous Exposure on the Site 10 Years After Shutdown

Organ of Reference/ Maximum Year[(b)]	Radionuclide[(c)]	Maximum Annual Dose (rem) via: Ingestion	Inhalation	External	All Pathways
Total Body/Year 37	^{60}Co	4.6×10^{-5}	7.6×10^{-10}	1.6×10^{-2}	1.6×10^{-2}
	$^{90}Sr+D$[(d)]	8.7×10^{-1}	1.3×10^{-6}	8.0×10^{-5}	8.7×10^{-1}
	$^{137}Cs+D$	3.7×10^{-3}	9.8×10^{-8}	8.4×10^{-2}	8.8×10^{-2}
	Totals	8.7×10^{-1}	4.8×10^{-6}	2.1×10^{0}	2.2×10^{0}
Bone/Year 42	^{60}Co	---[(e)]	---	8.3×10^{-3}	8.3×10^{-3}
	$^{90}Sr+D$	3.5×10^{0}	5.4×10^{-6}	7.1×10^{-5}	3.5×10^{0}
	$^{137}Cs+D$	3.7×10^{-3}	1.7×10^{-7}	7.5×10^{-2}	7.9×10^{-2}
	Totals	3.6×10^{0}	5.6×10^{-6}	8.3×10^{-2}	3.6×10^{0}
Lungs/Year 11	^{60}Co	---	5.8×10^{-6}	4.9×10^{-1}	4.9×10^{-1}
	^{134}Cs	8.5×10^{-6}	9.2×10^{-10}	4.1×10^{-3}	4.1×10^{-3}
	^{137}Cs	6.3×10^{-4}	6.5×10^{-8}	1.5×10^{-1}	1.5×10^{-1}
	Totals	6.4×10^{-4}	5.9×10^{-6}	6.5×10^{-1}	6.5×10^{-1}
Thyroid/Year 11	^{60}Co	---	---	4.9×10^{-1}	4.9×10^{1}
	^{134}Cs	---	---	4.1×10^{-3}	4.1×10^{-3}
	$^{137}Cs+D$	---	---	1.5×10^{-1}	1.5×10^{-1}
	Totals	---	---	6.5×10^{-1}	6.5×10^{-1}

Continuous Exposure on the Site 30 Years After Shutdown

Organ of Reference/ Maximum Year[(b)]	Radionuclide[(c)]	Maximum Annual Dose (rem) via: Ingestion	Inhalation	External	All Pathways
Total Body/Year 59	^{60}Co	2.5×10^{-6}	4.2×10^{-11}	8.8×10^{-4}	8.8×10^{-4}
	$^{90}Sr+D$[(d)]	5.4×10^{-1}	8.0×10^{-7}	4.6×10^{-5}	5.4×10^{-1}
	$^{137}Cs+D$	2.2×10^{-3}	5.9×10^{-8}	5.1×10^{-2}	5.3×10^{-2}
	Totals	5.4×10^{-1}	4.8×10^{-6}	2.1×10^{0}	2.2×10^{0}
Bone/Year 62	^{60}Co	---[(e)]	---	5.9×10^{-4}	5.9×10^{-4}
	$^{90}Sr+D$	2.2×10^{0}	3.2×10^{-6}	4.3×10^{-5}	2.2×10^{0}
	$^{137}Cs+D$	2.3×10^{-3}	1.0×10^{-7}	4.7×10^{-2}	4.9×10^{-2}
	Totals	2.2×10^{0}	3.4×10^{-6}	4.8×10^{-2}	2.2×10^{0}
Lungs/Year 31	^{60}Co	---	3.3×10^{-7}	3.5×10^{-2}	3.5×10^{-2}
	^{134}Cs	---	1.4×10^{-8}	9.5×10^{-5}	9.5×10^{-5}
	^{137}Cs	4.0×10^{-4}	3.3×10^{-8}	9.7×10^{-2}	9.7×10^{-2}
	Totals	4.0×10^{-4}	3.8×10^{-7}	1.3×10^{-1}	1.3×10^{-1}
Thyroid/Year 31	^{60}Co	---	---	3.5×10^{-2}	3.5×10^{-2}
	$^{137}Cs+D$	---	---	9.7×10^{-2}	9.7×10^{-2}
	Totals	---	---	1.3×10^{-1}	1.3×10^{-1}

Continuous Exposure on the Site 50 Years After Shutdown

Organ of Reference/ Maximum Year[(b)]	Radionuclide[(c)]	Maximum Annual Dose (rem) via: Ingestion	Inhalation	External	All Pathways
Total Body/Year 79	^{60}Co	1.8×10^{-7}	3.0×10^{-12}	6.3×10^{-5}	6.3×10^{-5}
	$^{90}Sr+D$[(d)]	3.3×10^{-1}	4.9×10^{-7}	2.8×10^{-5}	3.3×10^{-1}
	$^{137}Cs+D$	1.4×10^{-3}	3.1×10^{-8}	3.2×10^{-2}	3.3×10^{-2}
	Totals	3.3×10^{-1}	5.2×10^{-7}	3.2×10^{-2}	3.6×10^{-1}
Bone/Year 82	$^{90}Sr+D$	1.3×10^{0}	2.0×10^{-6}	4.1×10^{-7}	1.3×10^{0}
	$^{137}Cs+D$	1.5×10^{-3}	6.6×10^{-8}	3.0×10^{-2}	3.2×10^{-2}
	Totals	1.4×10^{0}	2.1×10^{-6}	3.0×10^{-2}	1.4×10^{0}
Lungs/Year 51	^{60}Co	---[(e)]	2.4×10^{-8}	2.5×10^{-3}	2.5×10^{-3}
	$^{90}Sr+D$	---	8.7×10^{-9}	5.6×10^{-5}	5.6×10^{-5}
	^{137}Cs	2.5×10^{-4}	2.1×10^{-8}	6.1×10^{-2}	6.1×10^{-2}
	Totals	2.5×10^{-4}	5.3×10^{-8}	6.4×10^{-2}	6.4×10^{-2}
Thyroid/Year 51	^{60}Co	---	---	2.5×10^{-3}	2.5×10^{-3}
	$^{137}Cs+D$	---	---	6.1×10^{-2}	6.1×10^{-2}
	Totals	---	---	6.4×10^{-2}	6.4×10^{-2}

Continuous Exposure on the Site 100 Years After Shutdown

Organ of Reference/ Maximum Year[(b)]	Radionuclide[(c)]	Maximum Annual Dose (rem) via: Ingestion	Inhalation	External	All Pathways
Total Body/Year 129	$^{90}Sr+D$[(d)]	9.8×10^{-2}	1.5×10^{-7}	8.5×10^{-6}	9.8×10^{-2}
	$^{137}Cs+D$	4.5×10^{-4}	1.2×10^{-8}	1.0×10^{-2}	1.0×10^{-2}
	Totals	9.9×10^{-2}	1.5×10^{-7}	1.0×10^{-2}	1.1×10^{-1}
Bone/Year 82	^{63}Ni	9.8×10^{-3}	---[(e)]	---	9.8×10^{-3}
	$^{90}Sr+D$	4.0×10^{-1}	5.9×10^{-7}	7.9×10^{-6}	4.0×10^{-1}
	$^{137}Cs+D$	4.6×10^{-4}	2.1×10^{-8}	9.5×10^{-3}	9.5×10^{-3}
	Totals	4.1×10^{-1}	6.2×10^{-7}	9.5×10^{-3}	4.2×10^{-1}
Lungs/Year 101	$^{90}Sr+D$	---	2.6×10^{-9}	1.7×10^{-5}	1.7×10^{-5}
	^{137}Cs	8.0×10^{-5}	6.5×10^{-9}	1.9×10^{-2}	1.9×10^{-2}
	Totals	8.0×10^{-5}	9.1×10^{-9}	1.9×10^{-2}	1.9×10^{-2}
Thyroid/Year 101	$^{137}Cs+D$	---	---	1.9×10^{-2}	1.9×10^{-2}
	Totals	---	---	1.9×10^{-2}	1.9×10^{-2}

TABLE F.4-3. Estimated Maximum Annual Dose to the Maximum-Exposed Individual Residing on the Decommissioned Test Reactor Site at Various Times After Shutdown

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APPENDIX G

DECOMMISSIONING METHODS

Methods and equipment that can be used to accomplish various decommissioning activities at the reference research and test (R&T) facilities are discussed in this appendix. These activities, in order of their presentation, are:

- system draining
- equipment deactivation
- contamination control
- decontamination
- equipment disassembly
- radioactive waste packaging and shipping
- quality assurance
- environmental surveillance.

G.1 SYSTEM DRAINING

At final shutdown of the reference R&T reactors, significant volumes of radioactively contaminated water remain in the reactor systems. This contaminated water (including any subsequent flushes) must be drained, treated, and disposed of before the equipment and piping can be decontaminated, dismantled, and shipped offsite for disposal.

The contaminated water drained from the reactor systems must be treated to remove the contaminants before it is released to the environs. Most of the water in the reference R&T reactor systems is demineralized water with low concentrations of radionuclides. This water is cleaned up for disposal to surface streams by passing the water through ion-exchange demineralizers. The existing water cleanup demineralizers and/or portable demineralizers are used for this water cleanup operation. Water containing relatively large amounts of dissolved solids will exhaust the ion-exchange (IX) resins too rapidly to be cost-effective; therefore, these solutions are filtered first or concentrated by

evaporation. The evaporator condensate is passed through the ion-exchange columns and is disposed of to surface streams with the demineralized water. The concentrated evaporator bottoms and IX resins are solidified by concreting.

Sediment is anticipated to be present in some of the tanks (e.g., the hot retention tanks for the reference test reactor). This sediment or slurry, which is contaminated with radionuclides, is removed from the hot retention tanks during the decommissioning process.

A double-diaphragm pump is used for transferring the slurry to the liquid waste processing sytems in the Waste Handling Building. These pumps are air-powered, self-priming and can be used completely submerged.

G.2 EQUIPMENT DEACTIVATION

When equipment and systems are no longer needed during decommissioning, they are placed in a condition that provides maximum safety with minimum maintenance. When possible, equipment is left in a condition that permits startup or salvage at a later date. Deactivation and isolation techniques include closing and securing installed valves, installing blank flanges as required, and disconnecting electrical power and other utilities. Equipment deactivation procedures are coordinated with system decontamination and draining operations. In some instances, decontamination must be carried out before equipment deactivation, while in others the opposite approach may be necessary. A safety audit of all systems is performed to ensure that all flammable and other potentially hazardous materials are removed. All deactivated equipment and systems are tagged out.

The particular method used to deactivate each system or piece of equipment is identified during the planning and preparation phase. In general, all systems not necessary to prevent the spread of contamination are deactivated when empty. All equipment, valves, circuit breakers, etc., are tagged when deactivated. These tags identify the piece of equipment, its system, and its condition.

The first step in equipment deactivation is a safety audit of all pumps and pipes used for chemicals or radioactive materials to ensure that all

hazardous or corrosive materials have been removed. Electrical service is disconnected from all equipment not required for operation later in the decommissioning process.

Systems inside the reference R&T reactor facilities are deactivated by a variety of methods. Many piping systems are isolated using the installed valves, with handles or valve operators removed. Other pipes that have contained contaminated materials are blanked where flanges are readily accessible. Other systems are drained and left open to the atmosphere. Cranes and other electrical equipment not required later in decommissioning or subsequent demolition are disabled by disconnecting electrical service to prevent their unauthorized use.

G.3 CONTAMINATION CONTROL

Many decommissioning activities, particularly the cutting operations required for equipment disassembly, have the potential for generating significant amounts of airborne radioactive contamination. In addition, decommissioning can involve operations in areas with smearable radioactive contamination, and the movement of personnel, equipment, and materials in these areas can result in the further spread of radioactive contamination. To minimize the personnel hazard and the potential for widespread contamination of work areas, contamination control is required.

Radioactive contamination control can be divided into three basic approaches:

- local mitigation of contamination sources
- collection of contamination
- isolation of contaminated areas.

These approaches are discussed in the following subsections.

G.3.1 Local Mitigation of Contamination Sources

Mechanical or physical measures can be used to limit the spread of radioactive contamination. Two methods that have been successfully used are: 1) water sprays to reduce airborne dust dispersion, and 2) painting of contaminated surfaces to prevent smearing.

The wetting of dust with water or other liquids is one of the oldest methods of contamination control and can be very effective if properly used.[1] Water sprays are widely used to control fugitive dust emissions from construction sites. The spraying of water containing detergent (as a wetting agent) has been used in the nuclear industry to reduce dust concentrations in air.[2] To be effective, the liquid application must be designed to blanket the dust source completely and to wet the dust particles thoroughly. Various types, sizes, and patterns of spray nozzles are used, depending on the physical properties of the dust, type and size of the dust source, and the degree of control desired. Water sprays can be used in combination with other contamination control techniques, and are commonly used for dusty operations such as concrete removal.[3]

Nonflammable, strippable paint can be used to seal porous surfaces (e.g., concrete) to prevent penetration of contamination into the surfaces. Paint can also be used to seal smearable contamination already present on surfaces to prevent subsequent contamination spread.[3] Spraying is generally the easiest and quickest method of application. Painting is especially useful in high-traffic areas, where smearable contamination is likely to be picked up and spread around on shoe covers and equipment wheels.

G.3.2 Collection of Contamination

Collection of radioactive contamination before it can be dispersed (preferably as it is generated) reduces the need for cleanup subsequent to some decommissioning activities. Various collection methods can be used. Vacuum collection and portable ventilation systems are discussed below.

G.3.2.1 Vacuum Collection

Contaminated materials can be collected as they are generated, using vacuum systems. A dust shield with a vacuum attachment can be installed on the tool (e.g., concrete spaller or scrubber) being used.[3,4] As the contaminated dust is generated, it is drawn into the vacuum system and deposited in a collection drum. The outlet air is filtered (with roughing and HEPA filters) to prevent the collected contamination from being expelled.

Various designs for vacuum collection systems are possible, depending on the required operating characteristics. One such system, shown schematically in Figure G.3-1, is described in Reference 5. This system, originally designed for collection of contaminated soil, uses a standard 0.21-m^3 waste drum to collect the contaminated material. When the drum is filled, it is capped and sealed for disposal. A special, commercially available vacuum lid, employing a cyclone baffle arrangement to enhance dust settling, is modified to accept an inexpensive, disposable roughing filter. A HEPA filter and power/vacuum-blower unit, mounted on a steel pallet, complete the system. The system is reported to be capable of pulling up to ~28 m^3/min of air at 110-mm-Hg vacuum, and is estimated to cost less than $5000.

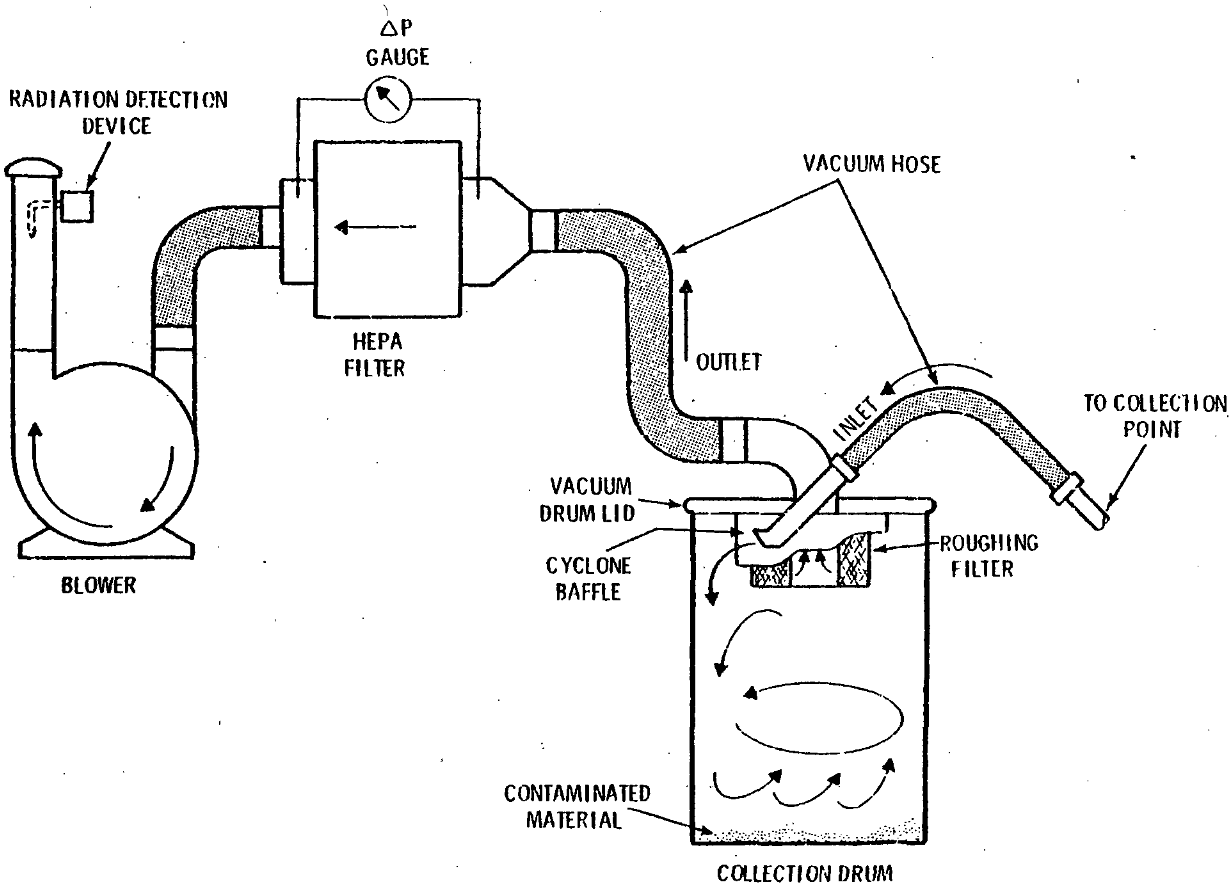

FIGURE G.3-1. Vacuum Collection System Schematic Diagram

G.3.2.2 Portable Ventilation Systems

Portable ventilation systems can be used to confine and collect airborne particulates generated during decommissioning operations.[3] General design information concerning such systems is discussed at length in Reference 1. Two portable ventilation systems, a work enclosure and a fume exhauster, are discussed here.

Portable Filtered Ventilation Enclosure. A typical portable filtered ventilation enclosure unit is illustrated in Figure G.3-2. A large squirrel-cage blower is coupled with a high-efficiency particulate air (HEPA) filter preceded by a glass-fiber roughing filter, all mounted on a wheeled cart. A flexible duct couples the cart unit to the enclosure unit that surrounds the work area and collects the materials being emitted. Roughing filters are installed at both the inlet and the outlet of the enclosure unit. The enclosure unit may have whatever shape best performs the required function at a particular location. A simple, rectangular open-faced box will suffice for many applications.

Radiation detection devices are used to monitor the buildup of radioactive material on the filters. A differential pressure gauge is installed across the

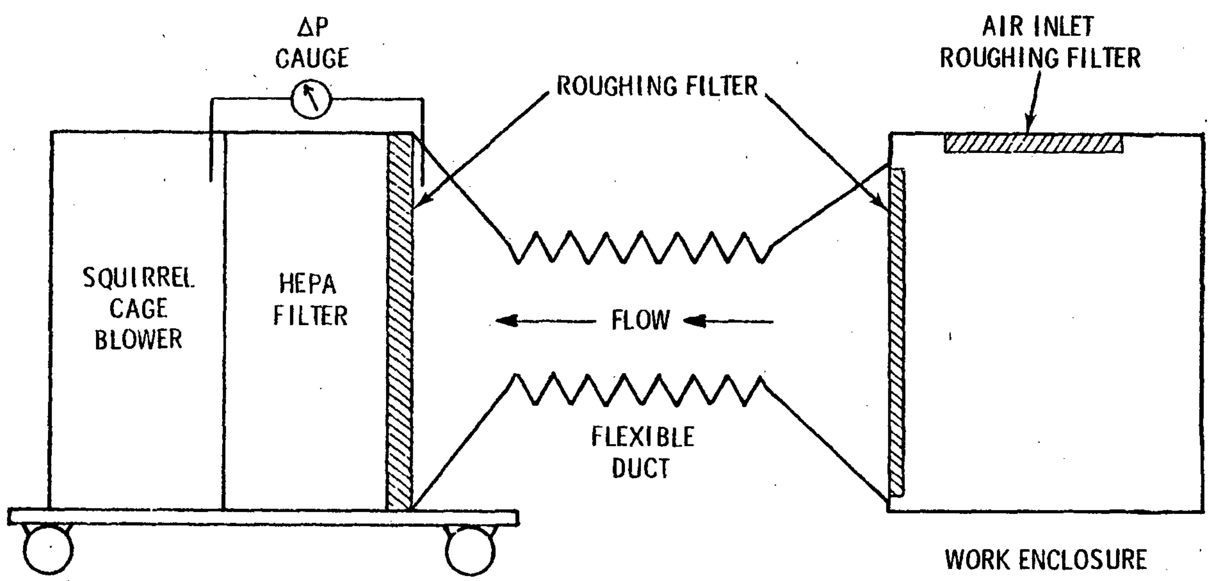

FIGURE G.3-2. Portable Filtered Ventilation Enclosure

HEPA filter to monitor the increasing pressure drop as particulates build up on the filter. Filters are changed when either the dose rate from the collected radioactive particles or the differential pressure across the HEPA filter reaches a predetermined level.

Portable Filtered Fume Exhauster. Another type of portable filtered ventilation system, a fume exhauster, is illustrated in Figure G.3-3. This system has an electrostatic precipitator coupled with a roughing filter, HEPA filter, air-handling motor, squirrel-cage blower, and one or two free-standing

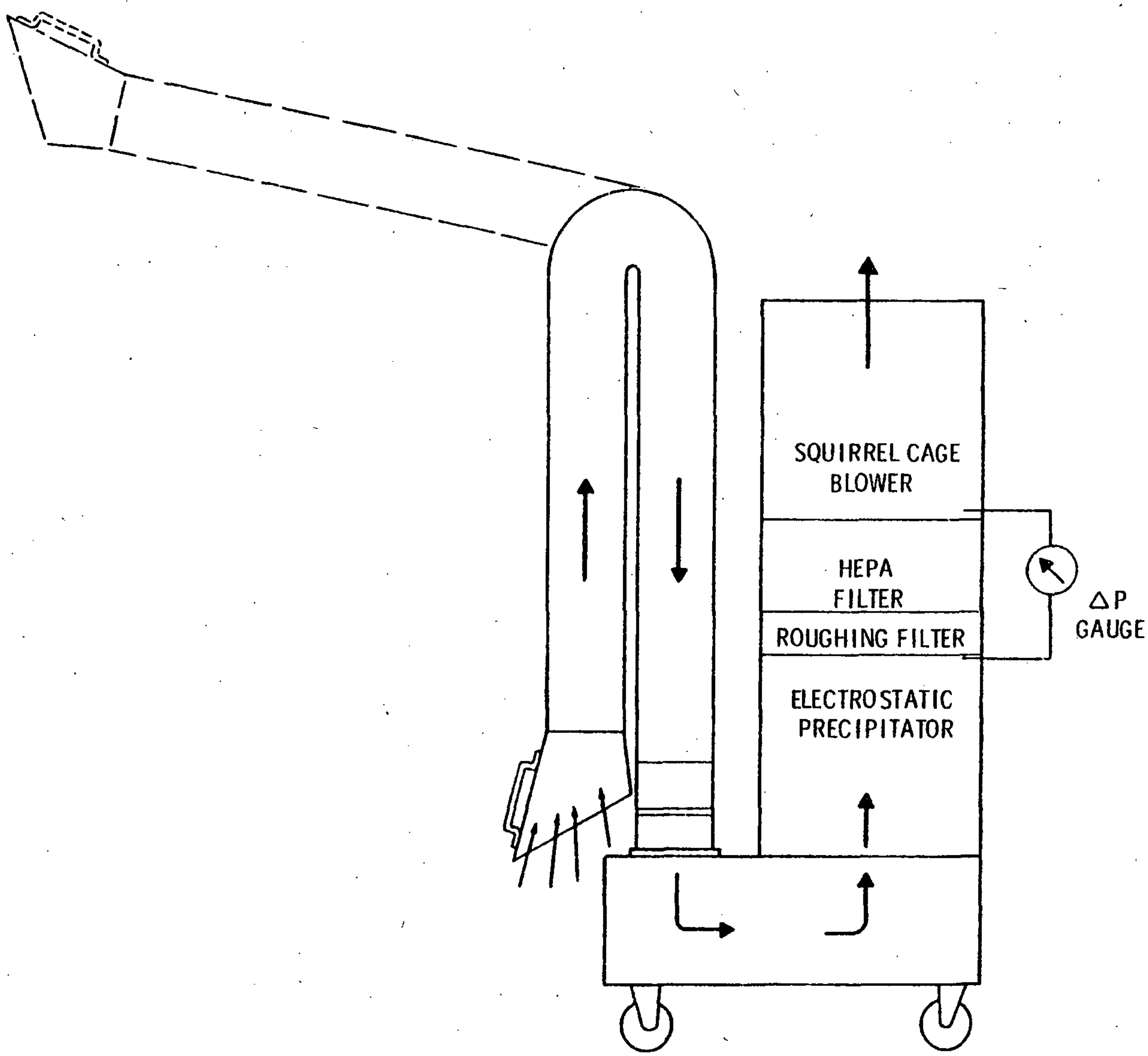

FIGURE G.3-3. Portable Filtered Fume Exhauster

intake ducts. The fume exhauster is used to collect radioactive and non-radioactive particulates at the point of generation. This high-volume ventilation system captures all types of particulate matter with efficiencies of greater than 97% for the electrostatic unit and at least 99.95% for the HEPA filter. The advantages of this unit are its portability, its ability to handle large volumes of particulate-laden air, and its generation of relatively small amounts of solid wastes (HEPA filters).

Buildup of radioactive materials on the precipitator and filters is monitored by radiation detection devices. To monitor the buildup of particulates on the filter, a differential pressure gauge is installed across the HEPA filter. The electrostatic precipitator is flushed and the filters are changed when either the dose rate from collected radioactive particles or the differential pressure across the HEPA filter reaches a predetermined level.

G.3.3 Isolation of Contaminated Areas

One method of controlling contamination is the use of barriers to isolate contaminated areas from those with lesser or no contamination. Isolation is an important tool during continuing care as well as during active decommissioning.

G.3.3.1 Isolation During Active Decommissioning

One type of barrier commonly used in the nuclear industry to isolate contaminated areas is a "greenhouse." A greenhouse is constructed by covering a framework, usually steel scaffolding or wood frame, with plastic sheeting and sealing all joints. Overlapping flaps of plastic are generally used for the door. The greenhouse is connected either to the plant ventilation system or to a portable system (see Section G.3.2.2), which prevents outward leakage of contamination by drawing a slight vacuum on the greenhouse.[3] Greenhouses can be semi-permanent, portable structures that can be moved from one location to another as needed, but are more often temporary confinement structures that are dismantled and discarded after each job.

In many cases, construction of a complete greenhouse is unnecessary. A simple plastic curtain partitioning off one section of a room may be all that is required to isolate a contaminated area. The type and degree of isolation

required depends on the equipment or structures involved, the associated level and mixture of radioactive contamination, and the decommissioning operation being performed.

G.3.3.2 Isolation During Continuing Care

Prior to the continuing care period, portions of the facility containing significant amounts of radioactive contamination that are not removed during active decommissioning must be isolated from the remainder of the facility. Potential pathways for the migration of contamination from these areas are blocked by the installation of physical barriers. Besides acting as contamination control barriers, the barriers are designed to discourage unauthorized personnel entry into contaminated areas. Structurally substantial barriers are used.

The preferred method of barrier installation is by welding. This method is used on piping, ventilation ductwork, and equipment penetrations. Piping is cut and plates are welded over the open ends. Rectangular plates are welded in inlet ventilation ductwork. All welds are inspected by appropriate techniques. A polysulfide adhesive is used to seal the perimeter of pipes and other penetrations through the concrete walls into contaminated areas. Care must be taken during the sealing procedures not to interfere with any fire-alarm sensors that may be present in the ducts.

The polysulfide adhesive is also used whenever seals must be formed on concrete surfaces or on metal surfaces where welding is not possible. This adhesive forms a strong, durable bond with these materials. After it has cured, the adhesive remains flexible, permitting differential thermal expansion of the two bonded surfaces without breaking the seal. This adhesive can be applied with a spatula or extrusion gun and will not run after application.

Metal doors are secured by welding them to the metal door frame. Spot welds are used where the intent is simply to make the door inoperative. Continuous welds are made or sealant is used where an airtight seal is required. Where the door frame meets the wall, it is sealed with the polysulfide adhesive.

Pressure equilization lines are installed between the outside environment and the interior of the Reactor Building, Reactor Containment Vessel, and other supporting buildings. These lines prevent pressure differentials (due to temperature or atmospheric pressure changes) from developing between the insides of the buildings and the outside atmosphere. The lines are equipped with replaceable absolute filters to prevent contamination from being entrained in the air flow out of the buildings.

G.4 DECONTAMINATION

Three basic methods can be used to remove radioactive materials from contaminated surfaces: 1) dissolution of the surface film containing the radionuclides, 2) physical cleaning of the surface, and 3) physical removal of the contaminated structural material. The various techniques used for each of these methods are discussed below.

G.4.1 Surface Film Dissolution

Two methods can be used to dissolve surface films containing radionuclide contamination: 1) chemical, and 2) electrochemical. The chemical method is generally used to decontaminate the internal and (in some cases) external surfaces of in-place piping and equipment. The electrochemical method (electropolishing) can sometimes be used in situ for internal decontamination, but it is more commonly applied to disassembled or segmented piping and equipment.

G.4.1.1 Chemical Decontamination

Several chemical decontamination methods are available for removing radioactive contamination from piping sytems, equipment, and external surfaces.

It should be remembered that in-situ chemical decontamination of the piping system of the reference R&T reactors being conceptually decommissioned in this study is not postulated. A history of no fuel failures, coupled with constant use of high-purity demineralized water, negates the need for chemical decontamination of piping systems in the reference R&T reactors. The in-situ chemical decontamination information in the following subsection is only presented here for completeness. The second subsection, Chemical Decontamination of External Surfaces, has direct application to the decommissioning of the reference R&T reactor facilities.

Chemical Decontamination of Piping Systems. Most contaminated piping systems are arranged so that a decontaminating solution can be recirculated through all or part of the system. However, other piping systems may be arranged such that recirculation of the decontaminating solution is not possible (e.g., contaminated drains). Chemical decontamination systems are designed to accommodate either of these situations, recirculating the solution through a pipe loop or accomplishing the decontamination in a single pass through the system. Selected chemical decontamination methods for piping systems, with their associated advantages and disadvantages, are presented in Table G.4-1. More detailed information on chemical decontamination is available in References 6 through 9.

TABLE G.4-1. Selected Chemical Decontamination Methods for Piping Systems

Methods	Advantages	Disadvantages
Recirculatory (loop-type)		
Alkaline permanganate; citric/oxalic acid (citrox)	Demonstrated effective solutions, nondamaging to components, decontamination factor (DF) ~10 to 20	Generally requires repeated cycles to obtain good decontamination. Generates large volumes of waste solutions requiring processing. Large tankage volume required for chemical preparation and handling.
Ethylenediaminetetraacetic acid (EDTA) and citrox, 5% solution	Only one cycle necessary. Contaminants can be removed by a sidestream through an online cation resin bed, if desired. Chemicals can be added directly to system water. DF ~10 to 20 at 80° to 100°C	Recirculation time ~2 days, minimum. Potentially damaging to system components. Produces large volumes of waste solutions requiring processing and/or radioactive ion-exchange resins requiring disposal. Additional R&D needed to determine optimum conditions.
EDTA, nitrilotriacetic acid (NTA), and reducing agent(s)	Produces relatively small volumes of waste solutions requiring processing. DF ~2 to 10	Recirculation time from 1 to 7 days. Requires elevated temperature (~150° to 180°C) to be effective.
Single-Pass		
Ammonium citrate and ammonium oxalate, 4 to 10% solution.	No preconditioning required, DF ~5 to 15 at 80°C	Attacks carbon steel, requiring waste solutions to be neutralized prior to long-term storage. Requires up to 20 hours for maximum effectiveness. Produces large volumes of waste solutions.
Inhibited phosphoric acid (H_3PO_4), ≤10% solution	Demonstrated fast (10 to 20 min.) and effective method, DF ~5 to 10 at 80°C	Produces large volumes of waste solutions. Possible redeposition of contaminants. Most effective on carbon steel.
Hydrochloric acid (HCl), ~10% solution	Fast, effective method for decontamination of carbon steel. (No reliable data on DF)	Incompatible with stainless steel. Possible redeposition of contaminants. Potentially more hazardous than other decontamination chemicals.

A mobile, shielded, decontamination unit can be used for decontaminating piping and equipment that is not amenable to system-wide internal decontamination. The unit can be used for both chemical recirculation and rinse-water flushing and, after use is itself decontaminated by backflushing. The unit consists of two parts: 1) a remotely controlled operating unit containing a recirculating pump, an expansion tank, a thermostatically controlled electric heater, and a valve manifold; and 2) a control unit, connected by an electrical cable to the operating unit, located an appropriate distance away from the operating unit to minimize radiation dose to the operators.

After isolation, the system being decontaminated is connected (with short-run flexible hoses) to the mobile decontamination unit to form a recirculation loop. Chemicals are injected into the loop through the valve manifold at the unit, valving is aligned, and recirculation is begun. The required solution temperature is automatically maintained by the unit's electric heater during the entire recirculation period. A minimum of operator attention is required during recirculation.

Chemical Decontamination of External Surfaces. A variety of chemical decontamination agents have been used for cleaning external surfaces in nuclear facilities and laboratories. These agents range from mild reagents such as detergents and household cleansers to harsh solutions of acids or alkalies. Decontaminating agents recommended for different surface materials are listed in Table G.4-2. Several commercially available proprietary compounds have also been used to decontaminate reactor surfaces and equipment. Water and steam are the most widely used reagents for gross decontamination of surfaces. The use of high-pressure steam or water is described in Section G.4.2.

A variety of household cleansers are effective in removing radioactive contamination from various surfaces. When it is necessary to remove contaminated paint or floor tile, paint stripping compound can be used for removing the paint and "dry ice" (solid CO_2) can be used to chill the floor tile for easy removal. Acids and alkalies remove surface contamination by dissolving a thin film of the surface to which the contamination is fixed. Redeposition of

TABLE G.4-2. Chemical Decontaminants for Various Surfaces

Surface	Decontaminant
All Surfaces	Steam or water and detergent Organic solvents
Stainless Steel	20% HNO_3 - 3% HF, by weight 20% sodium hydroxide - 2% tartaric acid Complexing agents (EDTA, oxalates, carbonates, citrates) Alkaline permanganate - ammonium citrate
Carbon Steel	Commercial rust removers Inhibited H_3PO_4, 1 Molar Complexing agents (EDTA, oxalates, carbonates, citrates)
Aluminum	Dilute NaOH Mixture of citric acid and detergent Complexing agents (EDTA, oxalates, carbonates, citrates)
Copper, Brass	Dilute HNO_3 Household and industrial cleaners of copper and brass Complexing agents (EDTA, oxalates, carbonates, citrates)
Lead	Dilute HNO_3 Concentrated HCl
Glassware	Chromic acid Concentrated HNO_3
Floor Tile	Ammonium citrate Trisodium phosphate Household cleaners containing grit or pumice
Painted Surfaces	Commercial paint removers Trisodium phosphate Household cleaners containing grit or pumice Complexing agents (EDTA, oxalates, carbonates, citrates)

the contaminants on the clean surface can frequently be avoided by adding a complexing agent such as ethylenediaminetetraacetic acid (EDTA) to the decontamination solution.

Mechanical action is an important factor in increasing the efficiency of chemical decontamination. For cleaning external surfaces, reagent decontamination should be accompanied by mechanical agitation with a scrub brush or swab that keeps the solution in motion. For decontaminating equipment items that can be immersed in a solvent, effective mechanical agitation is provided by the ultrasonic generator.

One effective method of applying chemical reagents involves spraying the solution on the surface to be decontaminated. For decontaminating transuranium-contaminated glove boxes, two holes are drilled through the glove box wall just large enough to snugly accept 6-mm plastic tubing. A meter or two of tubing is pushed into the box through the holes, and the outside end of one tube is connected to a small-capacity acid-proof pump. A solution of 30% nitric acid and 1% hydrofluoric acid is pumped into the box, using about 0.001 m^3 for a 2-m^3 internal volume. The operator constricts the end of the tubing within the box and directs the spray to all internal surfaces. When all surfaces have been wetted, the box is allowed to stand for 1 hour to permit the acid to dissolve the transuranic nuclides wedged into crevices and corners of the box. After about an hour, the exterior end of the second tube is connected to a vacuum system and the acid solution is sucked from the box into a 0.20-m^3 plastic carboy. The interior of the box is then rinsed with water pumped through the first tubing, and the rinse water is removed by suction into the carboy. The box is rinsed two or three times by this method. An average decontamination factor of about 100 is reported.[10]

G.4.1.2 <u>Electropolishing</u>

Electropolishing is an electrochemical process using electrolytic dissolution to produce a smooth, polished surface. It has been demonstrated effective in decontaminating a wide variety of equipment items with varying metal alloy compositions.[11]

The electropolishing process, shown schematically in Figure G.4-1, produces a two-fold result: 1) surface material is dissolved and removed (together with any contaminants), and 2) the new surface of the material is smoothed, minimizing recontamination from the electrolytic solution and allowing a simple water rinse (the final process step) to produce a contamination-free surface. Electropolishing offers a number of potential advantages over other decontamination processes:

- It can be used on a wide variety of metals and their alloys, equipment geometries, and types of contamination.

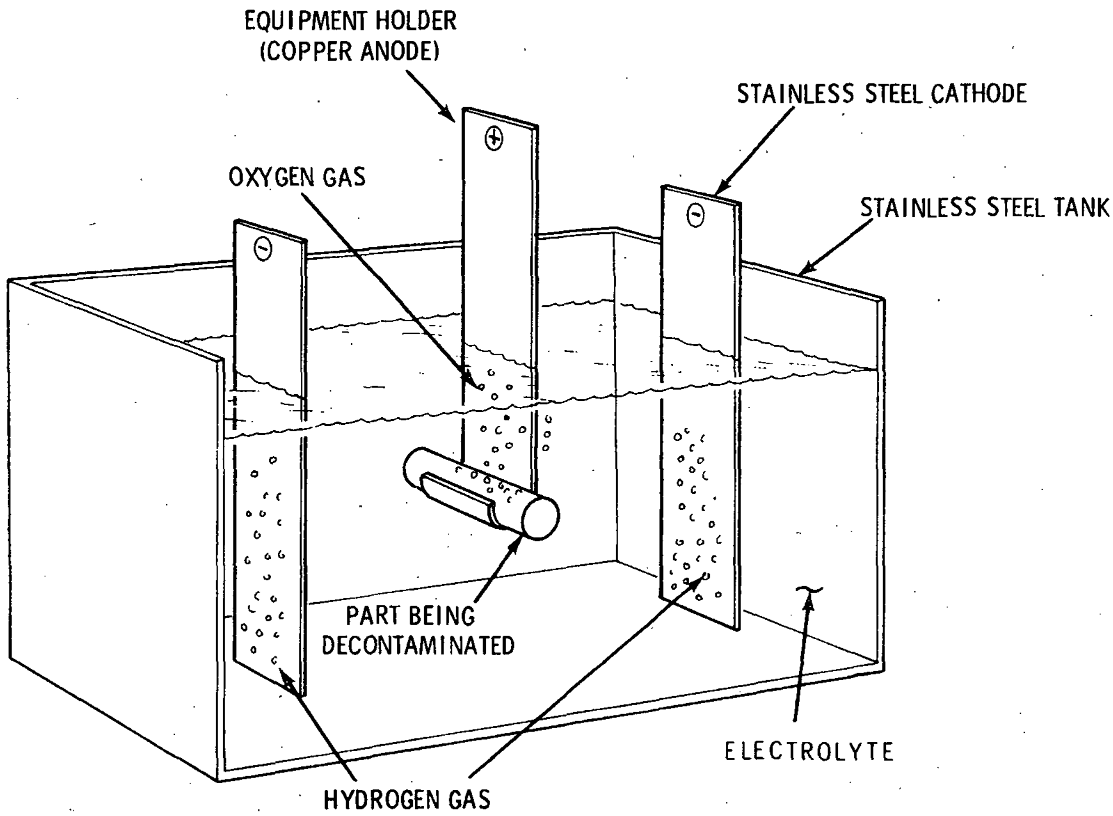

FIGURE G.4-1. Schematic Diagram of an Electropolishing Cell

- It reduces personnel radiation exposure and decontamination costs.
- It is rapid and effective.
- It can significantly reduce the amount of solid waste requiring expensive controlled storage, transportation, and disposal.
- It permits reclamation, reuse, and/or recycling of valuable materials.

For decontamination of disassembled (or segmented) piping and equipment, an electropolishing station is required. It consists of several open-top tanks: one contains the electrolyte (typically phosphoric and/or sulfuric acid) and appropriate fixtures, and the others are used for rinsing. A large constant-current, constant-voltage, DC power supply, capable of producing up to 10,000 A at about 10 V, is required to provide a current density of 500 to 1500 A/m^2 at the anode surface. A recirculation system constantly mixes the

electrolyte and also provides online filtration and ion exchange to remove suspended solids and dissolved metal ions from the electrolyte. When the capacity of the ion-exchange column is exhausted, it is backflushed and regenerated, with the resulting waste solutions going to the inplant liquid radwaste system for processing. An air sweep must be maintained over the electrolyte tank to sufficiently remove and dilute the hydrogen gas evolved by electrolysis during the cleaning process. The electropolisher is designed to automatically shut down if the ventilation system is inoperative, in order to prevent the accumulation of explosive concentrations of hydrogen gas.

Electropolishing can also be used for in-situ decontamination of the internal surfaces of cylindrical tanks. The electrolyte is sprayed onto the tank wall from nozzles mounted on rotating arms. A power supply provides current to these arms and, via the electrolyte stream, to the tank wall. Contaminated electrolyte is collected at the tank bottom and returned to an electrolyte handling system for cooling, filtration, and radiation monitoring. When the electrolyte becomes too contaminated, it is cleaned by processing it through the ion-exchange/filtration system of the electropolishing station and is then reused.

G.4.2 Physical Cleaning

Removal of smearable radioactive contamination from surfaces such as walls, floors, and tank exteriors can be accomplished using a variety of techniques. For small quantities of loose contamination on floors, vacuuming or simple sweeping is often effective. For more tenacious contaminants, various cleaning compounds are used in combination with handwiping and scrubbing techniques. Several proprietary decontamination solutions are available. Ordinary household detergents are quite effective but produce sizable quantities of waste water that may require special processing. Aerosol-type foaming cleaners are effective and eliminate the waste-water problem, but their use produces sizable quantities of contaminated wiping material. For suitably sized items, ultrasonic cleaning methods can be used.

Trichloroethylene, Freon-113, and other solvents are effective degreasing agents that can be used to decontaminate equipment surfaces. Organic solvents

have the advantage of not being corrosive to equipment and electrical connections. However, their use generates contaminated organic solutions that must be processed. Use of organic solutions can cause degradation of plastic materials and can produce sizable quantities of contaminated wiping materials.

High-pressure steam, either alone or mixed with chemicals or detergents, is effective in attaining high decontamination factors for some surfaces. In general, any surface that can be decontaminated by wiping or scrubbing can be cleaned in less time and just as effectively with high-pressure steam. An advantage of steam cleaning is the small volume of liquid waste generated. An arrangement for mixing high-pressure steam with a detergent[12] is shown schematically in Figure G.4-2. The tank containing the detergent is pressurized with steam, forcing the detergent through a mixing valve into the main steam stream.

Systems using a high-pressure-steam jet are commercially available. Equipment of this type is well suited for remote operation and for cleaning large surface areas. The steam jet, or ejector, pumps the detergent solution

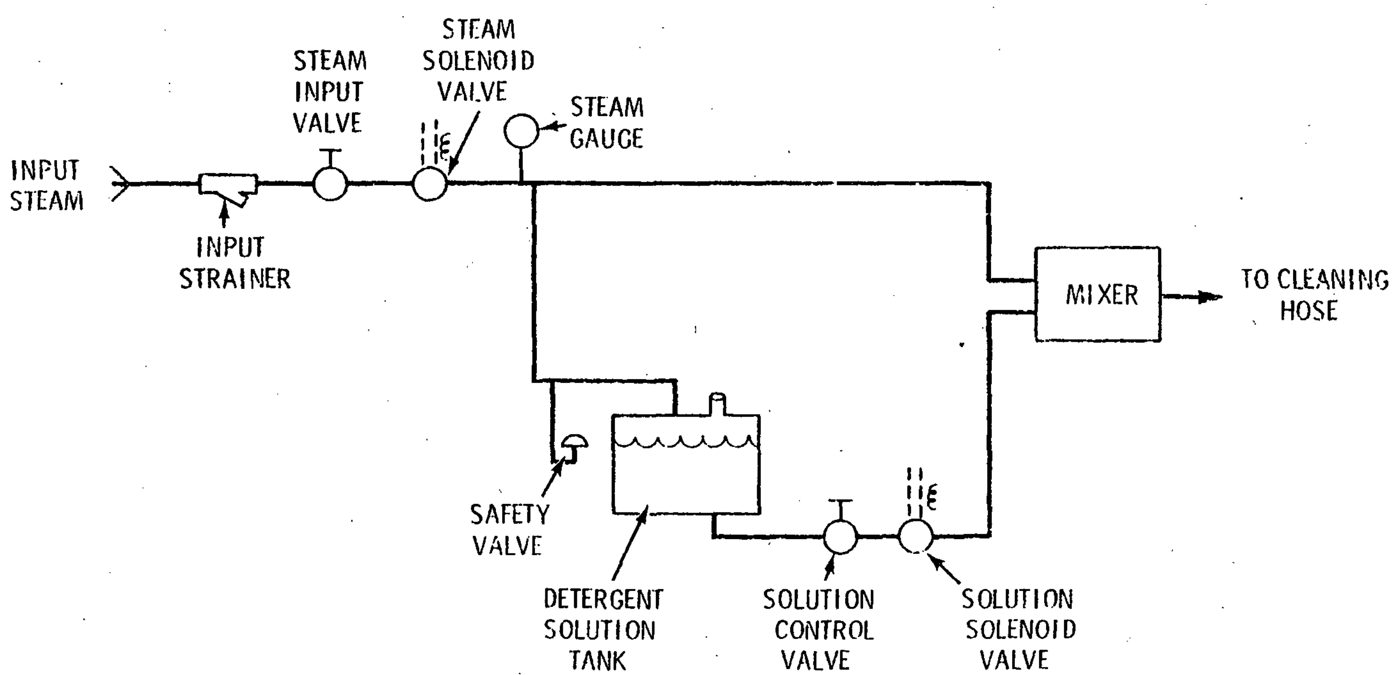

FIGURE G.4-2. Equipment for Mixing Detergent with High-Pressure Steam

using high-pressure steam as the pumping fluid and discharges the detergent solution against the surface to be cleaned. Jet cleaning devices can be operated by one person using a hand-held jet lance. High-pressure jet cleaning has the disadvantage of spreading contamination over a large area during the cleaning process.

Variable-pressure, high- or low-velocity liquid jets are quite effective for some types of surface decontamination work.(13) These devices utilize hand-held lances that can be operated by one man and can operate at pressures up to about 30 MPa. Where effective, the liquid jet is a very rapid decontamination method. However, it produces significant volumes of contaminated liquid requiring processing for offsite disposal.

An ultrasonic cleaning unit employs an ultrasonic generator, a tank containing a solvent, and a transducer to transmit the ultrasonic energy to the solvent. The compression-rarefaction cycle of the waves in the solvent, when striking an object in the fluid, causes localized cavitation and implosion at the surface of the object, scrubbing the surface and removing surface contaminants. A typical ultrasonic cleaner installation is shown in Figure G.4-3, with the tank contained in a ventilated enclosure.

G.4.3 Removal of Structural Material

During facility decontamination, removal of both metal and concrete surfaces may be required. However, the techniques for metal-surface removal are the same as those for equipment disassembly, discussed in Section G.5. The present discussion is limited to concrete removal.

Some concrete in nuclear facilities is contaminated below the surface and cannot be decontaminated to release levels by physical surface cleaning alone. The contaminated concrete must be physically removed and disposed of offsite to sufficiently decontaminate the facility.

Several criteria must be considered when selecting a material-removal method for a particular location. The method chosen should minimize personnel radiation exposure and airborne contamination dispersion. In addition, the size and weight of removed materials must facilitate packaging and shipping for offsite disposal.

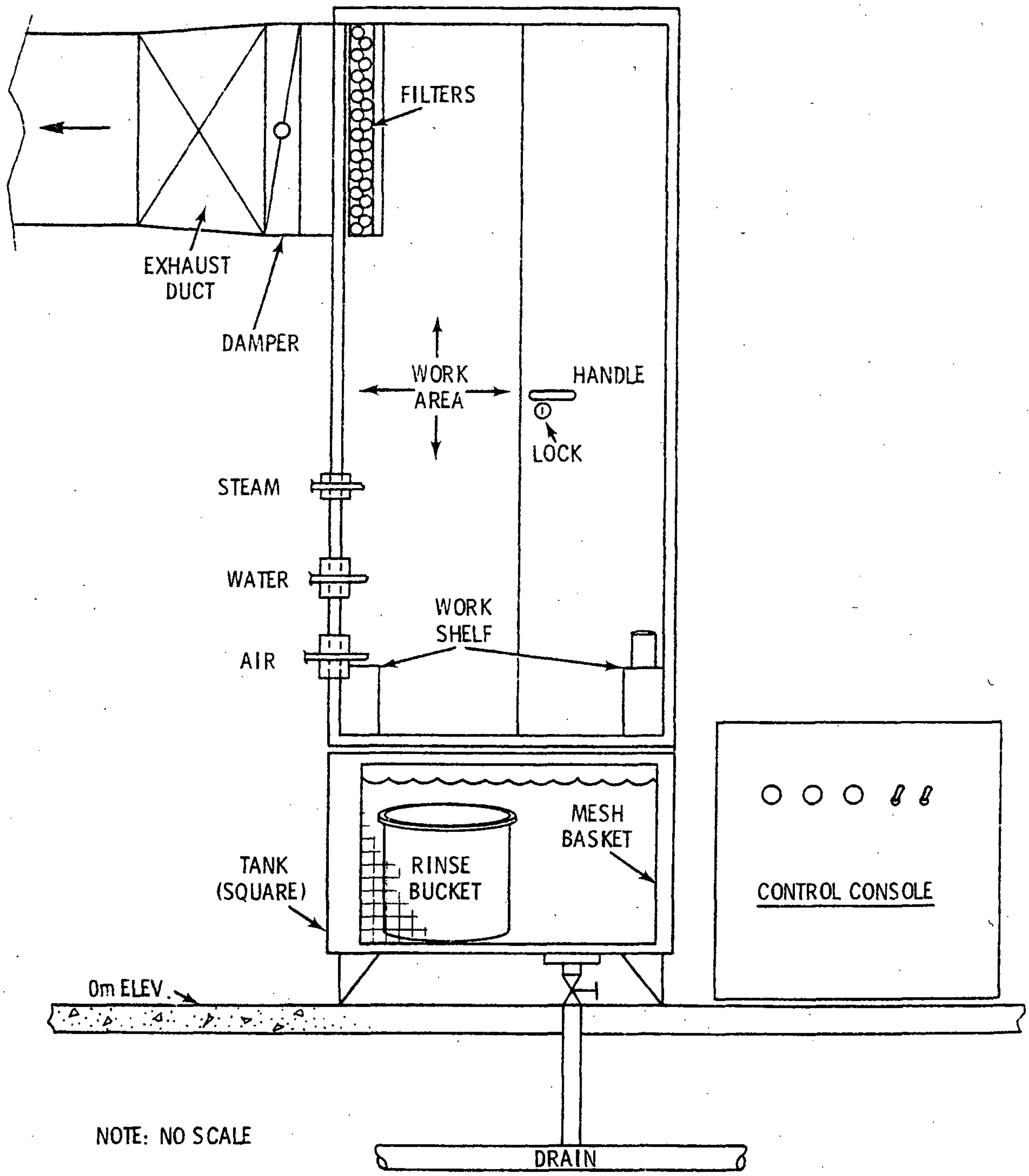

FIGURE G.4-3. Ultrasonic Cleaning Unit

TABLE G.4-3. Concrete Removal Methods

Removal Method	Advantages	Disadvantages
Blasting	Fast material removal rate where adjacent material damage is not of concern	May cause vibrations and shocks (vibra-(tional control possible by limiting maximum charge per delay)
	Adaptable for cutting irregular surfaces	Requires oxayacetylene torch for cutting exposed reinforcing steel
	Adaptable to remote operation	High noise level
	Adaptable to control of size and weight of material removed	Generates moderate quantities of dust that must be controlled
	Selective removal of radioactive contamination possible	
Core Boring and Rock-Splitting	Proven techniques used extensively in conventional mining operations	Reinforcing steel slows cutting speed and damages core drill
	Low noise level	Requires oxycetylene torch for cutting exposed reinforcing steel
	Low vibration/shock level	Cooling water required for drilling may have to be filtered and recycled
	No toxic gases generated	Not ideally suited for operation on irregular surfaces or in cramped quarters
	Cooling water spray minimizes airborne particulate material	
	Moderate-to-slow cutting speeds	
	Controlled rate of material removal	
Flame Cutting	Concrete and reinforcing steel may be cut in one operation	Generates copious quantities of toxic gases and smoke
	Fast cutting speed	Consumes large quantities of bottled oxygen
	Adaptable for remote operation	Smoke can potentially spread gross contamination throughout contamination-control envelope unless adequately controlled
	Adaptable for pivoted circumferential cutting	Requires a through starter-hole made by a core drill to provide an outlet gas-flow path
	Adaptable for cutting on irregular surfaces	
	No vibration/shock	
	Low noise level	
	Controlled rate of material removal	
Thermic Lance Cutting	Cuts both concrete and steel without difficulty	Generates moderate quantities of toxic gases and smoke
	Fast "hole-punching" speed	Requires rock jack to break away cut concrete sections
	Ideally suited for cutting irregular surfaces	Consumes large quantities of bottled oxygen
	Remote operation is possible for up to approximately 3 meters	Smoke can potentially spread gross contamination throughout contamination-control envelope unless adequately controlled
	No vibration/shock	
	Low noise level	
	Controlled rate of material removal	

The various methods available for concrete removal, with associated advantages and disadvantages, are listed in Table G.4-3. Of particular interest are the blasting techniques for use in bulk removal of concrete and the rock-splitting techniques for localized removal of contaminated concrete surfaces. Two techniques of lesser interest, primarily because of the copious quantities of toxic gases and/or smoke produced, are flame cutting and thermic lance cutting.

Flame cutting uses commercially available, 3-m-long by 19-mm-diameter, thin-wall-conduit burning bars. Each bar contains up to seven dissimilar metals in the form of strips or strands, encased in an outer tube into which oxygen is forced during the burning process. Once ignited, the entire hand-held device is consumed as it burns. After 4 to 5 minutes, with about 0.5 m of the bar left, the oxygen supply is shut off and the exothermic reaction stops. The remaining portion of the spent bar can be attached to a second bar and the burning process repeated, thus conserving bar material.

The thermic lance is also hand-held and uses powders of materials similar to those used in the aforementioned burning bar. The powders, together with oxygen, are forced through a tube or nozzle and ignited to form a high-temperature cutting medium.

G.4.3.1 Concrete Surface Removal

A number of techniques can be used to remove contaminated concrete surfaces in nuclear R&T reactor facilities. A comparison of the various techniques is presented in Table G.4-4.[4]

Sand blasting, where the surface is mechanically eroded away, and flame spalling, where intense heat is applied to concrete surfaces, remove only a minimal surface thickness and produce large quantities of small, contaminated particles. Sand blasting primarily removes paint and a little of the concrete surface but does not effectively remove contamination in the pores of the concrete or at expansion joints.[3] A large exhaust and air filtration system is needed with both of these methods. These two techniques are relatively slow if the contamination penetrates beyond a thin surface layer.

Two surface removal methods used more extensively than the rest are jack hammers and impactors. Jack hammers, powered by compressed air, are readily

TABLE G.4-4. Comparison of Concrete Surface Removal Techniques

Technique	Limitation	Type of Rubble Produced	Size of Air Filtration System Required	Relative Removal Speed
Sand Blasting	Contamination embedded in pores not effectively removed	Small particles	Large	Slow
Flame Spalling	Heat may cause undesirable chemical reactions	Small particles	Large	Slow
Jack Hammer	Awkward to use on walls	Medium-sized pieces and small particles	Medium	Medium-Fast
Impactor (powered pneumatically or hydraulically)	Limited to large accessible facilities	Medium-sized pieces and small particles	Medium	Fast
Scrubber	Awkward to use on walls	Small pieces and particles	Medium	Fast
Water Cannon				
Handheld Modified 458 Magnum Rifle	Gun powder combustion products are produced	Small pieces coated with glycerine and gun powder combustion products	Small	Slow
Rapid-Fire Model	Limited to large accessible facilities	Small pieces coated with water	Small	Fast
Concrete Spaller with Air Drill to Make Holes	Awkward to use on irregular surfaces or in cramped quarters	Medium-sized pieces and small particles	Small	Medium-Fast

available and are easily operated by one man. They are used to chip off the surface material deep enough to remove the contamination.[3] Because they are difficult to position on walls and ceilings, jack hammers are used primarily on floors. Impactors (or mounted impulse breakers), similar in operation to jack hammers but much larger, have been used successfully in several decontamination projects.[3,4] An impactor uses a pick chisel point that is driven into the concrete surface with high-energy impacts several times per second. Impactors are powered either pneumatically or hydraulically, and are held and positioned with linkages typical of those on tractor-mounted backhoes. A medium-size air filtration system is necessary to control the dust produced by both of these surface removal methods.

Another tool used to chip away concrete surfaces is a "scabbler," a hand-held tool with a gang of carbide-tipped bits. The bits rapidly impact the surface, causing small concrete pieces to be spalled. A dust shield with a vacuum attachment is placed around the bits to remove the dust generated. Dust removal increases this method's effectiveness by preventing the tool from pounding contaminated particles back into the concrete surface.[3] The "scabbler" is relatively slow and is difficult to use on walls and ceilings.

The water cannon removes concrete surfaces by shooting very-high-pressure liquid jets at the surface, causing it to spall. This method has three advantages: 1) no initial surface preparation is needed, 2) the equipment does not contact the surface, and 3) the resulting rubble is composed of small pieces coated with liquid, thus minimizing dust generation. Two versions of the water cannon can be used. The first is a modified 0.458 magnum rifle that shoots solidified glycerine through a nozzle; the device is hand-held and typically removes less than 20 mm of concrete from the surface. The second version uses stored, compressed gas to drive a piston that forces water through a small-diameter nozzle; this large unit is mounted on a backhoe or excavator, and is therefore limited to use in large, readily accessible areas.

The last technique, use of a concrete spaller, is selected in this study as the principal method for removal of contaminated concrete surfaces. This device permits localized concrete removal to depths of 50 to 75 mm with no explosions and very little dust. (The principal source of dust is the drilling

of the hole into which the splitting tool is inserted.) Surface-removal rates in the range of 5 to 7 m^2/hr have been reported.[4] A dust shield with a vacuum attachment minimizes the spread of contaminated dust and can be used to collect all but the largest pieces of rubble. The concrete spaller, mounted on a spaller platform, is illustrated in Figure G.4-4.

The concrete spaller consists of three basic parts: a hydraulic cylinder, a push rod, and a bit with expanding wedges. The hydraulic cylinder is attached at one end and activates the push rod, which is installed inside the bit. A schematic of the device is shown in Figure G.4-5.

The spaller is operated by inserting the bit into a predrilled hole about 55 mm deep and 25 mm in diameter. The hydraulic cylinder is then activated, forcing the push rod toward the end of the bit. The wedges in the bit are forced radially outward into the walls of the hole. When the tip of the push rod reaches the bottom of the hole, it forces the wedges away from the bottom and causes the concrete to be spalled off.

The spaller is small, lightweight, and fully portable and can be readily adapted to remote operation. It is connected to the hydraulic power source with hydraulic hoses. The spaller is about 0.4 m long, with a mass of about 10 kg.

For rapid removal of large surface areas, a number of the devices can be ganged together with a corresponding set of concrete drills and operated as a unit. The spacing between holes (0.2 m optimum) and the pattern of the holes (triangular array) are important parameters in the effectiveness of this technique. Arrangement of the concrete drills and the splitting tools into a fixed-geometry array ensures a relatively uniform removal pattern. Combining these ganged units with a vacuum transfer system for rubble removal results in a fast and dust-free concrete removal method, one ideally suited to nuclear facility decommissioning applications.

G.4.3.2 Contaminated Sewer and Soil Removal

The reference test reactor has an Emergency Retention Basin (or pond) that is serviced by several concrete sewer lines, ranging in size from 0.3 m to 0.9 m. These sewer lines and the Emergency Retention Basin are lightly contaminated with radionuclides (see Appendix E).

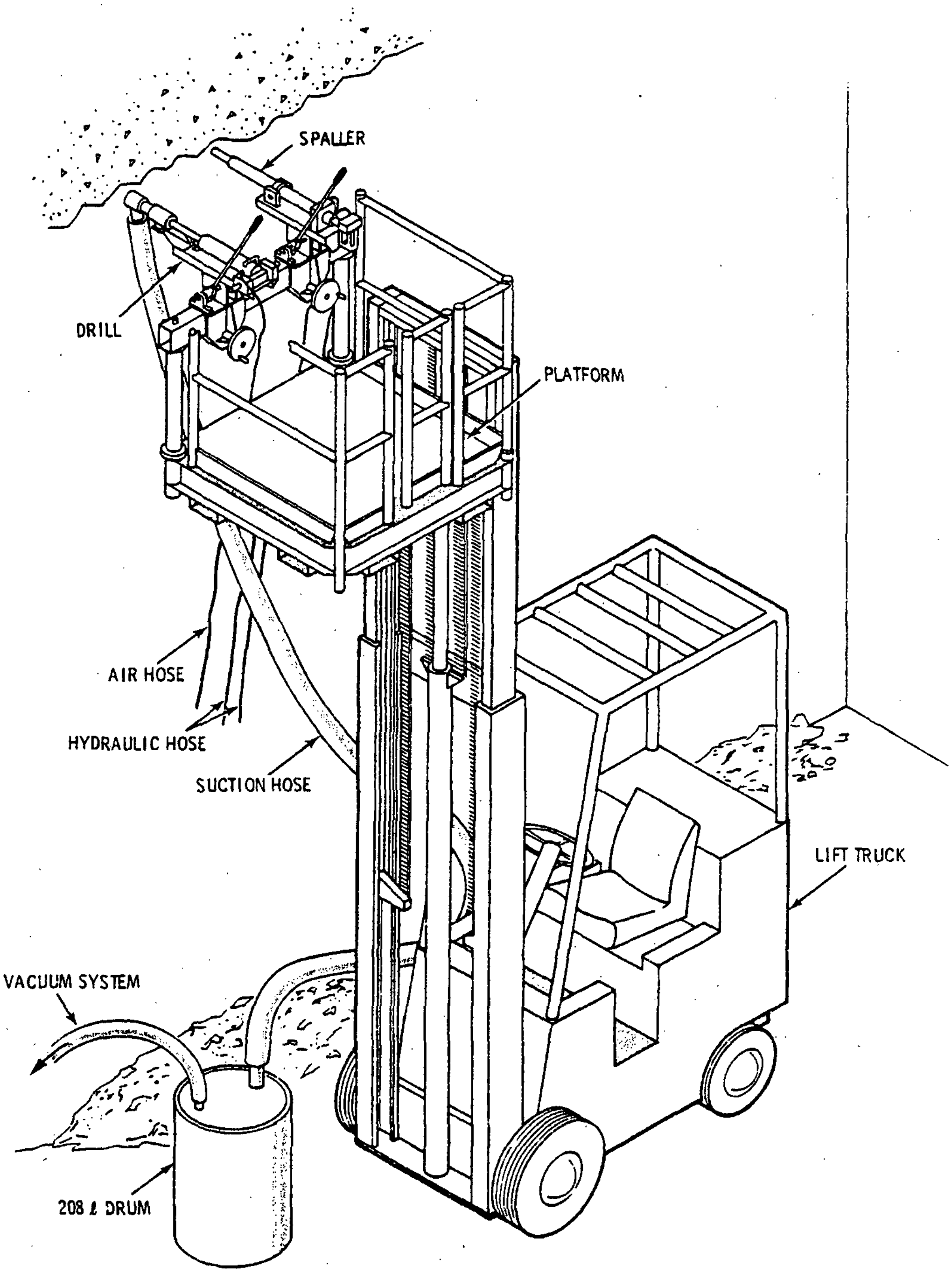

FIGURE G.4-4. Mounted Concrete Drill and Spaller Unit

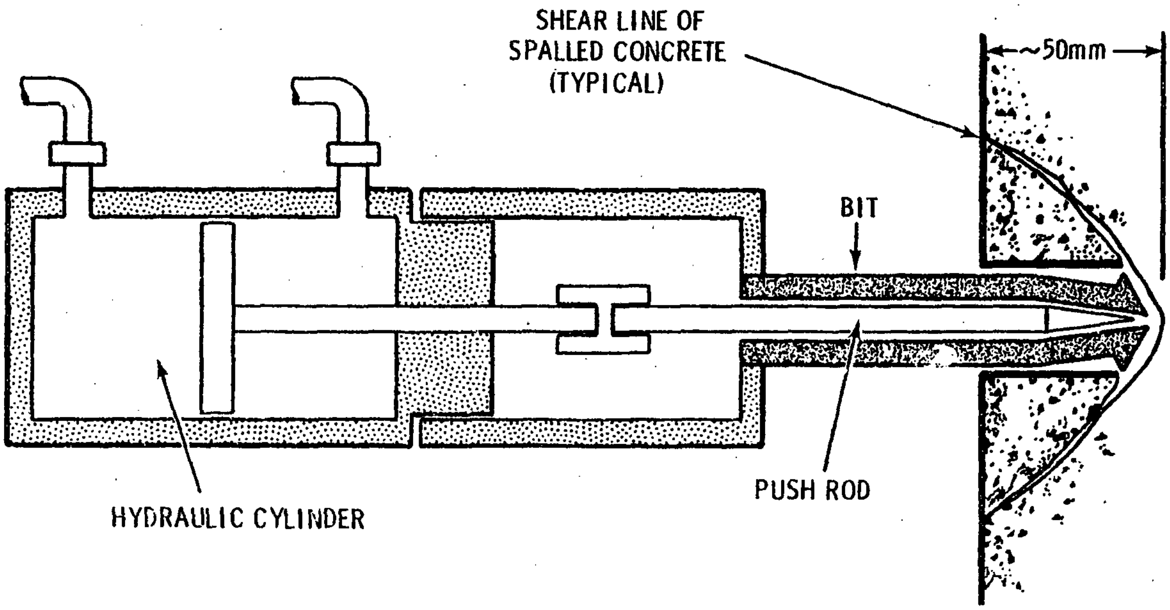

FIGURE G.4-5. Diagram of Concrete Spaller

No special tools are required for removal of the concrete pipe. The joints can be loosened and the 0.9-m sections can be hoisted out of the trench and placed in burial boxes.

G.5 EQUIPMENT DISASSEMBLY

Decommissioning of nuclear R&T reactor facilities requires the disassembly and removal of various contaminated equipment systems. The equipment must be segmented into pieces small enough to facilitate either onsite entombment or packaging for offsite shipment and disposal, depending on the decommissioning alternative selected.

Any of a number of methods can be used to disassemble and remove a particular piece of equipment. However, equipment-related parameters (e.g., size, location, design, and radioactive contamination and/or activation) and manpower/cost efficiencies of the various methods dictate the appropriate method for any given situation. In some cases, the required tools are available as part of the facility's normal operating-equipment complement; in others, the tools are readily available commercially (e.g., boltcutters, pipe saws, tubing cutters, and impact wrenches). However, some methods require the use of

unique specialized tools and equipment. These specialized items are discussed in the following subsections and include an underwater manipulator, cutting equipment for sectioning large items, and linear-shaped explosives for certain cutting applications.

G.5.1 Underwater Disassembly

The neutron-activated core structural components are sufficiently radioactive that they must be disassembled and packaged under water for shipment to an offsite disposal facility. Special tools needed for this underwater work include:

- powered socket drives to operate on either a horizontal or vertical axis
- extension wire cutters
- long-handled grapples
- mechanical fingers mounted on long handles
- jigs for breaking beryllium plates and pieces
- vises or jigs to hold components while they are sawed or cut with a torch.

An underwater manipulator is needed to hold and guide the cutting equipment used for disassembly and segmentation of the core structural members.

G.5.2 Cutting Equipment

The principal equipment anticipated to be used for cutting activated and/or contaminated items are the oxyacetylene torch, the plasma-arc torch, and the arc saw. This equipment can be used either under water or in air. The oxyacetylene torch is a relatively common device and, therefore, it is not discussed here; cutting rates and other information are available in Reference 14. Descriptions, cutting characteristics, and requirements of the plasma-arc torch and the arc saw follow.

G.5.2.1 Plasma-Arc Torch

Plasma-arc cutting employs an extremely high-temperature, high-velocity, ionized-gas arc between the electrode in the torch and the piece to be cut.

The arc is constricted by passing it through a nozzle with a small-diameter orifice. This localizes the arc so that its energy is concentrated on a small area on the work piece, where its intense heat melts the metal. The melted metal is continuously removed by the jet-like action of the gas stream to form a kerf. The heat and force of the arc stream produce a high-quality, saw-like cut. If inert gases are used, the cutting process is dependent on thermal action alone. When cutting ferrous materials such as mild steel or cast iron, increased cutting speeds can be achieved using oxygen-bearing gases. (In this case, the chemical energy obtained by the combination of oxygen with the base material is added to the arc heat to permit higher cutting speeds.) The process can be used to cut any metal. The plasma-arc torch is shown schematically in Figure G.5-1.

The electrical circuit is basically the same as that used for gas tungsten-arc welding, with the exception that an electrical lead is taken from the grounded work piece and passed through a resistor to the nozzle of the torch. A high-frequency spark is used to complete the circuit between the electrode and the nozzle, thereby producing a pilot arc that initiates the main arc from the electrode to the work piece.[15]

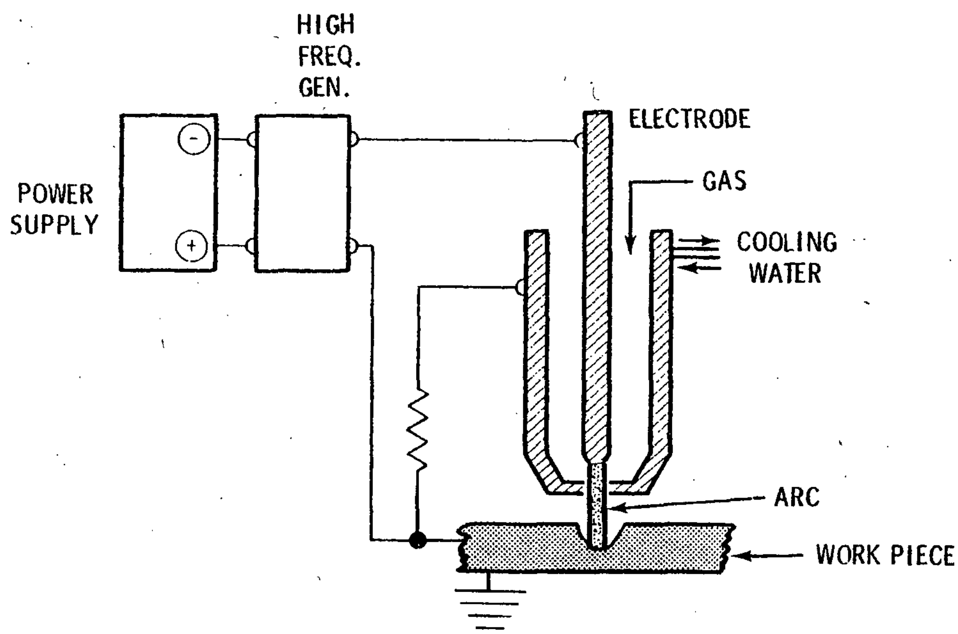

FIGURE G.5-1. Plasma-Arc Torch Schematic Diagram

There are two types of electrode holders and cutting controls, one designed for manual cutting and the other for mechanized cutting with the torch mounted on an automatic travel mechanism. Water cooling is used to prolong the operating life of the torch. Torch surfaces are insulated to protect the operator from electrical shock. Gas-pressure regulators and flowmeters control the cutting gas. A contactor, usually built into the power supply, is required for turning the power on and off.

Arc-constricting nozzles are available in a variety of diameters and shapes. The required shape and diameter of the nozzle depends on the application and the current to be used; high currents require larger-diameter orifices. Both single-port and multiport nozzles are available. Multiport nozzles generally have auxiliary ports arranged in a circle around a central orifice.

The power supplies used have open-circuit voltages in the range of 120 to 400 VDC. The high open-circuit voltage is required for heavy-duty cutting, such as severing thick plates and piercing metal as thick as 50 mm.

Mixtures of argon or nitrogen with hydrogen are generally used for cutting stainless steel, as well as aluminum and other nonferrous metals. Nitrogen and oxygen, supplied separately to the torch and mixed in the nozzle, are used for cutting carbon steel, cast iron, and alloy steels.

For mechanized operations in air, the torch standoff distance is set between 6 and 16 mm. The cutting current, gas type, and gas flow rate are set at values recommended by the equipment manufacturer. These values are primarily determined by the type and thickness of the material being cut. After the operator initiates the cutting arc, the sequence of operations is regulated by an automatic cutting control.

For manual operation, the operator selects the gas flow rate and cutting current from the table of conditions supplied by the manufacturer or from data obtained from prior testing. The torch is held over the work piece, the cutting arc is struck, and the torch is manually guided along the cut. At the end of the cut, the arc automatically extinguishes, and the control automatically opens the contactor and closes the gas valves. The operator can extinguish the arc at any time simply by moving the torch away from the work piece.

The plasma-arc cutting process can be used in air or under water. It is especially adaptable to automation and is thus useful when highly radioactive material is to be cut (e.g., the pressure vessels of the Elk River Reactor in Minnesota and the Sodium Reactor Experiment in California).[3,16] As it is not necessary to start the cut at the edge of the plate, the plasma-arc torch is particularly adaptable to cutting holes in large plates and vessels. It is also well adapted to gouging applications, including pad washing and scarfing. However, because of the short torch standoff distance, plasma-arc cutting is not suitable for some applications, particularly in tight spaces. Air carbon-arc cutting can be used for such work. Plasma-arc cutting is preferred where it is possible.

Some typical conditions for mechanized plasma-arc cutting of stainless steel in air are given in Table G.5-1. Recent tests with the plasma-arc torch have demonstrated its ability to cut 150-mm-thick stainless steel and 180-mm-thick carbon steel.[17]

G.5.2.2 Arc Saw

The arc saw, a state-of-the-art metal-cutting device, is currently being developed for contaminated-equipment segmentation, with initial development and demonstration work already completed.[18] Results to date indicate that the device holds great promise, and additional development work will undoubtedly provide a more sophisticated, commercially available unit in the near future. The cutting process is both economical and expedient relative to other methods.

TABLE G.5-1. Typical Conditions for Plasma-Arc Cutting of Stainless Steel in Air

Thickness (mm)	Speed (m/min)	Orifice Diameter (mm)	Current (amps)	Gas Flow (m^3/hr)
13	2.5	3.2	300	4.2 N_2
25	1.2	4.0	400	5.0 N_2
50	0.5	4.0	500	3.7 A and 2.0 H_2
75	0.4	4.8	500	3.7 A and 2.0 H_2
100	0.2	4.8	500	3.7 A and 2.0 H_2

Cutting speeds of around 10 m/min are anticipated for 10-mm-thick steel plate, with the speed being inversely proportional to the plate thickness.[19]

The prototype arc saw used a 11.2-kW motor to rotate a 0.91-m-diameter, 6.35-mm-thick blade made of copper or mild steel. Blade speed was approximately 880 rpm, and blade cooling was accomplished by a water-spray system installed in the blade guard. A material-cut/blade-wear ratio of 20 to 1 was reported for the 0.91-m-diameter blade. It is expected that increased blade diameters may reduce associated costs.[18]

Cutting can be accomplished remotely, either in air or under water, with automatic positioning and tracking of the saw blade during cutting operations. All equipment, with the exception of the blade, is commercially available, but modifications are necessary. Blades can be made in any well-equipped machine shop. The arc saw is shown schematically in Figure G.5-2.

The modified power source consists of a 7500-A, 80-V open-circuit, 50-V operating DC power supply of the constant-current, constant-voltage type. The controls include a servo controller for automatic positioning and tracking of the saw blade during cutting operations and a recorder to monitor voltage, amperage, blade travel, and servo current.

Low-voltage, high-amperage current is applied to the rotating blade, creating an electric arc discharge between it and the grounded work piece. The discharge spark-erodes both the blade and the work piece. The blade is advanced through the work piece to produce a kerf 1 to 4 mm wider than the blade thickness.

If, because of position change or vibration, a portion of the work piece falls against or pinches the blade, the point of contact spark-erodes away because the "electrical leading edge" of the blade is transferred to the point of contact. This reduces the potential for binding of the blade. Consequently, the arc saw can cut through a variety of materials, shapes, and loose components that would be difficult to cut with conventional saws.

The arc saw blade can be scaled to match the cutting requirements of the job. Blades up to 3 to 4 m in diameter (up to 25 mm thick) for cutting large-diameter vessels appear to be feasible. Heavier (thicker) blades are likely

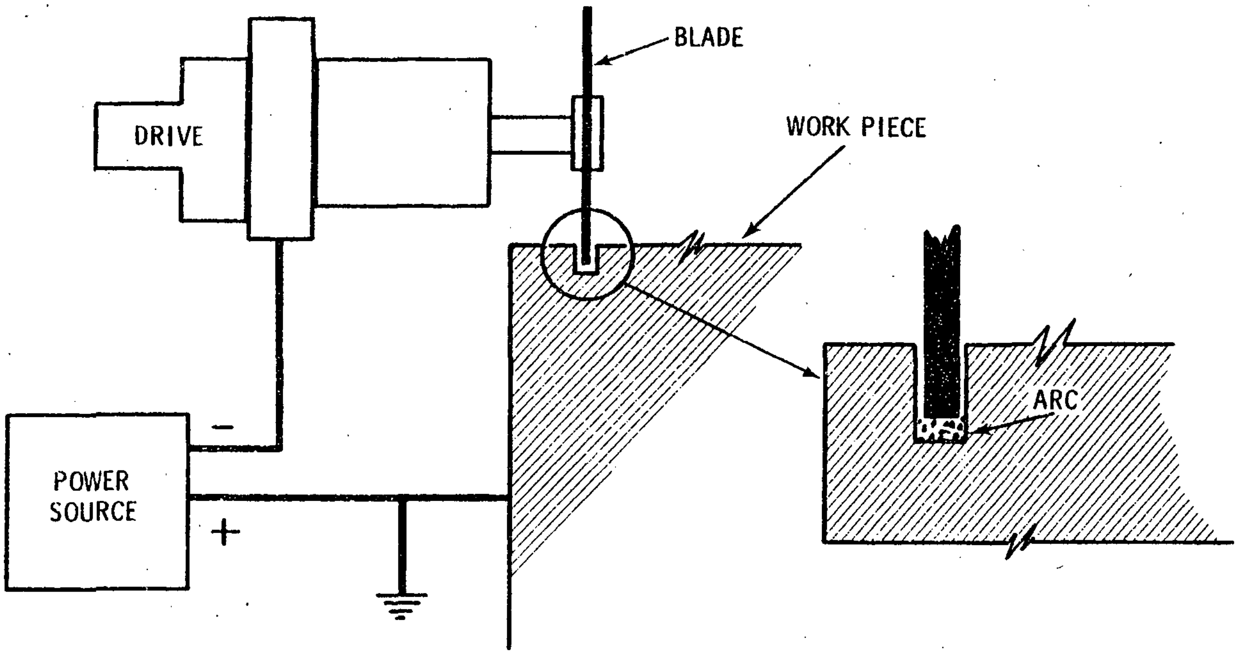

FIGURE G.5-2. Arc Saw Schematic Diagram

to tear most non-conducting materials (non-metals) they encounter, thus eliminating the need to remove them from the work piece before cutting. It also appears that several saws could be ganged together for simultaneous parallel cutting.

G.5.3 Linear-Shaped Explosive Charges

An economical and expedient method of reducing the physical size of equipment and piping to allow further processing or packaging for disposal is the use of self-contained, linear-shaped explosive charges. This method minimizes personnel radiation exposure and is particularly advantageous in areas with high radiation levels. Linear-shaped charges have been used extensively in the last 15 to 25 years.(20,21) Recently, such methods (i.e., the perforation and cutting of solids by ultra-high-velocity particles under explosive attack) have been used to segment and remove activated components and contaminated systems (e.g., fuel-pool liners and piping) that are not amenable to conventional removal techniques.(4,22) Figure G.5-3 shows a cross section of a typically positioned linear-shaped charge.

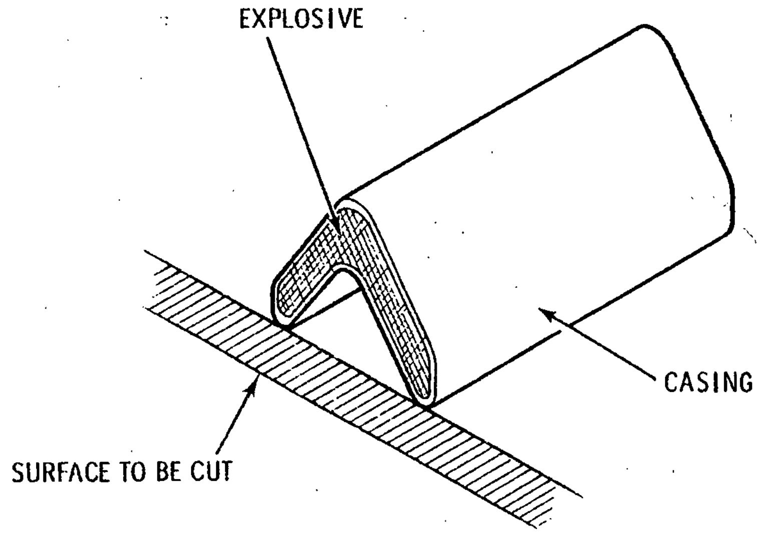

FIGURE G.5-3. Cross Section of Typically Positioned Linear-Shaped Charge

A linear-shaped charge consists of a V-shaped tubular casing filled with an explosive (cyclotrimethylenetrinitramine) commonly called RDX. The casing is generally copper, but other soft materials such as aluminum and lead can be used. The charges are equipped with detonators that are activated using a primacord lead having a high-explosive core of pentaerythritol tetranitrate (commonly known as PETN) that, in turn, is initiated with an electric blasting cap. Clamp-on charges, available commercially, eliminate many problems in placement, handling, and detonating. The number of charges that can be detonated at one time is limited only by the blast effect on nearby equipment. Shock-wave and fragment damage can be reduced appreciably by placing blast curtains or other barriers in the vicinity of the detonation to disrupt the shock wave and intercept the fragments.[23]

The principle of the linear-shaped charge is that, as the detonation wave collapses the inverted V, the casing material becomes a jet of extremely hot metal particles traveling at very high velocity. These particles then tear through the material to be cut. The melting and subsequent fusing of the casing material with the base material being cut, together with the ragged edges

of the finished cut, can make the electropolishing of those edges very difficult. Therefore, in-situ decontamination (either chemical or mechanical) prior to explosive cutting is recommended to minimize unnecessary waste of strategic materials.

The use of linear-shaped charges has been documented for cutting such diversely shaped objects as:(23,24)

- 0.61-m-O.D. by 16-mm-wall stainless steel pipe
- 100-mm by 100-mm by 10-mm-thick angle iron
- 25-mm-diameter solid stainless steel bar
- 152-mm by 10-mm-thick web "H" beam.

G.6 RADIOACTIVE WASTE PACKAGING AND SHIPPING

Radioactive wastes are generated during the decommissioning of a research or test reactor facility. The methods and requirements for packaging these wastes and shipping them to an authorized disposal site are described in this section.

G.6.1 Packaging of Radioactive Wastes

The radioactive wastes that result from decommissioning can be classified as follows:

- combustible or noncombustible
- activated or contaminated
- wet or dry.

The bulk of the decommissioning wastes from the reference R&T reactors are dry, noncombustible, and either activated or contaminated. They include the activated reactor vessel and core structural components, contaminated concrete from walls and floors, and contaminated piping and equipment. The contact radiation dose rates from these materials vary from a few mrem/hr to thousands of rem/hr. Different types of packaging and shielding are required, depending on the radiation levels involved.

Disposable, 0.28-m^3 (0.33-m^3 burial volume) steel cask liners are used for packaging, shipping, and burying the activated portions of the reactor vessel and its internals. These containers are estimated to cost $450 each. In

some cases, lead shielding must be added at an estimated average cost of \$1.23/kg, including labor and energy.

The bulk of the contaminated material is packaged in a standard container, illustrated in Figure G.6-1. The container is approximately 1.2 m by 1.2 m by 2.4 m, made from plywood, and internally reinforced and sealed with fiberglass to ensure confinement of loose contamination. Containers of this type are estimated to cost \$400 each. When necessary, containers of special sizes and shapes are fabricated. Some items do not require shielding but are larger and/or heavier than can be accommodated with plywood containers. In these cases, steel containers are specially fabricated at an average cost of about \$6,000 each.

Some contaminated equipment items (e.g., small heat exchangers and small tanks) are packaged by capping the piping connections with welded metal covers and using the item's outer shells as the containers.

Dry, combustible radioactive wastes include filter cartridges from liquid streams, HEPA ventilation filters, and miscellaneous cleaning mterials (e.g., rags, mops, anti-contamination clothing, and plastic sheeting and bags). These materials are compacted as necessry and packaged in 0.21-m^3 steel drums costing \$25 each.

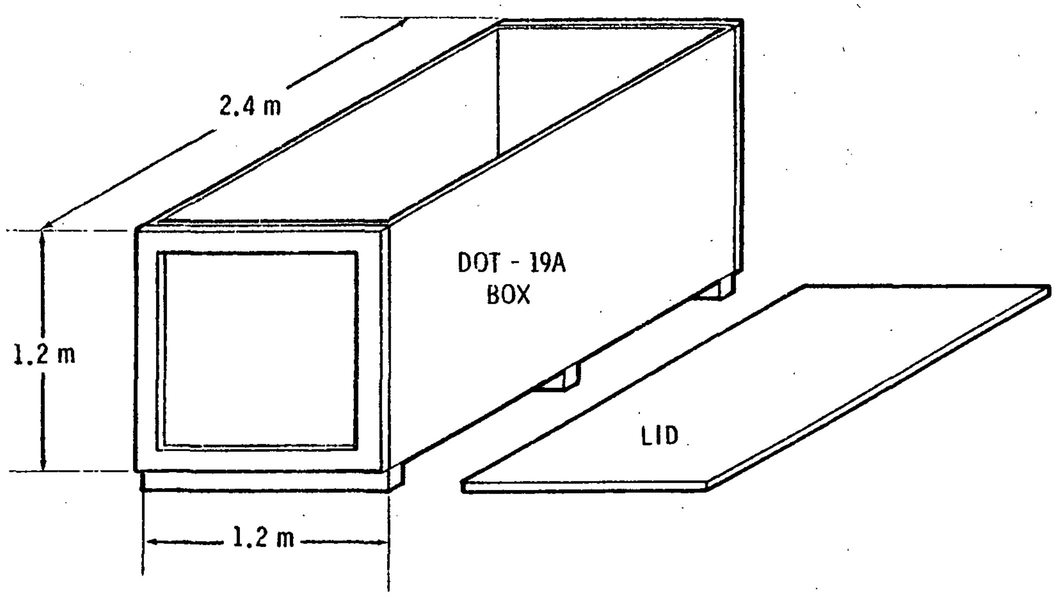

FIGURE G.6-1. Typical Plywood Shipping Container

Standard methods are in place for handling and disposing of the radioactive liquid wastes that are generated during decommissioning of the reference R&T reactors. The assumed practice of liquid waste treatment is through the process of ion exchange. Ion exchangers at the reference test reactor are augmented, when needed, with a low-capacity (<0.02 m^3/hr) evaporator unit. Dewatered, spent ion-exchange resins are solidified in containers prior to disposal. For this study, the containers are assumed to be 0.21-m^3 steel drums.

G.6.2 Shipment of Radioactive Wastes

All radioactive waste materials removed during decommissioning from the reference R&T reactors are assumed to be shipped in exclusive-use trucks to an authorized disposal site, via the interstate highway system. The hauler is assumed to have the appropriate NRC license and DOT permits before he handles the radioactive waste material. Several commercial transportation companies have the special capabilities and trained drivers for hauling radioactive wastes.

All shipments of radioactive material must be made in compliance with federal, state, and local regulations. Federal (DOT and NRC) transportation regulations establish container specifications, dose-rate limits, and handling procedures to ensure the safety of the public and the transportation workers during shipment of radioactive materials.[25] In addition, for highway transport, state agencies regulate vehicle sizes and weights and, in some cases, transportation routes and times of travel.

Dose rates for highway shipments in exclusive-use vehicles must not exceed the following values (49 CFR 173.393):[26]

- 1000 mrem/hr at 0.91 m from the external surface of a package shipped inside a closed vehicle
- 200 mrem/hr at any point on the external surface of a closed vehicle or an exposed shipping container (e.g., a shielding cask)
- 10 mrem/hr at 2 m from the external surface of the vehicle
- 2 mrem/hr at any normally occupied position in the vehicle.

These dose rate limits are illustrated in Figure G.6-2 for closed truck transport.[27] All of these criteria must be met for each shipment.

Some of the activated waste packages will exceed the package surface dose-rate limit, as will some of the solidified wet waste packages and dry combustible waste packages. These must be shipped in nondisposable shielded shipping containers, some typical examples of which are shown in Figure G.6-3.

All wastes are assumed to be transported to licensed commercial low-level waste burial grounds such as those currently available in Washington, Nevada, and South Carolina. For this study, the average one-way shipping distance is assumed to be 800 km for all shipments. Disposal site restrictions are assumed to limit the maximum quantity of radioactivity per shipment to 50,000 Ci. There are currently no limits on the dose rate at the surface of a disposable container, but burial costs increase with increasing surface dose rate.

A formal accident control and recovery plan is assumed to be developed before the first radioactive shipment of decommissioning waste is made. The

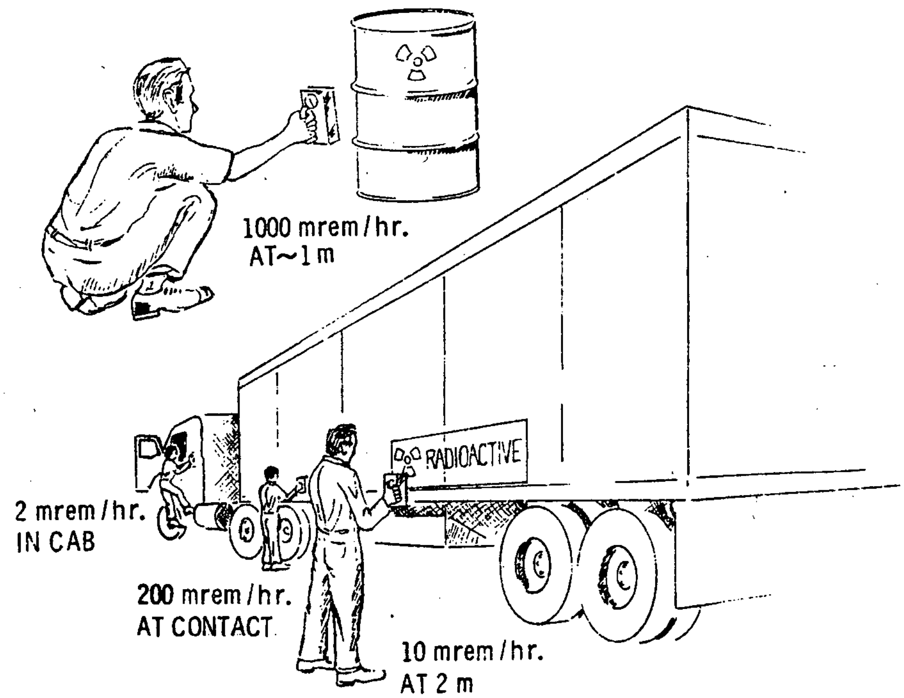

FIGURE G.6-2. Radiation Dose Limits for Closed Exclusive-Use Vehicles[27]

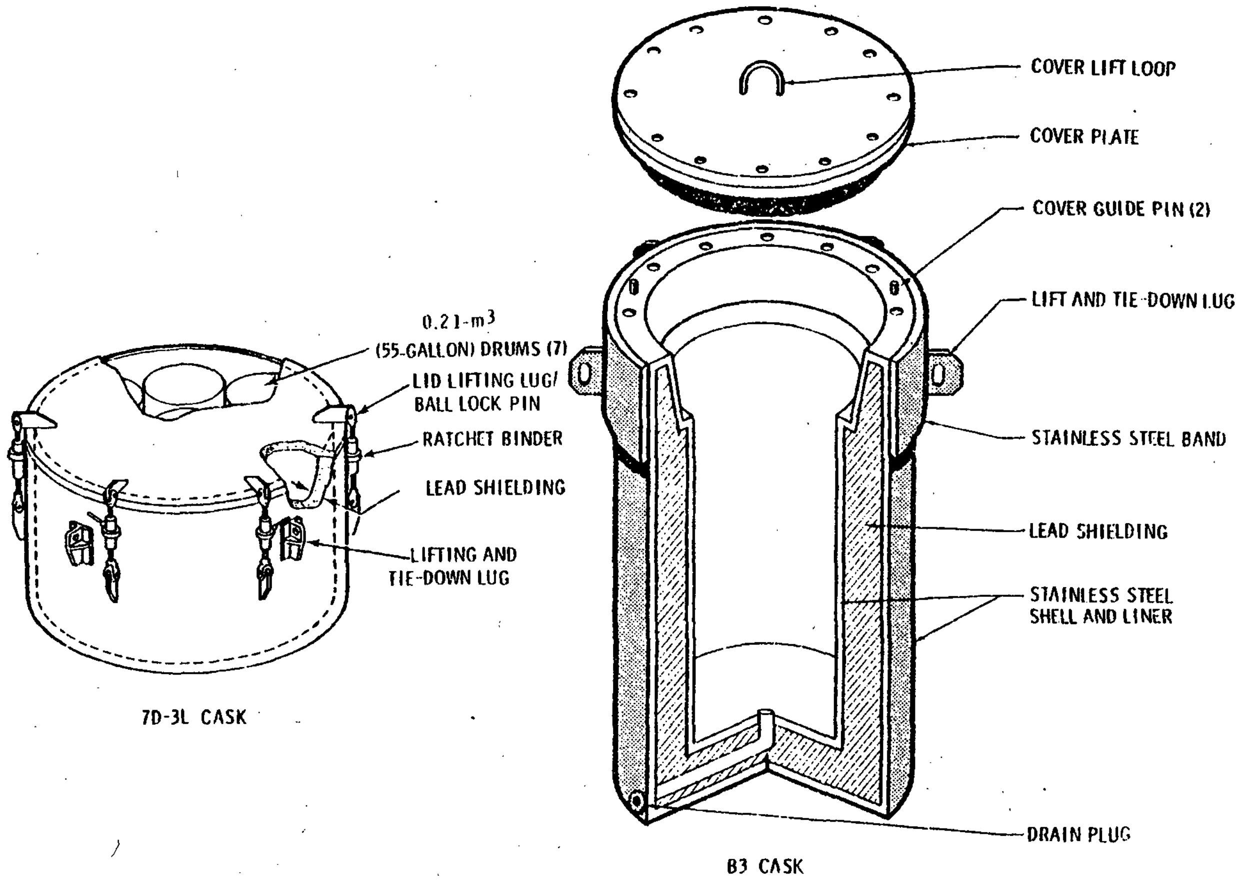

FIGURE G.6-3. Typical Shielded Shipping Containers

plan is to provide for rapid and orderly utilization of owner, carrier, state, and municipal emergency personnel, as well as NRC radiological assistance teams, as required in the event that any transportation accident occurs. Procedures for control of contamination, radiation exposure, bodily injury, and property damage are included in the recovery plan. Also included are procedures for salvage and recovery of the radioactive shipment.

G.7 QUALITY ASSURANCE

A reactor decommissioning project requires quality assurance (QA) planning from the earliest stages to meet the QA regulations imposed upon it. As each detailed procedure is developed, the QA portions are included. Current regulations and guides that could apply to decommissioning are discussed in Section 6.

G.7.1 Method Review

When a new procedure is written, a documented peer review is held. The review ensures that adequate consideration is given to physics, radiation, safety, accident, security, thermal, stress, QA/quality control, and hydraulic analyses. Other checks, tests, and formal design reviews may be required. Additional aspects that are reviewed include reliability criteria, the choice of test methods, and the choice of processes or materials. The conservatism in margins of safety established for procedures, equipment, or structures also receives documented review.

G.7.2 Procurement Document Control

The project's suppliers, including fixed-price contractors and shipping cask vendors, are selected from a QA list of qualified suppliers. QA inspections of a vendor's plant are performed as required to place new firms on the qualified list.

Procurement document control verifies that any QA requirements for specified hardware, materials, or services are clearly and accurately identified in procurement documents or purchase orders. This includes references to applicable drawings, specifications, standards, codes, and regulations; required records of test data, certifications, and qualifications of personnel and procedures; and special handling, shipping, or packaging requirements. This control is applied to any supplier of the specific devices and equipment vital to the project, such as environmental monitoring equipment, safety and shielding devices, radiation detection equipment, chemicals, and explosives.

G.7.3 Work Instructions, Procedures, and Drawings

Formal documentation of work instructions and procedures, drawings, and information management procedures is required, with responsible personnel clearly identified. These procedures verify the proper completion of activities or steps in the decommissioning process (e.g., the disassembly sequences that ensure that vital services remain intact). The latest revisions of all such documentation are required to be at the work location. Changes to such documents receive the same levels of review and approval as original planning.

G.7.4 Document Control

Document Control verifies that all essential documentation receives the proper review, approval, release, change, and distribution control. An established procedure is used to eliminate obsolete or erroneous information.

G.7.5 Identification and Control of Procured Items

Inspection of purchased materials or items is performed at the time of receipt. Items must be sufficiently identified to allow proper selection for use, completion of processing, or segregation of nonconforming items. The shelf-lives and the required storage and use conditions of chemicals must be clearly stated.

G.7.6 Special Processes

Decommissioning may involve several tasks requiring new or unusual work procedures (e.g., chemical decontamination work and the use of explosive charges). Special provisions are made to identify and control any unusual processes or procedures, using reviews, written procedures, and audits.

G.7.7 Inspection and Test

All items produced for the project must be properly inspected or verified. Inspections in the vendor's plant may be necessary for some items. Inspection instructions are required, and records of inspections are prepared and retained. Inspection of each essential operation is also required.

G.7.8 Calibration

A system of calibration control is used to ensure the validity of instrument readings. Accuracy requirements and tolerances for project measurements are identified. Equipment with calibration requirements includes radiation detection devices and environmental monitoring systems. Calibration procedures for instruments that monitor and control vital services are reviewed and revised as necessary.

G.7.9 Shipping and Receiving

Special instructions are prepared for the packaging, shipping, and receiving of components and materials for the project. Included are radioactive

shipments requiring QA procedures for each package type specifying all steps in the package inspection, preparation, loading, and closing. Signed checklists are used. Seals must be carefully verified on internally contaminated units having no additional outside packaging.

G.7.10 Operating Status

Identification procedures are used to prevent inadvertent operation of systems temporarily or permanently removed from service. Restricted use may be necessary for equipment (such as pumps) that have not been fully removed from service.

G.7.11 Nonconformance and Corrective Action

All nonconformances and resulting corrective actions are documented and the reports retained as part of the project records.

G.7.12 Special Quality Records

An index of records and their required retention periods is maintained. These records include operating logs; results of reviews, inspections, tests, audits, material analyses, and work-performance monitoring; orientation, training, and qualifications of personnel; procedures; drawings and specifications; criteria documents; procurement documents; shipping records; and nonconformance and corrective action reports. Extensive documentation of dose rate surveys is also included.

G.7.13 Quality Auditing

Regular audits verify that each QA procedure is being followed. Examples include witnessing radioactive shipments, auditing calibrations, verifying that up-to-date procedures are at all work locations, and reviewing inspection records.

G.8 ENVIRONMENTAL SURVEILLANCE

The required levels of environmental surveillance for active decommissioning differ from those for continuing care. The environmental surveillance programs for both active decommissioning and continuing care follow.

G.8.1 Environmental Surveillance During Active Decommissioning

The environmental surveillance program for the operating reactor can be revised for active decommissioning activities. The following objectives are relevant:[28]

- detection of sudden changes and evaluation of long-term trends of radionuclide concentration in the environment, with the intent to detect failure to adequately control releases and then to initiate appropriate actions
- assessment of the actual or potential exposure of man to radioactive materials or radiation present in his environment, or estimation of the probable upper limits of such exposure
- determination of the fate of contaminants released to the environment, especially with the intent of detecting previously unconsidered mechanisms of exposure
- demonstration of compliance with applicable regulations and legal requirements concerning releases to the environment.

Methods, procedures, and performance criteria for environmental surveillance at nuclear facilities are discussed in detail in Reference 29.

Basic radiation-exposure and radioactive effluent-release criteria applicable to population exposure are given in Title 10 of the Code of Federal Regulations.[26] For non-radioactive contaminants, consideration must be given to applicable standards such as water quality criteria and the EPA's ambient air quality standards.[30,31] In any case, local or state air quality criteria presumably apply on a site-specific basis.

In addition, the interfaces (of the environmental monitoring program) between the plant owner, the appropriate state agencies, and the EPA should be stipulated in the application for the amended license.

A suggested minimum program of environmental radiological surveillance, conducted outside the plant for the purpose of establishing population dose, is shown in Table G.8-1. This minimum program continues until all fuel and source material is shipped from the site.

TABLE G.8-1. Basic Environmental Monitoring Program for the Active Decommissioning Period

Sample Type	Frequency	Number of Sampling Stations Onsite	Offsite	Analysis	Analytical Detection Limit[a]
Terrestrial Samples					
Air Particulate	Weekly	2	3	Gross Beta	0.002 pCi/m^3
				Gross Alpha	0.002 pCi/m^3
				Gamma Scan[b]	0.3 pCi/m^3/isotope
Air Radioiodine	Weekly[c]	2	3	^{131}I	0.1 pCi/m^3
Direct Radiation	Quarterly	4	5	TLD[d]	1.25 mrem/quarter increase
Rainfall	Monthly	1	2	Gross Beta	0.5 pCi/ℓ
				Tritium	1000 pCi/ℓ
				Gamma Scan[e]	26 pCi/ℓ/isotope
Soil	Semiannually	2	2	^{89}Sr, ^{90}Sr	0.01 pCi/g (dry)
				Gamma Scan	0.1 p Ci/g/isotope (dry)
Vegetation	Semiannually	1	3	^{89}Sr, ^{90}Sr	5.0 pCi/kg (wet)
				Gamma Scan	50 p Ci/kg/isotope (wet)
Milk	Monthly	---[f]	3	^{89}Sr, ^{90}Sr	1.0 pCi/ℓ
				^{131}I	0.5 pCi/ℓ
				Gamma Scan	50 pCi/ℓ/isotope
Aquatic Samples					
Surface Water	Monthly	2	2	Gross Beta	0.5 pCi/ℓ
				Tritium	1000 pCi/ℓ
				Gamma Scan	25 pCi/ℓ/isotope
Bottom Sediment	Semiannually	---	2	Gamma Scan	0.1 pCi/g/isotope (dry)
Fish	Semiannually	---	3	^{89}Sr, ^{90}Sr	5 pCi/kg (wet)
				Gamma Scan	100 pCi/kg/isotope (wet)

(a) Analytical detection limit is that concentration that is three standard deviations above the average concentration in a blank sample, ensures accuracy of ±25%.
(b) Performed if gross beta exceeds 0.1 pCi/m^3.
(c) ^{131}I analyses only for first month after shutdown.
(d) Thermoluminescent dosimeter.
(e) Performed if gross beta exceeds 10 pCi/ℓ.
(f) Indicates no sample taken.

The analytical detection limits given in the table are based on practicable routine radioactivity-measurement techniques, and in all cases should be sufficient to quantify radionuclide concentrations resulting in conservatively estimated whole-body and individual-organ doses of <50 mrem/year, depending upon the values derived for the disposition criteria for specific nuclides. These nuclides are identified on completion of the operational monitoring

program. They are then defined in terms of concentrations in environmental media at levels that, if sustained, result in doses exceeding acceptable levels for unrestricted public use. The derivation of these levels corresponds with the site-specific method of assessing offsite doses from radioactive materials in estimated gaseous and liquid effluents (if any), and is assumed to be consistent with the recommendations of the Federal Radiation Council and the International Commission on Radiological Protection.[32,33] The monitoring program is designed to integrate fully with any ongoing programs of the state where the reactor is located. Sample collections and radiation measurements required to meet the schedule suggested in Table G.8-1 are assumed to be conducted according to specifications established during the operation of the reference reactor. The program, and any changes thereto, are documented with the NRC (and with other appropriate regulatory agencies) as environmental technical specifications, in accordance with the terms of the ammended license.

G.8.2 Environmental Surveillance During Continuing Care

An abbreviated version of the environmental monitoring program conducted during plant operation is carried out during continuing care. The purpose of the program is to identify and quantify releases of radioactivity to the environment. There is no intent to provide a surveillance program adequate for all potential nonroutine or accident releases, although the proposed program is useful in evaluating most lapses of control. For emergency situations involving radionuclide releases (from events such as fire or malicious acts) requiring prompt emergency actions to minimize public risk, special surveillance requirements would apply.

Changes in background radiation levels, in environmental radiation accumulations (e.g., fallout from nuclear weapons testing), and especially in land usage and population distribution may, over a period of years, justify modifications in the post-decommissioning surveillance program. Experience will indicate, and the owner should be alert for, trends in environmental results that would reasonably permit reduction of the programs.

The entire program is anticipated to be reviewed at the following post-decommissioning milestones:

- after all fuel and source material are shipped from the site
- ten years after plant shutdown
- after 10 half-lives of ^{60}Co decay (approximately 53 years), economic advantages of further decommissioning effort are ascertained by the licensee, and environmental monitoring could conceivably be eliminated.

The suggested minimum program of environmental surveillance shown in Table G.8-1 continues until all fuel and source material is shipped from the site. It can then be reduced in scope. As experience is gained during the continuing care period, further reductions and/or changes are likely.

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APPENDIX H

GENERIC DECOMMISSIONING INFORMATION

While every reactor decommissioning project is somewhat unique due to the individual characteristics of the particular plant, there are many aspects that are common to all reactor decommissioning projects. For example, a fairly detailed listing of the many functional responsibilities of the licensee during decommissioning is given in Table H.0-1. To satisfy these responsibilities, the licensee must proceed in a logical sequence with development of the following considerations:

- planning and preparation
- decommissioning staff
- essential systems and services
- specialty contractors.

Each of these areas is discussed for the reference research and test (R&T) reactors in subsequent sections of this appendix.

H.1 PLANNING AND PREPARATION

Effective planning and preparation is vital to successful completion of decommissioning activities at a nuclear facility. The safety and cost effectiveness of the project could be compromised if planning and preparation is inadequate. Ideally, planning and preparation is scheduled to be completed by the time the reactor is shut down; however, R&T reactor programs have frequently been terminated with little advance notice so that planning and preparation for decommissioning the facilities could not be completed by the time of reactor shutdown. For licensed, government-owned reactors, rapid termination of the test program may virtually rule out DECON as a viable decommissioning alternative since decommissioning funds must be obtained by the operating agency by preparing a budget request and securing approval of the request via the normal channels used to obtain operating funds. Budget requests need to be prepared well in advance of the planned date of decommissioning to allow adequate time

TABLE H.0-1. Summary of the Functional Responsibilities of the Licensee[a]

Predecommissioning	During Decommissioning
Establish a decommissioning activities team.	Operate and maintain the support systems.
Provide technical and design development.	Prepare detailed procedures of the decommissioning activities to ensure safe accomplishment of work.
Perform facility assessment of support systems, radioactivity, and structures.	Perform the facility decontamination and dismantling operations.
Prepare environmental assessment, decommissioning plan, and safety analysis of the decommissioning activities.	Provide safety and health physics training to all workers to ensure safe operations.
Obtain license revisions and dismantling order (if required) from the NRC.	Provide staffing for all functions.
Perform preliminary engineering review of all activities.	Provide equipment and supplies necessary to implement the program.
Prepare management plan.	Perform all other decontamination efforts.
Prepare engineering and activity specifications, design limits, safety standards, system operating limits and sequence of removal operations.	Perform all waste disposal functions.
	Provide radioactive laundry services.
Prepare support systems reactivation criteria (only for deferred decontamination).	Provide personnel monitoring and facility, station, and environmental monitoring.
Prepare decommissioning requirements for access to contaminated equipment.	Provide janitorial services.
Develop funding profile and schedule.	Provide security visitor control, plant protection, and damage control services.
Determine disposition of equipment.	Provide industrial safety, first aid, industrial hygiene, and health physics services.
Administer health and safety and health physics programs: implement the policy to maintain radiation exposures as low as is reasonably achievable (ALARA) in the health physics program.	Review quality assurance measurement results on a periodic basis and take appropriate corrective actions.
Administer quality assurance program, including an effective measurement system to determine the degree of success achieved with regard to program goals.	Stop work whenever radiological or industrial safety considerations require corrective action to safeguard personnel or the environment.
Specify design and drafting tasks for support systems (e.g., water, fire protection, electrical power, lighting, heating and ventilating, and decontamination and waste handling).	Provide periodic reports of progress to the licensee headquarters.
	Administer a safe work permit program.
Provide safety review of all aspects of decommissioning by the decommissioning safety committee, the radiation safety officer, and management.	Provide a final radiological survey and report following decommissioning.
Ensure that the resources needed to achieve successful completion of the program, including ALARA, are available.	

(a) It should be recognized that although the licensee is, in all cases, ultimately responsible for the functions listed, many of the functions and/or services (e.g., laundry, janitorial, health physics, etc.) may be subcontracted by the licensee to others who specialize in a particular area. This is determined on a case-by-case basis for optimum cost-effectiveness of the total decommissioning project.

for the approval process. Because a budget request is often initiated 2 or 3 years before the actual expenditure of the funds, it would be necessary to make adequate provision for cost escalation and inflation.

For the conceptual decommissioning of the reference research reactor analyzed in this study, the planning and preparation phase is postulated to require 1 year for either DECON, SAFSTOR, or ENTOMB. For the conceptual decommissioning of the reference test reactor analyzed in this study, the planning and preparation phase is postulated to span 2 years for either DECON or ENTOMB and 18 months for SAFSTOR. In either case, the time required for this phase may be heavily influenced by the time it takes to obtain an amended license and an NRC dismantling order. Historically, however, this has not been a problem for R&T reactors.

Planning and preparation includes the following concerns:

- regulatory requirements
- data gathering and analyses
- development of detailed work plans and procedures
- design, procurement, and testing of special equipment
- staff selection and training
- selection of specialty contractors
- removal of accumulated spent fuel and unneeded spent fuel storage racks.

These concerns are discussed in the following subsections.

H.1.1 Regulatory Requirements

Prior to the start of actual decommissioning tasks, the licensee must comply with applicable regulatory requirements. The current status of such requirements is detailed in Section 6 of Volume 1. This subsection summarizes the activities undertaken during planning and preparation to meet these requirements.

The major requirement is to provide the necessary documentation for amending the facility operating license to restrict the licensee to possess but not operate the facility and, if required, for obtaining an NRC dismantling order. In requesting an amended license, the licensee must provide:

- a description of the current facility status
- an inventory of the onsite radioactive materials
- a description of the proposed decommissioning activities
- a description of the proposed measures to prevent criticality and to minimize radioactive releases
- any proposed changes to the technical specifications (e.g., deletion of specifications relating solely to the plant operation)
- safety analyses of both the proposed activities and the specification changes.

This information becomes the decommissioning plan for SAFSTOR.

In addition to the aforementioned requirements for an amended license, an NRC dismantling order is required for DECON or ENTOMB. A request for such an order must include a decommissioning plan providing:

- a description of the ultimate facility status
- a description of the decommissioning activities (including radioactive material disposal and site decontamination) and the associated environmental and safety precautions
- a safety analysis of the decommissioning and any resultant releases
- a safety analysis of the plant in its ultimate status.

In addition, the licensee must submit a radioactive waste handling plan, a quality assurance plan, an environmental report, and security and safeguards plans. Updated information concerning the financial qualification of the licensee may also be required. (See Section 6.2 of Volume 1 for further details.)

H.1.2 Data Gathering and Analyses

A large body of data is gathered and analyzed during the planning and preparation phase of decommissioning. These data provide the bases for satisfying the regulatory requirements discussed above, particularly the inventory of radioactive materials and the various safety analyses. They also provide the bases for planning decommissioning tasks and selecting appropriate methods and equipment.

Included in this activity is a comprehensive survey of radiation dose rates and contamination levels within the facility. This survey, taken after final reactor shutdown, provides information for determining the need for decontamination and temporary shielding. It also provides initial data on radiation dose rates likely to be encountered during the various decommissioning tasks.

H.1.3 Development of Detailed Work Plans and Procedures

Detailed work plans and procedures are developed and provided to the NRC with the license amendment/dismantling order request, based on the information developed during data gathering and resultant analyses. These detailed plans and procedures contain all the information required to actually carry out the decommissioning tasks. They address the following items:

- decommissioning methods
- schedules and sequences of events
- radioactive waste management
- contamination control
- radiological and industrial safety
- equipment requirements.

Quality assurance, security, and environmental constraints are also considered. The plans and procedures cover all aspects of the decommissioning project.

H.1.4 Design, Procurement, and Testing of Special Equipment

Any special equipment required to complete the decommissioning project is identified during planning and preparation. Designs and specifications are prepared for each item required. When the item is procured, it is inspected to verify that it meets specifications and complies with applicable quality

assurance and safety requirements. It is then tested to ensure that it performs as required. The testing also serves to train personnel in the use of the equipment and to provide pertinent data on its operation.

H.1.5 Staff Selection and Training

At the start of planning and preparation, a decommissioning organization is created for the facility. Staff requirements are identified, and critical positions are filled with key engineering and operating personnel. The personnel are trained as required to fulfill their roles in the organization; special emphasis is given to the use of new and unique equipment and procedures. The decommissioning staff is discussed in detail in Section H.2.

H.1.6 Selection of Specialty Contractors

During planning and preparation, the decommissioning staff identifies the needs for any specialty contractors required to decommission the facility. These contractors perform unique services outside of the expertise or capability of the staff. After the needs are identified, contractors are invited to bid on the required work packages. Contractual agreements are concluded prior to the start of the actual decommissioning, if possible, to ensure the uninterrupted completion of the project. Specialty contractor requirements are discussed in Section H.4.

H.1.7 Removal of Accumulated Spent Fuel and Unneeded Spent Fuel Storage Racks

Any spent fuel stored from previous research and testing activities is removed from the spent fuel storage pool(s) and shipped to a repository, to another licensee,[a] or to a reprocessing facility. In addition, for DECON and ENTOMB, the spent fuel storage racks in excess of those required for final reactor defueling are removed and shipped offsite. By removing these excess items prior to the start of the actual decommissioning, extra space can be made available in the spent fuel storage pool(s) for the interim storage and packaging of activated materials removed from the R&T facilities, as required.

(a) For this alternative, either the receiver's license must already stipulate that he can legally accept the fuel assemblies (or other nuclear materials) or he must apply for an amendment to his existing license to receive, store, and/or use the additional nuclear material.

H.2 DECOMMISSIONING STAFF ORGANIZATIONS

The staff organizations postulated for performing the decommissioning activities at the reference R&T reactors are described in this section. The organization structures for the two types of facilities are somewhat similar but differ in several details because of the business nature of the licensee (university versus corporation or government agency), and because of differences in the physical nature of the individual facilities.

H.2.1 Research Reactor Decommissioning Staff

The postulated organization chart for the decommissioning organization at the reference research reactor is shown in Figure H.2-1. Ultimate responsibility for decommissioning activities rests with the university administration (the licensee). It is postulated that, for decommissioning of the reference research reactor, two staff committees oversee the operations and safety tasks. The operations branch, under a decommissioning superintendent, plans and performs the decommissioning activities while overseeing financial, security, and safety functions. The safety branch, under a health physicist, plans and conducts radiological and industrial safety programs. As shown in Figure H.2-1, the quality assurance supervisor interacts with both the operations and safety personnel while reporting to the staff committees, but he is directly responsible to the university administration.

The crew on the basic working unit includes: a crew leader, a utility operator, a laborer, and the necessary craftsmen and health physics technicians.

To the extent possible, decommissioning staff positions are filled with facility operations and maintenance personnel already familiar with the reference research reactor. In this way, effective and efficient task performance is obtained. Use is made of student labor where knowlegable personnel are available.

The specific crew makeup for a given decommissioning task will be tailored to fit the need. Specific crew assignments are described throughout the appendicies. Specialty contractors and consultants are employed, where necessary, to assist in areas outside the staff's expertise or capability.

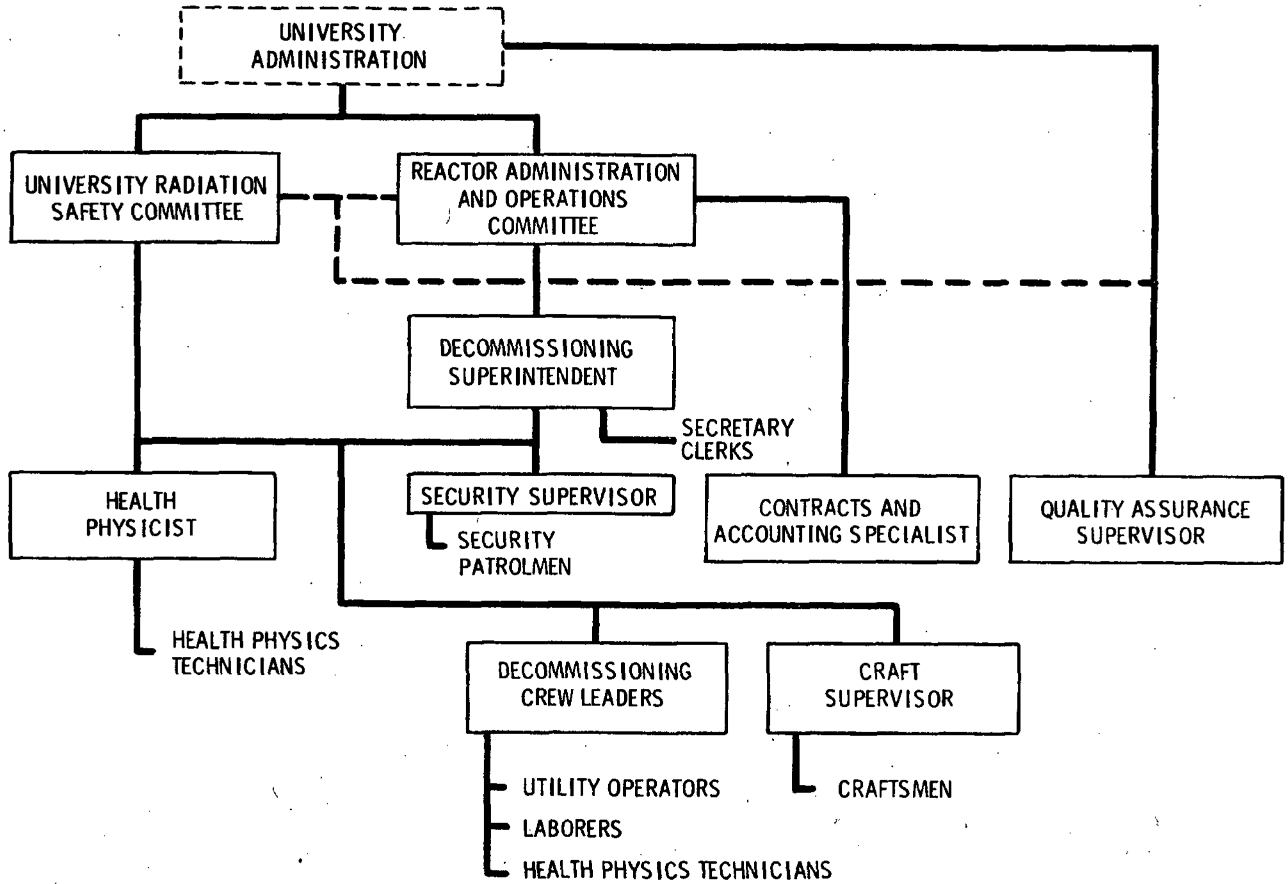


FIGURE H.2-1. Decommissioning Staff Organization for the Reference Research Reactor

The personnel interactions, activities, and responsibilities of key staff members are based on the reference research reactor's current administrative structure and are described below.

Reactor Administration and Operations Committee

This committee advises university administration on matters under its jurisdiction. Its main function is to provide overall planning and direction to the decommissioning superintendent and financial branch while interacting with the other facets of the organization.

Decommissioning Superintendent

This person plans and oversees all day-to-day decommissioning activities. Responsibilities include directing crew leaders, security supervisors, and the health physics branch.

Decommissioning Crew Leader

This individual directs a work crew in the performance of the actual decommissioning tasks.

Craft Supervisor

This person is responsible for maintenance of essential plant equipment and services as well as for assigning craft labor to particular decommissioning tasks. He instructs craftsman in their assigned tasks and ensures the availability of tools and supplies.

Security Supervisor

This person is responsible for site security during decommissioning. This includes supervising the security personnel and, if necessary, providing liaison with offsite civil authorities. The security shift supervisor directs shift activities.

Contracts and Accounting Specialist

An experienced accountant, this individual is responsible for the financial aspects of the project. He prepares procurement documents and contracts and, with approval from the reactor administration and operations committee, disburses funds. Responsibilities include the maintenance of up-to-date financial accounts, while providing the committee with regular summary reports.

Quality Assurance Supervisor

Responsible for preparing and implementing the quality assurance plan for decommissioning, this person works with all branches of the organization to implement the plan. To ensure the independence of the quality assurance program, he reports directly to the university administration. He supervises a quality assurance unit, which maintains audit and job performance records and verifies that established safety review procedures are followed. (See Section G.7 of Appendix G for further discussion of quality assurance functions.)

University Radiation Safety Committee

This committee advises university administration on matters of radiological and industrial safety. It provides overall planning and direction to the

health physicist and interacts with the decommissioning superintendent on matters of safety. Coordination is made with the reactor administration and operations committee on interrelated matters.

Health Physicist

This person recommends and enforces safety policy, both radiological and industrial. Responsibilities include maintenance of radiation exposure records, implementation of the environmental survey program, ensuring compliance with work procedures, and training and assigning health physics technicians to specific work tasks.

H.2.2 Test Reactor Decommissioning Staff

For the reference test reactor, the decommissioning staff is organized as shown in Figure H.2-2, and has five branches under a decommissioning superintendent. The project engineering branch, under a decommissioning engineer, develops detailed procedures of the decommissioning activities and performs the actual decommissioning activities. The support services branch provides craftsmen who assist the decommissioning crew leaders and perform plant maintenance as required. Support services also provides security patrolmen for plant security. The radiological services branch plans and conducts both radiological and industrial safety programs. The quality assurance branch maintains audit and job performance records and verifies that established safety review procedures are followed. The financial branch is responsible for the financial aspects of the project.

The basic working unit is the shift, which is supervised by a shift engineer. The crew on each shift includes: a crew leader (typically a reactor operator), utility operators, and laborers, plus craftsmen (e.g., welders, pipefitters, electricians, and air-balance technicians) and health physics technicians assigned as needed. Craftsmen and health physics technicians on the support crews report directly to the crew leaders because, on the third shift and on weekends, crew leaders are the only supervisory personnel on plant. Craftsmen and health physics technicians assigned to the regular decommissioning crews report to the crafts supervisor and the senior health physics technician on the day and swing shifts, respectively.

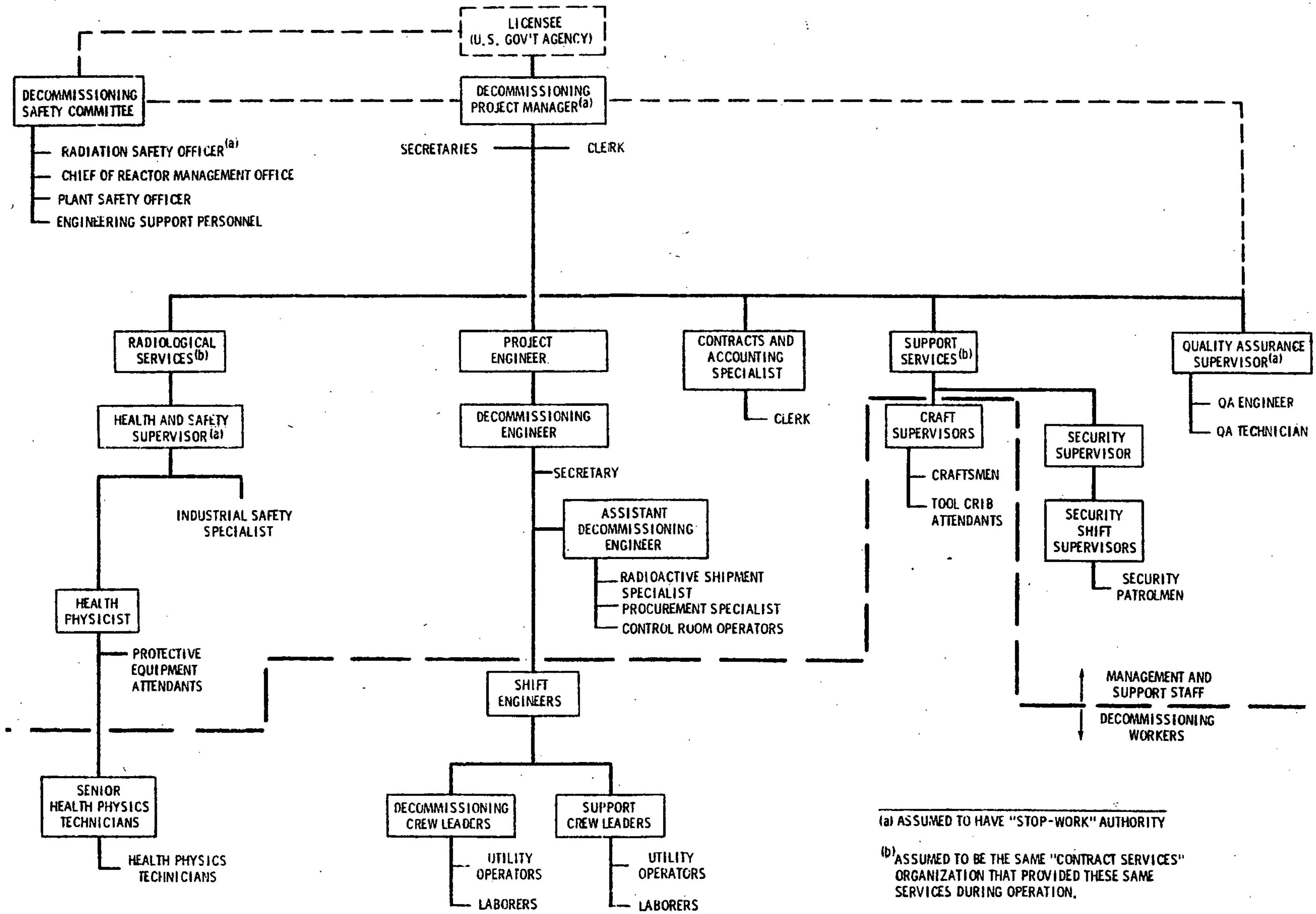


FIGURE H.2-2. Decommissioning Staff Organization for the Reference Test Reactor

The number and specific makeup of crews required for a given decommissioning alternative are discussed in the specific appendix detailing that alternative (Appendix I, J, or K).

Detailed knowledge of and familiarity with the reference test reactor increases the effectiveness of the decommissioning staff. Consequently, staff positions are filled with facility operations and maintenance personnel to the maximum extent possible. Specialty contractors and consultants are hired as needed to assist in areas outside the staff's expertise or capability.

It is postulated that an outside consultant, a registered architect, is required for those alternatives (SAFSTOR and ENTOMB) that involve a delayed decontamination phase. This person is responsible for developing a long-range planned maintenance program, based on a visual inspection and a review of construction drawings.

In general, hot cell operations at the reference test reactor are conducted by specialists. These same specialists should be retained for both the planning and preparation phase and the operational phase of decommissioning the hot cells. Their special operative talents should prove invaluable and cost-effective in the actual hot cell decontamination and dismantling activities.

Key decommissioning staff members perform the functions described below.

Decommissioning Superintendent

Directly responsible to management, the superintendent coordinates and oversees all decommissioning activities. He directs the decommissioning engineer and the health and safety supervisor, as well as support services (security, craftsmen), quality assurance, and contracts and accounting, to ensure the safety and cost-effectiveness of the decommissioning project. He provides necessary liaison with regulatory agencies and management.

Decommissioning Engineer

This individual plans, coordinates, and supervises the actual decommissioning tasks. He provides the engineering services and detailed procedures required to carry out the decommissioning plan in a safe and cost-effective manner. He prepares all routine and special reports as well as a chronological history of the project.

Assistant Decommissioning Engineer

This person supervises the decommissioning support personnel and assists the decommissioning engineer in developing detailed work procedures. He writes specifications for special equipment and tools that must be procured or fabricated. He also prepares reports requested by the decommissioning engineer.

Shift Engineer

Responsible for carrying out the actual decommissioning work during a shift, this person supervises the crew leader and craft supervisor. He reports to the decommissioning engineer. As he supervises the day-to-day performance of the shift, he recommends changes in procedures and schedules to improve the safety and/or cost-effectiveness of the project.

Crew Leader

Reporting to the shift engineer, this individual directs the work crews in the performance of the actual decommissioning tasks.

Craft Supervisor

This person is responsible for maintenance of essential plant equipment and services as well as for assigning craft labor to particular decommissioning tasks. He instructs craftsmen in their assigned tasks and ensures the availability of required tools and supplies.

Security Supervisor

This person is responsible for site security during decommissioning. He supervises the security personnel and, if necessary, provides liaison with off-site civil authorities. The security shift supervisor directs shift activities.

Contracts and Accounting Specialist

An experienced accountant, this individual is responsible for the financial aspects of the project. He prepares procurement documents and contracts and, with approval from the decommissioning superintendent and the decommissioning engineer, disburses funds. He maintains up-to-date financial accounts and provides the decommissioning superintendent with regular summary reports.

Health and Safety Supervisor

This person (typically a senior health physicist) recommends and enforces safety policy, both radiological and industrial. He advises the decommissioning superintendent on all safety matters. He maintains the occupational radiation exposure records, and also develops and implements the environmental survey (via a specialty contractor) and the emergency preparedness programs. He supervises and is assisted by the industrial safety specialist and the health physicist.

Health Physicist

This individual is responsible for ensuring compliance with radiation work procedures. He directs the activities of the health physics technicians who monitor all decommissioning activities, measure and record on-the-job radiation dose information, and operate the plant laboratory facilities, including sampling and analysis. The senior health physics technician assigns and trains the others on the shift.

Quality Assurance Supervisor

This person is responsible for preparing the quality assurance plan for decommissioning and works with the decommissioning engineer to implement it. To ensure the independence of the quality assurance program, he reports directly to corporate headquarters. He supervises a quality assurance unit, which maintains audit and job performance records and verifies that established safety review procedures are followed. (See Section G.7 of Appendix G for further discussion of quality assurance functions.)

H.3 ESSENTIAL SYSTEMS AND SERVICES

All or parts of certain facility systems and services must remain in place and in service until all radioactive material is either removed from the facility or secured on the site, to prevent the release of significant quantities of radionuclides (or other hazardous materials) to the environment. Some systems and services are required for cleanup and disassembly activities. Other

systems provide personnel health and safety protection. The required systems and services are listed in Table H.3-1, together with the justification for retaining each.

As areas within the reference R&T reactor facilities are readied for demolition or secured for storage, the essential systems and services in these areas are deactivated and, if contaminated, removed as required. Continuous service to the remaining work areas is maintained as long as necessary.

TABLE H.3-1. Systems and Services Required During Decommissioning

System or Service	Justification
Normal and Emergency Electric Power	Operation of electrical equipment including HVAC, lighting, and radiation monitoring
HVAC Systems	Ventilation and confinement of radioactive contamination
Demineralized Water System	Maintain purity of reactor tank water during defueling and reactor vessel/internals removal
Service Water System	Decontamination, cleanup, fire protection, and potable water
Compressed Air Systems (control and service)	Operation of pneumatic controls and tools; personnel fresh air supply
Communications Systems	Facilitate and coordinate decommissioning activities
Radwaste Systems	Treatment of radioactive liquids, solids, and gases
Fire Protection System	Health and safety
Security Systems considerations	Public safety and plant protection
Radiation Monitoring System	Personnel safety
Anti-Contamination Clothing Laundry Facilities	Health and safety

H.4 SPECIALTY CONTRACTORS

During decommissioning, specialty contractors are employed to provide services beyond the capability of the research or test reactor's decommissioning staff. Use of these contractors increases the overall cost-effectiveness of the project by improving the efficiency of specialty operations and reducing the need for specialized staff training. In addition, specialized experience gained from similar projects is directly applied to the decommissioning by these contractors, eliminating the mistakes and wasted effort inherent in learn-as-you-go situations.

If salvage of special hot cell components is to be attempted, proper equipment, operated by personnel skilled in its use, should be used for the decontamination and extraction process. For example, technique is extremely important when removing the hot cell shielding windows. Also, the surfaces of shielding slabs are easily scratched, chipped, etched or stained. Currently, several vendors of hot cell equipment are available to assist in the decontamination of hot cells.

The specialty contractors used during decontamination of the reference R&T reactors are:

- environmental monitoring specialists to implement the environmental surveillance program outlined in Appendix G
- excavation contractors to remove buried tanks and piping
- explosive specialists to break up concrete
- hauling contractors to transport packaged radioactive materials to a disposal site
- temporary radwaste handling and solidification support to handle radwaste and perform final cleanup after the installed radwaste handling systems are decontaminated.

If, following decontamination, the research or test facilities are demolished and the sites restored, demolition and landscaping contractors are also required.

APPENDIX I

DETAILS OF DECON

DECON is the decommissioning alternative used to remove the radioactive materials from the reference research and test (R&T) reactors shortly after cessation of facility operations. After the radioactive materials are removed from the reference sites, the facilities can be released for unrestricted use.

The information in this appendix forms the basis for the activities, costs, and occupational radiation doses for DECON at the reference R&T reactors presented in Sections 10, 11, and 12 in Volume 1, respectively.

The first half of this appendix is concerned with DECON at the reference research reactor, while the last half is concerned with DECON at the reference test reactor. The format is the same for both reference reactors; it includes the disassembly methods, schedules and manpower requirements, costs, and external occupational radiation doses. The facility descriptions given in Appendices B, D, and E for the reference research reactor and the facility descriptions given in Appendices C, D, and E for the reference test reactor, respectively, provide the basic information for the development of the tasks, schedules, manpower loadings, costs, and occupational radiation exposure estimates. Additional design details pertinent to dismantlement of specific equipment items come from engineering drawings, manufacturer's data, and onsite inspection and consultation with operating personnel.

I.1. METHODS POSTULATED FOR DECON AT THE REFERENCE RESEARCH REACTOR FACILITY

This section contains detailed information on decommissioning the reference research reactor facility described in Appendix B via the DECON alternative. Unrestricted use of the site is postulated upon completion of the DECON tasks; therefore, all radioactive materials are anticipated to be removed, packaged, and shipped to an authorized radioactive waste disposal site. The information in this appendix forms the bases for the activities, estimated costs, and occupational doses for DECON presented in Sections 10 and 11 of Volume 1.

I.1.1. DECON Details and Supporting Information

After completing the planning and preparations necessary to provide an efficient and orderly transfer from facility operations to facility decommissioning and after reactor shutdown, the decommissioning tasks begin. A comprehensive radiation survey is conducted at each of the buildings. These surveys aid in finalizing plans for decontaminating equipment and systems and also serve to help lower personnel exposures by directing traffic around radiation zones when possible.

Next, a general cleanup is accomplished and a total inventory of equipment is taken to determine its use to the decommissioning project. Following this, the task of discharging and shipping the fuel commences. This time-consuming task is initiated as early as possible since a number of other essential decommissioning tasks must await the removal of fuel from the reactor pool area before proceeding. Following defueling, high priority is given to fuel shipment to a government reprocessor, so that technical specification restrictions associated with the fuel being onsite can be eliminated and security forces can be reduced.

A number of planning and preparation steps must be accomplished before proceeding with decommissioning operations. These steps are described as follows:

1. Responsible management and safety personnel approve the plans and procedures for safety, material handling, decontamination, and emergency contingencies.
2. Disposition is predetermined for all equipment. The equipment can be decontaminated for reuse, sold as scrap, buried in the local landfill dump, partially decontaminated for use at another restricted plant, or shipped to a licensed burial ground for disposal.
3. All ventilation equipment, personnel protection systems, emergency power systems, fire protection systems, and radiation monitoring equipment in the building and onsite are in service and fully functional.
4. All personnel and contractors are adequately trained to perform their jobs.

5. Appropriate occupational safety equipment and continuous air sampling equipment are available during facility disassembly, transfer, and cleanup tasks.

6. Temporary portable cleaning chambers for decontaminating equipment are available (e.g., greenhouse with tank for water and steam rinsing of equipment, washing tanks, degreasers, etc.).

7. Packaging materials and shipping containers are available.

8. All equipment for dismantlement and decontamination operations is available, including contamination control envelopes for personnel and HEPA filtered vacuum pumps and power tools.

9. A comprehensive radiation survey is completed, with all results mapped and used as a basis for each building, room, and area's work plan.

10. The system and procedures for the functions of special nuclear material accountability measurements and measurement control are established.

11. All unneeded process material supplies (e.g., bottled gases, acids, and caustics) are disconnected from the plant and disposed of.

12. A quality assurance plan is in place and personnel are aware of its requirements.

At reactor shutdown most of the radioactive inventory at the reference facility is in the reactor pool associated with the reactor core and internals. A logical pattern for cleanup, decontamination, dismantlement, packaging and shipment is followed, with tasks associated with the reactor scheduled as early as possible. The possibility of recontaminating an area previously cleaned is minimized by decommissioning areas of high-level activity first and low-level areas last. Certain areas within the reactor room are designated for staging materials from other buildings and these must also be cleaned early. Tasks associated with buildings other than the Reactor Building are undertaken early in the schedule and their radioactive materials prepared for shipment in the reactor room area designated for this purpose.

Unneeded equipment is removed first to provide access to the remaining areas. As equipment and piping are disconnected from their positions they are

capped or fitted with seals to prevent spread of internal contamination. Where possible, equipment is removed first, followed by all external pipes, conduits, controls, and readout devices attached to the process equipment. Equipment is disconnected from exhaust ductwork and the open duct end is sealed to preclude contamination spread from back pressures.

Equipment is sealed with plastic, packaged, and removed to the staging area. Equipment that cannot be packaged at its origin is packaged in the staging area. If radiation levels are low enough to allow release for unrestricted use, the parts are packaged and shipped to a local dump. Parts that cannot be cleaned sufficiently for release are packaged in plywood boxes and shipped to a licensed burial ground.

Finally, the ceiling, walls and floors are decontaminated, as necessary, by cleaning or by using surface removal techniques. Where residual contamination persists, concrete is chipped, spalled and vacuumed. Metal surfaces may be water-jet cleaned. It is anticipated that external radiation doses to workers will be low and most decontamination operations will be efficient, hands-on activities. A final survey is conducted prior to releasing the area for unrestricted use.

All DECON activities are conducted in accordance with the decommissioning quality assurance plan (see Section G.7 of Appendix G) and are checked by the quality assurance specialist. Environmental monitoring at the facility is performed by members of the reference University staff. This is simply a continuation of the radiological services provided during facility operation.

I.1.1.1 Decommissioning Methods for DECON at the Reference Research Reactor

The disassembly methods proposed for DECON at the reference research reactor employ techniques that have been successfully used and are described generically in Appendix G.

Reactor Building. At reactor shutdown all of the neutron-activated materials and some of the contaminated materials at the reference reactor are located within the reactor structure. The reactor cooling and cleanup systems are kept operating continuously until all the fuel has been removed from the

reactor and the reactor internals are removed. It is postulated that the reactor internals can be removed, essentially intact and placed in a heavily shielded cask for shipment and disposal at a low-level waste burial site. This technique will significantly reduce the cutting tasks usually postulated for decommissioning larger cores and will result in manpower cost savings and radiation exposure reduction benefits. The reactor vessel is cut into segments that will allow it to also fit into this same large cask and serve to support the core into position for shipment.

An auxiliary water filtration system is utilized for radioactive material removal during underwater cutting operations. In addition, a contamination control envelope is utilized to constrain the spread of contamination during operations above the reactor pool. In addition to its cooling and cleanup capabilities, the normal operating water level is maintained in the reactor pool for personnel shielding purposes during decommissioning activities within the pool. Then, the reactor pool water is drained and decontaminated by passage through a demineralizer bed. The thermalizing and thermal columns are disassembled and the graphite bricks, liners, and other components are packaged for disposal. The associated piping and equipment interconnections with the Heat Exchanger Building are decommissioned and packaged for shipment. Several Reactor Building tasks can then proceed as planned. These tasks include contaminated concrete removal from the reactor structure by spalling techniques, equipment decommissioning, piping and drain removal, and the decommissioning of three fuel storage pits. When tasks that may create airborne dispersion of radioactive materials are completed, the HVAC exhaust system is disassembled and decontaminated.

Completion of DECON tasks for all other buildings and the staging of their radioactive materials in the reactor room takes place in advance of the time necessary for completion of Reactor Building tasks. Therefore, packaging and shipment for disposal of radioactive wastes from all other buildings as well as those from the Reactor Building is a continuing task that ends with the completion of Reactor Building decommissioning.

<u>Other Contaminated Buildings and Areas</u>. There are four other buildings/ areas at the reference research reactor that require DECON activites. Upon

completion of the comprehensive radiation survey and general cleanup, DECON activities commence in these areas as follows:

- Pump House Building - Remove waste fluid retention tank using jackhammer and forklift. Move to staging area using forklift. Remove pumps and piping. Decontaminate walls and floors using chemical decontamination and water-jet techniques.
- Heat Exchanger Building - Remove pumps, piping and equipment. Weld caps on disconnected heat exchanger piping and move to staging area with forklift.
- Annex - Decontaminate Hot Cell utilizing chemical decontamination, water-jet techniques, and concrete spalling techniques as necessary.
- Radiation Center Building - Decommission waste handling and treatment room. Remove sinks, drains, and piping.

The techniques to dismantle the piping and equipment and to remove concrete surfaces in these buildings/areas are essentially the same as those described previously for the Reactor Building.

I.1.1.2 Special Tools and Equipment

Several special tools and pieces of equipment needed for DECON at the reference research reactor are not normally used in routine plant operation. They are anticipated to be made available by purchase or rental. A listing of special tools and equipment needed to complete the DECON tasks is given in Table I.1-1.

I.1.1.3 Summary of Disassembly Methods

A summary of DECON data for the reference research reactor is given in Table I.1-2. In the table, DECON tasks which result in the removal of materials that contain neutron-activated products or which are radioactively contaminated are identified, cutting data are presented, and removal conditions and methods are given.

TABLE I.1-1. Special Tools and Equipment for DECON at the Reference Research Reactor

Item	Function
Underwater Plasma-Arc Torch	Sectioning vessel internals
Underwater Oxyacetylene Torch	Sectioning reactor vessel
Remote Operations Tool	Sectioning within reactor pool
Portable Oxyacetylene Torch	Cutting carbon steel, aluminum, welding carbon steel
Power Hacksaw	Sectioning piping and equipment
Nibbler	Cutting sheet metal
Jackhammer	Breaking concrete
Submersible Pump & Filter System	Auxiliary water cleaning
High-Pressure Water Jet	Surface decontamination of walls and equipment
Wet-Dry Vacuum Cleaner (HEPA filtered)	Removing contaminated debris
Supplied-Air Plastic Suit	Providing respiratory protection for personnel
Forklift	Packaging of contaminated materials and truck loading
Concrete Drill with HEPA Filter	Drilling holes in concrete for spalling
Concrete Surface Spaller	Removing contaminated concrete surfaces
Portable Filtered Ventillation Enclosure	Contamination control during cutting operations
Paint Sprayer	Contamination control during equipment removal
HEPA Filter System	Contamination control

TABLE I.1-2. DECON Removal Data and Methods for the Reference Research Reactor

Location/Task	Estimated Cutting Length (m) or Area (m^2)	Nominal Material Thickness (m)	Estimated Time (hr)	Method
Reactor Building				
Cut Drive & Position Shaft	2.7×10^3 m^2	3×10^{-3}	4.0×10^0	Remote, Plasma Arc Torch.
Cut Tow Chamber Guide Tube Holders	6.6×10^{-4} m^2	6.4×10^{-3}	1.0×10^0	Remote, Plasma Arc Torch.
Disassemble & Section:				
Rabbit & Terminal	1.2×10^{-2} m^2	2.5×10^{-3}	3.0×10^0	Hack Saw & Remote, Plasma Arc Torch.
Central Thimble	2.5×10^{-3} m^2	2.1×10^{-3}	1.0×10^0	Remote Arc Saw.
Ion Chambers	4.4×10^{-3} m^2	3.1×10^{-3}	1.0×10^0	Remote Hack Saw.
Control Rod Guides & Mechanism	5.6×10^0 m	3.1×10^{-3}	2.0×10^0	Hack Saw & Remote Plasma Arc Torch.
Storage Racks	1.1×10^{-3} m^2	1.9×10^{-2}	1.5×10^0	Hack Saw.
Reflector Platform	6.1×10^{-1} m	1.0×10^{-2}	2.0×10^0	Remote, Plasma Arc Torch.
Reactor Vessel	5.2×10^0 m	6.4×10^{-3}	8.0×10^0	
Beam Tubes	4.1×10^{-2} m^2	6.4×10^{-3}	4.5×10^0	Remote Hack Saw.
Remove Contaminated Concrete	1.1×10^3 holes	4.8×10^{-3} m^3 Waste Volume	1.6×10^2	Drill Holes for Removal of 0.05-m Contaminated Depth.
	9.1×10^1 m^2 Surface Area	4.6×10^0 m^3 Rubble Volume	2.7×10^2	Spall Surface Layer With Hydraulic Spaller. Package.
Remove HVAC & Electrical From Reactor Building	1.7×10^2 m	6.4×10^{-3}	9.1×10^0	Cut Ductwork, Fans & Hangers. Package.
Remove Primary Reactor Water Piping, Drains and Equipment	6.6×10^1 m	1.0×10^2	4.7×10^1	Hack Saw and Decontaminate as Required. Cut Into 2-m Segments. Include Small Valves. Cut off Large Valves. Package.
Annex				
Remove Contaminated Concrete	1.6×10^2 holes	7.1×10^{-4} m^3 Waste Volume	2.4×10^1	Drill Holes for Removal of 0.05-m Contaminated Depth.
	1.4×10^1 m^2 Surface Area	6.8×10^{-1} m^3 Rubble Volume	4.0×10^1	Spall Surface Layer with Hydraulic Spaller. Package.
Heat Exchanger Building				
Remove Piping & Equipment	3.1×10^0 m	6.4×10^{-3}	2.0×10^0	Cut & Weld Heat Exchanger Openings.
Pump House				
Remove Piping & Drain Lines	1.5×10^0 m	1.3×10^{-2}	1.0×10^1	Decontaminate as Required. Segments Include Small Valves. Package.
Remove Retention Tank	7.6×10^0 m	1.3×10^{-2}	5.3×10^0	Cut & Weld Openings.
Radiation Center Building				
Remove Piping & Equipment	7.0×10^0 m	1.3×10^{-2}	5.0×10^0	Decontaminate as Required. Segment. Package.

I.1.2 Schedules and Manpower Requirements for DECON at the Reference Research Reactor Facility

The postulated schedules and manpower requirements necessary to terminate the licence requirements via the DECON alternative are presented in this section. First, the logical sequence in which the tasks must be performed is determined. Next, the task time requirements and the number and skills of decommissioning workers required for each task are estimated. Based on the estimated dose to accomplish each task, the number of workers needed to complete the radiation zone work in the allotted time and within the assumed radiation dose limits is determined. Whole-body radiation doses to the decommissioning workers are limited in accordance with 10 CFR 20.101.[1] The supervisors, utility operators, and health physics technicians are assumed to be long-time radiation workers whose annual exposure is limited to 5 rem/yr by the formula 5(N-18) of 10 CFR 20.101(b)(2). The craftsmen and laborers are assumed to have had little prior radiation exposure and, therefore, under 10 CFR 20.101(b)(1) and (2) may receive up to 3 rem per quarter, within the limitation of the formula 5(N-18) rems where "N" equals the individual's age in years at his last birthday. In those instances where the manpower estimated for physically accomplishing a task results in an average dose for each person in excess of these limits, additional people are assigned to the tasks to keep the individual occupational doses below these limits. In the manpower tables presented in this section, the manpower shown is adequate both to accomplish the task and to meet these occupational dose limits.

Discussions on the task schedules and decommissioning worker requirements and on the management and support staff for DECON at the reference research reactor are presented in the following sections.

I.1.2.1 Reactor Building and Other Buildings/Areas

The Reactor Building, containing all of the neutron-activated materials and most of the other radioactive materials, requires most of the effort to accomplish the DECON tasks. The development of parameters for estimating work tasks, schedules and manpower requirements requires applying engineering judgments based on previous reactor decommissioning studies[2-4] and engineering

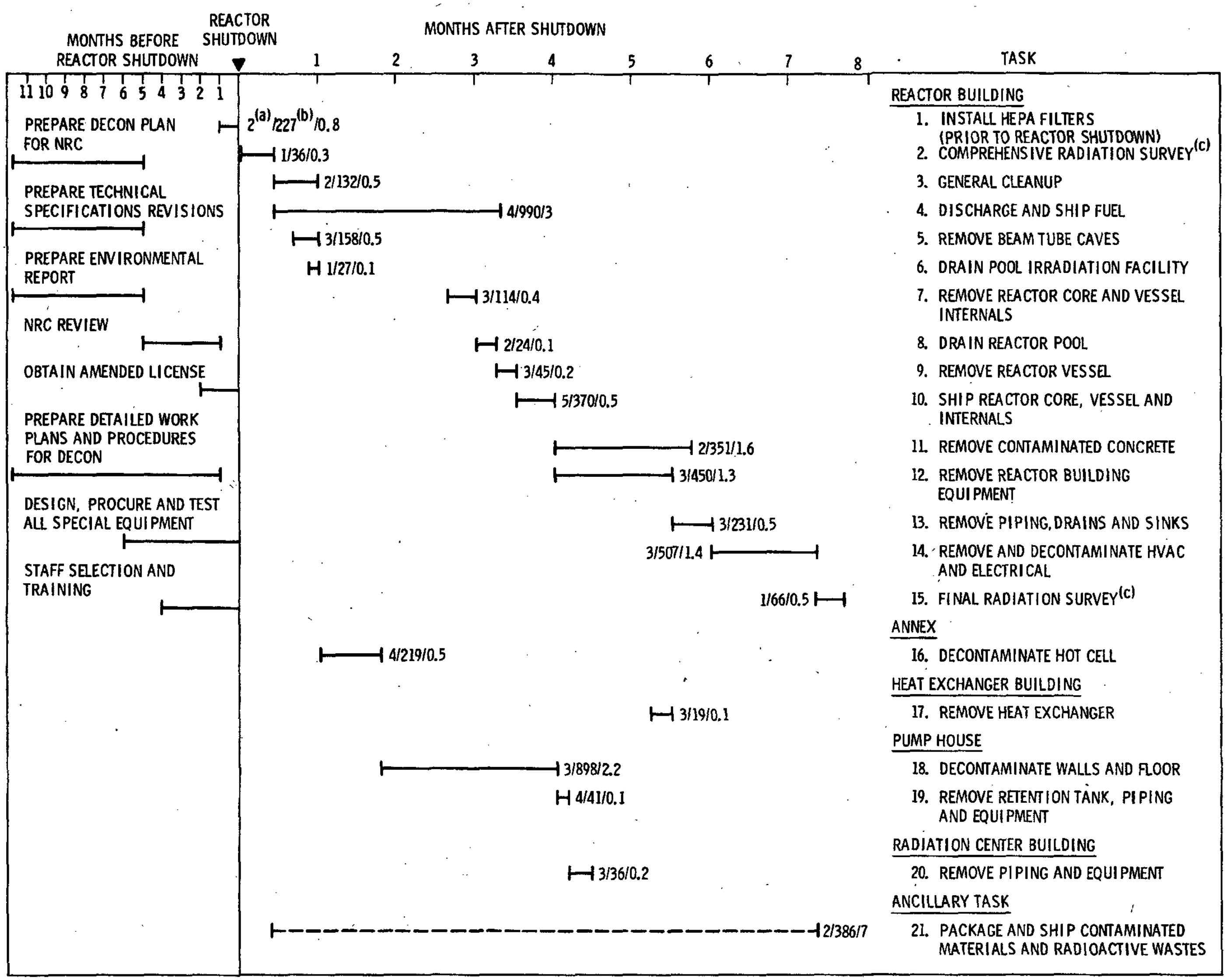
estimates applied to tasks unique to the reference research reactor. Using this information and the decontamination methods discussed previously in Section I.1, as well as the occupational dose estimates presented later in Section I.1.4, the task schedule and sequence and the decommissioning worker requirements for DECON at the reference facility are developed as presented in Figure I.1-1. DECON at the reference research reactor is estimated to be accomplished in approximately 8 months following reactor shutdown and is estimated to require 40.4 man-months of decommissioning worker effort.

I.1.2.2 <u>Overall Schedule and Manpower Requirements</u>

The overall schedule and sequence for DECON at the reference research reactor is shown in Figure I.1-1. The planning and preparation tasks (see Section H.2 of Appendix H) for the estimated 12-month period preceding reactor shutdown are also included. Decontamination of the building is estimated to be accomplished in about 8 months after final reactor shutdown.

DECON tasks, with a few exceptions, are performed on a single 8-hr shift, 5 days per week. Each task in Figure I.1-1 postulates a crew size that will provide a reasonably constant manpower loading for the bulk of the project. The overall dedicated manpower requirements for each DECON task are given in Table I.1-3. The overall decommissioning worker requirements for the period following reactor shutdown are also shown in Table I.1-3, and include 7000 man-hours of "hands-on" effort.

The total staff labor requirements for DECON at the reference research reactor are given in Table I.1-4. These requirements are given in equivalent man-years for the 12 months before and the 8 months following final reactor shutdown, and include the management and support staff as well as the decommissioning workers. Almost 13 man-years of effort are estimated for DECON at the reference research reactor, including approximately 9 man-years for the management and support staff and about 4 man-years for the decommissioning workers. In addition to the analyses already described for estimating task manpower requirements, the following assumptions are used in making this estimate:



LEGEND:

|——| CONTINUOUS OPERATIONS OVER THE TIME SPAN SHOWN

|- - -| INTERMITTENT OPERATIONS OVER THE TIME SPAN SHOWN

TOTALS:		
	PERSON-MONTHS	40
	PERSON-HOURS	7100
	EXPOSURE-HOURS	5326

(a) TASK INFORMATION NUMBERS INDICATE IN SEQUENCE: DIRECT STAFF PER DAY/EXPOSURE HOURS/CALENDAR MONTHS DURATION. WORKERS DEDICATING 15% OR LESS OF THEIR TIME TO THE TASK ARE NOT INCLUDED IN THE DIRECT STAFF PER DAY NUMBER

(b) THE NUMBER INCLUDES ALL WORKER EXPOSURE TIME AND IT IS ASSUMED THAT 75% OF THE WORKING DAY IS IN THE RADIATION ZONE

(c) TASK INCLUDES ALL BUILDINGS

FIGURE I.1-1. Overall Task Schedule and Sequence for DECON at the Reference Research Reactor

TABLE I.1-3. Dedicated Manpower Requirements for DECON at the Reference Research Reactor

Location/Task	Task Duration (months)	Dedicated Manpower Requirements (man-months)					
		Supervisors	Utility Operators	Laborers	Craftsmen	Health Physics Technicians	Totals
Reactor Building							
1. Install HEPA Filters[a]	0.80	0.12	--[b]	--	1.60	--	1.72
2. Comprehensive Radiation Survey[c]	0.27	--	--	--	--	0.270	0.27
3. General Cleanup	0.50	--	--	1.00	--	--	1.00
4. Discharge and Ship Fuel	3.00	3.0	2.5	0.50	--	1.50	7.50
5. Remove Beam Tube Caves	0.50	0.15	0.25	0.75	--	0.05	1.20
6. Drain Pool Irradiation Facility	0.12	0.03	--	0.12	0.05	0.01	0.21
7. Remove Reactor Core and Vessel Internals	0.36	0.16	0.02	0.39	0.25	0.04	0.86
8. Drain Reactor Pool	0.07	0.02	--	0.14	--	0.02	0.18
9. Remove Reactor Vessel	0.14	0.04	0.09	0.14	0.05	0.02	0.34
10. Ship Reactor Core, Vessel and Internals	0.50	1.00	1.20	--	--	0.60	2.80
11. Remove Contaminated Concrete	1.60	--	--	2.5	--	0.16	2.66
12. Remove Reactor Building Equipment	1.30	0.43	0.14	2.41	0.30	0.13	3.41
13. Remove Piping Drains and Sinks[d]	0.52	0.22	--	1.00	0.48	0.05	1.75
14. Remove and Decontaminate HVAC and Electrical	1.40	--	--	2.7	1.00	0.14	3.84
15. Final Radiation Survey[c]	0.50	--	--	--	--	0.50	0.50
Annex							
16. Decontaminate Hot Cell	0.50	--	--	1.40	--	0.26	1.66
Heat Exchanger Building							
17. Remove Heat Exchanger	0.05	0.01	0.02	0.05	0.05	0.01	0.14
Pump House							
18. Decontaminate Walls and Floor	2.20	--	--	6.60	--	0.20	6.80
19. Remove Retention Tank Piping and Equipment	0.10	0.04	0.10	0.10	--	0.07	0.31
Radiation Center Building							
20. Remove Piping and Equipment From Waste Process Room	0.16	0.03	--	0.16	0.06	0.02	0.27
Ancillary Tasks							
21. Package and Ship Contaminated Materials and Radioactive Wastes[c]	7.0	--		2.73	--	0.02	2.75
TOTALS		5.25	4.32	22.64	3.84	4.07	40.17

(a) Performed before reactor shutdown.
(b) Denotes no manpower dedicated to task.
(c) Includes all buildings.
(d) Includes Heat Exchanger Building.

TABLE I.1-4. Staff Labor Requirements for DECON of the Reference Research Reactor

Position	Prior to Shutdown -12[a]	After Shutdown +2	+4	+6	+8	Total Staff Labor Required (man-years)
	Staff Labor Requirements (man-months)					
Management & Support Staff:						
Decommissioning Superintendent	4	2	2	2	2	1.00
Secretary	4	2	2	2	2	1.00
Clerk & Procurement Specialist	3	2	2	2	1	0.83
Contracts & Accounting Specialist	1.5	2	2	2	2	0.79
Security Supervisor	0	2	2	2	1.5	0.63
Security Patrolman[b]	0	6	6	6	4.5	1.88
Armed Guards[b]	0	6	4	0	0	0.83
Health Physicist & Shipment Specialist	3	2	2	2	2	0.92
Industrial Safety Specialist	1	2	2	2	1.5	0.71
Control Room Operator	0	2	1.6	0	0	0.30
Quality Assurance Specialist	1	2	2	2	1	0.67
Subtotals	17.5	30	27.6	22	17.5	9.56
Decommissioning Workers:[c]						
Crew Leader	0	2	3	1.0	0	0.50
Utility Operator	0	1.5	1.9	0.7	0	0.34
Laborer	0	6	7.5	7.0	3.0	1.96
Craftsman	1.6	0.1	0.3	1.1	0.8	0.33
Health Physics Technician	0	2	2	2	2	0.67
Subtotals	1.6	11.6	14.7	11.8	5.8	3.80
Totals	19.1	41.6	42.3	33.8	23.3	13.36

(a) Time relative to reactor shutdown.
(b) Based on information supplied by personnel at the reference reactor, when fuel dose is ≤100 rem/hr within 1 meter of the material's surface. Both response and access-control personnel are necessary on a 3-shift, 7-day basis. This requirement is applicable when 70%-enriched fuel is present at the site.
(c) Requirements following reactor shutdown are based on Table I.1-3.

- supervisors, including crew leaders, are staffed one per shift.
- Laborers and craftman are readily available from the University maintenance staff or the open job market and are hired as required.
- A minimum staff of one health physics technician is required regardless of the dedicated manpower requirements at the time.
- Utility operators are available on an as-needed basis from the reference University staff.

The total staff labor given in Table I.1-4 for the decommissioning workers during the DECON tasks exceeds the overall decommissioning requirements presented in Table I.1-3 because of some of the above assumptions. This "excess" manpower is assumed used for the numerous, small, unspecified work items that accompany a project of this complexity.

An overall summary of the monthly workload for all decommissioning workers is presented in Table I.1-5. The total monthly workload is reasonably constant at 5 to 6 full-time workers over the term of most of the project.

TABLE I.1-5. Overall Decommissioning Worker Requirements for DECON at the Reference Research Reactor

Position	Man-Months per Working Month — Months Before or After Reactor Shutdown									Total Man-Months
	-1	1	2	3	4	5	6	7	8	
Crew Leader	--	0.9	1.0	1.1	1.3	0.3	0.7	--	--	5.3
Utility Operator	--	0.7	0.8	0.9	1.0	0.2	0.5	--	--	4.1
Laborer	--	2.2	3.0	4.0	3.5	4.0	3.0	2.0	1.0	22.7
Craftsman	1.6	--	0.1	0.2	0.1	0.3	0.8	0.7	0.1	3.9
Health Physics Technician	--	0.70	1.0	0.6	0.8	0.2	0.3	0.3	0.5	4.4
Total	1.6	4.5	5.9	6.8	6.7	5.0	5.3	3.9	1.6	40.4

I.1.3 Cost of DECON for the Reference Research Reactor

The costs of DECON for the reference research reactor are presented in this section. These costs are all adjusted to early-1981 costs. The estimated costs and the other possible costs of DECON for the facility are summarized in Table I.1-6.

The total cost of DECON is estimated to be about $0.84 million, including a 25% contingency. The major contributor to the cost is staff labor, representing 78.4% of the total cost. Disposal of radioactive materials is the only other major contributor to the total cost for DECON at the reference research reactor.

The total cost of other optional tasks which may be necessary to complete DECON is estimated to be $0.32 million, including a 25% contingency. This cost includes the demolition of the Reactor Building and the concrete reactor structure as well as the shipping of the spent fuel to a proper repository.

Detailed cost data for the individual cost categories shown in Table I.1-6 are presented and discussed in the following subsections.

I.1.3.1 Disposal of Radioactive Materials

Three distinct types of radioactive materials in the reference research reactor require disposal during DECON: 1) neutron-activated materials, 2) contaminated materials, and 3) radioactive wastes. The total cost for disposal of these materials is about $86,000 and is approximately 13% of the total DECON cost. The disposal cost includes the containers, transportation, and burial costs, but does not include the direct labor costs for removing and packaging these materials. Labor costs are incurred regardless of the decommissioning activity and are discussed later (Section I.1.3.2). Table I.1-6 contains a summary of the number of shipments, the burial volumes, and the disposal costs for the radioactive materials.

The costs of disposal for the three types of radioactive materials are discussed in more detail in the following subsections.

TABLE I.1-6. Summary of Estimated Costs of DECON for the Reference Research Reactor

Cost Category	Estimated[a,b] Costs ($)		Percent of[c] Total
Disposal of Radioactive Materials			
Neutron-Activated Materials	16 610		
Contaminated Materials	60 060		
Radioactive Wastes	9 620		
Total Disposal Costs		86 290	12.8
Staff Labor		530 570	78.4
Energy		13 790	2.0
Special Tool & Equipment		21 150	3.1
Miscellaneous Supplies		6 210	0.9
Nuclear Insurance		4 620	0.7
License Fees		13 950	2.1
Subtotal		676 580	100.0
Contingency (25%)		169 150	
Total, DECON Costs		845 730	
Other Possible Costs			
Spent Fuel Shipment		60 980[d]	
Facility Demolition & Site Restoration		196 750[e]	
Subtotal		257 730	
Contingency (25%)		64 430	
Total, Other Cost Options		322 160	

(a) 1981 costs used.
(b) The number of significant figures shown is for computational completeness.
(c) Individually rounded to the nearest 0.1%.
(d) Includes cost of containers, overpacks and 800-km transportation only.
(e) Based on Table L.3-1.

Neutron-Activated Materials. All of the neutron-activated materials in the reference research reactor are located in the reactor pool. As discussed previously in Section E.1 of Appendix E, the reference research reactor vessel and internal structures are estimated to contain the majority of the neutron-activated materials. The reactor core internals are removed essentially intact after only minor remote cutting for disconnection from experimental facility components. The cask, which is large enough in diameter to take the complete reactor core internals intact, has considerable excess length when compared with the internals. This extra space is used to package segments of the reactor vessel. It is postulated that the neutron-activated components are transported in one shipment for a total cost of $16,610. The various charges associated with this shipment are included in Table I.1-7, along with information on the radioactive content, burial volumes and material mass.

Contaminated Materials. Contaminated materials in the reference research reactor are assumed to include much of the piping and equipment located in the Reactor Building and the other buildings/areas described in Section B.1 of Appendix B. In addition, some concrete surfaces are assumed to be contaminated and require surface removal to a depth of about 50 mm. Information on the disposal costs for contaminated materials is given in Table I.1-8.

Approximately 2 Ci of radioactivity are removed with the contaminated materials. Transportation requirements for these materials are estimated to necessitate three shipments to a shallow-land burial site. The burial volume of all the contaminated materials is estimated to be 133 m^3.

Shipping requirements for the additional small volumes of radioactive wastes are satisfied by their inclusion into the shipments required for disposal of the contaminated materials.

Almost all of the contaminated materials are packaged in 34 wooden boxes specially made for this purpose. There are two pieces of larger equipment that must be transported along with these boxes: the heat exchanger found in the Heat Exchanger Building and the retention tank found in the Pump House. Their volume, mass, and transportation costs are included in Table I.1-7.

TABLE I.1-7. Costs of Disposal for Radioactive Materials for the Reference Research Reactor

Radioactive Type	Estimated Mass (Mg)	Radionuclide Content (Ci)[(a)]	Number of Pieces	Number of Disposable Containers	Container Cost ($)[(b)]	Cask Rental Cost ($)[(c)]	Number of Shipments	Transportation Costs ($)[(d)]	Fuel Surcharge ($)[(e)]	Handling Surcharge ($)[(f)]	Burial Volumes (m^3)[(g)]	Burial Costs ($)[(f)]	Liner Surcharge ($)[(f)]	Curie Surcharge ($)[(f)]	Total Disposal Costs ($)
Neutron-Activated Materials	1.6	1 457	1	1	2 000	6 000	1	2 640	480	1 360	6	1 840	1 460	830	16 610
Contaminated Materials	57.0	2	36	34	13 600	--[(h)]	3	3 550	640	--	133	42 270	--	--	60 060
Radioactive Wastes	2.0	<1	6	6	2 400	--	--[(i)]	550	100	--	21	6,570	--	--	9 620
Totals	60.6	1 459	43	41	18 000	6 000	4	6 740	1 220	1 360	160	50 680	1 460	830	86 290

(a) Based on Tables E.1-2, E.1-3 and E.1-4.
(b) Based on Table M.2-1.
(c) Based on Table M.2-2.
(d) Based on Table M.3-1.
(e) Based on Discussions in Section M.3 of Appendix M.
(f) Based on Table M.4-1.
(g) Includes the disposable container rounded to the nearest whole number.
(h) Denotes cost data is not applicable.
(i) Included in contaminated materials shipment.

TABLE I.1-8. Costs of Disposal for Contaminated Materials

Location/Task	Estimated Mass (Mg)	Number of Disposal Containers[a]	Container Costs ($)[b]	Transportation Costs ($)[c]	Burial Volumes (m^3)[d]	Burial Costs ($)[e]	Total Disposal Costs ($)
Reactor Building							
General Cleanup	10.2	4	2 000	760	16	5 000	7 760
Remove Beam Tube Caves	19.5	5	2 000	1 430	19	6 010	9 440
Drain Pool Irradiation Facility & Reactor Pool (IX resins, decontamination eqpt)	0.3	0.5	200	20	2	670	890
Remove Reactor Vessel	0.9	1	400	70	4	1 340	1 810
Remove Contaminated Concrete	10.9	2.5	1 000	800	10	3 000	4 800
Remove Reactor Building Equipment	4.3	10	4 000	320	40	12 700	17 020
Remove Piping Drains & Sinks	2.9	4	2 000	210	14	4 540	6 740
Annex							
Decontaminate Hot Cell	1.6	0.5	200	120	2	670	990
Heat Exchanger Building							
Remove Heat Exchanger	2.0	1	0	140	4	1 340	1 480
Pump House							
Decontaminate Walls & Floor	0.05	0.2	80	0	1	330	410
Remove Retention Tank, Piping, and Equipment	3.3	3.4	950	250	14	4 340	5 540
Radiation Center Building							
Remove Piping & Equipment	1.0	2	800	70	7	2 330	3 200
Totals	57	34	13 600	4 190	133	42 270	60 060

(a) Assumed to be 1.2-m by 1.2-m by 2.4-m plywood boxes unless otherwise indicated.
(b) Based on Table M.2-1.
(c) Based on Table M.3-1 including the fuel surcharge of 18%.
(d) Based on Table M.4-1.
(e) Includes the disposable container where required.

The total cost of disposal of contaminated materials from the DECON alternative is estimated to be about $60,000.

Radioactive Wastes. Radioactive wastes (radwastes) result directly from DECON of the reference research reactor. Two categories of radioactive wastes are defined: wet solid wastes and dry solid wastes.

Wet solid wastes result from the processing of chemical decontamination solutions and contaminated water volumes. These wastes include filter sludges and spent demineralizer resins, as well as the neutralized chemical solutions from decontamination of the contaminated drain systems. These wastes are assumed to be mixed with a cement solidifying agent and encapsulated in steel drums prior to shipping to a shallow-land burial facility. Only a few DECON tasks at the reference research reactor will produce such wastes.

Dry solid wastes include discarded contaminated materials such as plastic sheeting, rags, and anticontamination clothing. They are expected to occur as a result of most of the DECON tasks and are estimated on a taskwise basis. The dry solid wastes are compacted as much as possible to reduce their volume. It is postulated that 15% of the total volume of contaminated materials and radioactive wastes will be attributed to wet and dry radioactive wastes.

It is assumed that DECON activities at the reference research reactor will create only small amounts of radioactive waste materials. For example, from operational data at the reference research reactor, cleanup of the primary cooling water system requires the change out of ion exchange beds in the demineralizer only two to six times per year. This creates an annual waste volume of less than 1 m^3. Estimates of the radioactive wastes from all DECON activities are included in Table I.1-7 and indicate a total estimated waste volume of 21 m^3. The total costs of transportation and disposal of these materials packaged in six containers is estimated to be $9,620. Shipment of these six containers to a low-level waste burial site is included in the truck shipments of contaminated materials previously discussed.

I.1.3.2 Staff Labor

The cost of staff labor during DECON is shown in detail in Table I.1-9. More than 78% of the total DECON cost is associated with staff labor (see Table I.1-5). A total staff labor cost of about $0.5 million is estimated for DECON of the reference research reactor. Specialty contractor labor is not included in this total.

The dedicated manpower costs of the DECON tasks are given in Table I.1-10. These costs are attributed to manpower that is specifically assigned to the tasks, and do not include either non-dedicated personnel or management and support staff.

It is apparent from an analysis of Table I.1-9 that the management and support staff costs of $0.41 million represent the largest portion (77%) of the overall staff labor. This latter cost is very sensitive to the length of time postulated for support staff activities both before and after reactor shutdown. The task time needed after reactor shutdown is dependent upon the schedule and sequencing postulated in Figure I.1-1. Positive efforts to reduce this time interval will be cost effective; however, compressing the decommissioning worker load into a shorter time span to accomplish this will disrupt the constant crew size shown in Table I.1-4.

I.1.3.3 Energy

The cost of energy used during DECON is presented in Table I.1-11, together with the estimated usage of electricity. The usage and cost of energy is based upon data supplied by the reference research reactor personnel concerning annual consumption and costs since plant startup. These data were applied to the time frame estimated for DECON tasks (i.e., about 8 months) and the 1981 cost of energy ($0.025/kWh) at the reference research reactor.

A total of 551 MWh of electricity, costing $13,790, is estimated to be used during DECON. This cost represents about 2% of the total DECON cost.

I.1.3.4 Special Tools, Equipment and Miscellaneous Supplies

Based upon the requirements for special tools and equipment presented in Table I.1-1 and their estimated cost of acquisition, primarily by short-term

TABLE I.1-9. Staff Labor Costs of DECON for the Reference Research Reactor

Position	Time Relative to Final Reactor Shutdown (months) -12	+2	+4	+6	+8	Total Staff Labor Costs ($)
	Staff Labor Costs ($)[a,b]					
Management & Support Staff						
Decommissioning Superintendent	29 700	14 850	14 850	14 850	14 850	89 100
Secretary	8 060	4 030	4 030	4 030	4 030	24 200
Clerk & Procurement Specialist	7 940	5 290	5 290	5 290	2 650	26 460
Contracts & Accounting Specialist	5 890	7 850	7 850	7 850	7 850	37 290
Security Supervisor	0	9 320	9 320	9 320	6 990	34 940
Security Patrolman[c]	0	12 700	12 700	12 700	9 530	47 630
Armed Guards[c]	0	15 000	10 000	0	0	25 000
Health Physicist & Shipment Specialist	3 910	7 820	7 820	7 820	7 820	42 990
Industrial Safety Specialist	4 370	8 730	8 730	8 730	6 550	37 120
Control Room Operator	0	5 750	4 600	0	0	10 350
Quality Assurance Specialist	3 910	7 820	7 820	7 820	3 910	31 280
Subtotals	71 610	99 160	93 010	78 410	64 170	406 360
Decommissioning Workers:[d]						
Crew Leader	0	7 400	11 100	3 700	0	22 200
Utility Operator	0	4 010	5 080	1 880	0	10 970
Laborer	4 280	15 450	19 310	18 030	7 720	60 510
Craftsman	0	270	800	2 940	2 140	10 430
Health Physics Technician	0	5 020	5 030	5 020	5 030	20 100
Subtotals	4 280	32 150	41 320	31 570	14 890	124 210
Totals	75 890	131 310	134 330	109 980	79 060	530 570

(a) Calculated as the product of the data given in Table I.1-4 and the corresponding data given in Table M.1-1.
(b) The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest $10.
(c) Based on information obtained from personnel at the reference reactor when fuel dose is at 1 m ≤100 mR/hr. Both response and access-control personnel are necessary on a three shift, 7-day-week basis. This requirement is applicable when 70% enriched fuel is present.
(d) Requirements following reactor shutdown are based on Table I.1-4.

TABLE I.1-10. Dedicated Manpower Costs of DECON for the Reference Research Reactor

Location/Task	Dedicated Manpower Costs ($)[a]
Reactor Building	
1. Install HEPA Filters	4 750
2. Comprehensive Radiation Survey	680
3. General Cleanup	2 580
4. Discharge & Ship Fuel	23 450
5. Remove Beam Tube Caves	3 310
6. Drain Pool Irradiation Facility	590
7. Remove Reactor Core & Vessel Internals	2 450
8. Drain Reactor Pool	490
9. Remove Reactor Vessel	950
10. Ship Reactor Core, Vessel & Internals	8 620
11. Remove Contaminated Concrete	6 840
12. Remove Reactor Building Equipment	9 400
13. Remove Piping Drains & Sinks	4 840
14. Remove & Decontaminate HVAC & Electrical	9 980
15. Final Radiation Survey	1 250
Annex	
16. Decontaminate Hot Cell	4 260
Heat Exchanger Building	
17. Remove Heat Exchanger	380
Pump House	
18. Decontaminate Walls & Floor	17 500
19. Remove Retention Tank, Piping & Equipment	870
Radiation Center Building	
20. Remove Piping & Equipment	740
Ancillary Task	
21. Package & Ship Contaminated Materials & Radioactive Wastes	7 080
Total	111 010

(a) Calculated as the sum of the products of the data presented in Table I.1-3 (converted to man-years) and the corresponding data presented in Table M.1-1 in Appendix M; rounded to the nearest $10.

TABLE I.1-11. Costs of Energy During DECON for the Reference Research Reactor

System or Equipment[a]	Estimated Electrical Energy Usage (kWh)[b]	Costs ($)[c]
General System (crane, etc.)	2 700	69
HVAC	14 160	3 543
Lighting	23 000	574
Control Room	2 600	67
Fire Protection	300	8
Security	5 900	147
Communications	900	22
Domestic Water	36 300	908
Reactor Water	23 400	585
Compressed Air	5 700	137
Building Heating	302 600	7 565
Decommissioning Equipment	6 530	163
Total Energy Usage	551 300	
Total Energy Cost		13 790

(a) All buildings are included in this analysis.
(b) Based on present and past electrical energy usage at the reference research reactor and an analysis of individual component requirements.
(c) Estimated from the required equipment listed in Table M.5-1 and the present rate per kWh at the reference reactor of $0.025.

rental, about $21,000 will be spent for such items. The major contributors to this cost are the rental costs for the forklift and the remote-operations tool. This expenditure represents approximately 3% of the total estimated cost of DECON.

The variety of decommissioning tasks performed for DECON requires a number of associated supplies to adequately complete the tasks. These include protective clothing, hand tools, drill bits, sweeping compounds, mops, brushes, plastic sheeting, bags and sealing tools, demineralizer resin, saw blades, compressed gases, filters, steel materials, welding supplies, and cleaning

supplies. The estimated cost of these items is derived from janitorial data from the reference research reactor operating personnel and the expected consumption rates for the materials listed. The total cost for miscellaneous supplies is estimated to be about $6,000. This represents about 1% of the total DECON costs.

I.1.3.5 Nuclear Insurance

The cost of nuclear liability insurance at the reference research reactor during DECON is estimated from the current annual operating premium of $7,700. Based upon an 8-month pro-rata premium and a 10% credit on the remaining premium during the first year of reactor shutdown, the cost of nuclear insurance is $4,620. This represents less than 1% of the total DECON costs.

I.1.3.6 Licensing Fees

The fees charged for licensing services performed by the NRC are delineated in 10 CFR Part 170.23.[5] Two license amendments are needed to the operating license. The first amendment is needed to allow possession but not operation of the reactor after shutdown and the second amendment is termination of the license to allow unrestricted site use after all DECON activities are complete. In addition, two inspections are anticipated to occur during the 8-month DECON task span.

A summary of these costs is given in Table I.1-12. The total cost of license fees is about $14,000 and represents approximately 2% of the total cost of DECON.

I.1.3.7 Other Possible Costs

Two additional categories of costs could figure into the total DECON cost depending on how they are classified. In this study, they are presented separately since they cannot be clearly identified as belonging to DECON. The tasks that require these costs are: 1) shipment of the spent fuel to an offsite repository, and 2) demolition of the structures and restoration of the site.

TABLE I.1-12. Cost of Licensing Fees During DECON for the Reference Research Reactor[a]

Licensing Service Category	Fee ($)	Total Costs ($)[b]
Facility License Amendment (Class IV)	6 000	12 000
Health Safety and Environmental Inspection	650/yr	650[c]
Safeguards Inspection	1 300/2 yr[d]	1 300
Total		13 950

(a) Code of Federal Regulations, 10 CFR Part 170.
(b) Based on two license amendments, one to obtain a possession-only license prior to dismantlement and one to terminate that license following DECON.
(c) Based on a minimum annual fee for the 8-month DECON task.
(d) Based on a minimum fee for the 8-month DECON task.

Spent Fuel Shipment. Because of the northwestern location of the reference research reactor, it is assumed that the fuel will be shipped no farther than 800 km for either:

- shipment to an offsite repository
- or reuse.

Based upon cost information from Westinghouse personnel at Hanford, Washington, who are planning to remove, ship and reuse a complete fuel loading from the reference research reactor, the cost involved in the two spent fuel shipments is estimated to be about $61,000. This cost is presented on Table I.1-6.

Facility Demolition and Site Restoration. The cost of demolishing all of the unneeded decontaminated and uncontaminated structures at the reference research reactor by an independent contractor is discussed in Appendix L and is summarized in Table L.3-1 and included in Table I.1-6. The total cost of $196,750 (without contingency) includes labor, supplies, overheads and profits, but not extraordinary insurance premiums or bonding.

Property Taxes. In the case of privately owned research reactors, it should be noted that an anticipated cost from property tax assessments can also be charged to DECON; however, at this time specific values are considered outside the scope of this study due to site-specific assessment rates.

I.1.4 Estimated External Occupational Doses for DECON at the Reference Research Reactor

Estimates are made of the external occupational radiation doses that are accrued by the workers conducting decommissioning tasks during DECON at the reference research reactor. These estimates are based on the task analysis to determine the man-hours of effort required in radiation zone work, as discussed in Section I.1.2.2, and the anticipated dose rates associated with each task. Basic assumptions made in developing these estimates are: 1) every effort is made to minimize the personnel exposure to radiation (ALARA philosophy) while accomplishing a task, by the use of temporary shielding and remote handling techniques and by staying out of radiation fields when not actively participating in work in zones so identified, 2) careful, prompt accounting of radiation doses is maintained to rapidly identify jobs that are causing excessive dose accumulations so corrective action can be taken, and 3) ^{60}Co is considered to be the principal source of the radiation dose.

The estimated dose for each task is corrected for radioactive decay with a decay factor calculated using the half-life of ^{60}Co and decayed to the midpoint of the time line for the given task (shown in Figure I.1-1).

The results of these analyses and corrections are presented in Table I.1-13. The total corrected external occupational radiation dose for DECON at the reference research reactor is a little over 18 man-rem. Of the total, 78.4% or 14.4 man-rem results from DECON at the Reactor Building, 17.3% from DECON at all other buildings/areas remaining on the site, and 4.3% from ancillary DECON tasks which are attributed to all the buildings on the site.

I.2 METHODS POSTULATED FOR DECON AT THE REFERENCE TEST REACTOR FACILITY

The details presented in this section are specifically for decommissioning the reference test reactor facility described in Appendix C of this report by

TABLE I.1-13. Estimated Occupational Radiation Doses for DECON at the Reference Research Reactor

Location/Task	Supervisors[a] Exposure (man-hr)	Supervisors[a] Dose (man-rem)	Utility Operators Exposure (man-hr)	Utility Operators Dose (man-rem)	Laborers Exposure (man-hr)	Laborers Dose (man-rem)	Craftsmen Exposure (man-hr)	Craftsmen Dose (man-rem)	Health Physics Technicians Exposure (man-hr)	Health Physics Technicians Dose (man-rem)	Task Totals Exposure (man-hr)	Task Totals Dose (man-rem)	Task Totals Decay Factor[b]	Task Totals Corrected Dose[c] (man-rem)
Reactor Building														
1. Install HEPA Filters	16	0.016	--[d]	--	--	--	221	0.211	--	--	227	0.227	1.00	0.227
2. Comprehensive Radiation Survey	--	--	--	--	--	--	--	--	36	0.36	36	0.036	0.989	0.036
3. General Cleanup	--	--	--	--	132	0.132	--	--	--	--	132	0.132	0.989	0.131
4. Discharge & Ship Fuel	396	2.772	304	2.128	66	0.462	--	--	224	1.568	990	6.930	0.978	6.778
5. Remove Beam Tube Caves	20	0.040	33	0.066	99	0.198	--	--	6	0.012	158	0.316	0.989	0.313
6. Drain Pool Irradiation Facility	4	0.008	--	--	16	0.032	6	0.012	1	0.002	27	0.054	0.978	0.053
7. Remove Reactor Core & Vessel Internals	21	0.084	4	0.016	51	0.204	33	0.132	5	0.020	114	0.456	0.968	0.441
8. Drain Reactor Pool	3	0.006	--	--	18	0.036	--	--	3	0.006	24	0.048	0.957	0.046
9. Remove Reactor Vessel	6	0.060	12	0.121	18	0.180	7	0.070	2	0.020	45	0.450	0.957	0.431
10. Ship Reactor Core, Vessel & Internals	132	0.462	158	0.553	--	--	--	--	80	0.280	370	1.295	0.957	1.239
11. Remove Contaminated Concrete	--	--	--	--	330	1.650	--	--	21	0.105	351	1.755	0.946	1.660
12. Remove Reactor Building Equipment	57	0.342	18	0.108	318	1.908	40	0.240	17	0.102	450	2.700	0.946	2.554
13. Remove Piping Drains & Sinks	29	0.029	--	--	132	0.132	63	0.063	7	0.007	231	0.231	0.936	0.216
14. Remove & Decontaminate HVAC & Elec.	--	--	--	--	356	0.178	132	0.066	19	0.010	507	0.254	0.925	0.238
15. Final Radiation Survey	--	--	--	--	--	--	--	--	66	0.017	66	0.017	0.916	0.016
Annex														
16. Decontaminate Hot Cell	--	--	--	--	185	1.850	--	--	33	0.330	219	2.219	0.978	2.170
Heat Exchanger Building														
17. Remove Heat Exchanger	1	0.001	3	0.003	7	0.007	7	0.007	1	0.001	19	0.019	0.936	0.018
Pump House														
18. Decontaminate Walls & Floor	--	--	--	--	871	0.871	--	--	27	0.027	898	0.898	0.968	0.869
19. Remove Retention Tank, Piping & Equip.	6	0.012	13	0.026	13	0.026	9	0.018	--	--	41	0.082	0.957	0.078
Radiation Center Building														
20. Remove Piping & Equipment	4	0.004	--	--	21	0.021	8	0.008	3	0.003	36	0.036	0.946	0.034
21. Package & Ship Contaminated Materials & Radioactive Wastes[e]	--	--	--	--	360	0.720	--	--	26	0.052	386	0.772	0.957	0.739
Subtotals	695		545		2993		516		577		5326			18.287
Ancillary Tasks														
22. Routine Radiation Surveys	--		--		--		--		238		238	0.060	0.916	0.055
Totals	695		545		2993		516		815		5[illegible]54			18.342

(a) Includes shift engineers, crew leaders, craft supervisors, and senior health physics technicians.
(b) Based on the half-life of ^{60}Co; calculated at the midpoint of the task timelines shown in Figure I.1-1.
(c) The number of significant figures shown is for computational convenience and does not imply accuracy to the nearest millirem.
(d) Indicates that category of decommissioning staff is not involved in the particular task.
(e) Task includes all buildings.

the DECON alternative.[a] DECON is the removal from the reference test reactor site of all materials having radioactivity levels greater than permitted for unrestricted use of the property. Thus, all radioactively contaminated equipment, tanks, pumps and piping; the reactor vessel and internals; the MUR; activated and contaminated concrete; and contaminated soil must be removed, packaged, and shipped to an authorized radioactive waste disposal site. In addition, it is postulated that a statistically sound sampling and monitoring methodology acceptable to the NRC is used and a site survey plan is submitted to the NRC prior to conducting final sampling and monitoring.[6] The details presented here amplify the information presented in Sections 10 and 11 of Volume 1 and provide the basic information needed to assess the safety of decommissioning the reference test reactor facility.

I.2.1 DECON Details and Supporting Information

Following completion of planning and preparation (see Section H.1 of Appendix H) and cessation of facility operations, a comprehensive radiation survey of the reference test reactor facility is conducted. This survey is required to finalize plans for draining and flushing contaminated process systems and for installing temporary shielding for personnel protection during the dismantlement of piping and equipment. Next, a general cleanup is accomplished and a total inventory of equipment is taken to determine usefulness of specific equipment to the decommissioning project. Equipment not so designated is identified for later disposal.

Following the final inventory cleanout, fuel shipments commence, and equipment removal and final decontamination operations are initiated. The following conditions are met before commencing these operations.

1. Responsible management and safety personnel approve the following plans and procedures:

(a) It should be recognized that two licenses are involved with the reference test reactor facility--one for the reference test reactor and the second for the Mock-Up Reactor (MUR).

- radiation work, industrial safety, and emergency procedures
- equipment and material handling, disassembly, cleaning, and packaging and shipping procedures
- equipment and facility decontamination plans and procedures.

2. Disposition is predetermined for all equipment. The equipment can either be decontaminated for reuse, sold as scrap, buried in the local landfill dump, partially decontaminated for use at another restricted plant, or shipped to a licensed burial ground for contaminated materials.
3. All ventilation equipment, personnel protection systems, emergency power systems, fire protection systems, and radiation monitoring equipment in the building and onsite are in service and fully functional.
4. All personnel and contractors are adequately trained to perform their jobs.
5. Appropriate occupational safety equipment and continuous air sampling equipment are available for facility disassembly, transfers, and cleanup.
6. Temporary portable cleaning chambers for decontaminating equipment are available (e.g., greenhouse with tank for water and steam rinsing of equipment, washing tanks, degreasers, etc.).
7. Packaging materials and shipping containers are available.
8. All equipment for dismantlement and decontamination operations is available.
9. A comprehensive radiation survey is completed, with all results mapped and used as a basis for each building, room, and area's work plan.
10. The system and procedures for the functions of special nuclear material accountability measurements and measurement control are established.

11. All unneeded process material supplies (e.g., bottled gases, acids, and caustics) are disconnected from the plant and disposed of.

In this study it is assumed that most equipment items are disassembled and decontaminated in their original rooms or areas, if possible.

The equipment is disassembled and decontaminated in-situ as much as practicable to meet final disposition requirements. Portable greenhouses and tanks can be used to perform detergent washing and steam or water rinsing operations. Ends of piping, conduit, ductwork, etc., are always sealed or plugged with caps. Equipment is sealed with plastic, packaged, and removed to one of the staging areas for storage and shipment. Equipment that cannot be packaged in the original room is packaged in the staging areas. If released for unrestricted use, the parts are packaged and shipped to a local dump. Parts that cannot be cleaned sufficiently for release are packaged in plywood boxes and shipped to a licensed burial ground.

The pattern for final cleanup and decontamination of individual rooms is dictated by doing the most contaminated rooms first and improving the accessibility to staging areas. The need for certain rooms (e.g., scrap or waste handling facilities and hot maintenance) during DECON operations will delay their decommissioning.

For an individual room, accessibility to the equipment dictates the order of equipment removal. It is also desirable to remove the least-contaminated equipment from the room earliest. To provide access, the equipment nearest the exit area of the room is generally removed first. Where possible, equipment is removed first, followed by all external pipes, conduits, controls, etc., attached to the process equipment. The equipment is disconnected from exhaust ductwork and the open duct end is sealed or fitted with a filter to preclude contamination spread from back pressures. The final step is to carefully survey for radiation and decontaminate the ceiling, walls, and floor with detergent cleanser, powered brushes, and handwipes. Floor tiles or strippable seal coat are removed. Where hot spots occur, the concrete floor is chipped and vacuumed to ensure complete decontamination. A final survey is conducted prior to releasing the area for unrestricted use.

Strict accountability records are kept for equipment and appurtances being decontaminated and shipped. Because external radiation doses to workers are relatively low, many decontamination operations are hands-on operations. Plastic tents are constructed when appropriate to contain contamination where equipment is being disassembled. Disassembly of piping, valves, pumps, ancillary equipment, and ductwork external to equipment and hoods is conducted using strict contamination-control methods. Proper precautions and steps (e.g., in the Office and Laboratory Building) are taken to prevent spills or to confine contamination spills that may inadvertently occur. Health physics personnel observe and monitor operations to ensure that safe radiation and industrial practices are followed at all times.

In addition, all DECON activities are conducted in accordance with the decommissioning quality assurance plan (see Section G.7 of Appendix G) and are checked by the licensee quality assurance staff. Environmental monitoring (see Section G.8 of Appendix G) during DECON is performed by the same "contract services" organization that provided these radiological services during operation.

At reactor shutdown,[a] the reactor cooling and cleanup systems and the quadrant and canal (Q&C) systems contain the normal operating inventory of radioactively contaminated water. During DECON at the reference test reactor (and the MUR), the contaminated water inventories are processed repeatedly through the in-plant liquid radwaste systems for treatment before final disposition. It should be recognized that the Q&C water is postulated to be used over and over again for the various decontamination activities anticipated during DECON. Since these large reservoirs of deionized water are amenable to cleanup and reuse in this manner, this method reduces the total volume of radioactively contaminated water that is cleaned up.

(a) The term "reactor shutdown" used in this portion of the study always refers to both the reference test reactor and its associated mock-up reactor (i.e., the MUR), located in Canal H.

The primary cooling water system (PCWS) is drained and flushed after the fuel is removed from the test reactor core. In addition, after the fuel is removed from the MUR and all fuel and experimental hardware is removed from Canal H, then the MUR is removed and Canal H is drained, cleaned, dried, and further decontamination work completed.

Following defueling of both reactors, high priority is given to shipping spent fuel to a government reprocessor so that technical specification restrictions associated with the fuel being onsite can be eliminated. In addition, guard forces can be reduced. The MTR-type fuel is prepared for shipment by cutting the aluminum box ends off each spent fuel assembly before loading a critically safe complement of fuel plates into the spent fuel casks. The aluminum box end pieces are packaged for shipment to the burial ground.

However, once the irradiated fuel from the reference test reactor and the MUR either has been placed in storage in Canal G or Canal H, respectively, or shipped from the site and once the PCWS and associated fluid handling systems have been drained and/or flushed, disassembly, disposal, and further decontamination can proceed promptly. Work begins in the Reactor Building/Containment Vessel, proceeds through the Hot Laboratory Building, and concludes with the Waste Handling Building. With the exception of the exhaust stack, the auxiliary structures described in Section C.2.12 of Appendix C are assumed to be uncontaminated.

I.2.1.1 Decommissioning Methods for DECON at the Reference Test Reactor Facility

The disassembly methods proposed for DECON at the reference test reactor and the MUR employ techniques that have been used successfully and are described generically in Appendix G. Generic descriptions for the dismantlement of equipment located in each building are not repeated after they are first discussed (i.e., for piping and equipment, contaminated concrete surfaces, and HVAC and electrical equipment). The removal sequence of equipment and systems gives priority to removal of material having high radiation and/or high contamination levels in order to minimize exposure to the personnel by

attaining low background levels as soon as practicable. DECON methods are discussed in the following subsections for each of the buildings/areas containing radioactive materials.

Reactor Building, MUR, and Containment Vessel--DECON at the Reactor Building (RB) and the Containment Vessel (CV) includes the removal and packaging for shipment of the following activated or contaminated equipment and materials:

- The reference test reactor internals, test reactor vessel, and associated sub-pile room equipment.
- The Mock-Up Reactor (MUR).
- The fuel storage racks for both the reference test reactor and the MUR, located in Canals G and H, respectively.
- Canal H experimental hardware storage rack.
- Assorted experimental hardware from the quadrants and canals; specifically, Quadrants A, B, C, and D and Canals E, F, G (after all spent fuel is shipped), and H (after all spent fuel is shipped).
- Contaminated equipment, tanks, pumps and piping, exhaust air handling equipment and ductwork, and activated and contaminated concrete.
- Reactor Building contaminated materials and equipment located outside the CV.

Once the Q&Cs are drained, cleaned, and dried, the contaminated concrete is removed by mechanical means, primarily scarfing, from these areas (see Table D.2-3, Appendix D for estimated volumes).

All of the neutron-activated materials, as well as some of the radioactively contaminated materials in the reference test reactor and in the MUR, are located in the Primary Containment Vessel and Canal H of the Reactor Building, respectively. As discussed previously in Section E.2 of Appendix E, the reference test reactor vessel internals and portions of the test reactor vessel are estimated to contain the majority of neutron-activated materials ($\sim 3.7 \times 10^5$ Ci), while the total inventory of radioactivity in the MUR is

estimated to be quite small (<2 Ci). Neutron-activated components of the reference test reactor are cut into pieces that will fit steel liners for the B3 shielded shipping cask or other specially made wooden boxes 1.27 m by 1.27 m by 3.04 m long, depending upon the level of neutron activation (see descriptions in Section G.6 of Appendix G).

A conceptual generalized approach to dismantling the reference test reactor is described below, while a detailed step-by-step sequence for removing the reactor tank internals and the reactor tank is presented in Reference 7. The generalized procedure includes the following:

- The reactor tank is flooded with water to provide shielding. Deionized water is used to promote optical clarity and to facilitate decontamination of waste water by ion exchange techniques.
- Because of the bolted construction, most of the reactor core is disassembled remotely under water using mechanical devices. The various horizontal and vertical test ports and horizontal beam ports are constructed of aluminum and are sectioned and removed mechanically. High temperatures are not applied to lockalloy and beryllium pieces due to the high inventory of tritium entrapped within the metal. Each core component is cut into pieces small enough to transfer safely to shielded shipping casks. Most transfers are performed under water. Air transfer is permitted only when health and safety controls can ensure compliance with the exposure limits.
- After the beryllium pieces with their contained tritium inventory are cut and removed from the tank, then more conventional high-temperature cutting devices (e.g., plasma-arc torch) are used to facilitate further sectioning of in-tank components.
- Following removal of the reactor core, dismantling proceeds to the upper and lower flow guides and the control rod drive housing.
- Following removal of all in-tank components, underwater suction cleaners are used to clean any chips of debris from the bottom of the reactor tank. Then, the tank is gradually drained of its water

and monitored as the water is removed to verify the presence of reasonable dose rates for further work in air. At this point, the thermal shields are either sectioned in-tank or removed and sectioned elsewhere in the facility.

The feasibility of this generalized procedure is supported by the fact that considerable in-tank work was done throughout the operation of the reference test reactor. Substantial replacement of key portions of the reactor core box took place on three occasions during the reactor's operating history. This work was performed remotely with water shielding over the core and supports the assumption that dismantling of the core and core support structure can likewise be accomplished remotely with water shielding.(7)

After all in-tank components have been removed, the inner reactor tank surfaces are decontaminated before cutting and removal operations begin. It is assumed that tank removal is accomplished with a cutting torch. A conceptual generalized procedure for reactor tank removal includes the following nine steps:

1. The upper flange is cut off.
2. An adjustable-height work platform is installed in the tank. It is anticipated that the platform is supported from the lily pad.
3. Each gooseneck-shaped experiment penetration is cut circumferentially and removed from the trench.
4. A lifting eye is welded on the top edge of the reactor tank and a small crane is attached in preparation for lifting that section.
5. As-large-as-practical sections of the reactor tank are cut out and removed. Where the cutout section contains a tank penetration, the penetration is circumferentially cut and the pipe left in place to be removed later.

   NOTE: Since the activity level is not high, the size of the pieces removed is controlled largely by convenience of handling and appropriate shipping regulations.

6. Proceeding around the tank, similar sections are cut and removed as described in steps 4 and 5.

7. Thus, levels of the reactor tank are removed section-by-section until the core support is reached. The remaining portion of the core support is burned off and removed from the reactor tank.

8. The four instrument thimbles that extend upward into the reactor tank are cut off and removed.

9. The bottom of the reactor tank is cut into manageable-sized pieces and removed. Also, the tank support legs that are attached to the outside of the reactor tank and buried in the biological shield concrete are cut and removed at this time.

Contaminated piping and equipment inside the CV is removed, segmented, and packaged in standard or specially constructed shipping boxes for disposal. Cutting is accomplished with oxyacetylene or plasma arc cutting torches or portable power hacksaws. In general, the removal of a piping system, once started, is followed through to completion. For the more complex systems, this removal procedure avoids confusion and/or inadvertent spills.

Contaminated concrete surfaces in the Reactor Building/Containment Vessel are removed using the concrete spaller described in Section G.4.3.1 of Appendix G. A surface layer about 50 mm thick is assumed to be removed. The rubble is packaged in standard shipping boxes for disposal. Canal floor drains and the four sumps are removed.

The contaminated HVAC systems in the Reactor Building/Containment Vessel are cut up into segments that are packaged in standard shipping boxes for disposal. Oxyacetylene or plasma arc torches, power hacksaws, and/or nibblers are used for cutting. Contaminated piping in the Reactor Building is cut into pieces that are packaged in standard shipping boxes. Cutting is accomplished with oxyacetylene or plasma arc torches or portable hacksaws. Contaminated equipment in the Reactor Building is removed and packaged in standard shipping boxes. The spent fuel storage racks in Canals G and H and the experiment storage rack in Canal H are packaged in boxes for shipment to the burial ground.

Disassembly of the MUR--The MUR core structure and all other equipment in Canal H are removed, including the MUR cleanup system at the -4.6-m elevation.

The entire MUR core box with beryllium, beam tube mock-ups, flow guide, rod box, and support frame is packaged in one wooden box, 1.83 m by 1.83 m by 1.83 m, for shipment to a licensed disposal site.

Hot Laboratory Building--The following contaminated areas within the Hot Laboratory Building (HLB) are anticipated to require cleanup (see Figures C.2-7 and C.2-8 in Appendix C for details):

- Hot Cells 1 through 7
- Hot Dry Storage
- Canal J
- Canal K
- Off-Gas Cleanup Room
- Valve Pit
- Hot Pipe Tunnel
- Hot Handling Room 17
- Hot Work Area Room 16
- Decontamination Room 23
- Repair Shop Room 22
- Sump.

The total amount of contaminated material and equipment is dependent on what remains in the reference test facility at the time of final shutdown. It is assumed that all unneeded, stored equipment in the HLB has been removed and disposed of before decommissioning starts. As part of the overall facility equipment inventory and radiation survey previously described, loose and salvageable equipment are identified early in the decommissioning schedule for later disposal.

The hot cells are decontaminated using remote acid rinsing, jet-cleaning equipment, and other physical cleaning methods described in Section G.4.2 of Appendix G. A conceptual generalized approach to decontamination of the hot cells includes:

- Remotely rinsing equipment and cell interiors with acidic solutions for the more easily removed contamination.
- Manned-entry into the cell with emplacement of local shielding to allow continued decontamination by physical cleaning methods (see Section G.4.2 in Appendix G).
- Use of water-jet equipment to decontaminate the inside of equipment and the cell interiors.
- Further decontamination using "safety solvents" may be required if the cell was used extensively to handle organic materials.

The moderate volume of decontaminating liquids is evaporated and processed as low-level waste using the in-plant radwaste systems.

Because of the relatively large radiation exposure potential estimated for hot cell cleanup, specialized hot cell training is anticipated for the decommissioning workers. To lead this effort, this training is conducted by a small cadre of hot cell experts who remain after final shutdown. The training includes specifics on hot cell entry with double coveralls, forced-air masks and their proper application, and in-cell work programs and "dry runs" for efficiency and speed (ALARA considerations).

The salvageable equipment (see Section C.3.2 of Appendix C) is cleaned and removed while the remaining equipment is packaged for burial. Removal of structural material, piping, and stainless steel liners is accomplished by scarfing and conventional torch cutting devices, respectively, as previously described for the RB/CV. A surface layer of concrete about 50 mm thick is assumed to be removed from about 40% of the total surface area of the hot cells, while all of the stainless steel liners are anticipated to require burial as radioactively contaminated material.

Unneeded salvageable equipment from other areas within the HLB, including cranes (i.e., if structure demolition does not follow DECON activities), cell shielding windows (see Section C.3.8.2 in Appendix C for recommended

precautions during removal), manipulators,[a] periscopes, and cameras are removed and disposed of. Other major items of substantial salvage value (not included as a cost offset of decommissioning the reference test reactor) that are either anticipated to be clean or capable of being decontaminated include the 76.2-Mg lead sliding door in the Hot Work Area and varying amounts of copper (from wiring and other electrical components) and aluminum and scrap steel. The off-gas cleanup system equipment is removed and packaged for burial as is the cladding from the decontamination room and the hot sump area.

Contaminated piping and equipment inside the HLB is removed, segmented, and packaged in standard or specially constructed shipping boxes for disposal. Cutting is accomplished with oxyacetylene or plasma arc cutting torches or portable power hacksaws.

Contaminated concrete surfaces in the HLB (see Table D.2-3 in Appendix D for details) are removed using the concrete spaller described in Section G.4.3.1 of Appendix G. A surface layer about 50 mm thick is assumed to be removed. The rubble is packaged in standard shipping boxes for disposal. Canal floor drains are removed.

The pipes and sump in the Hot Pipe Tunnel pertaining to only the HLB areas are removed and packaged for burial. The tunnel is decontaminated, including the scarfing of an estimated 50% of the total concrete surface area of the tunnel only after the piping and HVAC runs within the tunnel are no longer being used. The contaminated air-handling systems within the Hot Pipe Tunnel that can affect the safe functioning of other structures and areas are sequentially scheduled for removal as each structure or area is decommissioned. Therefore, the HVAC systems are the last item to be decommissioned. These systems are cut into segments that are packaged in standard shipping boxes for disposal. Oxyacetylene or plasma arc torches, portable hacksaws, and/or nibblers are used for cutting.

(a) It is estimated that ~1 man-week/manipulator of decontamination work is required to achieve unrestricted release levels. The end-piece hand is considered unsalvageable and is packaged for burial.

Other Contaminated Structures and Areas--Ten of the 21 structures/areas at the reference test reactor facility have major radiological involvement (see Figure C.1-1 in Appendix C). The two buildings with the largest involvement were described in the previous two subsections. Some of the radioactive materials are well defined and their disposition is straightforward. This material can be either decontaminated or shipped in accordance with applicable regulations for burial. As discussed in Appendix L and elsewhere in this study, other areas are not as well defined, and during an actual decontamination and dismantling process it is likely that differing quantities of materials might have to be removed than are postulated in this study. This is particularly true in the case of contaminated soil in the areas of site ditches and the Emergency Retention Basin.

The remaining structures and areas to be decontaminated and dismantled are as follows:

- Primary Pump House
- Office and Laboratory Building
- Emergency Retention Basin and Site Ditches
- Cold Retention Area
- Hot Retention Area
- Fan House
- Waste Handling Building
- Miscellaneous: Hot Pipe Tunnel and Stack.

Upon completion of a comprehensive radiation survey and following a general cleanout, DECON activities commence. These are summarized as follows:

1. The areas close to truck doors in the Waste Handling Building, Hot Laboratory Building, and Fan House are decontaminated first to provide a staging area for the storage, packaging, and shipment of equipment and materials from the plant. A nonflammable, strippable coating is placed on the concrete floors to ensure that any inadvertent contamination occurring at the staging areas can be easily cleaned up.

2. To improve accessibility, the remaining plant-controlled areas are decontaminated, generally in the order of decreasing amount of contamination and increasing distance from the staging area. Where practical, equipment is packaged in the original room. Some equipment requiring decontamination for unrestricted use is sent to the storage area for cleanup when in-situ cleanup is not practical.

3. The Fan House ventilation filter rooms and the Waste Handling Building are decontaminated near the end of the project, so they can be used to support the decommissioning operations.

4. The laundry room and change rooms are the final rooms to be decontaminated.

Concurrent with Steps 1 through 4 above, the Hot Retention Area, the Cold Retention Area, the site ditches, and the Emergency Retention Basin are dismantled and decontaminated.

To the maximum extent possible, the liquid and solid radwastes from these DECON activities are processed in the in-plant radwaste systems before the systems are dismantled. A temporary, portable radwaste system is provided by a specialty contractor and is used to process liquid waste generated after dismantlement of the in-house liquid radwaste systems is started. In addition, temporary laundry services are provided as needed by a specialty contractor after these in-house services are dismantled.

The techniques used in dismantling the piping and equipment and the removal of concrete surfaces and air-handling equipment in these structures/ areas are essentially the same as those described previously for the Reactor Building/Containment Vessel and the Hot Laboratory Building.

The final DECON activity is to conduct a thorough radiation survey and cleanup of all the uncontrolled areas in and around the reference test reactor facility.

I.2.1.2 Special Tools and Equipment

Because of the diversified nature of the operations at the reference test reactor, many of the tools and equipment needed for DECON are anticipated to

already be onsite at final shutdown. Some of the more unique devices used for DECON are described in Sections G.3, G.4, and G.5 of Appendix G. A listing of special tools and equipment for DECON and their functions is given in Table I.2-1.

I.2.1.3 Summary of Disassembly Methods

A summary of the dismantlement data for the reference test reactor facility is given in Table I.2-2. DECON tasks that result in the removal of materials that contain neutron-activation products or are radioactively contaminated are identified, cutting data are presented, and removal conditions and methods are given.

I.2.2 Schedules and Manpower Requirements for DECON at the Reference Test Reactor Facility

This section presents the postulated schedules and manpower requirements necessary to eliminate all license requirements of the reference test reactor via the DECON alternative. The development of these estimates is the result of: 1) applying engineering judgement based on previous decommissioning studies in this series;[2-4] and 2) analyzing, modifying, and combining limited historical quantitative data from decommissioning tasks actually completed while placing the reference test reactor into safe storage in early-1973[8] with selected information from a recent definitive study aimed at the delayed dismantlement of the reference test reactor based on its current safe storage status.[9] It should be recognized that while common information is used from the aforementioned references, the similarity ends there. For example, the study in Reference 9 is based on a task-wise analysis that is necessary to complete delayed dismantlement after about 5 years of safe storage and this section focuses on DECON at a reference test reactor, which by definition means decommissioning shortly after final shutdown--two somewhat different decommissioning projects.

Following fuel removal from the reference test reactor and the MUR, the sequence in which the various systems must be drained and/or flushed and dismantled is determined. Next, the task time requirements and the numbers and types of decommissioning workers required to accomplish each task in the

TABLE I.2-1. Special Tools and Equipment for DECON at the Reference Test Reactor

Item	Number Required[(a)]	Function
Underwater Plasma-Arc Torch	1	Sectioning Vessel Internals
Underwater Oxyacetylene Torch	1	Sectioning Reactor Vessel
Portable Plasma-Arc Torch	2	Cutting Stainless Steel Equipment, Piping, and Pool Liners
Portable Oxyacetylene Torch	3	Sectioning Reactor Vessel and Cutting Carbon Steel Equipment and Piping; Welding Carbon Steel
Guillotine Pipe Saw	4	Cutting Piping
Power-Operated Reciprocating Hacksaw	4	Sectioning Piping and Equipment
Nibbler	4	Sectioning Mild Steel and Stainless Steel Components
Closed Circuit TV System	As Required	Observation of Remote or Underwater Operations
Submersible Pump with Disposable Filter	3	Draining Operations
High-Pressure Water Jet	2	Surface Decontamination of Walls and Equipment
Powered Floor Scrubber	3	Decontamination
Wet-Dry Vacuum Cleaner (HEPA Filtered)	3	Removal of Contaminated Debris
Supplied-Air Plastic Suit	50	Provide Personnel with Maximum Respiratory and Body-Surface Protection from Radioactive Contaminants
Power-Operated Mobile Manlift	1	Safe Access to Heights; Scarfing Quadrants and Canals and Disassembly of Air Handling Equipment and Ductwork
9100-kg Mobile Hydraulic Crane	1	Removal and Packaging of Contaminated Piping and Equipment
9100-kg Forklift	1	Packaging of Contaminated Materials, and Loading of Trucks
Concrete Drill with HEPA Filtered Dust Collection System	2	Drilling Holes in Concrete as Required for Blasting or Surface Spalling
Concrete Surface Spaller	2	Removal of Contaminated Concrete Surfaces
Front-End Loader (Light Duty)	1	Cleanup and Packaging of Concrete Rubble
Portable Filtered Ventilation Enclosure	5	Contamination Control During Cutting of Contaminated Material
Filtered-Exhaust Fan Unit	2	Contamination Control and Personnel Comfort
Polyurethane Foam Generator	2	Contamination Control During HVAC Removal
Paint Sprayer	2	Contamination Control During Equipment Removal
HEPA Filter	50	Contamination Control
Roughing Filter	50	Contamination Control

(a) Based on a taskwise analysis of the DECON schedule.

TABLE I.2-2. DECON Removal Data and Methods for the Reference Test Reactor

Location / Task	Removal Data: Metal: Extimated Cutting Length (m)	Metal: Nominal Material Thickness (m)	Metal: Estimated Cutting Time (hr)	Concrete: Number of Holes	Concrete: Estimated Surface Area (m^2)	Concrete: Estimated Operational Time (hr)[a]	Methods
Reactor Building/MUR/Containment Vessel							
7. Remove, Package, and Ship MUR and Associated Hardware	8.0×10^0	--[b]	5.0×10^0				Plasma-arc. Most connections are bolted, so a minimum of cutting is required.
12. Remove RV Internals and Ship RV Internals — Stainless Steel	1.9×10^2	4.8×10^{-2}	9.5×10^0				Plasma-arc torch.
12. (cont.) — Aluminum	1.8×10^1	6.6×10^{-2}	9.0×10^{-1}				
13. Remove RV and Ship RV Segments — Stainless Steel	2.7×10^2	3.2×10^0	5.6×10^1				Plasma arch torch. RV drained and then cut in a contamination control envelope.
13. (cont.) — Aluminum	2.3×10^1	1.3×10^0	4.8×10^0				
14. Remove Bio-Shield Concrete				2.5×10^2	21×10^0	1.0×10^1	Drill holes for removal of a 0.051-m contamination depth. Spall surface layer with hydraulic spaller. Package.
15. Remove Fixed Equipment in CV (except HVAC)	2.7×10^1	--	2.7×10^1				Oxyacetylene torch. Unbolting and direct cutting operations.
16. Remove Fixed Equipment Outside CV	3.0×10^0	--	3.0×10^0				Oxyacetylene torch. Unbolting and direct cutting operations.
17. Remove Quadrant Piping	2.6×10^2	--	2.2×10^{-1}				Plasma-arc. Direct cutting in air. Cut into 2.1-m segments. Package.
18. Segment and Remove Sub-Pile Room	4.0×10^1	--	5.0×10^0				Oxyacetylene torch. Direct cutting in air.
19. Remove Lead Sheild from Below Reactor Cavitiy	3.0×10^1	0.3×10^0	3.8×10^0				Oxyacetylene torch. Direct cutting in air after surrounding concrete is removed.
20. Remove Pipes from Bio-Sheild	7.6×10^1	--	7.6×10^0				Plasma-arc. Explosives for concrete removal. Direct cutting in air.
21. Remove PCWS Piping from Bio-Shield	7.6×10^1	--	3.0×10^0				Plasma-arc. Direct cutting in air.
22. Remove RB/CV Contaminated Concrete				7.1×10^4	6.0×10^3	2.1×10^3	Drill holes for removal of a 0.051-m contamination depth. Spall surface layer with hydraulic spaller. Package.
23. Remove Q&C, and Misc. Contaminated Drains	3.2×10^2	--	3.2×10^1				Oxyacetylene torch. Direct cutting in air after surrounding concrete is removed.
24. Remove Contaminated HVAC from RB/CV	1.4×10^2	--	2.6×10^1				Oxyacetylene torch and nibbler. Cut ductwork and hangers. Package.
Hot Laboratory Building							
2. Remove and Package Hot Cell Equipment and Piping	1.9×10^3	--	1.6×10^2				Oxyacetylene torch. Direct cutting in air.
3. Remove and Package Hot Cell Stainless Steel Cladding	4.7×10^3	3.2×10^{-3}	1.9×10^2				SS cladding is removed by a portable plasma arc torch. Cut into 2.53-m^2 sections. Package.

TABLE I.2-2. (Contd)

Location / Task	Metal: Extimated Cutting Length (m)	Metal: Nominal Material Thickness (m)	Metal: Estimated Cutting Time (hr)	Concrete: Number of Holes	Concrete: Estimated Surface Area (m^2)	Concrete: Estimated Operational Time (hr)[a]	Methods
Hot Laboratory Building (contd.)							
4. Remove Contaminated Concrete from Hot Cells				6.9×10^3	2.3×10^2	2.1×10^2	Drill holes for removal of a 0.051-m contamination depth. Spall surface layer with hyudraulic spaller. Package.
8. Remove Fixed and Permanent Eqpt. (except HVAC) Including the Hot Pipe Tunnel	1.3×10^1	--	1.3×10^0				Oxyacetylene torch. Unbolting and direct cutting operations.
9. Remove Stainless Steel Cladding from Decon. Room 23	5.8×10^1	3.2×10^{-3}	2.3×10^1				SS cladding is removed by a portable plasma arc torch. Cut into 2.53-m^2 segments. Package.
11. Remove and Package HLB Contaminated Concrete				1.4×10^4	1.2×10^3	4.2×10^2	Drill holes for removal of a 0.051-m contamination depth. Spall surface layer with hydraulic spaller. Package.
12. Remove and Package Contaminated HVAC from HLB	2.8×10^2	--	5.1×10^1				Oxyacetylene torch and nibbler. Cut ductwork and hangers. Package.
Other Contaminated Structures & Areas							
2. PPH							
Remove Fixed Equipment (except HVAC)	4.2×10^1	--	4.2×10^0				Oxyacetylene torch.
Remove Contaminated Concrete				2.1×10^3	1.8×10^2	6.4×10^1	Drilling/spalling.
Remove & Package Contaminated HVAC	2.7×10^2	--	5.0×10^1				Oxyacetylene torch.
3. OLB							
Remove Contaminated Hoods & Sinks	1.3×10^2	--	5.3×10^1				Oxyacetylene torch.
Remove Contaminated Concrete				6.0×10^2	5.1×10^1	1.8×10^1	Drilling/Spalling
5. CRA							
Remove Contaminated Concrete				1.4×10^4	1.2×10^3	4.2×10^2	Drilling/Spalling
6. HRA							
Remove & Package Contaminated Piping	1.5×10^2	--	1.5×10^1				Oxyacetylene torch.
Remove & Package HRA Tanks, Floor Plates, & Partitions	3.0×10^3	1.5×10^{-2}	1.2×10^3				Oxyacetylene torch. Cut into 2.53-m^2 sections. Package.
Remove Contaminated Concrete				2.5×10^3	2.1×10^2	7.5×10^1	Drilling/spalling.
7. Fan House							
Remove Contaminated Concrete				7.2×10^2	6.2×10^1	2.2×10^1	Drilling/spalling.
Remove Fixed Equipment	1.8×10^1	--	7.2×10^0				Oxyacetylene torch.
Raze Stack, Segment, & Remove	1.9×10^2	--	7.6×10^1				Oxyacetylene torch. Cut in a contamination control envelope.
8. Waste Handling Building							
Remove Fixed Equipment Including Evaporator & HVAC	1.3×10^1	--	1.3×10^0				Oxyacetylene torch.
Remove Contaminated Concrete				1.9×10^3	1.6×10^2	5.7×10^1	Drilling/spalling.

(a) Operational time includes both drilling and spalling but does not include set-up or clean-up time.
(b) Various material thickness involved in the cutting operation.

allotted time are estimated (the functions and organization of the decommissioning staff for a conceptual DECON at the reference test reactor are discussed in Section H.2 of Appendix H). Based on the estimated dose to accomplish each task, the number of workers needed to complete the radiation zone work in the allotted time and within the assumed radiation dose limits is determined. Whole-body radiation doses to the decommissioning workers are limited in accordance with 10 CFR 20.101.[1] The supervisors, utility operators, and health physics technicians are assumed to be long-time radiation workers whose annual exposure is limited to 5 rem/yr by the formula 5(N-18) of 10 CFR 20.101(b)(2). The craftsmen and laborers are assumed to have had little prior radiation exposure and, therefore, under 10 CFR 20.101(b)(1) and (2) may receive up to 3 rem per quarter, within the limitation of the formula 5(N-18) rems where "N" equals the individual's age in years at his last birthday. If a situation occurs where the manpower estimated for physically accomplishing a task results in an average dose for a person in excess of these limits, an additional person is anticipated to be assigned to the task to keep the individual occupational dose below these limits. In the manpower tables presented in this section, the manpower shown is adequate to both accomplish the task and to meet the occupational dose limits.

After all radiological work is completed, it is estimated that an additional 1680 man-days of effort (e.g., removing uncontaminated loose and fixed equipment) from the buildings at the reference test facility would be necessary before demolition of the structures could take place, if desired.

Task schedules and decommissioning worker requirements for DECON at the Reactor Building/MUR/Containment Vessel, the Hot Laboratory Building, and the other contaminated structures and areas are discussed and presented in the following three subsections. The decommissioning worker requirements given in these subsections do not include the management and support staff (see Figure H.2-2 in Appendix H). A fourth subsection presents the overall task schedule and decommissioning worker requirements and the total staff labor requirements for DECON at the reference test reactor facility.

The task completion schedules are presented on a calendar-month time scale and also present additional information. For example, the following:

|————————| 6 / 1488 / 3

means that this particular task is performed by six decommissioning workers and is estimated to require a total of 1488 man-hours to complete the task within a 3-calendar-month time span. Likewise the following:

|————————| Subcontract / 2

indicates that this task is performed by subcontracted labor over a span of 2 months.

I.2.2.1 Reactor Building, MUR, and Containment Vessel

All of the neutron-activated materials and a significant amount of contaminated materials as well are in the Reactor Building/Containment Vessel (RB/CV) complex. Based on the DECON methods discussed previously in Section I.2.1 and the occupational dose estimates presented later in Section I.2.4, the task schedule and sequence and the decommissioning worker requirements for DECON of the RB/CV (including the MUR) are presented in Figure I.2-1. The MUR is postulated to be removed first, before the reference test reactor, to provide a potential "lessons learned" dismantling basis. DECON of the RB/CV and MUR is estimated to be accomplished in about 21 months following final reactor shutdown and is estimated to require about 279 man-months of decommissioning worker effort.

I.2.2.2 Hot Laboratory Building

Hot Cells 1 and 2 have more deposited radioactive contamination per unit area than the other five cells (see Table E.2-10, Appendix E) and, therefore, have the highest radiation dose rates for DECON of the contaminated piping and equipment. In arranging the task schedule for DECON of the HLB, the work is planned so that lower exposure radiation work in this building is in progress at the same time as much of the higher exposure radiation work. In this way (by rotation of workers) more efficient use of manpower from an occupational dose consideration can be achieved.

TASK	REACTOR BUILDING/MUR/CONTAINMENT VESSEL: MONTHS AFTER SHUTDOWN (1–21)
1. COMPREHENSIVE RADIATION SURVEY FOR TOTAL FACILITY (i.e., ALL BUILDINGS)	6 / 528 / 1
2. DISCHARGE FUEL (INCLUDING MUR)	5 / 66 / 0.1
3. PREPARE AND SHIP SPENT FUEL	8 / 2500 / 3
4. GENERAL CLEANUP AND EQUIPMENT INVENTORY (i.e., ALL BUILDINGS)	14 / 924 / 0.5
5. DRAIN, CLEAN, DRY QUADRANTS A, B, C AND D AND CANALS E AND F	10 / 780 / 0.6
6. DRAIN, CLEAN, DRY CANAL H	10 / 180 / 0.14
7. REMOVE, PACKAGE, AND SHIP MUR AND ASSOCIATED HARDWARE	10 / 576 / 0.45
8. DRAIN, CLEAN, DRY CANAL G	10 / 180 / 0.14
9. REMOVE LOOSE EQUIPMENT IN Q AND C'S AND DRY ANNULUS	10 / 936 / 0.55
10. DRAIN AND FLUSH PCWS	6 / 72 / 0.1
11. ISOLATE RV AND ADD DEIONIZED WATER FOR SHIELDING	6 / 72 / 0.1
12. REMOVE RV INTERNALS AND SHIP ACTIVATED RV INTERNALS	16 / 10366 / 3.75
13. REMOVE RV AND SHIP RV SEGMENTS	16 / 2916 / 1.14
14. REMOVE ACTIVATED BIO-SHIELD CONCRETE	8 / 120 / 0.11
15. REMOVE FIXED EQUIPMENT IN CV (EXCEPT HVAC)	24 / 5890 / 1.5
16. REMOVE FIXED EQUIPMENT OUTSIDE CV	24 / 1728 / 0.55
17. REMOVE QUADRANT PIPING	12 / 360 / 0.23
18. SEGMENT AND REMOVE SUBPILE ROOM	6 / 252 / 0.32
19. REMOVE LEAD SHIELD FROM BELOW REACTOR CAVITY	6 / 252 / 0.32
20. REMOVE PIPES FROM BIO-SHIELD	8 / 1000 / 0.82
21. REMOVE PCWS PIPING TO PPH	10 / 2820 / 2.14
22. REMOVE RB/CV CONTAMINATED CONCRETE	10 / 3240 / 2.5
23. REMOVE Q AND C, AND MISCELLANEOUS CONTAMINATED DRAINS	15 / 2520 / 1.62
24. REMOVE CONTAMINATED HVAC FROM RB/CV	8 / 960 / 0.91
25. FINAL RADIATION SURVEY	2 / 100 / 1

ABBREVIATIONS:

CV	CONTAINMENT VESSEL
HVAC	HEATING, VENTILATION, AND AIR CONDITIONING
MUR	MOCK-UP REACTOR
PCWS	PRIMARY COOLING WATER SYSTEM
PPH	PRIMARY PUMP HOUSE
Q&C	QUADRANTS AND CANALS
RB/CV	REACTOR BUILDING/CONTAINMENT BUILDING
RV	REACTOR VESSEL

PERSONS PER DAY/EXPOSURE HOURS/CALENDAR MONTHS

|——| CONTINUOUS OPERATIONS OVER THE TIME SPAN SHOWN

TOTALS: PERSON-MONTHS 279
PERSON-HOURS 48,605
EXPOSURE-HOURS 39,338

FIGURE I.2-1. Task Schedule and Sequence and Decommissioning Worker Requirements for DECON at the Reactor Building/Primary Containment/MUR

An estimate of the main preparatory activities before equipment/systems dismantlement begins in the HLB is given in Table I.2-3. The task schedule and sequence and the decommissioning worker requirements are given in Figure I.2-2. DECON at the HLB is estimated to be completed in about 15 months following final reactor shutdown and is estimated to require approximately 88 man-months of decommissioning worker effort.

I.2.2.3 Other Contaminated Structures and Areas

After decommissioning of the RB/CV/MUR and HLB, the remaining structures and areas are dismantled and decontaminated. These are:

- Primary Pump House
- Office and Laboratory Building
- Emergency Retention Basin and Site Ditches
- Cold Retention Area
- Hot Retention Area
- Fan House
- Waste Handling Building
- Miscellaneous: Utility Tunnels and Stack.

The liquid and solid radwaste systems located in the Fan House and the Waste Handling Building are needed to process most of the contaminated liquids contained in the systems at final reactor shutdown and generated during DECON. In addition, continuous waste air-handling service to the remaining work areas is maintained as long as necessary.

An estimate of the preparatory activities before equipment/systems dismantlement of the aforementioned buildings and areas begins is given in Table I.2-4. The task schedule and sequence and the decommissioning worker requirements are given in Figure I.2-3. DECON at these remaining structures and areas starts in month 15 after final reactor shutdown and continues through month 25, thus requiring an estimated 11 months to complete. In addition, it is estimated to require approximately 192 man-months of decommissioning worker effort.

TABLE I.2-3. Postulated Preparatory Tasks Required Before Equipment/Systems Dismantlement Begins in the Hot Laboratory Building

Location	Preparatory Tasks
All	- Removal of unneeded equipment based on previous equipment inventory. - Perform a safety audit to ensure that all flammable materials and other hazardous materials are removed. - Deenergize and deactivate selected electrical circuits, when applicable per the main DECON schedule.
Hot Cells	- Close barrier doors and disconnect utility services. - Decontaminate tools and equipment and package for shipment. - Decontaminate all cells to lowest practical level. - Flush, remove filters from cell air systems. - Deenergize electrical circuits.
Hot Handling Room 17	- Remove, package, and ship waste. - Decontaminate to lowest practical level.
Hot Dry Storage	- Remove, package, and ship waste. - Decontaminate to lowest practical level.
Storage Room 24	- Decontaminate impact machine.
Repair Shop Room 22	- Decontaminate welding hood. - Remove filters.
Decontaminate Room 23	- Decontaminate to lowest practical level.
Controlled Work Area Room 16	- Decontaminate to lowest practical level.

TASK	HOT LABORATORY BUILDING: MONTHS AFTER SHUTDOWN (1–17)
NOTE: COMPREHENSIVE RADIATION SURVEY, GENERAL CLEANUP, AND EQUIPMENT INVENTORY HAVE BEEN COMPLETED (SEE FIGURE I.2-1)	
1. DECONTAMINATE HOT CELLS	12 / 624 / 0.91
2. REMOVE AND PACKAGE HOT CELLS EQUIPMENT AND PIPING	7 / 2146 / 2.5
3. REMOVE AND PACKAGE HOT CELLS SS CLADDING	10 / 480 / 0.36
4. REMOVE CONTAMINATED CONCRETE FROM HOT CELLS	10 / 336 / 0.25
5. DECONTAMINATE THE HLB (INCLUDING CRANES)	10 / 336 / 0.25
6. DRAIN, CLEAN, DRY CANALS J AND K	10 / 360 / 0.28
7. REMOVE LOOSE EQUIPMENT	6 / 913 / 0.91
8. REMOVE FIXED AND PERMANENT EQUIPMENT (EXCEPT HVAC), INCLUDING THE HOT PIPE TUNNEL	7 / 2146 / 2.5
9. REMOVE SS CLADDING FROM DECONTAMINATION ROOM 23	10 / 120 / 0.1
10. REMOVE HOT CELL WINDOWS	SUBCONTRACT / 1072 / 1.14
11. REMOVE AND PACKAGE HLB CONTAMINATED CONCRETE	10 / 795 / 0.6
12. REMOVE AND PACKAGE CONTAMINATED HVAC FROM HLB	10/ 1200 / 0.91
13. FINAL RADIATION SURVEY	2 / 88 / 0.5

PERSONS PER DAY/EXPOSURE HOURS/CALENDAR MONTHS

├──┤ CONTINUOUS OPERATIONS OVER THE TIME SPAN SHOWN

TOTALS:	
PERSON-MONTHS	88
PERSON-HOURS	15,465
EXPOSURE-HOURS	10,616

FIGURE I.2-2. Task Schedule and Sequence and Decommissioning Worker Requirements for DECON at the Hot Laboratory Building

TABLE I.2-4. Postulated Preparatory Tasks Required Before Equipment/Systems Dismantlement Begins in the Other Buildings and Areas of the Reference Test Facility

Location	Preparatory Tasks
Primary Pump House (PPH)	- All sampling and instrument lines from the PCWS to PPH Room 8 are drained and flushed. - The sampling hood and sinks are decontaminated. - The general area is decontaminated, including the hot sumps and drain. - Unneeded electrical and instrument leads are disconnected.
Office and Laboratory Building (OLB)	- All radioactive material and SNM are disposed of. - Filters are removed from the hood filter housings, the housings are decontaminated, and all roof openings are covered. - All hot drains are identified for removal later. - Unneeded utility lines and electrical leads are disconnected and tagged. - All hazardous, flammable materials and chemicals are removed. - Physics (MUR) Counting Room--all radioactive dosimetry materials are discarded. Unneeded utilities are disconnected and tagged. - Vault: 1) verification that all fueled materials are removed; 2) the vault area and all tools are decontaminated; 3) the criticality alarm reference source is removed and discarded and the alarm source drive equipment is decontaminated. - Sumps in the basement and the utility tunnel to the HLB are decontaminated to lowest practical level prior to removal.
Emergency Retention Basin	- Drain
Cold Retention Area (CRA)	- Decontaminate to lowest practical level prior to removal. - The ground-water sump pumps and level alarms are deactivated and the pump motors are disconnected.

TABLE I.2-4. (contd)

Location	Preparatory Tasks
Hot Retention Area (HRA)	- The exterior of the hot retention tank area is decontaminated to clean zone status.
	- Each of the HRA Tanks 1 through 8 and the hold tanks (4) are flushed and drained. The HRA tanks are then cleaned internally and externally and left dry. The four hold tanks are used as radwaste containers later (see Section L.3.2.6 of Appendix L).
	- The pumps are deactivated, electrical leads disconnected, and all valves are closed.
	- The HRA ground-water sump pumps and alarms are disconnected from service.
Waste Handling Building (WHB)	- All air and water filters are removed from housings and water and air systems are drained.
	- All hot floor drains are flushed.
	- All hot sumps are decontaminated and deactivated.
Fan House	- Decontaminated to lowest practical level, including the Hot Pipe Tunnel.
	- All air and water filters and deionizer resins are removed from housings and deionizers except as noted below under Contaminated Air Systems.
	- All water systems are drained.
	- All air systems are depressurized and the systems drained.
	- All hot drain sumps are cleaned, flushed with clean water, and deactivated.
Ancillary Systems: Hot Drain System	- All hot sumps, lines, and drains are cleaned, flushed, and deactivated/isolated to prevent accumulation or escape of contaminated liquids. Note: any hot drain that drains a clean zone is replumbed into a cold sump where possible.

TABLE I.2-4. (contd)

Location	Preparatory Tasks
Contaminated Air Systems	- In general, after the areas served by the various contaminated air systems are decommissioned and work in the areas is completed, the air-handling systems are shut down and prepared for dismantlement. Note: Dismantlement of a contaminated air-handling system begins at the farthest point from the fan/filter for that system and proceeds to the fan/filter housing. - All roughing filters, prefilters, and HEPA (absolute) filters are removed and disposed of by standard procedures. - The filter housings and accessible piping are vacuumed and washed down as appropriate. - Fan motors are deactivated and disconnected. - The service air electronic and electrical controls associated with the systems are deenergized and left in place. All air control lines are vented to the atmosphere.

I.2.2.4 Overall Schedule and Manpower Requirements

The overall task schedule and sequence for DECON at the reference test reactor is shown in Figure I.2.4. The planning and preparation tasks (see Section H.1 of Appendix H) for the estimated 2-year period preceding reactor shutdown are also included. Those tasks that are necessary to eliminate all license requirements of the reference test reactor via the DECON alternative are estimated to be accomplished in about 25 months after final reactor shutdown.

DECON tasks, with a few exceptions, are performed on two 8-hour shifts, five days per week. Shipment of spent fuel, neutron-activated reactor vessel internals and reactor vessel segments are conducted three shifts per day, 7 days per week, as required. Nearly optimum decommissioning worker requirements are met by using the staff breakdown given in Table I.2-5. It is estimated that each of the two regular shifts and the support crew for each shift shown in the table comprise a relatively constant manpower loading almost to the very end of the DECON project.

TASK	OTHER CONTAMINATED STRUCTURES AND AREAS: MONTHS AFTER SHUTDOWN (15–24)
NOTE: AN INITIAL COMPREHENSIVE RADIATION SURVEY, GENERAL CLEANUP, AND EQUIPMENT INVENTORY HAS BEEN COMPLETED (SEE FIGURE I.2-1)	
1. RADIATION SURVEY AND INVENTORY UPDATE (i.e., ALL "OTHER BUILDINGS/AREAS")	6 / 264 / 0.5
2. PRIMARY PUMP HOUSE (PPH):	
• PREPARATORY TASKS (SEE TABLE I.2-4)	14 / 420 / 0.23
• REMOVE FIXED EQUIPMENT, EXCEPT HVAC (SEE TABLE C.3-5)	14 / 2480 / 1.35; 14 / 1930 / 1.05
• DECONTAMINATE PPH AND REMOVE CONTAMINATED CONCRETE (SEE TABLE D.2-3)	10 / 180 / 0.14
• REMOVE AND PACKAGE CONTAMINATED HVAC (SEE TABLE C.5-1)	10 / 300 / 0.23
• FINAL RADIATION SURVEY	2 / 40 / 0.23
3. OFFICE AND LABORATORY BUILDING (OLB):	
• PREPARATORY TASKS (SEE TABLE I.2-4)	12 / 950 / 0.5
• REMOVE CONTAMINATED HOODS AND SINKS (SEE SECTION C.3.6 OF APPENDIX C)	12 / 519 / 0.33
• REMOVE CONTAMINATED CONCRETE (SEE TABLE D.2-3)	10 / 120 / 0.09
• FINAL RADIATION SURVEY	2 / 24 / 0.09
4. EMERGENCY RETENTION BASIN (ERB) AND SITE DITCHES	
• DRAIN THE ERB	(a)
• REMOVE AND PACKAGE CONTAMINATED PIPING AND SOIL FROM ERB AND SITE DITCHES (SEE TABLE C.4-1)	SUBCONTRACT[(b)]/1338 / 5.46
5. COLD RETENTION AREA (CRA)	
• REMOVE CONTAMINATED CONCRETE (SEE TABLE D.2-3) AND CONTAMINATED SOIL (SEE APPENDIX C, SECTION C.4)	SUBCONTRACT / 0.23; 10 / 1320 / 1; SUBCONTRACT / 0.23
• FINAL RADIATION SURVEY	2 / 36 / 0.14

(a) A MINOR TASK; VERY LITTLE MANPOWER REQUIRED. THIS TASK IS SHOWN FOR COMPLETENESS

(b) ONLY SINGLE SHIFT IS USED FOR THIS TASK.

FIGURE I.2-3. Task Schedule and Sequence and Decommissioning Worker Requirements for DECON at Other Contaminated Structures and Areas

TASK	OTHER CONTAMINATED STRUCTURES AND AREAS: MONTHS AFTER SHUTDOWN (CONTINUED) 19 20 21 22 23 24 25 26 27 28
6. HOT RETENTION AREA (HRA)	
• PREPARATORY TASKS (SEE TABLE I.2-4)	12 / 1584 / 1
• REMOVE AND PACKAGE CONTAMINATED PIPING (SEE TABLE C.3-11)	14 / 1008 / 0.55
• PROVIDE TANK ACCESS TO EIGHT HRA TANKS (SEE APPENDIX L, SECTION L.3.2.6)	SUBCONTRACT / 0.14
• REMOVE AND PACKAGE HRA TANKS 1-8, FLOOR PLATES, AND PARTITIONS (SEE TABLE C.3-10)	14 / 3108 / 1.68
• UNCOVER AND PREPARE HRA TANKS 9-12 FOR SHIPMENT (SEE TABLE C.3-10 AND APPENDIX L, SECTION L.3.2.6)	SUBCONTRACTOR / 0.55
• REMOVE CONTAMINATED CONCRETE (SEE TABLE D.2-3)	10 / 180 / 0.14
• FINAL RADIATION SURVEY	2 / 48 / 0.27
7. FAN HOUSE	
• PREPARATORY TASKS (SEE TABLE I.2-4)	12 / 792 / 0.5
• REMOVE CONTAMINATED CONCRETE (SEE TABLE D.2-3)	10 / 240 / 0.18
• REMOVED FIXED EQUIPMENT (SEE TABLE C.5-2)	12 / 1585 / 1; 12 / 935 / 0.59
• RAZE STACK, SEGMENT AND PACKAGE	SUBCONTRACT / 0.14
• FINAL RADIATION SURVEY	2 / 36 / 0.14
8. WASTE HANDLING BUILDING (WHB)	
• PREPARATORY TASKS (SEE TABLE I.2-4)	12 / 792 / 0.5
• REMOVE FIXED EQUIPMENT, INCLUDING EVAPORATOR (SEE TABLE C.3-4) AND HVAC (SEE TABLE C.5-1)	12 / 2376 / 1.25
• REMOVE CONTAMINATED CONCRETE (SEE TABLE D.2-3)	10 / 180 / 0.14
• FINAL RADIATION SURVEY	2 / 36 / 0.14

PERSONS PER DAY/EXPOSURE HOURS/CALENDAR MONTHS

├──┤ CONTINUOUS OPERATIONS OF THE TIME SPAN SHOWN

TOTALS:		
	PERSON-MONTHS	192
	PERSON-HOURS	30,823
	EXPOSURE-HOURS	22,821

FIGURE I.2-3. (contd)

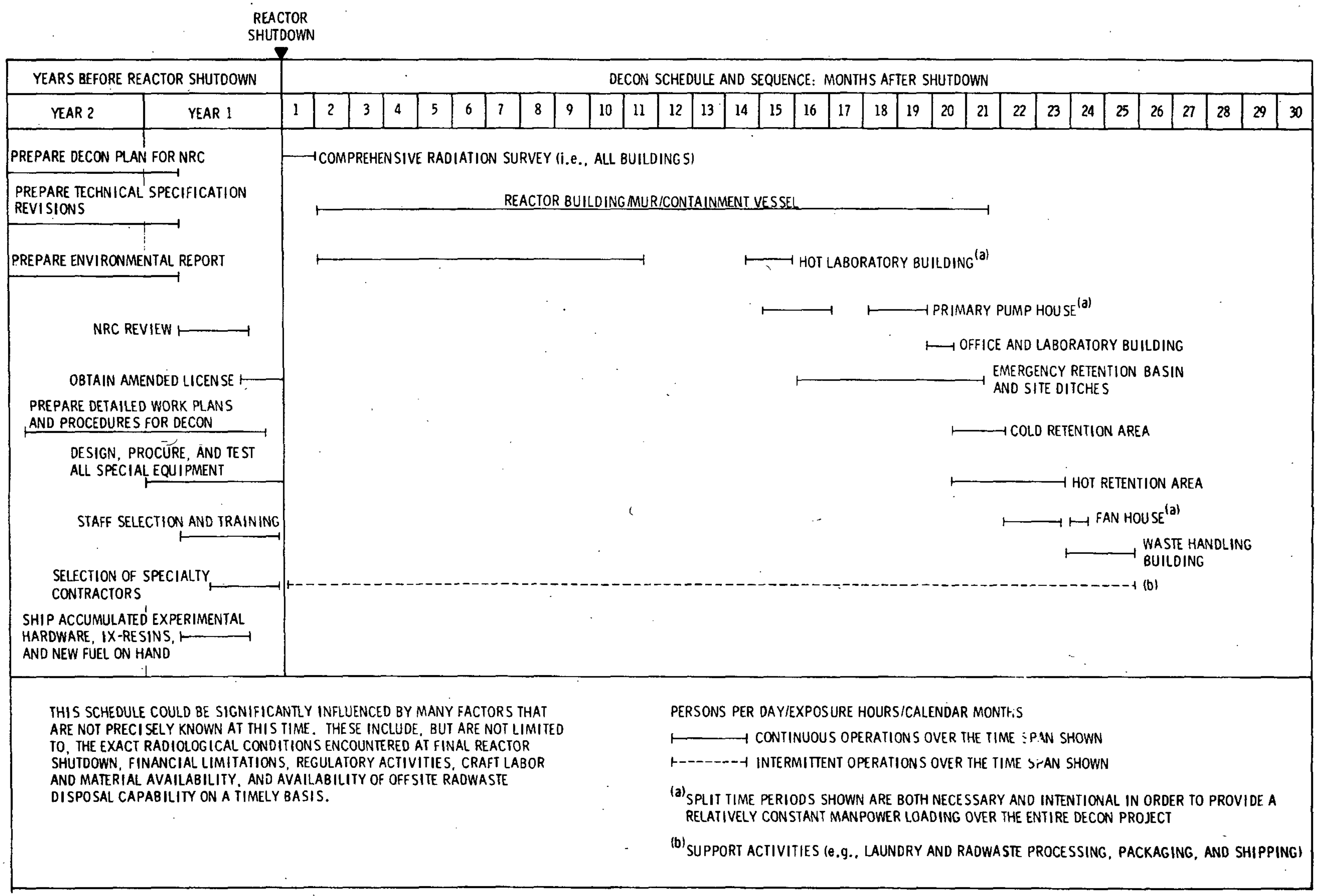


FIGURE I.2-4. Overall Task Schedule and Sequence for DECON at the Reference Test Reactor

TABLE I.2-5. Postulated Number and Specific Makeup of Crews Required for DECON at the Reference Test Reactor

First Shift	Second Shift	Support Crew for Both Shifts[a]	Total Number of Staff
1 Shift Engineer	1 Shift Engineer		2
2 Crew Leaders[b]	2 Crew Leaders[b]	1 Crew Leader	5
4 Utility Operators	4 Utility Operators	3 Utility Operators	11
2 Laborers	2 Laborers	3 Laborers	7
3 Health Physicists	3 Health Physicists	1 Health Physicist	7
5 Craftsmen	5 Craftsmen	—	10
17 Total	17 Total	8 total	42

(a) Anticipated to be equally divided in providing laundry services and radwaste processing services as well as backup service for the first and second shifts as required.
(b) Former Senior Reactor Operator (SRO) or hot cell specialist.

The total staff labor requirements for DECON at the reference test reactor are given in Table I.2-6. These requirements are given in equivalent man-years for the 2 years before and the 2.08 years following final reactor shutdown, and include management and support staff as well as the decommissioning workers. About 232 man-years of effort are estimated for DECON at the reference test reactor, including approximately 126 man-years for the management and support staff and about 106 man-years for the decommissioning workers.

The staff labor requirements shown in Table I.2-6 for the decommissioning workers during the 25 months following final reactor shutdown exceed the total manpower presented in Figures I.2-1 through I.2-3. This "excess" manpower is assumed to: 1) augment the basic work force on an as-needed basis; and 2) provide the manpower necessary for the numerous, small unspecified work items that accompany an activity of this magnitude.

I.2.3 Cost of DECON for the Reference Test Reactor

The cost of DECON for the reference test reactor is presented in this section. The data presented are all late-1980 or early-1981 costs. The

TABLE I.2-6. Staff Labor Requirements for DECON at the Reference Test Reactor

Position	Time Relative to Final Shutdown (year) -2	-1	1	2	3	Total Staff Labor Required (man-years)
	Annual Staff Labor Requirement (man-years)[a]					
Management and Support Staff						
Decommissioning Superintendent	0.3	1.0	1.0	1.0	0.5[b]	3.8
Secretary	1.0	2.0	3.0	3.0	1.0[b]	10.0
Clerk	0	1.0	2.0	2.0	0.5	5.5
Decommissioning Engineer	1.0	1.0	1.0	1.0	0.5[b]	4.5
Assistant Decommissioning Engineer	1.0	1.0	1.0	1.0	0.1	4.1
Radioactive Shipment Specialist	0	1.0	1.0	1.0	0.1	3.1
Procurement Specialist	0.3	1.0	1.0	1.0	0.1	3.4
Tool Crib Attendant	0	0	2.0	2.0	0.2	4.2
Control Room Operator[c]	0	0	5.0	5.0	0	10.0
Security supervisor	0	0	1.0	1.0	0.1	2.1
Security Shift Supervisor[d]	0	0	5.0	5.0	0.2	10.2
Security Patrolman[d]	0	0	16.0	12.0	0.4	28.4
Contracts and Accounting Specialist	0.3	1.0	1.0	1.0	0.5[b]	3.8
Clerk	0	1.0	1.0	1.0	0.5[b]	3.5
Health and Safety Supervisor	1.0	1.0	1.0	1.0	0.5[b]	4.5
Health Physicist	0	0.5	1.0	1.0	0.2	2.7
Protective Equipment Attendant	0	0	2.0	2.0	0.2	4.2
Industrial Safety Specialist	0.3	1.0	1.0	1.0	0.2	3.5
Quality Assurance Supervisor	0.3	1.0	1.0	1.0	0.5[b]	3.8
Quality Assurance Engineer	0.5	1.0	1.0	1.0	0.2	3.7
Quality Assurance Technician	0	0.5	2.0	2.0	0.4	4.9
Consultants (Safety Review Committee)	0.5	0.5	0.5	0.5	0	2.0
Subtotals, Management and Support Staff	6.5	15.5	50.5	46.5	6.9	125.9
Decommissioning Workers[e]						
Shift Engineer	1.0	2.0	2.0	2.0	0.2	7.2
Crew Leader	0	1.0	5.0	5.0	0.5	11.5
Utility Operator	0	3.0	11.0	11.0	0.6	25.6
Laborer	0	0	7.0	7.0	0.4	14.4
Craft Supervisor	0	0.5	2.0	2.0	0.2	4.7
Craftsman	0	5.0	10.0	10.0	1.0	26.0
Senior Health Physics Technician	0	1.0	2.0	2.0	0.2	5.2
Health Physics Technician	0	1.5	5.0	5.0	0.2	11.7
Subtotals, Decommissioning Workers	1.0	14.0	44.0	44.0	3.3	106.3
Totals	7.5	29.5	94.5	90.5	10.2	232.2

(a) Rounded to the next higher 0.1 man-year.
(b) Includes an additional 4 months following active decommissioning in order to complete the documentation and other unspecified license and contract termination requirements.
(c) Based on one operator per shift in the test reactor control room, three shifts per day, seven days per week.
(d) Based on 10 CFR 73; includes both response and access-control personnel on a three-shift, seven-day week basis.
(e) Requirements during the 2 years following reactor shutdown are based on a relatively constant manpower loading using the staff breakdown given in Table I.2-5.

estimated costs and the other possible costs for DECON at the reference test reactor are summarized and totaled in Table I.2-7.

The total cost of DECON is estimated at about $15.6 million including a 25% contingency. Costs for disposal of radioactive materials, staff labor, and specialty contractors (approximately 21%, 69%, and 5% of the total, respectively) are the major contributors to the total cost for DECON. Combined costs for special tools and equipment and for miscellaneous supplies make up an additional 4.5% of the total. Since the reference test reactor for this study is assumed to be federally owned, nuclear insurance and license fees do not contribute to DECON costs.

The total cost for other possible DECON requirements (i.e., spent fuel shipment, facility demolition, and site restoration) is estimated at approximately $3.1 million, including a 25% contingency.

Detailed cost data for the individual cost categories shown in Table I.2-7 are presented and discussed in the following subsections.

I.2.3.1 Disposal of Radioactive Materials

Three distinct types of radioactive materials in the reference test reactor require disposal during DECON: 1) neutron-activated materials, 2) contaminated materials, and 3) radioactive wastes. The total cost of disposal for these materials is about $2.6 million and is approximately 21% of the total DECON cost. The disposal cost includes the container, transportation, and burial costs, but does not include the direct labor costs for removing and packaging these materials. Labor costs are incurred regardless of the decommissioning activity and are discussed later (Section I.2.3.2). Table I.2-8 contains a summary of the number of shipments, the burial volumes, and the disposal costs for the radioactive materials.

The disposal costs of the three types of radioactive materials are discussed in more detail in the following subsections.

Neutron-Activated Materials--All of the neutron-activated materials in the reference test reactor and in the MUR are located in the Primary Containment and Canal H of the Reactor Building, respectively. As discussed previously in Section E.2 of Appendix E, the reference test reactor vessel

TABLE I.2-7. Summary of Estimated Costs of DECON for the Reference Test Reactor

Cost Category		Estimated Costs ($ millions)[a,b]	Percent of Total[c]
Disposal of Radioactive Materials			
Neutron-Activated Materials			
Reference Test Reactor	0.131		
Mock-Up Reactor (MUR)	0.004		
Contaminated Materials	2.338		
Radioactive Wastes[d]	0.099		
Total Disposal Costs		2.572	20.7
Staff Labor		8.63	69.3
Energy		0.076	0.6
Special Tools and Equipment		0.361	2.9
Miscellaneous Supplies		0.203	1.6
Specialty contractors[e]		0.616	4.9
Nuclear Insurance		---[f]	---
License Fees		---[g]	---
Subtotal		12.458	100.0
Contingency (25%)		3.115	
Total, DECON Costs		15.573	
Other Possible Costs			
Spent Fuel Shipment		0.204[h]	
Facility Demolition and Site Restoration		2.289[i]	
Subtotal		2.493	
Contingency (25%)		0.623	
Total, Other Possible Costs		3.116	

(a) 1981 costs.
(b) The number of significant figures shown is for computational accuracy and does not imply precision to the nearest $1000.
(c) Individually rounded to the nearest 0.1%.
(d) Includes both wet solid wastes and dry solid wastes.
(e) Includes selected demolition, explosives, temporary radwaste, and environmental monitoring services.
(f) Indemnity fees are currently $100/yr for each license (i.e., the test reactor license and the MUR license) at the reference test facility and are not included in this study since they represent only a small fraction of 1% of the total decommissioning cost.
(g) Because the reference test reactor is assumed to be federally owned, these fees are not applicable; however, where applicable for other nuclear R&T reactor facilities, the schedule of fees for license amendments and other approvals required by the license or NRC regulations is given in 10 CFR 170.
(h) Does not include costs for handling at the reactor or costs for handling and storage at the repository.
(i) This total cost is only for those demolition tasks remaining after license termination (see Section I.2.3.9 and Appendix L for details).

TABLE I.2-8. Summary of Information on Disposal of Radioactive Materials from DECON at the Reference Test Reactor

Radioactive Material Category	Shipments	Burial Volumes (m^3)[a]	Disposal Costs ($ millions)[b]
Neutron-Activated Materials			
Metal (Test Reactor)	15	56	0.131
Metal (MUR)	1	6	0.004
Concrete[c]	---	---	---
Contaminated Materials			
Metal	35	789	0.367
Concrete and Soil	163	2 779	1.352
Concrete	82	1 194	0.619
Radioactive Wastes			
Wet Solid Wastes	2	17	0.011
Dry Solid Wastes	12	93	0.087
Totals	310	4 934	2.571

(a) Includes disposable containers if required.
(b) The number of figures shown is for computational accuracy and does not imply precision to the nearest $1000.
(c) Irradiation testing of concrete core samples and certain biological shield experiments during reactor operations indicate that there are no significant radiation hazards due to activation products within the concrete shielding surrounding the reactor tank. It is anticipated that this radioactivity is removed by standard mechanical methods once the reactor tank is removed, with the generation of only low specific activity wastes.[7] This concrete is disposed of as contaminated waste material.

internals and portions of the test reactor vessel are estimated to contain the majority of neutron-activated materials ($\sim 3.7 \times 10^5$ Ci), while the total inventory of radioactivity in the MUR is estimated to be quite small (<1-1/2 Ci). Neutron-activated components of the reference test reactor are cut into pieces that will fit steel liners for the B3 shielded shipping cask or other specially made wooden boxes, 1.27 m by 1.27 m by 3.04 m long, depending upon the level of neutron activation (see descriptions in Section G.6 of Appendix G).

A detailed breakdown of the masses, the radioactivity contents, the number of packaging containers, and the disposal costs for the neutron-activated components in the reference test reactor is given in Table I.2-9. The packaged materials require an estimated 15 truck shipments to a shallow-land burial facility and occupy an estimated 56 m^3 of space at the burial facility. The total estimated cost for disposal of the reference test reactor's neutron-activated materials in a shallow-land burial facility is about $131,300.

<u>Disposal of the MUR</u>--A detailed breakdown of the disposal costs for the neutron-activated components in the MUR is given in Table I.2-10. The entire MUR core box with beryllium, beam tube mock-ups, flow guide, rod box, and support frame is packaged in one wooden box, 1.83 m by 1.83 m by 1.83 m. The packaged materials require one truck shipment to a shallow-land burial facility and occupy an estimated 6.12 m^3 of space at the burial facility. The total estimated cost for disposal of the MUR's neutron-activated materials in a shallow-land burial facility is about $4,000.

<u>Contaminated Materials</u>--Contaminated materials in the reference test reactor are assumed to include piping and equipment located in the Reactor Building/Containment Vessel/MUR, the Hot Laboratory Building, and eight other buildings and areas described previously in this appendix. In addition, many concrete surfaces in these buildings and areas are assumed to be contaminated and require surface removal to a depth of about 50 mm. Breakdowns of the disposal costs for contaminated material are given in Table I.2-11.

The breakdowns given in Table I.2-11 are based on the estimated reference inventories presented in Appendices C and D. With few exceptions these estimated inventories are presented as tables in the aforementioned appendices. It should be recognized that information gaps exist in several of the tables referred to simply because of insufficient data available to make a complete estimate. In addition, the estimates are highly dependent on the amount and types of experimental "loose" equipment in the reference test reactor facility at the time of final shutdown. As stated previously, the total amount of this type of equipment is assumed to be quite low since shutdown was anticipated and planned for 2 years in advance.

TABLE I.2-9. Costs of Disposal for Neutron-Activated Materials from the Reference Test Reactor

Component	Estimated Mass (Mg)	Estimated Radioactivity (Ci)	Number of Disposable Containers	Container Costs ($)	Cask[(a)] Rental ($)	Number of Shipments	Transport Costs ($)	Burial Volume (m^3)	Burial Cost ($)	Liner[(b)] Surcharges ($)	Handling Surcharges ($)	Curie Surcharges ($)	Total Disposal Cost ($)
Stainless Steel													
Upper Flow Guide	1.05	1.96×10^3											
Lower Flow Guide	0.514	1.27×10^{-1}											
Control Rod Upper Rollers	0.0125	1.78×10^3											
Miscellaneous Bolts	0.00194	3.35×10^2	4[(c)]	2 000	4 500	4	7,336	1.32	420	5 852	1 340	5 475	26 923
Metering Plate	0.592	3.93×10^2											
Leveling Plate	0.243	1.00×10^1											
Control Rod Drive Box	0.704	8.63×10^{-6}											
Miscellaneous Hardware	(d)	2.82×10^4											
Thermal Shield													
Upper Outer	5.65	2.03×10^{-3}											
Upper Middle	4.70	3.30×10^{-3}											
Upper Inner	3.76	4.51×10^{-3}											
Lower Outer	4.10	6.99×10^{-4}											
Lower Inner	3.80	1.85×10^{-3}	11[(e)]	7 850	NA	4[(f)]	6 226	49.84	15 839	NA	NA	NA	29 915
Pressure Vessel													
Cylindrical Wall	17.30	1.35×10^{-3}											
Removable Head	2.61	---[(g)]											
Hemispheric Bottom	2.30	2.88×10^{-7}											
Flanges	1.62	---[(g)]											
Flow Guide Support	1.70	2.60×10^{-5}											
Aluminum													
Lower Grid	0.168	2.26×10^4											
Reflector Top Grid (2 ea)	0.098	1.16×10^2											
Core Top Grid (2 ea)	0.090	3.89×10^3	1[(c)]	500	1 125	1	1 125	0.33	105	1 463	335	4 063	9 415
Core Box End Plates (2 ea)	0.100	7.43×10^3											
Beam Tubes	0.238	3.75×10^3											
South Box Plate	0.062	1.47×10^2											
Beryllium													
North Core Box Plate (2 ea)	0.084	8.89×10^3											
RD Blocks (8 ea)	0.163	4.40×10^2											
RC Blocks (8 ea)	0.163	2.22×10^3											
RB Blocks (8 ea)	0.113	1.23×10^4											
RA Blocks (8 ea)	0.113	5.61×10^4	6[(c)]	3 000	6 750	6	11 004	1.98	629	8 778	2 010	32 908	65 079
LI, II Blocks (8 ea)	0.054	4.00×10^4											
LA Blocks (19 ea)	0.129	1.09×10^5											
R and L Block Plugs (11R, 5L)	(d)	1.66×10^4											
Flow Divider Plate	0.041	1.76×10^4											
Be Control Rods (5 ea)	0.033	6.02×10^4											
Cadmium													
Cd Control Rods (6 ea)	(d)	9.17×10^1											
Totals	52.31	3.94×10^5		13 350	12 375	15	26 400	56.0	16 993	16 093	3 685	42 436	131 332

(a) Assume 5 days per shipment.
(b) Assume each cask liner has surface dose rate ≤100 R/hr.
(c) Steel cask liner, 0.63 m dia. x 1.02 m high, outside dimensions, 0.33 m^3, $500 each.
(d) Mass data not available.
(e) Wooden box, 1.27 m x 1.27 m x 3.04 m long outside dimensions, 4.53 m^3, $713 each.
(f) 3 shipments of 3 boxes, 1 shipment of 2 boxes.

TABLE I.2-10. Costs of Disposal for Neutron-Activated Materials from the MUR

Component	Estimated Mass (Mg)	Estimated Radio-activity (Ci)[b]	Number of Disposable Containers	Container Cost ($)	Number of Shipments	Transport Costs ($)	Burial Volume (m^3)	Burial Cost ($)[c]	Handling Surcharges ($)	Curie Surcharges ($)	Curie Surcharges ($)	Total Disposal Costs ($)
Mock-Up Reactor (MUR)	4545	1.5	1	803	1	1076	6.12	2118	None[c]	None[c]	None[c]	3997

(a) The entire MUR core box with beryllium, beam tube mock-ups, rod box, and support frame is packaged in one wooden box, 1.83 m x 1.83 m x 1.83 m.
(b) See Appendix E, Table E.2-8 for details.
(c) Based on Table M.4-1; based on an estimated <1 R/hr dose rate at the container surface.

TABLE I.2-11. Costs of Disposal for Contaminated Materials

Estimated Reference Inventory	Component (Quantity)	Location(a)	Estimated Mass (kg)(b)	Number of Disposal Containers(c)	Container Costs(d)	Number of Shipments(e)	Transport Costs ($)(f)	Handling Costs ($)(g)	Burial Volumes (m^3)	Burial Costs ($)(h)	Total Disposal Costs ($)
Appendix C, Table C.3-1	Equipment(i)	RB/CV	38 923	9	3 600	2(j)	2 781	0	32 8	10 430	17 481
	Drums	RB/CV	2 420	10 (drums)	0			0	2.1	670	
As Indicated	Reactor Vessel Appendages:	CV		2(k)	1 426	1(l)	1 076	0	9.1	2 900	5 402
	• Tank Support Legs (4)		493								
	• Through Tube Ports (4)		434								
	• Beam Tube Ports (3)		322								
	• Instrument Inlets (24)		1 009								
	• Rabbit Ports (4)		17								
	• Beam Tube Inner Thimble (3)		1 500								
	• Beam Tube Middle Thimble (3)		93								
	• Fuel Discharge Tube		275								
	• Thermal Column Assembly		3 779								
Appendix C, Figure C.3-1	Equipment(m)	CV, Dry Annulus @ -3 m & -4 m Level	---(m)	4	1 600	1(l)	1 076	0	14.6	4 640	7 316
Appendix C, Figure C.3-2	Equipment(m)	DV, Dry Annulus @ -7.6 m Level	---(m)	9	3 600	2(l)	2 152	0	32.8	10 430	7 316
Appendix C, Table C.3.3	Equipment(i,n)	Hot Cells	433	1	400	1(l)	1 076	0	36.4	11 570	16 646
Appendix C, Table C.3-4	Equipment:(i)	WHB									
	- Evaporator(o)		3 400	7	2 800						
	- Miscellaneous		1 060	2	800						

TABLE I.2-11. (contd)

Estimated Reference Inventory	Component (Quantity)	Location[a]	Estimated Mass (kg)[b]	Number of Disposal Containers[c]	Container Costs[d]	Number of Shipments[e]	Transport Costs ($)[f]	Handling Costs ($)[g]	Burial Volumes (m^3)	Burial Costs ($)[h]	Total Disposal Costs ($)
Appendix C, Table C.3-5	Equipment:	PPH									
	• Primary Pumps (3)		12 270	3[p]	1 845	1	1 240	0	12.3	3 910	6 995
	• Primary Heat Exchanger (2)		18 180	2[p]	2 800	1	1 365	580	19.3	6 140	10 885
	• Deionizers (2)		3 000	2[p]	1 256						
	• Degassifier System		17 160	8[q]	3 200	1	1 415	0	40	12 270	18 991
	• Miscellaneous		630	1	400						
Appendix C, Table C.3-6	Piping and Valves	PCWS	26 827	24	9 600	2	2 530	0	87.4	27 780	39 910
Appendix C, Table C.3-7	Piping and Valves	Q&C Systems	14 222	3	1 200	1	1 266	814	11	3 500	6 780
Appendix C, Table C.3-8	Sump Pumps (21)	Hot Drain Systems	6 839	5	2 000	1[l]	1 076	0	18.2	5 790	8 866
Appendix C, Table C.3-9	Piping and Valves	Hot Drain Systems	13 318	3	1 200	1	1 266	0	11	3 500	5 966
Appendix C, Table C.3-10	Equipment	HRA									
	• HRA Tanks 1-8		136 482	30	12 000	8	5 260	2 625	109.2	34 710	54 595
	• HRA Tanks 9-12		16,900	4[r]	0[r]	2[l]	2 152	0	114.2	36 300	38 452
	Structural Steel (Position & Floor Plates)		25 367	6	2 400	2	2 481	0	21.9	6 960	11 841
Appendix C, Table C.3-11	Piping and Valves	HRA & CRA	27 277	7	2 800	2[s]	2 505	0	25.3	8 040	13 345
Appendix C, Table C.5-1	HVAC Ductwork and Housings	All Contaminated Exhaust Ductwork & Housings	21 746	23	9 200	2[l]	2 152	0	84	26 700	39 522
Appendix C, Section C.5.5	Exhaust Filters	All Locations	---[m]	15 (drums)	450				3.2	1 020	
Appendix C, Table C.5-2	Equipment:	As Indicated									
	• CV Ventilation		7 020	---[t]	0	1[l]	1 076	0	16.5	5 250	12 566
	• Miscellaneous		1 735	4	1 060				14.6	4 640	
	• Off-Gas System		38 608	16	6 400	3	4 095	0	58.3	18 530	35 265
	• Stack		18 155	4	1 600				14.6	4 640	

TABLE I.2-11. (contd)

Estimated Reference Inventory	Component (Quantity)	Location[a]	Estimated Mass (kg)[b]	Number of Disposal Containers[c]	Container Costs[d]	Number of Shipments[e]	Transport Costs ($)[f]	Handling Costs ($)[g]	Burial Volumes (m^3)	Burial Costs ($)	Burial Costs ($)[h]	Total Disposal Costs ($)
Appendix C, Table C.4-1 and Section C.4	Concrete Piping Plus Soil	Ditches	276 882	350[u]	0[u]	43	56 545	0	500	158 900	158 900	215 445
	Concrete Piping	Ditches	49 602	45	18 000	3	3 945	0	164	52 120	52 120	74 065
	Soil	Ditches and ERB	1 711 672	447	178 800	90	122 850	0	1 627	517 060	517 060	818 710
	Soil	CRA	511 723	134	53 600	27	35 478	0	488	155 090	155 090	244 168
Appendix D, Table D.2-3	Concrete Waste Materials	All: see Table D.2-3	1 513 960	328	131 400	22	107 748	0	1 194	379 460	375 460	618 608
Totals (from all locations)			4 523 743[m]	1 129 Boxes, +25 Drums, +350[u]	455 977	280	364 606	4 019	4 761.8	1 513 400	1 513 400	2 338 002

(a) RB/CV is Reactor Building/Containment Vessel: WHB is Waste Handling Building; PPH is Primary Pump House; PCWS is Primary Cooling Water System; Q&C Systems is Quadrant and Canal Systems; HRA is Hot Retention Area and CRA is Cold Retention area; ERB is Emergency Retention Basin.
(b) Individual totals include container(s) mass.
(c) Assumed to be 1.2-m by 1.2-m by 2.4-m plywood boxes, unless otherwise noted. Fractional box requirements are rounded up to next whole box.
(d) Based on information in Section M.2 of Appendix M.
(e) Assumed to be overweight shipments, unless otherwise noted.
(f) Based on Table M.3-3.
(g) Based on Table M.4-1.
(h) Based on Table M.4-1; based on an assumed container surface dose rate of <0.20 R/hr; rounded to next highest $10.
(i) Assumed packing efficiency factor for equipment is 1.5, unless otherwise noted.
(j) 1 shipment of 5 boxes; 1 shipment of 4 boxes and 10 drums.
(k) Wooden box, 1.22-m x 1.22-m x 3.04-m long outside dimensions, 4.53-m^3, $713 each.
(l) Based on Appendix M, a legal-weight shipment.
(m) Some mass data unavailable; in those cases, the number of containers is based on only the individual volume data known for each component.
(n) Salvageable equipment is neglected (see Appendix C, Table C.3-3 for details).
(o) Efficiency packing factor has previously been applied (see Appendix C, Table C.3-4).
(p) Specially fabricated plywood boxes.
(q) Efficiency packing factor has previously been applied (see Appendix C, Table C.3-5).
(r) Tanks are shipped whole; 2 tanks per shipment.
(s) 1 shipment of 4 boxes, 1 shipment of 3 boxes.
(t) Assumed to be shipped as whole units; no other containers required.
(u) Different size concrete pipes are nested with the remaining internal volume filled with contaminated dirt, and the ends concreted to form individual burial packages.

To account for the potentially contaminated equipment for which data is unavailable by applying a contingency factor to the estimated volumes given in Table I.2-11 would be highly speculative at best, and is therefore neglected in this study. In any case, the estimated total volume of contaminated equipment and materials given in Table I.2-11, while known to be slightly on the low side because of the unavailability of data, appears to represent the majority or equipment and materials anticipated to require shipment to a licensed disposal site.

The contaminated materials presented in Table I.2-11 require an estimated 280 truck shipments to a shallow-land burial site and occupy an estimated 4,762 m^3 of space (including the disposable containers, as required). The total disposal cost for contaminated materials from the reference test reactor (including the MUR) is estimated to be about $2.3 million.

Radioactive Wastes--Radioactive wastes (radwastes) result directly from DECON of the reference test reactor. Two categories of radioactive wastes are defined: wet solid wastes and dry solid wastes.

Wet solid wastes result from the processing of chemical decontamination solutions and contaminated water volumes. These wastes include slurry from the cleanout of the HRA tanks, water filters, and spent demineralizer resins. Wet solid wastes are assumed to be mixed with a cement solidifying agent and encapsulated in 0.21-m^3 steel drums for shipment to a shallow-land burial facility. The disposal cost data for the wet solid wastes generated during DECON are presented in Table I.2-12. An estimated two truck shipments, 16.8 m^3 burial space, and $11,180 are required for disposal of the wet solid wastes. The total radioactivity content is estimated to be 2.8 Ci, or about 0.07 Ci per drum.

Dry solid wastes include discarded contaminated materials such as plastic sheeting, rags, and anticontamination clothing. They are expected to occur as a result of most of the tasks specified in Section I.2.2. The dry solid wastes are compacted as much as possible to reduce their volume. These volumes are assumed to be compacted five-fold and packaged in 0.21-m^3 standard steel drums.

TABLE I.2-12. Costs of Disposal for Wet Solid Wastes

Material	Assumed Surface Radiation Dose Rates (R/hr)	Number of Disposable Containers[(a)]	Container and Solidification Costs ($)[(b)]	Number of Shipments	Transportation Costs ($)[(c)]	Handling Costs ($)[(d)]	Burial Volume (m^3)[(e)]	Burial Costs ($)[(f)]	Liner Surcharges ($)[(d)]	Total Disposal Costs ($)[(g)]
Filters, Spent Demineralizer Resins, and Slurry from Hot Retention Tanks	0.2	80	3 360	2	2 480	0	16.8	5 340	0	11 180

(a) Assumed to be 0.21-m^3 standard steel drum.
(b) Based on Table M.2-1 in Appendix M; includes solidification costs based on Section M.2.3; rounded to next highest $10.
(c) Based on Table M.3-3.
(d) Based on Table M.4-1.
(e) Includes the disposable container; the average radioactivity content of each container is estimated at about 0.70 Ci/drum.
(f) Based on Table M.4-1; rounded to next highest $10.
(g) The number of figures shown is for computational accuracy and does not imply precision to that many significant figures.

Of the total number of drums, 140 are assumed to require shielding during shipment, with the remainder shipped unshielded in closed trucks. The two types are shipped separately, since, because of the ^{60}Co content, those requiring shielding also require Type B packaging and shipping. Table I.2-13 gives a breakdown of the disposal costs for the dry solid wastes. An estimated 10 shielded, overweight truck shipments and two unshielded legal-weight truck shipments are required to transport the compacted, packaged, dry solid wastes to a shallow-land burial facility, where they occupy an estimated 93 m^3 of space. The total disposal cost for the dry solid wastes from the DECON of the reference test reactor is estimated at $87,400.

The sum of the disposal costs for the wet solid wastes and the dry solid wastes is $98,580, which represents the disposal cost for the radioactive wastes.

I.2.3.2 Staff Labor

The cost of staff labor during DECON is shown in detail in Table I.2-14. More than 69% of the total DECON cost is associated with the staff labor requirements (presented in Table I.2-6). A total staff labor cost of about $8.6 million is estimated for DECON at the reference test reactor. Specialty contractor labor is not included in this total.

I.2.3.3 Energy

The cost of energy used during DECON is presented in Table I.2-15. The use of electricity and natural gas shown in the table is based on data (1978) supplied in Reference 9, adjusted for inflation to mid-1981, and applied to the time frame estimated for DECON tasks (i.e., about 25 months). The total cost of energy is about $76,250 and represents less than 1% of the total DECON cost.

I.2.3.4 Special Tools and Equipment

Based on the information presented in Table I.2-1, the estimated costs of the special tools and equipment that are required for DECON at the reference test reactor are presented in Table I.2-16. The estimated total cost for special tools and equipment is approximately $0.361 million and is approximately 3% of the total DECON cost.

TABLE I.2-13. Costs of Disposal for Dry Solid Wastes

Number of Disposable Containers(a)	Container Costs ($)(b)	Estimated Number Requiring Shielding	Cask Rental Costs ($)(c)	Number of Shipments(d) Shielded/Unshielded	Transportation Costs ($)(e)	Handling Costs ($)(f)	Burial Volumes (m^3)(g)	Burial Costs ($)(h)	Total Disposal Costs ($)(i)
440	13 200	140	22 500	10/2	14 802	6 700	93	30 200	87 402

(a) Based on a 5:1 compaction of estimated waste volumes into standard 0.21-m^3 steel drums; estimated on a taskwise assessment of expected dry solid waste generation rates.
(b) Based on Table M.2-1 in Appendix M.
(c) Based on Table M.2-2; assumes a maximum of seven containers per cask and five days per shipment.
(d) Assumes two cask loads per shipment.
(e) Based on Table M.3-3 for overweight shipments (10) and legal-weight shipments (2).
(f) Based on Table M.4-1, cask handling fee.
(g) Includes the disposable container; rounded to next whole m^3.
(h) Based on Table M.4-1; surface dose rates assumed to be 0.21 to 1.00 R/hr for those drums requiring shielding during shipment, <0.2 R/hr for all others; rounded to the next highest $10.
(i) The number of figures shown is for computational accuracy only and does not imply precision to that many significant figures.

TABLE I.2-14. Estimated Costs of Staff Labor During DECON at the Reference Test Reactor

Position	Time Relative to Final Reactor Shutdown (year) -2	-1	1	2	3	Total Staff Labor Costs[b] ($ thousands)
	Annual Staff Labor Costs ($ thousands)[a,b]					
Management and Support Staff						
Decommissioning Superintendent	26.8	89.1	89.1	89.1	44.6	338.7
Secretary	24.2	48.4	72.6	72.6	24.2	242.0
Clerk	0	24.2	48.4	48.4	12.1	133.1
Decommissioning Engineer	76.0	76.0	76.0	76.0	38.0	342.0
Assistant Decommissioning Engineer	52.4	52.4	52.4	52.4	5.3	214.9
Radioactive Shipment Specialist	0	39.3	39.3	39.3	4.0	121.9
Procurement Specialist	11.8	39.3	39.3	39.3	4.0	133.7
Tool Crib Attendant	0	0	55.6	55.6	5.6	116.8
Control Room Operator	0	0	172.5	172.5	0	345.0
Security Supervisor	0	0	55.9	55.9	5.6	117.4
Security Shift Supervisor	0	0	182.0	182.0	7.3	371.3
Security Patrolman	0	0	406.4	304.8	10.2	721.4
Contracts and Accounting Specialist	14.2	47.1	47.1	47.1	23.6	179.1
Clerk	0	24.2	24.2	24.2	12.1	84.7
Health and Safety Supervisor	59.8	59.8	59.8	59.8	29.9	269.1
Health Physicist	0	23.5	46.9	46.9	9.4	126.7
Protective Equipment Attendant	0	0	55.6	55.6	5.6	116.8
Industrial Safety Specialist	15.8	52.4	52.4	52.4	10.5	183.5
Quality Assurance Supervisor	15.7	52.2	52.2	52.2	26.1	198.4
Quality Assurance Engineer	23.5	46.9	46.9	46.9	9.4	173.6
Quality Assurance Technician	0	13.9	55.6	55.6	11.2	1 6.3
Consultants (Safety Review Committee)	50.0	50.0	50.0	50.0	0	200.0
Subtotals	370.2	738.7	1780.2	1678.6	298.7	4866.4
Decommissioning Workers						
Shift Engineer	52.2	104.4	104.4	104.4	10.5	375.9
Crew Leader	0	44.4	222.0	222.0	22.2	510.6
Utility Operator	0	96.3	353.1	353.1	19.3	821.8
Laborer	0	0	216.3	216.3	12.4	445.0
Craft Supervisor	0	23.5	93.8	93.8	9.4	220.5
Craftsman	0	160.5	321.0	321.0	32.1	834.6
Senior Health Physics Technician	0	39.3	78.6	78.6	7.9	204.4
Health Physics Technician	0	45.0	150.0	150.0	6.0	351.0
Subtotals	52.2	513.4	1539.2	1539.2	119.8	3763.8
Totals	422.4	1252.1	3319.4	3217.8	418.5	8630.2

(a) Calculated as the product of the data given in Table I.2-6 and the corresponding data given in Table M.1-1 in Appendix M; rounded to next higher $100.
(b) The number of figures shown is for computational accuracy and does not imply precision to the nearest $100.

TABLE I.2-15. Estimated Costs of Energy During DECON at the Reference Test Reactor

Energy Form	Estimated Energy Usage Months	Estimated Average Cost/Month ($)[a]	Costs($)
Electricity	25	2 000	50 000
Natural Gas	25	1 050	26 250
Total Energy Cost			76 250

(a) Based on data (1978) supplied in Reference 9 and adjusted for inflation to mid-1981.

I.2.3.5 Miscellaneous Supplies

A variety of supplies are used during DECON. These include expendable glass-fiber and HEPA filters, anticontamination clothing, cleaning and contamination control supplies (chemical agents, sweeping compounds, rags, mops, and plastic bags and sheeting), expendable hand tools, cutting and welding supplies (saw blades, torch gas, and welding rods), and decontamination chemicals, as well as office supplies. The estimated costs for these items are given in Table I.2-17. The total estimated cost for miscellaneous supplies during DECON of the reference test reactor is about $0.2 million and represents less than 3% of the total DECON cost.

I.2.3.6 Specialty Contractors

The estimated costs of specialty contractors are given in Table I.2-18. As discussed in Section H.4 of Appendix H, these specialty contractors perform explosives work, temporary radwaste handling, and environmental monitoring. The costs for a hauling contractor are not shown in Table I.2-18, but are shown as "transport costs" in Section I.2.3.1 for disposal of radioactive wastes.

The total cost of specialty contractors during DECON, excluding the hauling contractor, is $615,650, which is about 5% of the total DECON cost.

TABLE I.2-16. Estimated Costs of Special Tools and Equipment for DECON at the Reference Test Reactor

Item	Estimated Number Required[a]	Estimated Costs ($ thousands)
Underwater Plasma-Arc Torch	1	20
Underwater Oxyacetylene Torch	1	5
Portable Plasma-Arc Torch	2	40
Portable Oxyacetylene Torch	3	3
Guillotine Pipe Saw	4	16
Power-Operated Reciprocating Hacksaw	4	3.2
Nibbler	4	4[b]
Closed Circuit TV System	As Required	15[c]
Submersible Pump with Disposable Filter	3	6
High-Pressure Water Jet	2	40
Powered Floor Scrubber	3	0.9
Wet-Dry Vacuum Cleaner (HEPA Filtered)	3	9[d]
Supplied-Air Plastic Suit	50	2.5
Power-Operated Mobile Manlift	1	40
9100-kg Mobile Hydraulic Crane	1	28
9100-kg Forklift	1	28
Concrete Drill with HEPA Filtered Dust Collection System	2	4
Concrete Surface Spaller	2	10
Front-End Loader (Light Duty)	1	20
Portable Filtered Ventilation Enclosure	5	30[d]
Filtered-Exhaust Fan Unit	2	10
Polyurethane Foam Generator	2	10[e]
Paint Sprayer	2	1.6
HEPA Filter	50	10 Total
Roughing Filter	50	5 Total
Total Cost		361.2

(a) Based on Table I.2-1.
(b) Remote operating extensions would add to this cost.
(c) Estimated for modifications of existing systems.
(d) Depends on size and capacity.
(e) Depends on capacity of system.

TABLE I.2-17. Estimated Costs of Miscellaneous Supplies During DECON at the Reference Test Reactor

Item[a]	Estimated Costs ($ thousands)
Filters[b]	9 000
Office Supplies	11 050
Graphic Reproduction	19 500
Maintenance and Repair	4 900
Service Supplies	23 400
Protective Clothing[c]	61 700
Electronic Repair Parts	18 200
Lab Supplies	13 000
Decon Supplies	26 000
Radiological Waste Supplies	13 000
Unclassified	3 600
Total	203 350

(a) Based on data (1978) in Reference 9; adjusted for inflation to mid-1981 dollars; plus, proportional adjustments based on the two-shift operation for the estimated 25-month DECON time frame, unless otherwise noted.
(b) Exhaust air-handling unit filters are assumed to be changed out once during the DECON period.
(c) Estimated at four changes per day per decommissioning worker. See Reference 10.

The close working relationship anticipated between licensee and a demolition subcontractor during DECON at the reference test reactor, which is discussed in Appendix L, manifests itself clearly in the cost estimates for "demolition" in Table I.2-18. In addition, the need envisioned for this working relationship is evidenced further in the DECON task schedule and sequence figures (see Figures I.2-2 and I.2-3).

TABLE I.2-18. Estimated Costs of Specialty Contractors During DECON at the Reference Test Reactor

Specialty Contractor[a]	Cost Estimating Basis ($/unit)	Estimated Costs ($)
Demolition:[b]		
ERB	---[c]	143 300[d]
Site Ditches	---[c]	254 100[d]
CRA	---[c]	40 300[e]
Remove Hot Cell Windows	---[f]	23 400
HRA (Provide Access to Tanks)	---[g]	1 500
HRA (Remove Tanks No. 9 through 12)	---[g]	16 000
Stack	---[h]	1 600
Explosives	715/day[i]	22 450
Temporary Radwaste Handling	13 000/mo[j]	13 000
Environmental Monitoring	50 000/yr[k]	100 000
Total Cost		615 650

(a) Does not include the hauling contractor, which is shown as "transport costs" in Tables I.2-9 through I.2-13, with a total cost of approximately $0.41 million.
(b) Refers to subcontractor tasks requirements in Figures I.2-2 and I.2-3.
(c) Based on data (1978 dollars) in Reference 4, adjusted for inflation to mid-1981 dollars and proportioned to the estimated volumes of soil and concrete pipe at the reference test reactor presented in Appendix C, Section C.4.
(d) Includes a subcontractor's fee based on 8% of the sum of the subcontractor's charge for manpower, equipment, and materials. (Does not include waste management costs described in Section I.2.3.1).
(e) Includes $27,000 to remove the CRA roofs (see Appendix L, Section L.3.2.7).
(f) Based on data (1978) in Reference 9 and adjusted for inflation to mid-1981 dollars.
(g) See Appendix L, Section L.3.2.6 for cost breakdown.
(h) See Appendix L, Section L.3.2.4 for cost breakdown.
(i) Based on information supplied by Mr. Jerry Curry of JC Drilling and Contracting, Bellevue, Washington, for the BWR study in Reference 3. For this study the base data are updated and includes material costs for explosives ($\sim$\$1000) and labor costs for one explosives specialist, plus helper, for 30 days.
(j) Based on information presented on Page 10-4 of Reference 2 and adjusted upward to account for inflation to 1981 dollars.
(k) Based on information presented on Page G-37 of Reference 2 and adjusted upward to account for inflation to 1981 dollars.

I.2.3.7 Nuclear Insurance

Indemnity fees are currently $100/yr for each license (i.e., the test reactor license and the MUR license) at the reference test facility and are not included in this study since they represent only a small fraction of 1% of the total decommissioning cost.

I.2.3.8 Licensing Fees

The fees charged for licensing services performed by the NRC are delineated in 10 CFR 170.[5] The costs of licensing fees during DECON at the federally-owned reference test reactor (also see Section I.2.3.7) are not included in this study since the federal government does not charge itself for these inspections.

I.2.3.9 Other Possible Costs

Two additional categories of costs could figure into the total DECON cost depending on how they are classified. In this study, they are presented separately since they cannot be clearly identified as belonging to DECON. The tasks that require these costs are:

- shipment of the reactor fuel to an offsite fuel reprocessor
- demolition of the structures and restoration of the site.

Discussions of the costs associated with these tasks are presented in the following subsections.

Fuel Shipment--The ultimate disposition of the reactor fuel is assumed to be the DOE Savannah River Operations Office (SROO). This disposition is based on historical data supplied by NASA. All fuel movement was made by routine operating procedure, with one exception. To expedite the shipment of 328 irradiated fuel elements, three NASA-owned shipping casks and the rented service of National Lead Industries (NLI) were used. The rental was served by government contract, and agreement was made for reprocessing of the irradiated fuel with DOE (then AEC) SROO under an interagency agreement.

Seven shipments were made in the NLI casks and eight shipments were made in the NASA-owned casks. The approximate cost (1973 dollars) for shipping the irradiated fuel were:

Transportation Furnished by NLI	$31,090
Transportation of NASA-Owned Casks	9,600
Estimated Reprocessing Cost	210,000
Total	$250,690

Based on the foregoing historical information, the estimated transportation costs in mid-1981 dollars for the same 15 shipments of irradiated fuel (assuming the same conditions) are presented in Table I.2-19. Current costs for reprocessing this same amount of irradiated reactor fuel are not included in this study.

It is assumed for the calculations used to generate Table I.2-19 that the reprocessor is located 2400 km[a] by road from the reference test reactor (see Appendix A for generic site location) and requires 15 days round trip. Daily rental of the cask is estimated to be $800. Transportation by sole use truck is based on current rates given in Appendix M. The total cask rental cost for seven shipments is $100,800, and the total transportation cost for all 15 shipments is $103,470. Therefore, the total estimated cost for shipping the irradiated reactor fuel to the reprocessor is $204,270. This does not include either handling costs at the reactor or handling and reprocessing costs at the reprocessor.

TABLE I.2-19. Estimated Fuel Shipment Costs

Type of Cask	Number of Shipments	Cask Rental Costs ($)	Transport Costs ($)[a]	Total Cost ($)
NLI	7	100 800	48 286	149 086
NASA	8	---[b]	55 184	55 184
Totals	15	100 800	103 470	204 270

(a) Based on Table M.3-3; assumed to be overweight, 15-day/round-trip and requiring a second driver. Additional costs for armed guards, if required, is neglected in this study.
(b) No cost incurred; assumed to be government-owned casks.

(a) For comparative purposes, this assumed shipping distance is consistent with that used in previous decommissioning studies in this series.

The 15-day-round-trip requirement as well as cask-use assumptions stated above, are necessary in order to meet the schedule shown in Section I.2.2.

Facility Demolition and Site Restoration--The cost of demolishing all of the decontaminated and uncontaminated structures at the reference test reactor is discussed in detail in Appendix L and is summarized in Table L.3-3. However, because of the physical layout (i.e., significant quantities of buried contaminated piping, soil, and tanks) at the reference test facility, a modified demolition approach is used to effect cost savings through the use of this same demolition contractor's services for selected tasks. These selected tasks are given in Table I.2-18; they are: 1) provide access to HRA tanks; 2) remove HRA Tank Nos. 9 through 12; and 3) raze the stack so that it can be segmented and packaged for burial as low-level radioactive waste.[a] The total cost of these three tasks is $19,100. Therefore, this cost is subtracted from the total costs given in Table L.3-3 of Appendix L to arrive at the total cost for demolition and site restoration of the remaining uncontaminated structures at the reference test reactor facility. The total cost of $2,289,025 (without contingency) includes labor, supplies, overheads, and profit, but not extraordinary insurance premiums, bonding, or state sales tax. Details of the cost estimates for this task, which is not required for termination of the facility nuclear license, are given in Section L.3.2 of Appendix L.

I.2.4 Estimated External Occupational Radiation Doses from DECON at the Reference Test Reactor

Estimates are made of the external occupational radiation doses that are accumulated by the decommissioning workers during DECON at the reference test reactor. The estimates are based on a task analysis to determine the man-hours of effort required in radiation zone work and the anticipated dose rates associated with each task. Basic assumptions made in developing these estimates are: 1) every effort is made to minimize the personnel exposure to radiation (ALARA philosophy) while accomplishing a task, by the use of

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(a) It is assumed that a separate specialty contractor completes the other four demolition tasks with radiological consequences given in Table I.2-18.

temporary shielding and remote handling techniques and by staying out of radiation fields when not actively participating in the work; 2) careful, prompt accounting of radiation doses is maintained to rapidly identify jobs that are causing excessive dose accumulations so corrective action can be taken; and 3) ^{60}Co is the dominant source of radiation dose.

The estimated total dose for each task is corrected for radioactive decay with a decay factor calculated using the half-life of ^{60}Co and the midpoint of the timeline for the given task (shown in Figures I.2-1, I.2-2, and I.2-3).

The results of these analyses and corrections are presented in Table I.2-20. The total corrected external occupational radiation dose for DECON at the reference test reactor (including MUR) is 322 man-rem. Of the total, 46% results from DECON at the RB/MUR/CV, about 19% from DECON at the HLB, about 21% from DECON at all the other buidings/structures remaining on the site, and approximately 14% from ancillary tasks.

Two other tasks not shown in Table I.2-20 provide the potential for external radiation dose and occur as a direct result of DECON: 1) transporting the radioactive materials to the disposal site, and 2) handling these materials at the disposal site. A total of 310 shipments of radioactive materials must be transported to, and handled at, the disposal site. The external radiation dose from transportation is discussed in Section N.5 of Appendix N. The external radiation dose from handling these materials at the disposal site is beyond the scope of this study, since it is considered under another nuclear license.

TABLE I.2-20. Estimated Occupational Radiation Doses for DECON at the Reference Test Reactor

Location / Task	Exposure (man-hr)	Dose (man-rem)	Decay Factor[a]	Corrected Dose (man-rem)[b]
Reactor Building/MUR/Primary Containment				
1. Comprehensive Radiation Survey for Total Facility (i.e., all buildings)	528	2.64	0.995	2.626
2. Discharge Fuel (including MUR)	66	0.75	0.988	0.741
3. Prepare and Ship Spent Fuel	2 500	6.0	0.963	5.781
4. General Cleanup and Equipment Inventory (i.e., all buidings)	924	1.848	0.986	1.823
5. Drain, Clean, Dry Quadrants A, B, C and D and Canals E and F	780	7.8	0.980	7.648
6. Drain, Clean, Dry Canal H	180	1.8	0.946	1.702
7. Remove, Package, and Ship MUR and Associated Hardware	576	10.0	0.944	9.436
8. Drain, Clean, Dry Canal G	180	1.8	0.940	1.693
9. Remove Loose Equipment in Q&Cs and Dry Annulus	936	9.36	0.937	8.774
10. Drain and Flush PCWS	72	0.36	0.934	0.336
11. Isolate RV and Add Deionized Water for Shielding	72	0.072	0.932	0.067
12. Remove RV Internals and Ship Activated RV Internals	10 366	51.83	0.914	47.349
13. Remove RV and Ship RV Segments	2 916	14.58	0.889	12.966
14. Remove Bio-Shield Concrete	120	0.060	0.882	0.053
15. Remove Fixed Equipment in CV (Except HVAC)	5 890	29.45	0.874	25.750
16. Remove Fixed Equipment Outside CV	1 728	1.728	0.865	1.495
17. Remove Quadrant Piping	360	0.72	0.863	0.622
18. Segment and Remove Subpile Room	252	6.3	0.860	5.416
19. Remove Lead Shield from Below Reactor Cavity	252	0.54	0.857	0.463
20. Remove Pipes from Bio-Shield	1 000	1.6	0.852	1.363
21. Remove PCWS Piping to PPH	2 820	5.64	0.839	4.733
22. Remove RB/CV Contaminated Concrete	3 240	3.24	0.825	2.675
23. Remove Q&C, and Miscellaneous Contaminated Drains	2 520	5.04	0.822	4.142
24. Remove Contaminated HVAC from RB/CV	960	2.4	0.810	1.943
25. Final Radiation Survey	100	0.050	0.801	0.040
Subtotals[b,c]	39 338	166		150

TABLE I.2-20. (contd)

Location / Task	Exposure (man-hr)	Dose (man-rem)	Decay Factor[a]	Corrected Dose (man-rem)[b]
Hot Laboratory Building				
1. Decontaminate Hot Cells	624	17.119	0.973	16.656
2. Remove and Package Hot Cells Equipment and Piping	2 146	21.46	0.956	20.506
3. Remove and Package Hot Cells SS Cladding	480	4.8	0.942	4.519
4. Remove Contaminated Concrete from Hot Cells	336	0.336	0.938	0.315
5. Decontaminate the HLB (including cranes)	336	0.336	0.935	0.314
6. Drain, Clean, Dry Canals J and K	360	3.6	0.933	3.360
7. Remove Loose Equipment	913	1.826	0.926	1.691
8. Remove Fixed and Permanent Equipment (except HVAC), Including the Hot Pipe Tunnel	2 146	7.15	0.900	6.436
9. Remove SS Cladding from Decontamination Room 23	120	0.240	0.896	0.215
10. Remove Hot Cell Windows	1 072	0.536	0.932	0.500
11. Remove and Package HLB Contaminated Concrete	795	0.795	0.894	0.711
12. Remove and Package Contaminated HVAC from HLB	1 200	4.8	0.860	4.126
13. Final Radiation Survey	88	0.044	0.843	0.037
Subtotals[b,c]	10 616	63		60
Other Contaminated Structures and Areas				
1. Radiation Survey and Inventory Update (i.e, all "other buildings/areas")	264	0.66	0.855	0.565
2. Primary Pump House (PPH):				
• Preparatory Tasks (see Table I.2-4)	420	2.1	0.852	1.789
• Remove Fixed Equipment, Except HVAC (see Table C.3-5)	4 410	30.87	0.834	25.735
• Decontaminate PPH and Remove Contaminated Concrete (see Table D.2-3)	180	0.18	0.818	0.147
• Remove and Package Contaminated HVAC (see Table C.5-1)	300	0.6	0.816	0.490
• Final Radiation Survey	40	0.020	0.815	0.016

TABLE I.2-20. (contd)

Location / Task	Exposure (man-hr)	Dose (man-rem)	Decay Factor[a]	Corrected Dose (man-rem)[b]
3. Office and Laboratory Building (OLB):				
• Preparatory Tasks (see Table I.2-4)	950	0.475	0.814	0.387
• Remove Contaminated Hoods and Sinks (see Section C.3.6 of Appendix C)	519	0.260	0.810	0.211
• Remove Contaminated Concrete (see Table D.2-3)	120	0.060	0.808	0.048
• Final Radiation Survey	24	0.012	0.807	0.010
4. Emergency Retention Basin (ERB) and Site Ditches:				
• Drain the ERB	0	0	0	0
• Remove and Package Contaminated Piping and Soil from ERB and Site Ditches (see Table C.4-1)	1 338	0.134	0.823	0.110
5. Cold Retention Area (CRA):				
• Remove Contaminated Concrete (see Table D.2-3) and Contaminated Soil (see Appendix C, Section C.4)	1 320	6 618	0.801	5.298
• Final Radiation Survey	36	0.018	0.793	0.014
6. Hot Retention Area (HRA):				
• Preparatory Tasks (see Table I.2-4)	1 584	4 752	0.802	3 812
• Remove and Package Contaminated Piping (see Table C.3-11)	1 008	5.04	0.796	4.013
• Provide Tank Access to Eight HRA Tanks (see Appendix L, Section L.3.2.6)	0	0	0	0
• Remove and Package HRA Tanks 1 through 8, Floor Plates, and Partitions (see Table C.3-10)	3 108	15.54	0.785	12.204
• Uncover and Prepare HRA Tanks 9 through 12 for Shipment (see Table C.3-10 and Appendix L, Section L.3.2.6)	0	0	0	0
• Remove Contaminated Concrete (see Table D.2-3)	180	0.180	0.779	0.140
• Final Radiation Survey	48	0.024	0.778	0.019

TABLE I.2-20. (contd)

Location / Task	Exposure (man-hr)	Dose (man-rem)	Decay Factor[a]	Corrected Dose (man-rem)[b]
7. Fan House:				
• Preparatory Tasks (see Table I.2-4)	792	3.96	0.791	3.132
• Remove Contaminated Concrete (see Table D.2-3)	240	1.2	0.789	0.946
• Remove Fixed Equipment (see Table C.5-2)	2 520	6.3	0.781	4.918
• Raze Stack, Segment and Package[d]	---	---	---	---
• Final Radiation Survey	36	0.18	0.773	0.139
8. Waste Handling Building (WHB):				
• Preparatory Tasks (see Table I.2-4)	792	2.376	0.775	1.841
• Remove Fixed Equipment, Including Evaporator (see Table C.3-4) and HVAC (see Table C.5-1)	2 376	2.376	0.767	1.823
• Remove Contaminated Concrete (see Table D.2-3)	180	0.180	0.762	0.137
• Final Radiation Survey	36	0.018	0.761	0.014
Subtotals[b,c]	22 821	84		68
Ancillaries				
1. Radwaste Handling and Laundry Operations	3 178	14.7	0.876	12.877
2. Routine Radiation Surveys	618	1.854	0.876	1.624
3. Miscellaneous[e]	---	---	---	29.0
Subtotals, Ancillaries[b,c]	3 796	17		44
Totals[b]	76 571	330		322

(a) Based on the half-life of ^{60}Co; calculated at the midpoint of the task times shown in Figures I.2-1, I.2-2, and I.2-3.
(b) The number of figures shown is for computational accuracy and does not imply precision to the nearest millirem.
(c) Dose totals are rounded to the nearest whole number.
(d) The occupational dose for segmenting and packaging the stack is included in "removal of fixed equipment" for the Fan House.
(e) Consists of an allowance of 10% of the total explicitly estimated task radiation dose to account for any omissions and uncertainties in the analysis.

REFERENCES

1. U.S. Code of Federal Regulations, Title 10, Part 20. "Permissible Doses, Levels and Concentrations," 1980.

2. R. I. Smith, G. J. Konzek and W. E. Kennedy, Jr., Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0130, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1978.

3. H. D. Oak, et al., Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1980.

4. E. S. Murphy, Technology, Safety and Costs of Decommissioning Reference Non-Fuel-Cycle Nuclear Facilities, NUREG/CR-1754, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, February 1981.

5. U.S. Code of Federal Regulations, Title 10, Part 170, "Fees for Facilities and Materials Licenses and Other Regulatory Services," 1980.

6. Docket Nos. 50-30 and 50-185, Letter to Mr. D. E. Benedict (NASA) from Mr. R. W. Reid (NRC) regarding, "Radiation Levels for Release of Plum Brook Reactor Facility," dated February 11, 1981.

7. Dismantling Plan--Plum Brook Reactor Dismantling, NASA Lewis Research Center, February 1980.

8. NASA-PBRF Monthly Report, 1973 (February through July).

9. "OPTIONS" Teledyne Isotopes Study (July 1978).

10. C. A. Pelletier and P. G. Voillegue, Potential Benefits of Reducing Occupational Radiation Exposure, AIF/NESP-010, Atomic Industrial Forum, Inc., Washington, DC, May 1978.

APPENDIX J

DETAILS OF SAFSTOR

SAFSTOR encompasses those activities required to place (preparations for safe storage) and maintain (safe storage) the reference research and test (R&T) reactor facilities in such condition that the risk to safety is within accept- able bounds and that the facilities can be safely stored under the conditions of the amended nuclear license. Since materials having radioactivity levels above unrestricted release levels are still onsite, the amended nuclear license remains in force throughout the SAFSTOR period. SAFSTOR is completed by sub- sequently decontaminating the facility to levels that permit release of the facilities for unrestricted use (deferred decontamination).

The information in this appendix forms the basis for the activities, costs, and occupational radiation doses for SAFSTOR at the reference R&T reactors presented in Sections 10, 11, and 12 in Volume 1, respectively.

The first half of this appendix is concerned with the reference research reactor, while the last half is concerned with the reference test reactor. The format is the same for both reference reactors; it includes the decommissioning methods, schedules and manpower requirements, costs, and external occupational radiation doses. The facility descriptions given in Appendices B, D, and E for the reference research reactor and the facility descriptions given in Appendices C, D, and E for the reference test reactor, respectively, provide the basic information for the development of the tasks, schedules, manpower loadings, costs, and occupational radiation exposure estimates. Additional design details pertinent to dismantlement of specific equipment items come from engineering drawings, manufacturer's data, and onsite inspection and consultation with operating personnel.

J.1 METHODS POSTULATED FOR SAFSTOR AT THE REFERENCE RESEARCH REACTOR FACILITY

Detailed information concerning decommissioning of the reference research reactor facility via the SAFSTOR alternative is presented in this section.

The procedures described in the following subsections are those expected to be used where appropriate to dismantle, decontaminate, and secure all radioactive materials within the Reactor Building at the reference research reactors. It is postulated that decontamination and dismantlement of all the smaller auxiliary buildings is appropriate to minimize the long-term restrictions on university and urban lands. It is estimated that any attempt to place these small buildings into a SAFSTOR condition would incur initial costs equal to their DECON-type costs, as well as add to the annual cost burden during the safe storage period. Therefore, these tasks and several more associated with the Reactor Building are identical to those of DECON, having identical manpower requirements and costs. Once these small buildings are decontaminated and dismantled, the Reactor Building is secured to provide protection to the public and the environment with limited maintenance and surveillance.

J.1.1 Preparing the Reference Research Reactor for Safe Storage

The work tasks necessary to prepare the reference research reactor for safe storage are discussed in this section. The small size of the reactor and the uncomplicated design of the facility make these tasks relatively straightforward. These tasks include preparatory activities, decontamination, deactivation, sealing, spray painting, contaminated material transfer, isolation procedures and the coordinating of planning for special equipment, systems and services.

J.1.1.1 Preparatory Activities

Subsequent to completion of the planning and preparation and reactor shutdown, a comprehensive radiation survey of the reference facility is completed for planning radiation work-zone procedures. A general cleanup follows and equipment is identified for decommissioning use, reuse elsewhere, disposal, or onsite storage.

Cleanup activities around the reactor are scheduled first to allow the removal and shipment of fuel to commence rapidly. At this time the complete decommissioning plan must be approved and in place. This includes training in decontamination procedures, contingent emergency plans, as well as the needed equipment and supplies for the ultimate disposition of radioactive materials.

A logical pattern for cleanup, decontamination, dismantlement and storage is one in which tasks associated with the reference research reactor come first and areas of lesser contamination follow. Tasks associated with the other buildings are undertaken early in the schedule because of their independent locations; however, two factors affect their exact timing. First, the distribution of the Reactor Building workload among the decommisioning staff makes it impractical to start until some of the early Reactor Building tasks are completed, and second, Reactor Building areas ultimately intended for contaminated waste storage (such as the pool irradiation facility) must first be cleared and made acceptable for safe storage.

Unneeded equipment is removed as early as possible to provide easy access to the remaining areas. As equipment and piping are disconnected from their positions, they are capped or fitted with seals to prevent spread of internal contamination. Decontamination of the ceilings, walls and floor is followed by a radiation survey to verify the effectiveness of the task. A record is kept of the radiation level of each piece and its disposition into the storage areas located in the Reactor Building. Essentially all tasks are hands-on operations, since only very low levels of radiation exposure are anticipated.

SAFSTOR activities are conducted in accordance with the decommissioning quality assurance plan and are checked by the licensee quality assurance specialist. Environmental monitoring during SAFSTOR is performed by the reference research reactor staff.

The reactor water cooling and cleanup system is kept in operation continuously until all the fuel has been removed from the core. Cleaning of contamination from the fuel removal task is, therefore, continuous and concentrated in the ion exchange resins. The fuel is then shipped to a storage or reprocessing facility offsite, thereby relieving the licensee of restrictions and regulations associated with fuel possession. The reactor pool is then drained and the associated piping and equipment interconnecting with the Heat Exchanger Building are dissassembled and stored. When tasks that may create airborne dispersion of radioactive materials are completed, the HVAC exhaust system is dissassembled and decontaminated. Next, all unnecessary utilities are deactivated, including electrical circuits, transformers, water

service, pneumatic systems and other systems. A complete security system is installed including a radiation alert system, a fire alarm system, and an intrusion identification system.

Safe storage begins when the preparations for safe storage are completed. During this period surveillance and maintenance is carried out to assure that the facility is secure and poses no unexpected risk to the public. In addition to the security system discussed, which incorporates remote monitoring, visual inspection and repair activities are continuously performed.

It is postulated that 12 months are needed for planning and preparation prior to reactor shutdown. Based upon detailed work plans, about 5 months are required to complete the tasks associated with preparations for safe storage. It is assumed that the facility will remain in safe storage until deferred decontamination takes place.

J.1.1.2 Decontamination, Deactivation and Sealing Methods

Decontamination, deactivation and sealing methods proposed for the reference research reactor are those in general use and suggested for similar tasks in decommissioning studies on other reactor types.[1,2] Except for the Reactor Building, all other areas are decontaminated to unrestricted use levels.

Noncombustible contaminated materials are stored within the Reactor Building. All equipment extraneous to the necessities of safe storage is deactivated. The possible spread of contamination outside the Reactor Building is eliminated by sealing all ports and entries from the reactor structure and the reactor room. Physical barriers are utilized to prevent access to the Reactor Building, except where necessary for the surveillance and maintenance activities. Air flow leaving the areas containing radioactive materials is filtered to prevent release of airborne contamination.

J.1.1.3 Spray Painting and Contaminated Material Transfer

Spray painting and plastic wrapping are two methods used to control contamination when transferring radioactive materials to the Reactor Building. Spray painting is also used on such things as structures, walkways, and ladders to immobilize contamination during the safe storage period in the Reactor Building.

J.1.1.4 Decontamination and Isolation Procedure

The following procedure is postulated to be used to prepare the Reactor Building for safe storage.

1. Conduct initial radiation survey.
2. Vacuum interior surface areas.
3. Deactivate nonessential systems and equipment.
4. Clean interior surface areas and exposed surfaces of equipment and piping.
5. Clean remaining hot spots.
6. Apply protective paint (determined on a case-by-case basis).
7. Transfer contaminated equipment and materials into the Reactor Building from other buildings as they are dismantled.
8. Decontaminate and seal vent systems.
9. Install HEPA-filtered vents in reactor structure and Reactor Building.
10. Deactivate remaining nonessential systems and equipment.
11. Install intrusion alarms; provide for offsite readout for intrusion, fire, and radiation survey.
12. Conduct final radiation survey.
13. Secure the structure.

J.1.1.5 Special Tools and Equipment

A list of special tools and equipment and their function as postulated for use in preparations for safe storage is given in Table J.1-1. These Items are in addition to tools and equipment already in general use and planned for use in decommissioning tasks at the reference research reactor.

J.1.1.6 Essential Systems and Services

Most facility systems (such as those that provide power, heat, water, communications and safety) must be maintained during preparations for safe

TABLE J.1-1. Special Tools and Equipment for Preparations for Safe Storage at the Reference Research Reactor

Item	Estimated Number Required	Function
Portable Oxyacetylene Torch	1	Cutting and welding steel
Nibbler	1	Sectioning steel materials
Power Hack Saw	1	Cutting piping
Submersible Pump & Filters	1	Unloading fuel
High-Pressure Water Jet	1	Decontaminating Pump House walls
2000-kg Forklift	1	Material and Retention Tank Handling
Concrete Drill with Dust Collection System	1	Drilling concrete for surface removal
Concrete Surface Spaller	1	Contaminated concrete surface removal
Supplied-Air Plastic Suits	1	Safety envelope for personnel
Paint Sprayer	1	Immobilization of contamination
HEPA Filters	1	Exhaust air contamination control

storage. As the decommissioning tasks become complete, the systems are deactivated and the services terminated in order to minimize the costs during the safe storage period.

Certain systems and services are, however, required during the safe storage period. These are given in Table J.1-2.

It is postulated that these systems are inspected, maintained and operated by members of the reference university staff.

TABLE J. 1-2. Systems and Services Required During the Safe Storage Period

System and Service	Justification
Electrical Power	Lighting, heat, surveillance monitoring, radiation monitoring and alarms
Fire Protection	Health and safety
Telephone Communications	Safety
Security	Public safety and plant protection

J.1.2 Activities, Schedules and Manpower Requirements for Placing the Reference Research Reactor in Safe Storage.

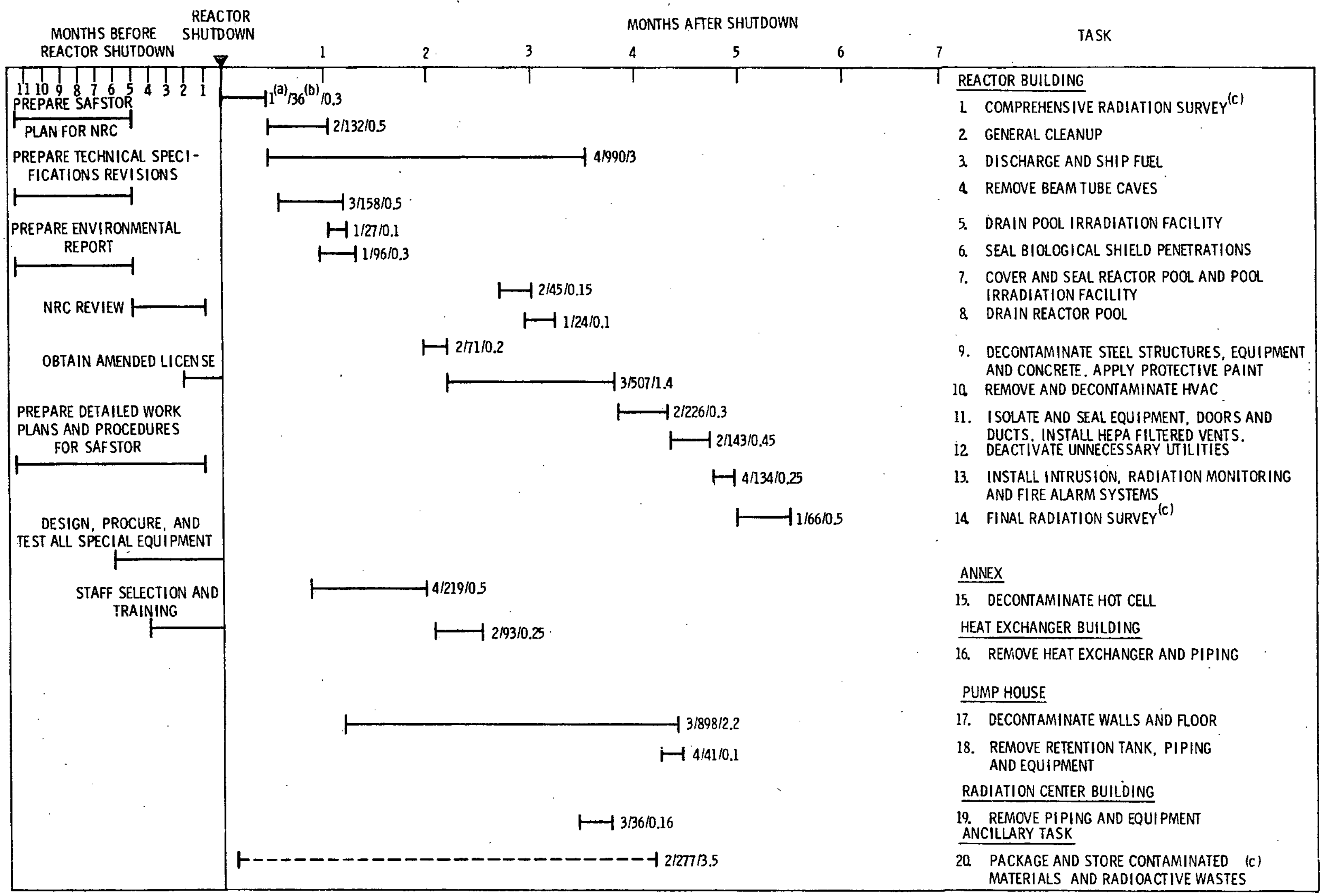
This section contains the postulated work tasks, schedules, and manpower requirements for placing the reference research reactor into safe storage.

The development of these parameters include the results of applying engineering judgement based on previous reactor decommissioning studies[1-3] and engineering estimates applied to tasks unique to the reference research reactor.

The task schedule and sequence for preparing the reference research reactor for safe storage is shown in Figure J.1-1. Also shown are schedules for several preliminary tasks needed prior to inception of the actual decommissioning. Planning and preparation is estimated to require 1 year, while the actual tasks for placing the reactor into safe storage require 5 months.

The tasks shown in Figure J.1-1 are performed on a single, 8-hour shift, 5-day-per-week basis. Each task in this figure postulates a crew size that will provide a reasonably constant manpower loading for the bulk of the project.

The dedicated manpower requirements for preparations for safe storage of the reference research reactor are shown in Table J.1-3. The compact size of the research reactor site makes it convenient to do all radiation surveys at the same time. From the table, it is apparent that rather small crew sizes are in order for most of the decommissioning tasks.



LEGEND:

|——| CONTINUOUS OPERATIONS OVER TIME SPAN SHOWN

|- - -| INTERMITTENT OPERATIONS OVER TIME SPAN SHOWN

TOTALS: PERSON MONTHS 32.0
PERSON HOURS 5625
EXPOSURE HOURS 4850

(a) TASK INFORMATION NUMBERS INDICATE IN SEQUENCE: DIRECT STAFF PER DAY/EXPOSURE HOURS/CALENDAR MONTHS DURATION. WORKERS DEDICATING 15% OR LESS OF THEIR TIME TO THE TASK ARE NOT INCLUDED IN THE DIRECT STAFF PER DAY NUMBER.

(b) THIS NUMBER INCLUDES ALL WORKER EXPOSURE TIME AND IT IS ASSUMED THAT 75% OF THE WORKING DAY IS IN THE RADIATION ZONE.

(c) TASK INCLUDES ALL BUILDINGS.

FIGURE J.1-1. Overall Task Schedule and Sequence for the Preparations for Safe Storage at the Reference Research Reactor

TABLE J.1-3. Dedicated Manpower Requirements for Preparations for Safe Storage at the Reference Research Reactor

Location Task	Task Duration (months)	Dedicated Manpower Requirements (man-months) Supervisors	Utility Operators	Laborers	Craftsmen	Health Physics Technicians	Total Man-Months
Reactor Building							
1. Comprehensive Radiation Survey[a]	0.27	--[b]	--	--	--	0.27	0.27
2. General Cleanup	0.50	--	--	1.00	--	--	1.00
3. Discharge and Ship Fuel	3.00	3.0	2.5	0.50	--	1.50	7.50
4. Remove Beam Tube Caves	0.50	0.15	0.25	0.75	--	0.05	1.20
5. Drain Pool Irradiation Facility	0.12	0.03	--	0.12	0.05	0.01	0.21
6. Seal Biological Shield Penetrations	0.30	0.10	--	0.40	0.10	0.05	0.65
7. Cover and Seal Reactor Pool and Pool Irradiation Facility	0.015	0.02	0.03	0.18	0.08	0.02	0.33
8. Drain Reactor Pool	0.07	0.02	--	0.14	--	0.02	0.18
9. Decontaminate Steel Structures Equipment Concrete: Apply Protective Paint	0.23	0.07	--	0.45	--	0.02	0.54
10. Remove and Decontaminate HVAC	1.40	--	--	2.70	1.00	0.14	3.84
11. Isolate and Seal Equipment-Doors-Ducts. Install HEPA Filtered Vents	0.30	0.14	--	0.60	0.84	0.14	1.72
12. Deactivate Unnecessary Utilities	0.45	0.14	--	0.45	0.45	0.05	1.09
13. Install Intrusion, Radiation Monitoring and Fire Alarm System	0.25	0.25	--	0.14	0.50	0.14	1.03
14. Final Radiation Survey[a]	0.50	--	--	--	--	0.05	0.05
Annex							
15. Decontaminate Hot Cell	0.50	--	--	1.40	--	0.26	1.66
Heat Exchanger Building							
16. Remove Heat Exchanger and Piping	0.25	0.05	0.10	0.25	0.25	0.05	0.70
Pump House							
17. Decontaminate Walls and Floor	2.20	--	--	6.60	--	0.20	6.80
18. Remove Retention Tank-Piping and Equipment	0.10	0.04	0.10	0.10	--	0.07	0.31
Radiation Center Building							
19. Remove Piping and Equipment	0.16	0.03	--	0.16	0.06	0.02	0.27
Ancillary Task							
20. Package and Store Contaminated Materials and Radioactive Wastes	3.50	--	--	2.00	--	0.10	2.10
Totals		4.04	2.98	17.94	3.33	3.16	31.45

(a) Includes all buildings.
(b) Denotes no manpower dedicated to task.

The total staff labor requirements during preparation for safe storage for both decommissioning workers and for management and support staff are given in Table J.1-4. The total labor requirements for decommissioning workers are slightly higher than those estimated for dedicated worker requirements. This is due to the necessity of maintaining certain personnel continuously, irrespective of their dedicated task load during each phase of the decommissioning project. This undedicated manpower is assumed to: 1) augment the basic work force on an as-needed basis, and 2) produce the manpower necessary for the numerous, small, unspecified work activities that accompany the overall decommissioning project.

The totals in Table J.1-4 include about 6 man-years of support staff (1 year of which is attributed to preparatory activities before reactor shutdown), and 2.8 man-years of decommissioning workers during the 5-month decommissioning project.

J.1.2.1 Preparations for Safe Storage Activities

This section describes the rationale and detail associated with the tasks and their timing. The overall task schedule and sequence for preparations for safe storage is given in Figure J.1-1.

It is important to remove the fuel from the core as early as possible and to complete the removal process as quickly as practical. Although fuel shipment activities will not interfere with other tasks, activities relating to reactor pool draining, sealing, and cleaning, as well as activities at the nearby pool irradiation facility, must be delayed until the fuel has been removed. The timing of reactor pool draining is coordinated with construction of the reactor pool cover. The reactor pool water serves as radiation shielding for workers constructing the pool cover until such time as it is necessary to drain the reactor pool.

After the fuel has been removed, all of the neutron-activated materials, representing over 99.8% of the total site inventory of radioactive materials, remain in the reactor pool. For this reason, several task steps are taken that will improve the isolation of the reactor structure within the Reactor Building. These steps involve sealing access ports and passageways into the

TABLE J.1-4. Staff Labor Requirements for Preparations for Safe Storage at the Reference Research Reactor

Position	Staff Labor Requirements (man-months)						Total Staff Labor Required
	Prior to Shutdown	After Shutdown					
Management and Support Staff:	-12[a]	+1	+2	+3	+4	+5	(man-years)
Decommissioning Superintendent	2	1	1	1	1	1	0.58
Secretary	2	1	1	1	1	1	0.58
Clerk and Procurement Specialist	2	1	1	1	1	0.5	0.54
Contracts and Accounting Specialist	2	1	1	1	1	1	0.58
Security Supervisor	0	1	1	1	1	1	0.42
Security Patrolman[b]	0	3	3	3	3	2	1.2
Armed Guards[c]	0	3	3	3	1	0	0.83
Health and Physicist and Safety Specialist	2	1	1	1	1	1	0.58
Control Room Operator	0	1	1	1	0.5	0	0.29
Quality Assurance Specilist	1	1	1	1	1	0.5	0.46
Subtotals	11	14	14	14	11.5	8	6.06
Decommissioning Workers[c]							
Crew Leader		0.9	1.2	1.1	0.5	0.3	0.33
Utility Operator		0.9	0.8	1.0	0.3		0.25
Laborer		2.3	5.1	6.1	4.2	0.4	1.50
Craftsman		0.1	.2	1.0	1.3	0.7	0.28
Health and Physics Technician		1.0	1.0	1.0	1.0	1.0	0.42
Subtotals		5.2	8.3	10.2	7.3	2.4	2.78
Totals	11	19.2	22.3	24.2	18.8	10.4	8.84

(a) Time relative to reactor shutdown.
(b) Based on information supplied by personnel at the reference research reactor, when fuel dose is ≤100 rem/hr within 1 meter of the material's surface. Both response and access-control personnel are necessary on a three-shift, 7-day basis. This requirement is applicable when 70%-enriched fuel is present.
(c) Requirements following reactor shutdown are based on Table J.1-3.

reactor pool and placing permanently sealed covers over the reactor pool and the pool irradiation facility. Since the pool irradiation facility is postulated as a storage area for radioactive materials, it is drained and filled with these materials before sealing. Contaminated materials removed from tasks associated with the other building, started early-on, are available for storage (see Figure J.1-1, Tasks 5 and 17).

It has been customary, in other reactor studies,[1,2] to chemically decontaminate primary reactor cooling system pipes and other contaminated piping. This task is not postulated for the reference research reactor because of the already low dose rates from the systems associated with the components. The small benefits derived from expected dose rate reductions due to decontamination of 0.002 R/hr to 0.0005 R/hr are not considered cost effective since the decommissioning task is already an efficient, hands-on activity.

Simultaneous decommissioning tasks at the Reactor Building and at the other buildings are coordinated so that space is available in the Reactor Building for the contaminated materials from the other buildings. This storage space is either in the pool irradiation facility or, after it is filled, in an area of the Reactor Building allocated for this purpose, preferably near the large external access doors of the reactor room. Once decontamination tasks are completed in each building (except for the Reactor Building), it is released for unrestricted use.

After cleanup and storage of loose materials in the Reactor Building, including the beam tube caves, the activities already described are conducted relative to the two pools in the reactor structure, including draining of the piping lines. Then, all equipment gratings, ladders, stairs and other miscellaneous items requiring decontamination are cleaned to their lowest practical levels. Protective paint coatings are applied to cleaned areas that still exibit residual contamination. Nonessential utilities are deactivated, including fans, electrical units, motors, pumps and the crane. The exhaust HVAC ductwork from the reactor room is removed and decontaminated. Remaining openings are sealed and a HEPA filtered vent unit installed in one of the openings to allow the reactor room to breathe without allowing airborne contamination to leave the Reactor Building. Similar HEPA units are placed on the

penetrating beam tube opening into the reactor pool and on a post designed into the top of the pool irradiation facility cover. All entryways into the reactor room and Reactor Building which give access to the safe storage area are barred and sealed permanently. Only the accessway necessary for surveillance activities is available for passage of authorized personnel.

J.1.2.2 Site Survey

It is recognized that radiation surveys and soil analyses are required to demonstrate the contamination levels, if any, surrounding the reference research reactor. No level of effort is included in this study for cleanup tasks of this nature. Annual atmospheric releases from the reactor within the site boundary and in close proximity outside of the boundary have demonstrated that no soil contamination has accumulated after 5 years of plant operation. It is postulated that this will be the situation at reactor shutdown.

J.1.3 Manpower Needs During the Safe Storage Period for the Reference Research Reactor

This section contains the postulated staff required and their task activities incurred during the safe storage period. The estimated manpower requirements during the safe storage period for the reference research reactor are summarized in Table J.1-5. A breakdown of the largest single item, estimated manpower for the environmental monitoring task, is shown in Table J.1-6. About one-half of a man-year is estimated to adequately safisfy the annual safe storage needs.

TABLE J.1-5. Estimated Annual Manpower Requirements During the Safe Storage Period for the Reference Research Reactor

Task	Estimated Annual Labor (man-years)
Surveillance and Maintenance	0.08
Secretarial	0.08
Repair	0.08
Security	0.12
Environmental and Radiological Monitoring	0.16
Inspection and Quality Assurance Verification	0.02
Total	0.54

TABLE J.1-6. Estimated Manpower Requirements for Environmental Monitoring During the Safe Storage Period for the Reference Research Reactor

	Personnel	Estimated Labor (man-years)
Sample Preparation, Storage and Archive. Data Collection and Archive.	Health Physicist	0.02
Data Analysis and Reporting	Health Physicist, Superviser	0.04
Sample Analysis	Radiochemist	0.04
Instrument Repair and Calibration	Instrument Specialist	0.04
Audit Functions	Quality Assurance Specialist	0.02
Total		0.16

Figure J.1-2 illustrates the postulated staff organization for the required activities during the safe storage period.

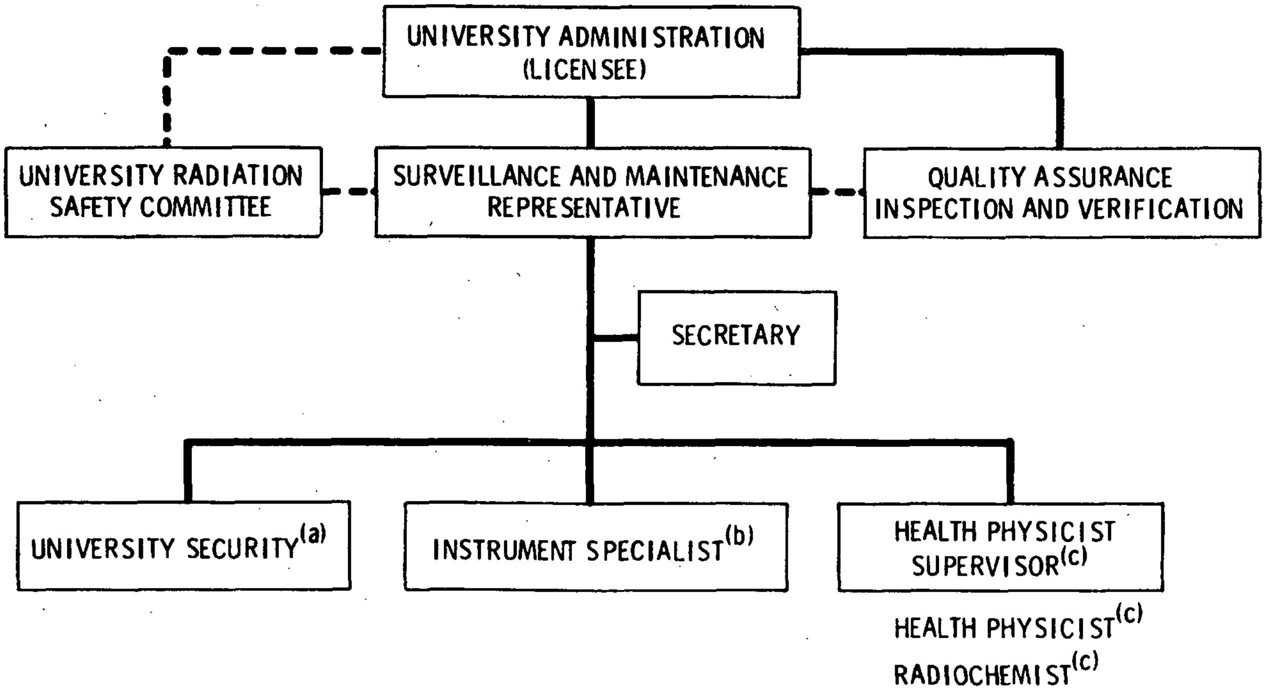
J.1.4 Costs of Safe Storage for the Reference Research Reactor

This section presents the estimated costs of both preparing the reference facility for safe storage and the annual costs during the safe storage period.

J.1.4.1 Preparations for Safe Storage

The estimated costs of preparations for safe storage for the reference research reactor are summarized in Table J.1-7. The total cost for this task is estimated at about $0.5 million. The major contributor to the total cost is the staff labor (at about 85% of the total cost). Sections that follow provide further analyses of this large proportion.

The costs of other optional tasks that may be appropriate at some time during SAFSTOR are also included in Table J.1-7. Their magnitude will produce a serious impact on the total costs involved in the decommissioning alternative selected.



[a] ASSIGNED TO SITE SECURITY TASK
[b] ASSIGNED TO EQUIPMENT MAINTENANCE TASK
[c] ASSIGNED TO ENVIRONMENTAL MONITORING TASK

FIGURE J.1-2. Postulated Staff Organization for the Safe Storage Period

J.1.4.2 Disposal of Radioactive Materials

Combustible radioactive wastes are postulated to be disposed of at a shallow-land burial ground. All other contaminated materials will be stored onsite, within the pool irradiation facility cavity or at a designated area of the Reactor Building. The packing and storage of these materials are included in the cost of an additional task, task 20, in Figure J.1-1. The cost of disposal for the radioactive wastes in a single shipment is $5530 and represents only 1.4% of the total preparatory costs.

J.1.4.3 Staff Labor

The cost of staff labor during preparations for safe storage is shown in Table J.1-8. As mentioned in Section J.1.4.1, this labor cost of $0.34 million represents over 85% of the total preparatory cost of safe storage and is by far

TABLE J.1-7. Summary of Estimated Costs of Placing the Reference Research Reactor in Safe Storage

Cost Category	Estimated Costs ($)[a,b]	Percent of Total[c]
Disposal of Radioactive Materials[d]	5 530	1.4
Storage of Radioactive Materials and Contaminated Wastes	11 200	2.8
Staff Labor	335 210	85.1
Energy	8 080	2.1
Special Tools and Equipment	2 340	0.6
Miscellaneous Supplies	15 000	3.8
Nuclear Insurance	2 890	0.7
License Fees	13 950	3.5
Subtotal	394 200	100.0
Contingency (25%)	98 550	
Total, Preparations for SAFSTOR	492 750	
Other Possible Costs		
Spent Fuel Shipment	60 980[e]	
Contingency (25%)	15 245	
Total, Other Possible Costs	76 225	

(a) 1981 costs.
(b) The number of significant figures shown is for computational completeness.
(c) Individually rounded to the nearest 0.1%.
(d) Includes dry solid wastes only.
(e) Includes cost of containers, overpacks and 800-km transportation, only.

TABLE J.1-8. Cost of Staff Labor While Preparing the Reference Research Reactor for Safe Storage

Position	Staff Labor Costs[a,b] -12[c]	+1	+2	+3	+4	+5	Total Staff Labor Costs ($)
Management and Support Staff:							
Decommissioning Superitendent	14 850	7 430	7 420	7 430	7 420	7 430	51 970
Secretary	4 020	2 020	2 020	2 020	2 020	2 020	14 120
Clerk and Procurement Specialist	4 040	2 020	2 020	2 020	2 020	2 020	13 130
Contracts and Accounting Specialist	7 860	3 930	3 930	3 930	3 930	3 920	27 500
Security Supervisor		4 660	4 660	4 660	4 660	4 600	23 300
Security Patrolman		6 350	6 350	6 350	6 350	4 230	29 630
Armed Guards		7 500	7 500	7 500	2 500		25 000
Health Physicist and Safety Specialist	7 820	3 910	3 910	3 910	3 910	3 910	27 370
Control Room Operator		2 880	2 880	2 880	2 430		11 070
Quality Assurance Specialist	3 910	3 910	3 910	3 910	3 910	1 950	21 500
Subtotals	42 500	44 600	44 600	44 610	39 150	29 130	244 590
Decommissioning Workers[d]							
Crew Leader		3 310	4 440	4 070	1 850	1 110	14 670
Utility Operator		2 420	2 150	2 690	810		8 100
Laborer		5 920	13 130	15 710	10 820	1 030	46 370
Craftsman		260	520	2 610	3 390	1 830	8 780
Health Physics Technician		2 510	2 510	2 510	2 510	2 510	12 700
Subtotals		14 420	22 750	27 590	19 380	6 480	90 620
Totals		59 020	67 350	72 200	58 530	35 610	335 210

(a) Calculated as the product of the data given in Table J.1-4 and the corresponding data given in Table M.1-1.
(b) The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest $10.
(c) Time relative to reactor shutdown.
(d) Requirements following reactor shutdown are based on Table J.1-2.

the most significant cost item. It is clear from an analysis of Table J.1-7 that the management and support staff costs of $0.24 million represent the largest portion (73%) of the staff labor. This latter cost is very sensitive to the length of time that the management and support staff is needed and is therefore sensitive to the schedule and sequencing times postulated in Figure J.1-1.

J.1.4.4 Energy

Table J.1-9 presents a summary of the energy usage and the incumbent costs accrued during preparations for safe storage at the reference research reactor. The energy usage is based upon a detailed analysis of the essential systems necessary during these decommissioning tasks as well as on the current usage and charge-out rates at the reference facility.

At the present cost of $0.025 per kWh at the reference research reactor, the cost of about 323 MWh of electricity is about $8100, representing 2.1% of the total preparatory costs of safe storage.

TABLE J.1-9. Cost of Energy During Preparations for Safe Storage of the Reference Research Reactor

System or Equipment[a]	Estimated Electrical Energy Usage (kWh)[b]	Costs ($)[c]
General System (crane, etc.)	1 580	40
HVAC	82 620	2 070
Lighting	13 420	336
Control Room	1 520	38
Fire Protection	180	5
Security	3 440	86
Communications	530	13
Domestic Water	21 180	530
Reactor Water	13 650	341
Compressed Air	3 320	83
Building Heating	176 400	4 410
Decommissioning Equipment	5 000	125
Total Energy Usage	322 840	
Total Energy Cost		8 077

(a) All buildings are included in this analysis.
(b) Based on present and past electrical energy usage at the reference research reactor and an analysis of individual component requirements.
(c) Estimated from the required equipment listed in Table J.1-1 and the present rate per kWh at the reference research reactor of $0.025.

J.1.4.5 Special Tools, Equipment and Miscellaneous Supplies

Based upon the requirements shown in Table J.1-1 and their estimated cost of acquisition, primarily by short-term rental, about $2300 is allocated for special tools and equipment. This represents less than 1% of the total preparatory costs of safe storage for the reference research reactor.

The variety of decommissioning tasks needed for safe storage requires a large number of associated material supplies to adequetly complete the tasks. These supplies include protective clothing, hand tools, drill bits, sweeping compounds, mops, plastic sheeting and bags, demineralizer resins, saw blades, torch gases, filters, steel cover plates, welding rods and a host of cleaning supplies. The cost of these items is estimated by comparison with other decommissioning studies,[1,2] based on the relative building size involved. About $15,000 is derived as the estimated cost of these miscellaneous items, which represents about 4% of the total preparatory costs of safe storage.

J.1.4.6 Nuclear Insurance

The costs of nuclear liability insurance during preparations for safe storage are estimated based on the annual operating premium of $7700. Based upon a 5-month pro rata premium and a 10% credit on the remaining premium during the first year of reactor shutdown, the cost for nuclear insurance is $2890. This represents less than 1% of the total preparations costs.

J.1.4.7 Licensing Fees

The fees charged for licensing services performed by the NRC are delineated in 10 CFR Part 170.23. Two amendments are needed from the operating license to enter the safe storage period. In addition, two inspections are anticipated to occur during the preparatory period. A summary of these costs is given in Table J.1-10. The total cost of licensing fees is about $14,000 and represents 3.5% of the total preparations for safe storage costs.

J.1.4.8 Other Possible Costs

A significant cost will accrue due to inclusion of the irradiated fuel shipment within the scope of the preparatory safe storage decommissioning

TABLE J.1-10. Costs of Licensing Fees During Preparations for Safe Storage at the Reference Research Reactor[a]

Licensing Service Category	Fee ($)	Cost ($)[b]
Facility License Amendment (Class IV)	6 000	12 000
Health Safety and Environmental Inspection	650/yr	650[c]
Safeguards Inspection	1 300/2 yr[d]	1 300
Total		13 950

(a) Code of Federal Regulations, 10 CFR, Part 170.
(b) Based on two license amendments, one to obtain a possession-only license prior to dismantlement and one to terminate that license following DECON.
(c) Based on a minimum annual fee for the 5-month DECON task.
(d) Based on a minimum fee for the 5-month DECON task.

tasks. The estimated cost for this task is about $61,000 as described previously for DECON (see Section I.1.3.7 of Appendix I).

The cost of demolishing all of the unneeded decontaminated and uncontaminated structures at the reference research reactor by an independent contractor is discussed in Appendix L and is summarized in Table L.3-1 and included in Table J.1-7. The total cost of $196,750 (without contingency) includes labor, supplies, overheads and profits, but not extraordinary insurance premiums or bonding.

In the case of privately owned research reactors, it should be noted that an anticipated cost from property tax assessments can also be charged to the preparations for safe storage; however, at this time specific values are considered outside the scope of this study due to site-specific assignment rates.

J.1.5 Costs During the Safe Storage Period for the Reference Research Reactor

Annual costs during the safe storage period for the reference research reactor are developed in this section. Based upon the estimated staff labor requirements given in Table J.1-4 and on other necessary expenditures described in the following sections, the total estimated annual costs during the safe storage period are presented in Table J.1-11.

TABLE J.1-11. Estimated Annual Safe Storage Costs for the Reference Research Reactor

Labor	Estimated Annual Cost ($)
Surveillance and Maintenance	2 480
Secretarial	1 940
Repair	2 570
Security	3 050
Environmental and Radiological Monitoring	7 960
Inspection and Quality Assurance Verification	940
Total, Labor Costs	18 940
Other Costs	
Reactor Building Roof Repair[a,b]	720
Reactor Building Roof Replacement[a,c]	1 420
Pest Control	400
Equipment and Supplies	500
Parts for Monitoring Intrusion and Fire System Including Video Monitoring and Remote Readout	1 000
Emergency Maintenance	500
Utilities	1 080
License Fee	650
Nuclear Liability Insurance	770
Total, Other Costs	7 540
Subtotal	26 480
Contingency (25%)	6 620
Total, Annual Safe Storage Costs	33 100

(a) Amortized on an annual basis.
(b) Based on estimated cost of $3,600 every 5 years.
(c) Based on estimated cost of $28,400 every 20 years.

The surveillance, maintenance and security forces postulated to take over operation of the reference research reactor are organized according to Figure J.1-2. Based on the personnel structure given in the figure, estimates are made of overall costs for periods of 10, 30, 50 and 100 years of safe storage.

J.1.5.1 Cost of the Environmental Monitoring Program During the Safe Storage Period

The estimated cost of the ongoing environmental monitoring program during the safe storage period at the reference research reactor is presented in Table J.1-12. It is postulated that personnel of the reference university staff carry out the monitoring program.

J.1.5.2 Security Costs During the Safe Storage Period

It is anticipated that members of the reference reactor staff are utilized both during the preparations for safe storage and throughout the safe storage period. The estimated costs of security are given in Table J.1-13. The activities required for security are:

- Routine surveys are conducted and personnel respond to intrusion and radiation alarms.

TABLE J. 1-12. Estimated Annual Cost of Environmental Monitoring During Safe Storage for the Reference Research Reactor

Task	Personnel	Estimated Cost ($)
Sample Preparation, Storage and Archive. Data Collection and Archive	Health Physicist	940
Data Analysis and Reporting	Health Physicist, Supervisor	2 400
Sample Analysis	Radiochemist	2 400
Instrument Repair and Calibration	Instrument Specialist	1 280
Audit Functions	Quality Assurance Specialist	940
Totals		7 960

TABLE J.1-13. Estimated Security Costs During Safe Storage for the Reference Research Reactor[a]

Activities	Cost ($)
Utilize Existing Video Monitoring System of Reactor Room and Add Remote Alarm Readout	12 000[b]
Routine Security Surveys, Equipment Maintenance, Monitoring Systems and Alarm Response	3 050[c]
Total (first year)	14 540
Total Annual Cost After First Year	3 050

(a) Includes onsite detection and notification systems for security, fire and radiation levels.
(b) One time cost only.
(c) From Table J.1-10.

- The existing video monitoring system is utilized with added onsite remote alarm signals.
- Locks and door alarms are installed on all entryways left unsealed.

The surveillance and maintenance representative is responsible for administrating the security program. His responsibilities include controlling access, upkeep, security breaches, violation reporting and records-keeping throughout the safe storage period.

J.1.6 Costs of Deferred Decontamination for the Reference Research Reactor

The cost impact of delaying decommissioning activities until after the safe storage period is evaluated in order to derive the cost differences for essentially the same tasks at a later date. It is assumed that the staff labor requirements are very similar for each remaining deferred decontamination task as it is for DECON (see Table I.1-3), except for reductions in decommissioning workers as a result of lower radiation doses for each task.

Significant reductions in radioactive waste volumes and in the absolute radionuclide quantities are also expected with time as the radioactive decay processes decrease the radionuclide quantity in the stored wastes. Estimates

of the cost savings due to this process are given in Table J.1-14 for safe storage intervals ranging from 10 to 100 years.

In comparing values from Table J.1-14 with those listed in Table I.1-6, it is apparent that the reduced quantities of radionuclides and the corresponding radioactive waste volumes can lower costs due to its effect on:

1. Burial ground curie surcharges assessed per curie.
2. Burial ground handling surcharges assessed per curie.
3. Burial ground liner surcharges assessed only when lines are needed.
4. Reduced cask rental due to reduced cask shielding requirements.
5. Container cost charges per unit volume of radioactive materials.
6. Transportation cost reductions due to weight and volume factors.

TABLE J.1-14. Cost Savings in Disposal of Radioactive Materials Due to Deferred Decontamination[a]

	Deferred Decontamination Time (years)			
	10	30	50	100
Neutron-Activated Materials	163[b]	830[b]	830[b]	830[b]
	0[c]	0[c]	750[c]	750[c]
	520[d]	1 460[d]	1 460[d]	1 460[d]
	0[e]	0[e]	4 775	4 775
Subtotal, Savings ($)	683	2 290	7 815	7 815
Contaminated Materials	0[f]	0[f]	3 600[f]	3 600[f]
	0[g]	22 815[g]	45 630[g]	45 630[g]
Subtotal, Savings ($)	0	22 815	49 230	49 230
Radioactive Materials	0[f]	0[f]	1 200[f]	1 200[f]
	0	3 505	7 010	7 010
Subtotal, Savings ($)	0	3 505	8 210	8 210
Total, Savings ($)[h]	680	28 610	65 260	65 260

(a) From Table I.1-6.
(b) Curie surcharge cost reductions due to radioactive decay.
(c) Handling surcharge cost reduction due to radioactive decay.
(d) Liner surcharge cost reductions due to reduced shielding requirements.
(e) Cask rental cost reductions due to cask differences from reduced shielding requirements.
(f) Container cost reductions due to reduced radioactive waste volumes.
(g) Transportation Cost reductions due to reduced waste shipments.
(h) Rounded to the nearest $10.

A summary of estimated costs for decontamination and dismantlement of the reference research reactor from its safe storage posture after 10, 30, 50 and 100 years is given in Table J.1-15. As a comparison, the costs of DECON, taken from Table I.1-5, are included. Small cost reductions with time are apparent due to:

1. Radionuclide quantity reductions discussed in this section.
2. Decommissioning worker reductions discussed in this section.
3. Energy reductions due to increased work efficiencies at lower dose rates.
4. Reduced remote control tool requirements.
5. Reduced supply requirements as tool requirements are eliminated.
6. Reduced insurance as the potential radioactive hazard diminishes.

TABLE J.1-15. Estimated Costs of Deferred Decontamination for the Reference Research Reactor

	Costs ($ thousands)				
		Decontamination Deferred (years)			
Cost Category	DECON[a]	10	30	50	100
Disposal of Radioactive Materials					
Neutron-Activated Materials	16.61	15.93	14.32	8.79	8.79
Contaminated Materials	60.06	59.87	37.05	10.64	10.64
Radioacive Wastes	9.62	9.41	5.90	1.20	1.20
Staff Labor	530.57	530.14	526.61	516.73	516.73
Energy	13.79	13.79	13.79	13.50	13.50
Special Tools and Equipment	21.15	21.15	4.45	4.28	4.28
Miscellaneous Supplies	6.21	6.21	5.15	5.15	5.15
Nuclear Insurance	4.62	0.64	0.64	0.64	0.64
License Fees	13.95	12.00	12.00	12.00	12.00
Subtotal	676.58	669.14	619.91	572.93	572.93
Contingency (25%)	169.15	167.29	154.98	143.23	143.23
Totals	845.73	836.43	774.89	716.16	716.16

(a) From Table I.1-6.

An interesting comparison of estimated total costs for DECON and various SAFSTOR alternatives is presented in Table J.1-16. These total costs for SAFSTOR include: 1) the initial costs of preparing the reference research reactor for the safe storage period, 2) the accumulated annual costs for the safe storage period selected, and 3) the costs to decommission the facility after the safe storage period. The estimated costs of DECON are taken from Table I.1-5 in Appendix I. The total SAFSTOR costs escalate rapidly with time, from about $1.6 million after 10 years to $4.5 million after 100 years. The largest factor affecting these costs is the accumulating annual costs during the safe storage period. Although, the $3300 annual cost is moderate, it accrues with increasing safe storage periods, eventually overwhelming the relatively small decontamination costs at the reference research reactor.

J.1.7 Estimated Occupational Radiation Doses for SAFSTOR at the Reference Research Reactor

Estimates are made of the external occupational radiation doses that are accumulated by the decommissioning workers during activities related to SAFSTOR. These activities include tasks necessary: 1) during preparations for safe storage, 2) throughout the safe storage period, and 3) to decontaminate and decommission the facility after the safe storage period.

TABLE J.1-16. Total Costs of Various Decommissioning Alternatives for the Reference Research Reactor

Years After Reactor Shut Down	Decommissioning Costs in Thousands of Dollars[a]: DECON	Preparations for Safe Storage	Deferred Period	Deferred Decontamination	Total
0	845.73	--	--	--	845.73
10	--	492.75	331.00	836.43	1 660.18
30	--	492.75	993.00	774.89	2 260.64
50	--	492.75	1 655.00	716.16	2 863.91
100	--	492.75	3 310.00	716.16	4 518.91

(a) 1981 costs.

These estimates are based on a task analysis to determine the man-hours of effort spent in radiation zones and the anticipated dose rates associated with each task. The analyses are performed in the same manner as those described for DECON in Section I.1 of Appendix I.

J.1.7.1 Estimated External Occupational Radiation Doses Accumulated During Preparations for Safe Storage

A number of tasks necessary for preparations for safe storage are identical with those postulated for DECON at the reference research reactor. This is particularly true of tasks related to buildings other than the Reactor Building and certain early basic tasks such as radiation surveys, general cleanup and fuel handling. For these tasks, the same accumulated radiation doses to workers are postulated (See Table I.1-12) for both decommissioning alternatives. These estimates are reflected in the dose figures that appear in Table J.1-17 for the preparatory safe storage activities. The dose values shown in the table assume that during a standard 8-hour work day, 6 hours are spent by the decommissioning worker in a radiation zone whose magnitude is equal to the average exposure during the 6-hour interval. Tasks requiring part-time participation are assessed at the same pro-rata exposure per day. The dose estimates for each task are corrected for radioactive decay that has occurred prior to the time the task is one-half completed (based upon the 5-year half-life of ^{60}Co). A total of 4457 man-hours are spent by decommissioning workers in the preparations for safe storage, which results in an accumulated radiation dose of approximately 13.1 man-rem.

J.1.7.2 Estimated Occupational Doses Accumulated During the Safe Storage Period

Based upon the manpower required to carry out the tasks associated with safe storage given in Table J.1-5 and the coordinating estimated radiation dose data, the cumulative occupational external radiation dose to workers is given in Table J.1-18. Estimates are calculated for intervals of safe storage activities for 10, 30, 50 and 100 years following preparations for safe storage. Diminishing dose rates, which reflect radionuclide decay, are incorporated into the estimates. Very little addition to the 30-year cumulative dose of 0.78 man-rem is apparent, even after 100 years.

TABLE J.1-17. Estimated Occupational Radiation Doses for Preparations for Safe Storage of the Reference Research Reactor

Task	Supervisors[a] Exposure (man-hr)	Supervisors[a] Dose[b] (man-rem)	Utility Operators Exposure (man-hr)	Utility Operators Dose (man-rem)	Laborers Exposure (man-hr)	Laborers Dose (man-rem)	Craftsman Exposure (man-hr)	Craftsman Dose (man-rem)	Health Physics Technicians Exposure (man-hr)	Health Physics Technicians Dose (man-rem)	Task Totals Exposure (man-hr)	Task Totals Dose (man-rem)	Task Totals Decay Factor[c]	Task Totals Corrected Dose (man-rem)[d]
Reactor Building														
1. Comprehensive Radiation Survey[e]	--	--	--	--	--	--	--	--	36	0.036	36	0.036	0.989	0.036
2. General Cleanup	--	--	--	--	132	0.132	--	--	--	--	132	0.132	0.989	0.131
3. Discharge and Ship Fuel	396	2.772	204	2.128	66	0.462	--	--	224	1.568	990	6.930	0.978	6.778
4. Remove Beam Tube Caves	20	0.40	33	0.066	99	0.198	--	--	6	0.012	158	0.316	0.989	0.313
5. Drain Pool Irradiation Facility	4	0.008	--	--	16	0.032	6	0.012	1	0.02	27	0.054	0.989	0.053
6. Seal Biological Shield Penetrations	18	0.141	--	--	53	0.422	18	0.141	7	0.053	96	0.768	0.989	0.760
7. Cover and Seal Reactor Pool and Pool Irradiation Facility	3	0.015	4	0.020	24	0.119	11	0.053	3	0.013	45	0.225	0.989	0.223
8. Drain Reactor Pool	3	0.006	--	--	18	0.036	11	--	3	0.006	24	0.048	0.978	0.047
9. Decontaminate Steel Structures Concrete: Apply Protective Paint	9	0.013	--	--	59	0.119	--	--	3	0.005	71	0.142	0.978	0.139
10. Remove and Decontaminate HVAC	--	--	--	--	356	0.178	132	0.006	19	0.010	507	0.254	0.968	0.246
11. Isolate and Seal Equipment-Doors-Duct. Install HEPA Filtered Vents	18	0.009	--	--	79	0.040	111	0.055	18	0.009	226	0.113	0.957	0.108
12. Deactivate Unnecessary Utilities	18	0.009	--	--	59	0.030	59	0.030	7	0.003	143	0.072	0.957	0.069
13. Install Intrusion, Radiation Monitoring and Fire Alarm Systems	33	0.033	--	--	18	0.018	66	0.066	18	0.018	135	0.135	0.957	0.129
14. Final Radiation Survey[e]	--	--	--	--	--	--	--	--	66	0.017	66	0.017	0.946	0.016
Annex														
15. Decontaminate Hot Cell	--	--	--	--	185	1.850	--	--	33	0.330	219	2.219	0.989	2.195
Heat Exchanger Building														
16. Remove Heat Exchanger and Piping	7	0.007	13	0.013	33	0.033	33	0.033	7	0.007	93	0.093	0.978	0.091
Pump House														
17. Decontaminate Walls and Floor	--	--	--	--	871	0.871	--	--	27	0.027	898	0.898	0.967	0.869
18. Remove Retention Tank Piping and Equipment	6	0.012	13	0.026	13	0.026	9	0.018	--	--	41	0.082	0.957	0.079
Radiation Center Building														
19. Remove Piping and Equipment	4	0.004	--	--	21	0.021	8	0.008	3	0.003	36	0.036	0.967	0.035
All Buildings														
20. Package and Store Contaminated Materials[e] and Radioactive Wastes	--	--	--	--	264	0.528	--	--	13	0.026	277	0.554	0.967	0.536
Subtotals	533		357		2340		427		488		4219	13.124		12.853
Ancillaries														
Routine Radiation Surveys	--	--	--	--	--	--	--	--	238		238	0.238	0.967	0.230
Totals	533		357		2340		427		726		4457			13.083

(a) Includes shift engineers, crew leaders, craft supervisors, and senior health physics technicians.
(b) Based on data presented in Figure D.1-1 in Appendix D.
(c) Based on the half-life of ^{60}Co; calculated at the midpoint of the task timelines shown in Figure 1.1-1.
(d) The number of significant figures shown is for computational convenience and does not imply accuracy to the nearest millirem.
(e) Task includes all buildings.
(f) Indicates that category of decommissioning staff is not involved in the particular task.

TABLE J.1-18 Estimated Cumulative Occupational External Radiation Dose During Safe Storage of the Reference Research Reactor

Task	Estimated Annual Exposure (man-hr)	First-Year Task Dose Rate (R/hr)	Cumulative Dose (man-rem)[a] 1 yr	10 yr	30 yr	50 yr	100 yr
Surveillance and Maintenance	9	0.001	0.009	0.05	0.07	0.07	0.07
Repair	38	0.001	0.038	0.19	0.28	0.29	0.30
Security	38	0.001	0.038	0.19	0.28	0.29	0.30
Environmental and Radiological Monitoring	24	0.0005	0.012	0.06	0.09	0.09	0.09
Inspection and Quality Assurance Verification	16	0.0005	0.008	0.04	0.06	0.06	0.06
Totals	125		0.105	0.53	0.78	0.80	0.82

(a) Calculated assuming ^{60}Co to be the dominant occupational dose contributor during the first 100 years of safe storage.

J.1.7.3 External Occupational Radiation Doses Accumulated During Deferred Decontamination.

This study postulates that safe storage of the reference research reactor is a temporary situation. Eventual disposition of the materials in safe storage is expected at a time that is appropriate and expeditious for the licensee. Then, the amended nuclear license may be terminated.

The remaining tasks associated with deferred decontamination are performed essentially the same way as those postulated for DECON. Therefore, the occupational radiation dose accumulated by workers in deferred decontamination may be calculated from the doses given in Table I.1-12 for DECON and allowing for radioactive decay (based on the 5-year ^{60}Co half-life) for the safe storage period selected.

These occupational dose estimates are given in Table J.1-19 for decontamination starting 10, 30, 50, and 100 years after reactor shutdown.

TABLE J.1-19 Estimated Occupational Radiation Doses from Various Decommissioning Alternatives for the Reference Research Reactor

Years After Reactor Shutdown	Occupational Radiation Dose (man-rem)				
	DECON	Preparations for Safe Storage	Deferred Period	Deferred Decontamination	Total
0	18.34	--	--	--	18.34
10	--	13.08	0.53	1.48	15.09
30	--	13.08	0.78	0.11	13.97
50	--	13.08	0.80	0.01	13.89
100	--	13.08	0.82	0.01	13.90

As a final comparison of dose savings to decommissioning workers when SAFSTOR is chosen over DECON as the preferred decommissioning alternative, the change in overall total accumulated dose to workers (for all SAFSTOR tasks) with time is also shown in this table. It is apparent that little, if any, added exposure benefits can be derived after 30 years of safe storage at the reference research reactor.

J.2 METHODS POSTULATED FOR SAFSTOR AT THE REFERENCE TEST REACTOR FACILITY

The details presented in this section are specifically for decommissioning the reference test reactor facility by the SAFSTOR alternative. These details amplify the information presented in Sections 10 and 11 of Volume 1 and provide the basic information needed to assess the safety of decommissioning the reference test reactor facility via the SAFSTOR alternative.

The procedures discussed in this section are postulated to be used to decontaminate, deactivate, and secure radioactive materials and contamination within the buildings/areas of the reference test reactor facility, to achieve a condition that provides protection to the public and the environment, with limited maintenance and surveillance.

J.2.1 Preparing the Reference Test Reactor for Safe Storage

The preparatory activities, decommissioning methods, special tools and equipment, and essential systems and services used in preparing the reference

test reactor (including the MUR) for safe storage are discussed in this section. These preparations, methods and considerations, in order of their presentation in the following subsections, are:

- preparatory activities
- decontamination, deactivation, and sealing methods
- spray painting
- transfer of contaminated equipment and materials
- decontamination and isolation procedure
- special tools and equipment
- essential systems and services.

J.2.1.1 Preparatory Activities

Following completion of planning and preparation (see Section H.1 of Appendix H) and cessation of facility operations, a comprehensive radiation survey of the reference test reactor facility is completed. These surveys are required to finalize plans for draining and flushing contaminated process systems and for installing temporary shielding for personnel protection during subsequent decommissioning operations. Next, a general cleanup is accomplished and a total inventory of equipment is taken to determine usefulness of specific equipment to the decommissioning project. Equipment not so designated is identified for later disposal, resuse elsewhere, or onsite storage.

Following the final inventory cleanout, fuel shipments commence, and final decontamination operations are initiated. The conditions outlined below are met before commencing these operations.

1. Responsible management and safety personnel approve the following plans and procedures:
   - radiation work, industrial safety, and emergency procedures
   - equipment handling, disassembly, cleaning, and packaging and shipping procedures
   - equipment and facility decontamination plans and procedures.
2. Disposition is predetermined for all equipment. The equipment can be decontaminated for reuse, sold as scrap, or buried in the local

landfill dump; partially decontaminated for use at another restricted plant, left in place, or shipped to a licensed burial ground for contaminated materials.

3. All ventilation equipment, personnel protection systems, emergency power systems, fire protection systems, and radiation monitoring equipment in the building and onsite are in service and fully functional.

4. All personnel and contractors are adequately trained to perform their jobs.

5. Appropriate occupational safety equipment and continuous air sampling equipment are available for equipment disassembly, transfer, and cleanup.

6. Temporary portable cleaning chambers for decontaminating equipment are available (e.g., greenhouse with tank for water and steam rinsing of equipment, washing tanks, degreasers, etc.).

7. Packaging materials and shipping containers are available.

8. All equipment for dismantlement (where required) and decontamination operations is available.

9. A comprehensive radiation survey is completed, with all results mapped and used as a basis for each building, room, and area's work plan.

10. The system and procedures for the functions of special nuclear material accountability measurements and measurement control are established.

11. All unneeded process material supplies (e.g., bottled gases, acids, and caustics) are disconnected from the plant and disposed of.

The pattern for cleanup and decontamination of individual rooms is dictated by doing the most contaminated rooms first and improving the accessibility to staging areas. The need for certain rooms (e.g., scrap or waste handling facilities and hot maintenance) during SAFSTOR operations will delay their decommissioning.

For an individual room, accessibility to the equipment dictates the order of decommissioning. It is also desirable to shield hot spots and the most contaminated equipment in the room earliest. Where possible, unneeded equipment is removed first to provide better overall access. Equipment is disconnected from exhaust ductwork and the end is sealed or fitted with a filter to preclude contamination spread from back pressures. The final step is to carefully survey for radiation and decontaminate the ceiling, walls, and floor with detergent cleanser, powered brushes, and handwipes to the lowest practical levels. Floor tiles or strippable seal coats may be removed. Where hot spots occur, the concrete floors and/or walls may be chipped and vacuumed to achieve the lowest practical level of decontamination. A final survey is conducted to comply with the historical record requirements.

Strict accountability records are kept for equipment and appurtences being decontaminated and shipped. Because external radiation doses to workers are relatively low, many decontamination opertions are hands-on operations. Plastic tents are constructed when appropriate to contain contamination where equipment is being disassembled. Disassembly of piping, valves, pumps, ancilliary equipment, and ductwork external to equipment and hoods (e.g., in the Office and Laboratory Building) is conducted using strict contamination-control methods. Precautions are taken to prevent spills or to confine contamination spills that may inadvertently occur. Health physics personnel observe and monitor operations to ensure that safe radiation and industrial practices are followed at all times.

In addition, SAFSTOR activities are conducted in accordance with the decommissioning quality assurance plan (see Section G.7 of Appendix G) and are checked by the licensee quality assurance staff. Environmental monitoring (see Section G.8 of Appendix G) during SAFSTOR is performed by the same "contract services" organization that provided these radiological services during operation.

At reactor shutdown,[a] the reactor cooling and cleanup systems and the quadrant and canal (Q&C) systems contain the normal operating inventory of

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(a) The term "reactor shutdown" used in this portion of the study always refers to both the reference test reactor and its associated mock-up reactor (i.e., the MUR), located in Canal H.

radioactively contaminated water. During the decommissioning of the reference test reactor (and the MUR), the contaminated water inventories are processed repeatedly through the in-plant liquid radwaste systems for treatment before final disposition, since this water is postulated to be used over and over again for the various decontamination activities. These large reservoirs of deionized water are amenable to cleanup and resuse in this manner, and this approach reduces the total volume of radioactively contaminated water that is cleaned up.

The primary cooling water system (PCWS) is drained and flushed after the fuel is removed from the test reactor core. In addition, after the fuel is removed from the MUR and all fuel and experimental hardware are removed from Canal H, then Canal H can be drained, cleaned, dried, and further decontamination work can be completed on the MUR.

Following defueling of both reactors, the irradiated fuel is shipped to a government reprocessing plant so that technical specifications associated with having the fuel onsite can be eliminated and guard forces can be reduced. The MTR-type fuel is prepared for shipment by cutting the aluminum box ends off each spent fuel assembly before loading a critically safe complement of fuel plates into the spent fuel cask. The aluminum box end pieces are packaged as radioactive waste for shipment to a burial ground.

Disassembly, disposal, and further decontamination begins in the Reactor Building/Containment Vessel, proceeds through the Hot Laboratory Building, and concludes with the Waste Handling Building. The auxiliary structures described in Section C.2.12 of Appendix C, with the exception of the exhaust stack, are assumed to be uncontaminated.

The continuing care period begins when the preparations for safe storage are completed. This period is characterized by surveillance and maintenance activities designed to ensure that the facility remains in a condition that poses minimum risks to the public. The activities included in this phase are limited to environmental and facility radiation monitoring and to inspecting and repairing the physical barriers, the structures and the instrumentation.

Security is provided by the fence around the immediate facility site (see Figure C.1-1), by high-security locks on entrance doors, and by electronic alarms.

It is assumed that approximately 18 months before final plant shutdown are devoted to planning and preparation. Based on detailed work plans, about 6-3/4 months (from the time of initiation of physical decommissioning tasks) are estimated to be required to prepare the facility for the period of continuing care. It is assumed that the facility will remain in safe storage until deferred decontamination takes place.

J.2.1.2 Decontamination, Deactivation, and Sealing Methods

Decontamination, deactivation, and sealing methods proposed for the reference test reactor and the MUR employ techniques that have been used successfully and are described generically in Appendix G. In general, areas of the facility that must be accessible during continuing care are decontaminated to unrestricted use levels. Noncombustible, contaminated materials that are not removed from the facility may be placed in the drained and cleaned quadrants and canals. These areas are isolated from the remainder of the facility by structurally substantial physical barriers. In any case, the primary concern is to ensure that no recontamination of clean areas occurs and that air leaving a given area flows through a filter system or, in the case of liquid effluents, through the existing contaminated waste systems.

The particular method used to decontaminate, deactivate, and seal each system or piece of equipment is identified during the planning phase. In general, all systems not necessary to prevent the spread of contamination are deactivated. Equipment deactivation, isolation of contaminated areas, and sealing techniques are described generically in Sections G.2 and G.3, respectively, of Appendix G. Generic decontamination methods used in the preparations for safe storage are described in Section G.4.

It is assumed that two of the preparatory methods that can be used for temporary contamination control before transferring equipment and materials are: wrapping the items in plastic, and spray painting. Spray painting and transfer techniques are described in the following subsections.

J.2.1.3 Spray Painting

After the readily removable contamination is removed by the physical cleaning methods described in Section G.4 of Appendix G, the rooms or areas and their associated equipment may be spray painted before isolation or removal procedures begin. When possible, all contaminated surfaces, both inside and outside, may be coated to prevent the entrainment of radioactivity in the air during the active decommissioning tasks or during subsequent surveillance and maintenance activities.

In general, if the contamination on a surface cannot be removed by wiping or washing using standard decontamination solutions, it may be painted to fix the contamination in place. An example is a concrete surface that has been penetrated by contaminated liquids. While the surface might be clean initially, the subsurface contamination can migrate to the surface and be dispersed by air movement and/or foot traffic. On protected, interior surfaces with essentially no traffic or adverse environment, such paint coatings can be expected to last almost indefinitely. Part of the surveillance program is to monitor painted areas for deterioration of the coatings and to recoat them as necessary.

J.2.1.4 Relocation of Contaminated Equipment and Materials

Unsalvageable, contaminated equipment and other miscellaneous noncombustible items may be relocated to other secured, onsite retrievable storage areas, as described in Section J.2.1.1. It is anticipated that before transferring small equipment items, the items are carefully wrapped in plastic and/or spray painted to immobilize any contamination. Freshly exposed surfaces may be immediately painted to prevent dispersal of contamination. The disconnected items are carefully bagged and transferred into a retrievable storage area. The equipment and ductwork remaining in the work area are physically decontaminated as described in Section G.4 and, in addition, may be spray painted as previously described.

J.2.1.5 Decontamination and Isolation Procedure

The 13-point procedure given below is postulated to be used to prepare the contaminated areas throughout the reference test reactor facility for safe storage:

1. Conduct initial radiation survey.
2. Vacuum interior surface areas.
3. Deactivate nonessential systems and equipment.
4. Clean interior surface areas and exposed surfaces of equipment and piping.
5. Clean remaining hot spots.
6. Apply protective paint (determined on a case-by-case basis).
7. Transfer, as feasible, contaminated equipment and materials.
8. Decontaminate and seal vent systems.
9. Install HEPA-filtered vents.
10. Deactivate remaining nonessential systems and equipment.
11. Install intrusion alarms; provide for offsite readout for intrusion, fire, and radiation survey.
12. Conduct final radiation survey.
13. Secure the structure.

J.2.1.6 Special Tools and Equipment

Fewer special tools and equipment are required for the preparations for safe storage than for DECON. No dismantlement of highly activated material or equipment is necessary, thus eliminating the need for and expense of special remote handling equipment.

A list of special tools and equipment postulated for use in preparations for safe storage, together with their functions, is given in Table J.2-1.

J.2.1.7 Essential Systems and Services

During preparations for safe storage, certain facility systems and services must remain in place and in service for contamination control, for industrial safety, and to aid in the completion of decommissioning tasks. These systems are the same as those described generically in Section H.3 and are not repeated here.

TABLE J.2-1. Special Tools and Equipment for Preparations for Safe Storage at the Reference Test Reactor

Item	Estimated Number Required[(a)]	Function
Portable Oxyacetylene Torch	2	Cutting carbon steel equipment and piping; welding carbon steel
Nibbler	2	Sectioning steel materials
Guillotine Pipe Saw	2	Cutting piping
Power-Operated Reciprocating Hacksaw	2	Cutting piping
Closed Circuit, High-Resolution Television	As Required	Remote observation of tasks
Submersible Pump with Disposable Filter	3	Draining operations
High-Pressure Water Jet	2	Surface decontamination of walls and equipment
Powered Floor Scrubber	2	Decontamination
9100-kg Capacity Forklift	1	Packaging of contaminated materials, and loading of trucks
Concrete Drill with HEPA-filtered Dust Collecting System	1	Drilling holes in concrete as required for surface spalling
Concrete Surface Spaller	1	Removal of contaminated concrete surfaces
Front-End Loader (highly maneuverable, light-duty)	1	Cleanup and packaging tasks
Wet-Dry Vacuum Cleaner (HEPA-filtered)	3	Cleanup tasks
Portable Ventilation Enclosure, Filtered	3	Contamination control during cutting of contaminated material
Filtered Exhaust Fan Unit	2	Contamination control and personnel comfort
Supplied-Air Plastic Suit	50	Provide personnel with maximum respiratory and surface protection from radioactive contaminants
Polyruethane Foam Generator	2	Contamination control during HVAC isolation tasks
Paint Sprayer	2	Immobilization of contamination
HEPA Filter	50	Contamination control
Roughing Filter	50	Contamination control

(a) Based on a taskwise analysis of the schedule, plus spares, as required.

As areas within the facility are secured for safe storage, the essential systems and services in these areas are deactivated. Continuous service to the remaining work areas is maintained as required.

After placing the facility in safe storage, certain systems and services are required during the safe storage period. The systems and services postulated to be required during safe storage are listed in Table J.2-2, together with the justification for retaining each. The equipment in these systems is inspected and renovated to ensure adequate reliability before the surveillance and maintenance period begins. In addition, the intrusion alarm systems within the facility and on the perimeter fence are modified to provide surveillance capability by a commercial security agency.

Electricity requirements at the facility are nominal after the preparations for safe storage are completed. It is assumed that the plant's large electrical service is changed to a small electrical service near the end of the active decommissioning period to provide for the needs of the essential systems and services described in Table J.2-2.

J.2.2 Activities, Schedules, and Manpower Requirements for Placing the Reference Test Reactor in Safe Storage

This section contains the postulated activities and schedules for placing the reference test reactor and MUR in safe storage, and the postulated manpower needs to achieve that condition.

TABLE J.2-2. Systems and Services Required During the Safe Storage Period

Systems and Services	Justification
Electrical Power	Operation of electrical equipment, including lighting, surveillance monitoring, and radiation monitoring systems and alarms
Fire Protection System	Health and safety
In-Plant Communications System (telephone)	Personnel safety considerations
Security Systems	Public safety and plant protection considerations

The development of these estimates is the result of: 1) applying engineering judgement based on previous decommissioning studies in this series,[1-3] and 2) analyzing and adapting limited historical quantitative data from decommissioning tasks actually completed while placing the reference test reactor into a modified version of safe storage in early-1973.[4,5]

The overall task schedule and sequence for preparing the reference test reactor (including the MUR) for safe storage is shown in Figure J.2-1, together with the total decommissioning worker requirements for the period following final reactor shutdown. The planning and preparation tasks (see Section H.1 of Appendix H) for the 18-month period preceding reactor shutdown are also included. The preparations for safe storage are estimated to be accomplished during the 6-3/4-month period following final reactor shutdown. This time frame includes a radiation survey of the site.

The task completion schedule shown in Figure J.2-1 is presented on a calendar month time scale and also presents additional information. For example, the following:

├──────────┤ 6/1488/3

means that this particular task is performed by 6 decommissioning workers and is estimated to require a total of 1488 man-hours to complete the task within a 3 calendar month time span. Likewise the following:

├──────────┤ subcontract/2

means that this task is performed by subcontracted labor over a span of 2 months.

The preparatory safe storage tasks, with few exceptions, are performed on two 8-hour shifts, five days per week. Shipment of spent fuel is conducted three shifts per day, 7 days per week, as required. Nearly optimum decommissioning worker requirements are met by using the staff breakdown given in Table J.2-3. It is estimated that each of the two regular shifts and the

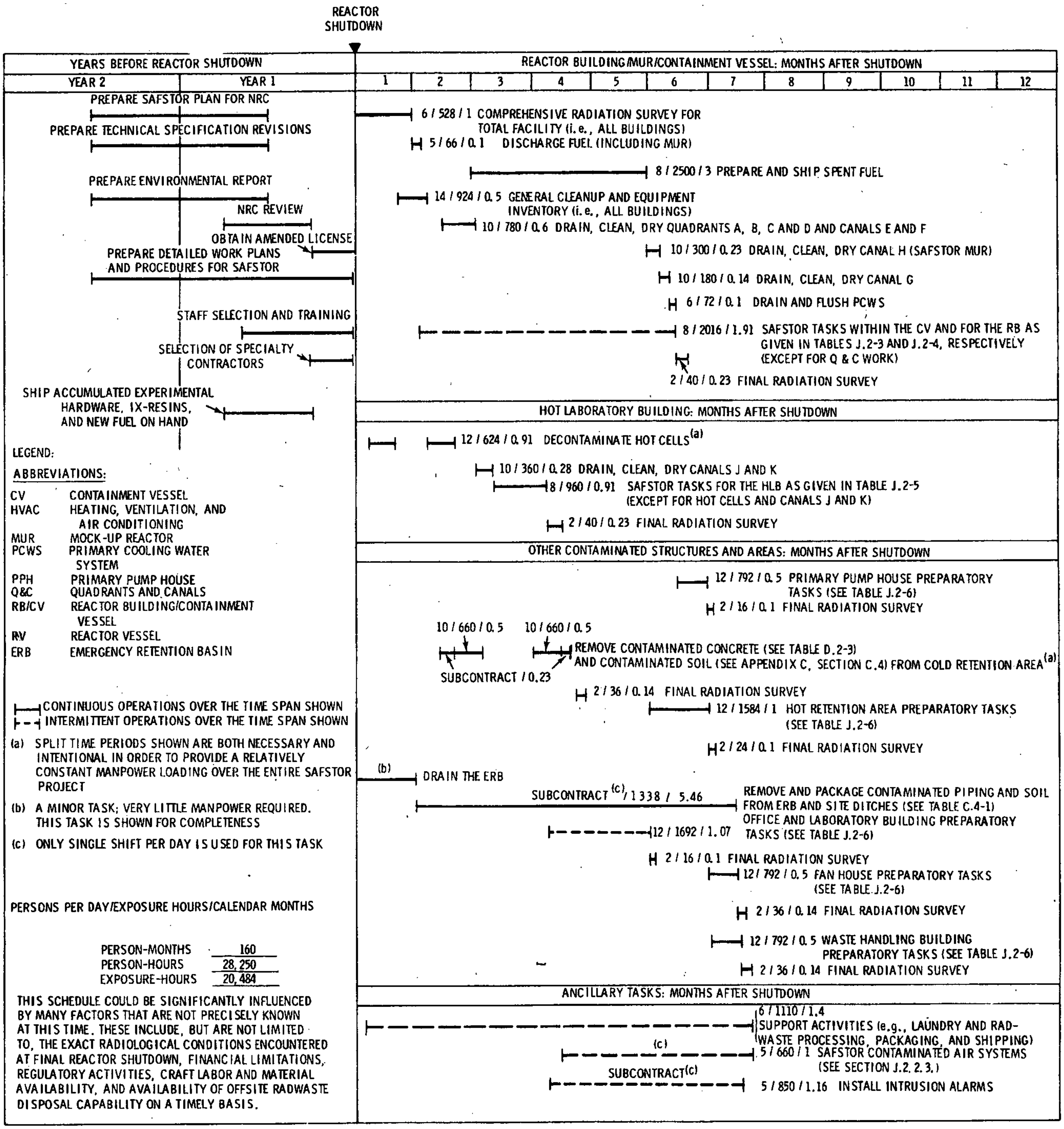

FIGURE J.2-1. Task Schedule and Sequence for the Preparations for Safe Storage at the Reference Test Reactor

TABLE J.2-3. Postulated Number and Specific Makeup of Crews Required to Prepare the Reference Test Reactor for Safe Storage

First Shift	Second Shift	Support Crew for Both Shifts[a]	Total Number of Staff
1 Shift Engineer	1 Shift Engineer		2
2 Crew Leaders[b]	2 Crew Leaders[b]	1 Crew Leader	5
2 Utility Operators	2 Utility Operators	2 Utility Opertors	6
2 Laborers	2 Laborers	1 Laborer	5
3 Health Physicists	3 Health Physicists	1 Health Physicists	7
5 Craftsmen	5 Craftsmen	_	10
15 Total	15 Total	5 Total	35

(a) Anticipated to be equally divided in providing laundry services and radwaste processing services, as well as backup service for the first and second shifts.
(b) Former senior reactor operator (SRO) or hot-cell specialist.

support crew for each shift shown in the table comprise a relatively constant manpower loading almost to the very end of the decommissioning project.

The total staff labor requirements for preparations for safe storage of the reference test reactor are given in Table J.2-4 in equivalent man-years for the 1.5 years before and the 0.56 years following final reactor shutdown, and include management and support staff as well as decommissioning workers. About 76 man-years of effort are estimated for preparing the reference test reactor for safe storage, including approximately 44 man-years for the management and support staff and about 32 man-years for the decommissioning workers.

The staff labor requirements shown in Table J.2-4 for the decommissioning workers during the 6-3/4 months following final reactor shutdown exceed the total manpower presented in Figure J.2-1. This "excess" manpower is assumed to: 1) augment the basic work force on an as-needed basis; and 2) provide the manpower necessary for the numerous, small unspecified work items that accompany an activity of this magnitude.

Following fuel removal from the reference test reactor and the MUR, the sequence in which the various systems must be drained and/or flushed is

TABLE J.2-4. Staff Labor Required to Prepare the Reference Test Reactor For Safe Storage

Position	Time Relative to Final Reactor Shutdown (year): -2	-1	1	Total Staff Labor Required (man-years)
	Annual Staff Labor Requirement (man-years)[a]			
Management and Support Staff				
Decommissioning Superintendent	0.5	1.0	0.73[b]	2.23
Secretary	0.5	1.0	0.73[b]	2.23
Clerk	0.0	1.0	0.57	1.57
Decommissioning Engineer	0.5	1.0	0.73[b]	2.23
Assistant Decommissiong Engineer	0.5	1.0	0.57	2.07
Radioactive Shipment Specialist	0.0	1.0	0.57	1.57
Procurement Specialist	0.2	1.0	0.57	1.77
Tool Crib Attendant	0.0	0.0	0.57	0.57
Control Room Operator[c]	0.0	0.0	0.57	0.57
Security Supervisor	0.0	0.0	0.57	0.57
Security Shift Supervisor[d]	0.0	0.0	2.85	2.85
Security Patrolman[d]	0.0	0.0	9.12	9.12
Contracts and Accounting Specialist	0.2	1.0	0.73[b]	1.93
Clerk	0.0	1.0	0.73[b]	1.73
Health and Safety Supervisor	0.5	1.0	0.73[b]	2.23
Health Physicist	0.0	0.5	0.57	1.07
Protective Equipment Attendant	0.0	0.0	1.14	1.14
Industial Safety Specialist	0.2	1.0	0.57	1.77
Quality Assurance Supervisor	0.2	1.0	0.73[b]	1.93
Quality Assurance Engineer	0.3	1.0	0.57	1.87
Quality Assurance Technician	0.0	0.5	1.14	1.64
Consultants (Safety Review Committee)	0.3	0.5	0.5	1.3
Subtotal	3.9	14.5	25.56	43.96
Decommissioning Workers[e]				
Shift Engineer	0.5	2.0	1.14	3.64
Crew Leader	0.0	1.0	2.85	3.85
Utility Operator	0.0	2.0	3.42	5.42
Laborer	0.0	0.0	2.85	2.85
Craft Supervisor	0.0	0.5	1.14	1.64
Craftsman	0.0	2.5	5.7	8.2
Senior Health Physics Technician	0.0	1.0	1.14	2.14
Health Physics Technician	0.0	1.5	2.85	4.35
Subtotal	0.5	10.5	21.09	32.09
Totals	4.4	25.0	46.65	76.05

(a) Rounded to next higher 0.01 man-year.
(b) Includes an additional 2 months following active decommissioning in order to complete the documentation and other unspecified license and/or contract termination requirements.
(c) Based on one operator per shift in the test reactor control room, three shifts per day, 7 days per week.
(d) Based on 10 CFR Part 73; includes both response and access-control personnel on a three shift, 7-day-week basis.
(e) Requirements during the 6-3/4 months following reactor shutdown are based on a relatively constant manpower loading utilizing the staff breakdown given in Table J.2-3.

determined. Next, the task time requirements and the numbers and types of decommissioning workers required to accomplish each task in the allotted time are estimated. The organization of the decommissioning staff is the same as that for DECON shown in Figure H.2-2 in Appendix H (the functions and organization of the decommissioning staff for the conceptual decommissioning of the reference test reactor via the SAFSTOR alternative are discussed in Section H.2 of Appendix H. Based on the estimated dose to accomplish each task, the number of workers needed to complete the radiation-zone work in the alloted time and within the assumed radiation dose limits is determined. Whole-body radiation doses to the decommissioning workers are limited in accordance with 10 CFR 20.101. The supervisors, utility operators, and health physics technicians are assumed to be long-time radiation workers whose annual exposure is limited to 5 rem/yr by the formula 5(N-18) of 10 CFR 20.101(b)(2). The craftsmen and laborers are assumed to have had little prior radiation exposure and, therefore, under 10 CFR 20.101(b)(1) and (2) may receive up to 3 rem per quarter, within the limitation of the formula 5(N-18). If a situation occurs where the manpower estimated for physically accomplishing a task results in an average dose for a person in excess of these limits, an additional person is anticipated to be assigned to the task to keep the individual occupational dose below these limits. In the manpower table presented in this section, the manpower shown is adequate both to accomplish the task and to meet the occupational dose limits.

Task schedules and decommissioning worker requirements for preparing the Reactor Building/MUR/Containment Vessel, the Hot Laboratory Building, and the other contaminated structures and areas for safe storage are discussed in the following subsections. The decommissioning worker requirements given in these subsections do not include the management and support staff (see Figure H.2-2 in Appendix H).

J.2.2.1 Reactor Building, MUR, and Containment Vessel Activities

All of the neutron-activated materials and many of the radioactively contaminated areas are located in the containment vessel (CV). For this reason, several task steps are taken that are designed to improve the isolation of this

significant substructure within the Reactor Building. These steps involve sealing all major accessways into the CV, as well as providing barriers around any high-radiation-level piping areas just outside of the CV.

At reactor shutdown, the operations proceed in the usual manner for discharging the existing fuel load. The fuel is stored and cooled as necessary and then transferred to fuel shipment casks for shipment to a reprocessor. The fuel shipment operations maintain an inventory of the fuel material for accountability purposes. Truck shipments convey the nuclear fuel to the appropriate reprocessor (see Section I.2.3.9 of Appendix I for details). Components of the reactor vessel that may have been removed during refueling are returned to their normal positions and secured as normally provided for.

Fluids from the primary cooling water system (PCWS) as well as other systems described earlier in Section J.2.1.1 are used repeatedly for flushing and cleaning purposes (with IX cleanup in between uses), and then sequenced into the existing radwaste systems for final cleanup.

Based on the methods discussed previously in Section J.2.1 and the occupational dose estimates presented later in Section J.2.6, the task schedule and sequence and the decommissioning worker requirements for the RB/CV (including the MUR) are developed as presented in Figure J.2-1. The basic MUR core structure and experimental facilities which are part of the MUR structure are stored dry in Canal H. Fuel storage baskets are left in Canals G and H. Canals F, G, and H are drained; the canals' walls, floors, and storage baskets are decontaminated to the maximum extent practical and the canals are left dry. All ladders and other means of access to the canal floor levels are removed and/or deactivated. Fencing or other means of barricading is provided to prevent access to radiation areas >2-1.2 mr/hr. The fence is locked and the appropriate radiation area signs are posted. Other radioactive equipment and/or materials of an experimental nature are assumed to have been removed from the Reactor Building and disposed of before the start of active decommissioning.

All loose items and selected items of fixed equipment in the quadrants, canals, and dry annulus are disposed of by one of the methods identified

earlier in Section J.2.1.1. Sumps are cleaned to lowest practical levels and sealed. Valves within the CV are closed in preparation for the structure's eventual isolation.

The task requirements within the CV and outside the CV (i.e., for the Reactor Building) are listed in Tables J.2-5 and J.2-6, respectively, together with their status for the safe storage period. In general, all equipment, gratings, ladders, stairs, HVAC, electrical, concrete, and other miscellaneous items requiring decontamination are cleaned to lowest practical levels. Long-lasting protective paint coatings are applied to selected cleaned areas to immobilize any residual radioactivity. Nonessential utilities are deactivated, including fans, electrical units, motors, cranes, and pumps. All piping is drained. The various areas of the building are isolated and sealed, where appropriate, by closing all piping valves, sealing interconnecting HVAC air passageways, and sealing rooms containing significant quantities of radiation (e.g., the sub-pile room).

The containment vessel ventilation system (CVVS) compressors are cleaned internally and preserved. The water jackets are drained, cleaned and preserved. The bearings and drive components are cleaned and preserved. The lubricating oil is drained and the system is flushed with new oil. The after-cooler is drained. The CVVS storage tanks are drained and the manual inlet, outlet, and drain valves are left closed.

No more than two CV penetrations are left open, through a filter, to permit free breathing of the CV.

Waste materials are packaged and shipped to the appropriate disposal sites. The RB is locked.

Placing the RB/CV and MUR in safe storage is estimated to be accomplished in about 5-1/2 months following final reactor shutdown and is estimated to require about 63 man-months of decommissioning worker effort.

J.2.2.2 Hot Laboratory Building Activities

An estimate of the main preparatory activities in the HLB is given in Table I.2-3 in Appendix I. These same preparatory activities are postulated

TABLE J.2-5. SAFSTOR Tasks Within the Containment Vessel[a]

Task or Item	Tasks and/or Status During Safe Storage
Airlock Personnel Doors;[b] Airlock Experiment Transfer Doors;[b] Double-Gasketed Truck Door; Singly Gasketed Canal E/F Door	Closed and sealed and/or locked in place after the interior areas are decontaminated. All penetrations from CV to the outside are plugged or capped.
Personnel and Experiment Transfer; Double Airlock Door Electrical; Interlock Systems and Vacuum/Pressure Systems	Deenergized.
Penetration Monitoring System; Remote Area Monitoring System	Deenergized.
All Plumbing Services, Liquid or Steam	Drained and vented to atmospheric pressure. Lines are broken outside the CV and the CV-side of the broken lines are either blank-flanged off with two nuts welded or capped off and welded.
Gas	Vented to atmospheric pressure.
Isolation Valves (outside the CV)	Closed and secured, unless the system can be isolated inside the CV.
Sumps and Sump Pumps	Cleaned to lowest practical levels with the sump pumps deactivated.
Unused CV Penetrations	Blank-flanged from inside the CV or two bolts welded if flanged from the outside.
Quadrants A, B, C, D, and Canal E	Drained, cleaned, and left dry. All ladders and other means of conveyances which provide personnel access to the quadrant and/or Canal E floors are removed, deactivated and barricaded. Fencing or other means of barricading is provided to prevent personnel access to radiation areas of >2-1/2 mr/hr.

TABLE J.2-5. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Sub-Pile Room	Decontaminated to lowest practical level, closed, and padlocked.
CV Elevator	Power is deactivated; elevator is left at top 0-m level, door is padlocked closed.
Canal E-F Door	Door is closed and dogged and power is turned off and locked out at the control panel.
Quad-Canal Doors (A-E and C-E)	Doors are closed per procedure.
Quad C and D Hot Cave	All radioactive materials, tools, and equipment are removed; the interior is decontaminated to the lowest practical level and cave doors and plugs are welded closed. All lines servicing the Hot Cave are drained and vented and the periscope and manipulators are secured from moving. The ventilation filter is replaced and all electrical services are disconnected. The exterior surfaces of the Hot Cave are decontaminated to lowest practical levels.
CV Dry Air System; Drain CV Helium and Argon Gas Systems	Shut down per existing procedures. Drain and vent valves are opened. All cooling water lines are flushed and drained. All electrical services are deenergized. Isolation valves are secured in a closed position.
CV Air Conditioning	Lines into CV are drained and isolated. Filters are removed and electrical supply is deenergized. Cooling coil is drained.
Normal Access Areas of the CV	Decontaminated to a clean area such that protective clothing and dosimeters are not required.
Electrical Systems	Shut down and deenergized inside and outside except for selected alarm systems and lighting.

TABLE J.2-5. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Heating Systems	Shut down and deactivated. Standby procedures will provide for heating and ventilation by an external system if required.
CV Pressure Equalization	Equalization of normal variations in atmospheric pressure and CV venting is through absolute filters. Inspection of filters and changeouts is assumed to be covered under safe storage procedures.
Cathodic Protection	Cathodic protection for the CV is postulated to remain operative and monitored under safe storage procedures.
Reactor Vessel (RV)	After all fuel elements, fueled experiments, and fuel-bearing components are removed from the RV, the RV is drained and flushed with fresh deionized water along with the PCWS. The RV is purged with air and isolated and leak-tested. Then, the RV is isolated with shielding and/or barriers to limit direct radiation levels at boundary points to ≤100 mrad/hr. Experiment facilities with built-in radiological shielding and/or shield plugs are installed in all horizontal through-hole ports, horizontal beam ports, and instrument thimble ports to limit radiation streaming into the quadrants. Pertinent equipment located on the RV dome and in the annulus surrounding the RV that may deteriorate due to the radiation environment is removed and all RV penetrations are closed. The RV hatch is installed, dogged, and padlocked.

TABLE J.2-5. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Reactor Vessel (contd)	Shrapnel shields (see Figure C.1-3 in Appendix C) are installed to provide radiological shielding from the reactor core with the lifting eye holes plugged and tack-welded. Additional shielding and/or appropriate barriers may be installed to either reduce radiation levels from the RV area and/or to limit access to these areas.
	Ion chambers are removed from the four thimbles in Quadrant D that contain them. All nine instrument test holes are shielded with new plugs and welded in place.
	Helium pressurization system (in-tank) piping is depressurized and all lines are plugged or capped. The associated electrical system is deenergized.
	HB Test Holes:[c]
	HB-1 thimble is left in place and a new shield plug is installed and welded in place; the gate valve is blocked open.
	HB-2 gate valve is blocked open and a shielding plug is installed and welded in place.
	HB-3 collimator is left in the hole. An external shield plug is installed and welded in place.
	HB-4, -5, -6 (thermal column) is loaded with white iron shot (5,500 kg) for shielding.

TABLE J.2-5. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Reactor Vessel (contd)	HT Test Holes: Existing plugs are installed and welded in place. VT Test Holes: VT-1 and VT-2 remain in place in the RV. Piping is flushed with deionized water, drained, and isolated using blank flanges where needed. Hydraulic Rabbit Tubes: Water is drained from the tubes. Shield plugs are fabricated and installed in the upper RV penetrations. The lower penetration in Quad A is shielded externally. Access to the lower penetration in Quad C is barricaded. All shields and barricades are welded in place.

(a) It is assumed that the CV is a closed, restricted area under lock-and-key control for the duration of the safe storage period.
(b) In addition, the outer or inner airlock door of one of the personnel airlocks and the inner door of the experiment transfer airlock are barred from the inside. One of the doors of the remaining personnel airlock is closed, padlocked, and monitored remotely by an intrusion door alarm.
(c) All experimental equipment associated with these systems is contaminated waste material (see Table C.3-1 in Appendix C) and is assumed to be left in place during safe storage.

TABLE J.2-6. SAFSTOR Tasks Within the Reactor Building

Task or Item	Tasks and/or Status During Safe Storage
Reactor Building (RB), General (outside of the CV)	The RB (with the exceptions of Canals F, G, & H) is decontaminated to unrestricted release levels. All equipment is shut down and/or deactivated and all piping is drained.
	All fuel elements (both irradiated and non-irradiated), all fueled-experiments, and all fuel-bearing components are removed from the RB.
	MUR: the basic MUR core structure and experiment facilities which are part of the MUR structure are decontaminated to lowest practical levels and stored dry in Canal H along with storage baskets. They are shielded as required to <100 mrad/hr and the shields are secured against inadvertent removal.
	The Canal H recirculation system is drained, the ion exchange beds are emptied, and the water supply and return lines are blank-flanged or capped.
	All electrical sources are deenergized and disconnected. The air conditioner is secured. The Canal H drain valve is secured and the drain is covered.
	All liquid process lines which enter the Canal H area are blank-flanged or capped. The access to the Canal H pool-site fence (described below) is locked and the appropriate radiation area signs are posted.
Helium and Argon Gas Systems (outside the CV)	The systems are depressurized and vented to atmospheric pressure. Argon purifiers are vented and capped; lines are capped. All electrical services are deenergized. Gas bottles are disposed of per existing procedures.
Quadrants and Canals (Q&C), General	The filter elements and deionizer resins are removed from the Q&C recirculating

TABLE J.2-6. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Quadrants and Canals (Q&C), General	system and disposed of as radioactive waste. The recirculating system is flushed and drained and the resin pits (see Figure C.2-9) are cleaned and sealed closed. Filters are removed from the Q&C drain system and disposed of as radioactive waste. The drain system is flushed and drained. All exposed external surfaces of the recirculating and drain system pipes, pumps, filter housings, deionizers, and associated equipment and all floor areas around the equipment are decontaminated. The pumps are deactivated and all inlet and outlet lines associated with these systems which pass through the CV are opened and blank-flanged off to effect isolation of the CV. The quadrant cooling water lines are flushed and drained, the Y-strainers are cleaned and all pumps are deactivated. The quadrant cooling supply and return headers are cut and blank-flanged off to effect isolation of the CV.
Canals F, G, and H	The fuel storage baskets are left in Canals G and H. Each canal is drained and cleaned simultaneously. The canal walls, floors, and storage baskets are decontaminated to the maximum extent practical and the canals are left dry. All ladders and other means of access to the canal floor levels are removed and/or deactivated. Fencing or other means of barricading is provided to prevent access to radiation areas >2-1/2 mr/hr. All other radioactive equipment and/or materials are postulated to be removed from the RB and disposed of.

to be accomplished for SAFSTOR and are not repeated here. Additional SAFSTOR tasks for the HLB are listed in Table J.2-7, together with their status for the safe storage period.

The hot cells are decontaminated using remote acid rinsing, jet-cleaning equipment, and other physical cleaning methods described in Section G.4.2 of Appendix G. A conceptual generalized approach to decontamination of the hot cells is presented in Section I.2.1.1 of Appendix I and is not repeated here.

Because of the relatively large radiation exposure potential estimated for hot cell cleanup, specialized hot cell training is anticipated for the decommissioning workers. This training is conducted by a small cadre of hot cell experts who remain after final shutdown. The training includes specifics on hot cell entry with double coveralls, forced-air masks and their proper application, and in-cell work programs and "dry runs" for efficiency and speed (ALARA considerations).

The task schedule and sequence and the decommissioning worker requirements are given in Figure J.2-1. Preparing the Hot Laboratory Building for safe storage is estimated to be completed in 3-1/2 months following final reactor shutdown and is estimated to require about 21-1/2 man-months of decommissioning worker effort.

J.2.2.3 Other Contaminated Structures and Areas Activities

The remaining structures and areas are prepared for safe storage concurrent with preparing the RB/CV/MUR and the HLB. These structures and areas include:

- the Primary Pump House
- the Office and Laboratory Building
- the Emergency Retention Basin and Site Ditches
- the Cold Retention Area
- the Hot Retention Area
- the Fan House
- the Waste Handling Building
- Miscellaneous: the Utility Tunnels and Stack.

TABLE J.2-7. SAFSTOR Tasks Within the Hot Laboratory Building[a]

Task or Item	Tasks and/or Status During Safe Storage
Off-Gas Cleanup & Storage System (OGCUS)	All radioactive gas is assumed to be purged from the system before final reactor shutdown.
	The OGCUS scrubber is drained and flushed with clean deionized water. All water drains are left open.
	The OGCUS is left intact with the containment dome and shielding plugs in place. All mechanical equipment, instrumentation, and controls are deenergized. All valves are left closed.
HLB Hot Drain System	The hot sumps, lines, and drains are cleaned, deactivated, and isolated (plugged and/or sealed) to prevent accumulation or escape of contaminated liquids.
	Sinks, floor drains, and traps are cleaned and flushed.
	All accessible external surfaces of the hot sumps are decontaminated to the lowest practical level.
Canals J and K	The canals are drained and the drains are plugged. The canal walls and floors are decontaminated to the maximum extent practical and the canals are left dry. Fencing or other means of barricading is provided to prevent access, and the barricades are posted as radiation zones and padlocked-closed.
76-Mg Lead Door	Doors are locked and the power-secured.
Hot Cells	All cells are decontaminated to the lowest practical level. All electrical services except lighting are deenergized. All flammables are removed from the cells. Cell doors are closed and secured with chains and padlocks. Cell air system filter housings are flushed; filters are removed; and the cell inlets and outlets are covered with sheetmetal and sealed. Tools are decontaminated and stored either in cells or above the hot dry storage or removed for reuse elsewhere. Periscopes and manipulators are decontaminated

TABLE J.2-7. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Hot Cells (contd)	per procedure. The outcell end is bagged with disiccant. All electrical leads are disconnected.
	General Mills manipulators (see Figure C.3-9 in Appendix C) are wiped down and the electrical leads are disconnected.
	Hot cell barrier doors are closed and electrical power is deenergized. The Cell 1 cut-off port door is closed and secured with a lock and chain; the power is deenergized.
	In general, equipment is covered and left in the cells with the power leads disconnected.
	The tops of the cells are decontaminated to the lowest practical level.
CO_2 Fire Control System	The CO_2 bottles and electrical squibs are removed and decontaminated. All nozzle and valve openings are taped shut.
Hot Dry Storage	All waste is removed and shipped. Any remaining radioactive material is logged and identified for future use. All power is secured.
Controlled Work Area (Room 16)	This area is decontaminated by using the physical cleaning methods described in G.4.2 of Appendix G. Power is deenergized.
Hot Handling Room (Room 17)	All waste is removed and shipped and the area is decontaminated to lowest practical levels.
Repair Shop (Room 22)	Tools are removed and disposed of. The welding hood is decontaminated and the filters are disposed of as radioactive waste. Air systems are blanked-off.
Decon. Room (Room 23)	The room is decontaminated, the decon-pump is secured, power is deenergized, and the doors are padlocked.
Storage Room (Room 24)	The impact machine is decontaminated and disposed of.

(a) Information derived from Reference 5.

The liquid and solid radwaste systems located in the Fan House and the Waste Handling Building are needed to process most of the contaminated liquids contained in the systems at final reactor shutdown and generated during cleanup activities. In addition, continuous waste air-handling service to the remaining work areas is maintained as long as necessary.

SAFSTOR tasks for the remaining contaminated structures and areas of the reference test reactor are summarized in Table J.2-8, together with their status for the safe storage period.

Contaminated Air Systems. After the areas served by the various contaminated air systems have been cleaned and work in the areas is completed, the air systems are shut down. All roughing filters, prefilters, and high-efficiency (absolute) filters are removed and disposed of by standard procedures. The filter housings and accessible piping are vacuumed and washed down as appropriate. New filters are installed in the system and clean, fresh air is circulated through the systems with the fans operating. The systems are then shut down and opened for inspection. The purge continues as necessary to ensure all loose contamination is collected by the filters; the systems are then shut down.

The contaminated air systems are left in a condition that minimizes convective air movement in the piping systems between interconnecting systems and/or areas, except that the vent systems servicing the HRA and the CV breathing system are left open through to the stack. New absolute filters are installed in these systems for the safe storage period. The fan motors are deenergized and breakers are secured. All filters are removed and housings are cleaned. Except for the HRA vent and CV breathing system, all manually operated valves and dampers are blocked-closed and all air-operated valves and dampers are fixed in the closed position. Where applicable, piping and ducts are blocked at their inlets and/or outlets. The service air, electronic, and electrical controls associated with the systems are deenergized and left in place. All air-control lines are vented to atmospheric pressure.

The task schedule and sequence and the decommissioning worker requirements are given in Figure J.2-1. Work on these remaining structures and areas starts essentially at final reactor shutdown with the draining of the ERB and

TABLE J.2-8. SAFSTOR Tasks for the Other Buildings/Areas of the Reference Test Reactor

Task or Item	Tasks and/or Status During Safe Storage
Primary Pump House (PPH)	Black iron piping and structures that are rusting are cleaned and painted. The roof plugs are closed, sealed against water leakage, and locked in place. The crain hoists are left in place and locked.
	All sampling and instrument service lines from the primary cooling water system to PPH Room #8 are flushed and drained. The sampling hood and sinks and area in general are decontaminated and left in a clean zone. All electrical equipment and instrumentation is deactivated.
	All high-pressure gas bottles are disconnected; gas lines are vented, and gas bottles are removed from the building.
	The fuel element test loop is drained and valves are closed after draining is complete.
	All floor hot drains are plugged and the hot sumps are deactivated, cleaned, and sealed off.
	All outside doors are closed and locked.
Office and Laboratory Building (OLB)	All radioactive material and SNM are disposed of.
	Filters are removed from the hood filter housings; the housings are decontaminated, and all roof openings are covered.
	All hot drains are flushed and sealed.
	Unneeded utility lines and electrical leads are disconnected and tagged.
	All hazardous, flammable materials and chemicals are removed.

TABLE J.2-8. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Office and Laboratory Building (OLB) (contd)	Physics (MUR) Counting Room - all radioactive dosimetry materials are discarded. Unneeded utilities are disconnected and tagged. Vault: 1) verification that all fueled materials are removed; 2) the vault area and all tools are decontaminated; 3) the criticality alarm reference source is removed and discarded and the alarm source drive equipment is decontaminated. Sumps in the basement and the utility tunnel to the HLB are decontaminated to lowest practical level. The pipe tunnel which leads from the basement of the HLB to the basement of the OLB is barricaded with a poured concrete wall to prevent access to the HLB. This wall extends all the way to the ceiling of the tunnel, is bonded to the floor and walls of the tunnel, and is watertight up to the pipe penetrations to prevent flooding. The wall is erected near the OLB end of the tunnel. All process piping passing through the tunnel is cut (or disconnected) and blank-flanged off and/or capped. The cuts or disconnects are made on the OLB side of the tunnel barrier wall so that the flanges and/or caps are accessible for periodic inspection.
Emergency Retention Basin (ERB)	The ERB is drained and the drain valve is left locked open. The trench manhole is left closed. Contaminated soil is removed, packaged, and shipped to a shallow-land burial facility.
Cold Retention Area (CRA)	The CRA tanks are decontaminated sufficiently to ensure that any effluent from the tanks would not exceed 1×10^{-7} uCI/mℓ. The tanks are left partially filled with water and are left in a "free flooding" condition, i.e., with the equalizer lines to ground-water

TABLE J.2-8. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Cold Retention Area (CRA) (contd)	open. Hatches to the CRA tanks and valve pit areas are closed and locked. The CRA ground-water sump pumps and level alarms are deactivated and pump motors are disconnected and removed to inside storage.
Hot Retention Area (HRA)	The exterior of the hot retention tank area is decontaminated and left a clean zone. Each of the HRA tanks 1 through 8 and the combined hold tanks are flushed and drained. The tanks are then cleaned and left dry. The pumps are deactivated and all valves are closed. All access plugs and openings are closed and secured against unauthorized entry except one entrance into the tunnel area, which is locked closed. The inlet louvers to the HRA pipe tunnel are blanked-off and the HRA is allowed to breathe via the vent system to the stack. Absolute filters are retained in the HRA vent system to remove particulate airborne contamination. The HRA ground-water sump pumps remain in service with water level alarm monitors active.
Waste Handling Building (WHB)	The WHB is left a white zone[a] as far as practical. All waste material is removed. All air and water filters are removed from housings and water and air systems are drained. The boiler is deactivated, flushed and drained.[b] The WHB evaporator is completely flushed and drained and all external surfaces of the evaporator and pit area are decontaminated as practical. The access doors to the evaporator room (located in the WHB basement) are closed and locked. The O'-m area around the evaporator is fenced. All hot floor drains are flushed and sealed closed, converted to cold drains,

TABLE J.2-8. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Waste Handling Building (WHB) (contd)	and all hot sumps are deactivated, cleaned and sealed closed. The cold sumps that collect storm water or water from cold floor drains shall remain in service with their water level alarm monitors active. The office air-conditioning system is de-energized, drained and placed in standby. No heat is provided to the building. All outside doors to the Waste Handling Building are closed and locked.
Fan House	The Fan House is left a white zone[a] except for areas in the hot pipe tunnel, waste treatment room, and specific areas that cannot be decontaminated and/or are subject to radioactivity from another area. All waste material is removed. All filters (air and water) and deionizer resins are removed from housings and deionizers except as noted in Section J.2.2.3. All water systems are drained. All air systems are depressurized and the systems drained. The hot pipe tunnel entrances are padlocked at each end (Fan House basement and Office and Laboratory Building). The hot pipe tunnel is decontaminated to the extent practical. The waste cleanup room (first floor and basement) is decontaminated to lowest radiation level practical. The entrance to each room is padlocked if required due to fixed radiation levels. The waste cleanup system deionizer resins and filter elements are removed and disposed of as radioactive waste. The system is flushed with fresh clean water. External surfaces of the waste cleanup system equipment and floor areas in the Fan House shielded cubicles are decontaminated as practical and the cubicle gate at the -3.7-m level is closed and locked.

TABLE J.2-8. (contd)

Task or Item	Tasks and/or Status During Safe Storage
Fan House (contd)	All cold-drain sumps that collect storm water or floor drains remain energized and accessible. Alarm monitors are provided for the sumps. All hot drain sumps are deactivated, cleaned, and flushed with clean water and left dry and sealed. Water which cannot be prevented from leaking into contaminated areas is prevented from escaping. The Fan House Building is locked.
Miscellaneous: Overhead Cranes and Hoists	The seven-point procedure given below is postulated to be used to prepare overhead cranes and hoists for safe storage: 1. Drain all gear boxes and replenish oil with preservative-type oil. 2. Operate several minutes to properly distribute the preservative. 3. Clean all shafts and coat with preservative. 4. Raise hook to extreme top portion. 5. Coat all cables and drum with preservative. 6. Cover drum assembly and cables with visqueen (except in contaminated areas). 7. Secure power to crane or hoist.

(a) In this study, a white zone means: 1) inside the reference test reactor fence area; 2) direct radiation (mrad/hr) <0.5 where practical and <2.5 otherwise; transferable contamination (dpm/100 cm^2) at 10 α, 1,000 $\beta\alpha$; fixed contamination (dpm 500 α and 1,500 $\beta\alpha$; and 3) no posting required, no health physicist required to enter, no film badge needed, and no protective clothing required to enter.
(b) A nonradiological consideration; included for completeness.

continues for 6-3/4 months to the end of the decommissioning project. Approximately 54 man-months of decommissioning worker effort are estimated to be required.

J.2.2.4 Site Survey

The site surfaces outside of the shaded areas anticipated to require decontamination (shown in Figure C.1-1 in Appendix C) but within the site perimeter fence are surveyed and assumed to be releasable for unrestricted use without further effort. However, it is recognized that extensive radiation surveys and analyses of soil surfaces and paved areas are required to demonstrate that these portions of the site are releasable. Determination of the level of effort actually necessary for potential cleanup of these portions of the site not already discussed and accounted for (see Section C.4 of Appendix C) is beyond the scope of this study because of the wide range of possible conditions. However, for this study, these activities are assumed to be accomplished by the health physics staff on a fill-in basis during the preparations for safe storage.

J.2.3 Staff, Activities, and Costs During Safe Storage

This section contains the postulated staff, activities, and estimated costs during the safe storage period for the reference test reactor.

J.2.3.1 Postulated Staff Organization During the Safe Storage Period

The staff organization, shown in Figure J.2-2, is postulated to take control of all surveillance, maintenance, and security activities for the duration of the safe storage period and thus adequately and safely satisfy anticipated safe storage requirements.

J.2.3.2 Security Costs During the Safe Storage Period

The information contained in this subsection forms the bases for the estimated security costs during the safe storage period. It is assumed that at the start of the safe storage period the guard force is disbanded and replaced by private security forces and electronic intrusion systems.

The costs of a private security agency using remote surveillance equipment are given in Table J.2-9. Consideration is given to the following parameters:

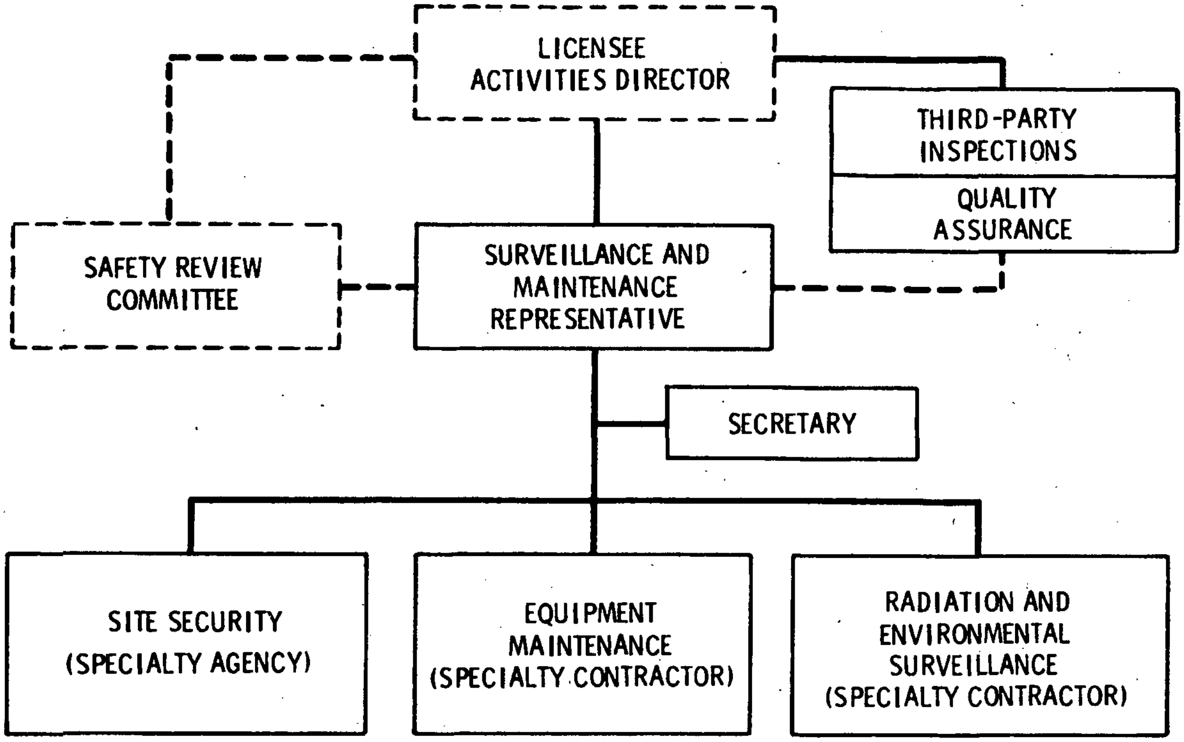

FIGURE J.2-2. Postulated Staff Organization for the Safe Storage Period

- renovation and utilization of existing site security and surveillance-related equipment
- possible relocation of the perimeter fence to include only those buildings housing activated and/or contaminated equipment areas
- addition of special equipment as indicated by the security agency's site survey
- installation of remote readout facilities either onsite or offsite.

Thus, security is provided during safe storage by several methods. Locks on gates around the decommissioned facility provide the first level of security. The fence is maintained in good condition throughout the safe storage period. The locks and secured entryways of the structures and the remote surveillance equipment and alarm system provide the final level of security.

It is arranged with the security agency to assure that the surveillance and maintenance (S&M) representative, who is thoroughly knowledgeable about

TABLE J.2-9. Estimated Security Costs During Safe Storage for the Reference Test Reactor[a,b]

Activities	Cost ($ thousands)
Site Security Survey	2.6[c]
Reduce Circumference of Existing Security Fence	13.0[c]
Utilize Portions of Existing Security Equipment; Add Others	52.0[d]
Install Remote Readout at Offsite Location	32.5[c,e]
Lease of Equipment, Maintenance, Monitoring, Response (includes temporary maintenance when required), Patrol (unscheduled, any hour), and Supervision	11.5
Total (first year)	111.6
Total/year (after first year)	11.5

(a) Includes onsite detection and notification systems for: security (intrusion), fire, and radiation levels.
(b) Based on data (1978) taken from Table H.4-3 of Reference 1 and adjusted for inflation to mid-1981 dollars.
(c) One-time cost only.
(d) One-time cost only; existing equipment allowances would be a contractural matter open to negotiation between the facility owner and the security agency. This dollar value includes installation of devices and hardware not normally found at the site. Such items include audible alarms, 24-hour video tape recorder, multiplex systems (self trouble-shooting), battery backup power where required, and capacitance perimeter fence protection.
(e) Includes installation charges for security agency hardware, which are primarily for the offsite installation costs of "slave" annunciator drops from existing plant security equipment.

the facility, can be contacted at any time. The S&M representative is designated by the licensee and is responsible for controlling authorized access into and movement within the facility. He is further charged with the

responsibilities of appropriate actions and notifications regarding breaches of security, upkeep of plant surveillance and maintenance programs, and administrative reporting of these events as required by state and federal regulations. A postulated administrative reporting process and security notification chain-of-command is shown in Figure J.2-3, delineating responsibilities for reportable events. It is imperative that such a notification and reporting process be kept current throughout the safe storage period.

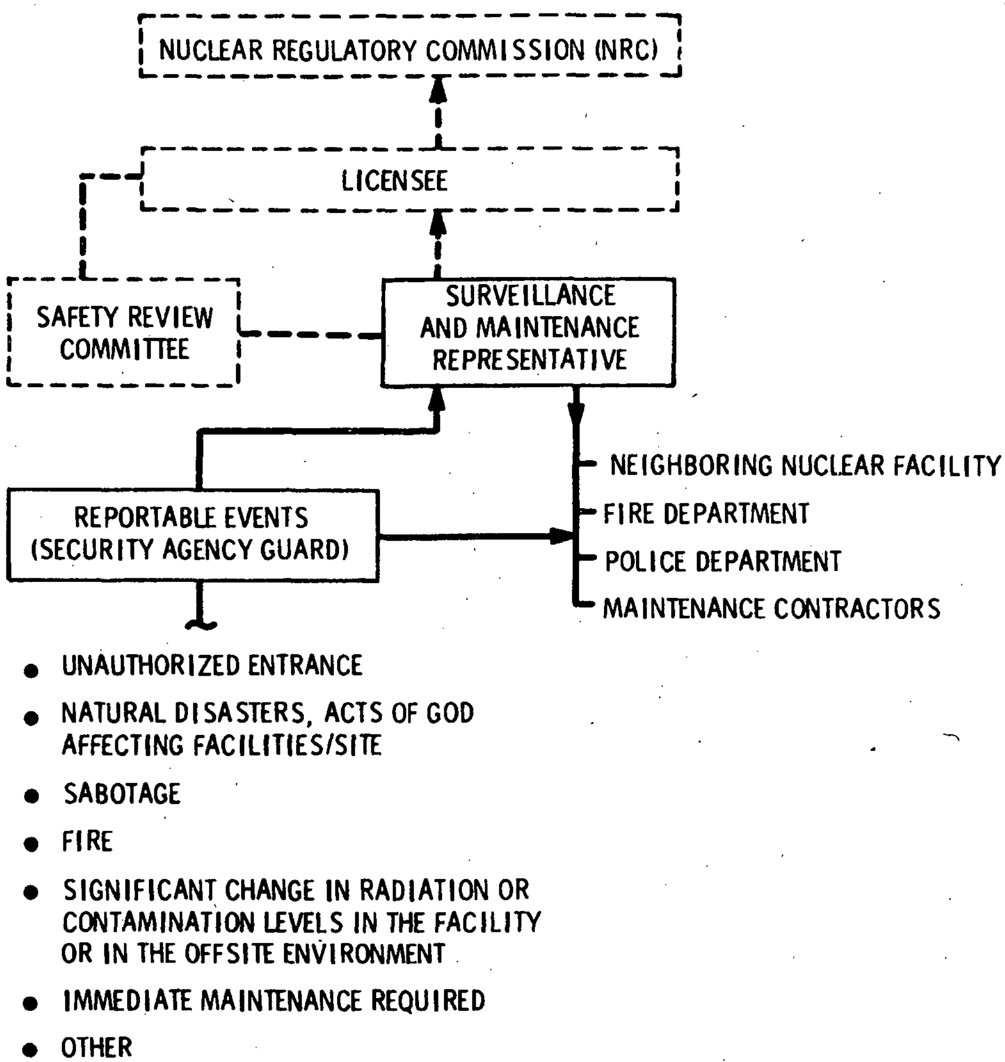

FIGURE J.2-3. Postulated Security and Administrative Notification and Reporting Process

J.2.3.3 Postulated Continuing Maintenance Program for the Reference Test Reactor Facility

A planned maintenance program was developed for the Plum Brook Reactor Facility (PBRF).[6] The program for PBRF was based on a visual inspection of the PBRF buildings and a review of construction photographs by a registered architect. The planned maintenance program was developed for decommissioning alternatives (i.e., safe storage and entombment) involving a 100-year delayed dismantling phase specifically analyzed for PBRF. A modified version of this program, with adjustments for inflation and the safe storage conditions described previously in this appendix for the reference test reactor, is presented in Table J.2-10. It can be seen from the table that those buildings having radiological significance (i.e., license implications) are included in the 100-year planned maintenance program for the reference test reactor facility used for this study. In addition, estimated costs of miscellaneous tasks (e.g., cathodic protection and emergency maintenance) are included for completeness.

J.2.4 Staff and Activities Required During Deferred Decontamination

It is possible that sometime after the reference test reactor is put in safe storage the owner will want to eliminate the continuing responsibilities and terminate the amended nuclear license. An amended nuclear license may be terminated (in accordance with present regulations) when the facility is sufficiently decontaminated to permit unrestricted release of the property. Depending on the length of the storage period, deferred decontamination may be required to permit termination of the nuclear license.

The same basic activities necessary for DECON are also assumed for deferred decontamination following safe storage. Neutron-activated materials in the reactor vessel internals and the reactor vessel still require removal and segmenting remotely under water, using the same techniques that are assumed for DECON.

The radioactive contamination of the piping systems, equipment, Q&C's floor and wall surfaces, etc., is primarily ^{60}Co. Thus, for safe storage periods of less than 50 years ($\sim$10 half-lives for ^{60}Co), it is assumed the

TABLE J.2-10. Postulated Maintenance Program for 100 Years of Safe Storage of the Reference Test Reactor[(a)]

Item	Cost Base ($)	Scheduled Maintenance (years) 0	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	90	95	100	Cumulative Total ($)
Reactor Building																							
Replace Roof	56 550	X				X				X				X				X					282 750
Top Dress Roof	7 540		X	X	X		X	X	X		X	X	X		X	X	X		X	X	X		113 100
Dome Roof Replacement	62 530	X																					62 530
Top Dress Dome	8 450	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		169 000
Refasten Siding	520	X		X		X		X		X		X		X		X		X		X			5 220
Cap Deep Wells	2 665	X																					2 665
Cathodic Protection	15 600	X		X		X		X		X		X		X		X		X		X			156 000
Painting	6 084	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		120 960
Subtotal																							912 225
Hot Laboratory																							
Roof Repair	4 810	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		96 200
Replace Roof	35 750		X				X				X				X				X				178 750
Fasten Siding	1 690	X		X		X		y		X		X		X		X		X		X			16 900
Repair Locks & Hardware	260/yr																						26 000
Paint Trim	5 980	X																					5 980
Paint Maintenance	1 560/yr																						156 000
Paint Roof Vents, etc.	1 820	X																					1 820
Maintenance on Above	520/yr																						52 000
Subtotal																							533 650
Fan House & Waste Handling Bldg.																							
Repair Roof	1 820	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		36 400
Replace Roof	18 590	X		X		X		X		X		X		X		X		X		X			185 900
Paint Trim & Doors	2 600	X																					2 600
Paint Building	16 640	X		X		X		X		X		X		X		X		X		X			166 400
Subtotal																							391 300
Office and Lab Building																							
Replace Roof	21 125	X				X				X				X				X					105 625
Reseal Roof	3 380		X	X	X		X	X	X		X	X	X		X	X	X		X	X	X		50 700
Caulk Gravel Stop	507		X	X	X		X	X	X		X	X	X		X	X	X		X	X	X		7 605
Caulk Windows and Doors	7 605		X	X	X		X	X	X		X	X	X		X	X	X		X	X	X		114 075
Paint Roof Vents	4 056		X	X	X		X	X	X		X	X	X		X	X	X		X	X	X		60 840
Masonry Painting	2 028		X	X	X		X	X	X		X	X	X		X	X	X		X	X	X		30 420
Subtotal																							369 265
Total for Buildings																							2 206 440
Miscellaneous																							
Grass and Snow Removal	2 600/yr																						260 000
Road Maintenance	1 300/yr																						130 000
Pest Control	1 300/yr																						130 000
Cathodic Protection	15 600	X		X		X		X		X		X		X		X		X		X			156 000
Fence	3 900		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		78 000
Emergency Maintenance	7 800/yr																						780 000
Total for Facility																							1 534 000
Total Maintenance																							3 740 440

(a) Based on information presented in Table F-7 in Reference 6 and adjusted for inflation to mid-1981 dollars.

surfaces remain radioactively contaminated at levels greater than those that permit unrestricted use of the material. It is also assumed that after 50 years of decay, the radioactive contamination on the bulk of the formerly contaminated material has decayed to levels that permit release for either salvage as scrap or disposal as nonradioactive waste.

A number of DECON tasks are accomplished during the preparations for safe storage (i.e., discharging and shipping the fuel; draining of contaminated liquid systems; removing, packaging, and shipping of contaminated soil from the ERB and buried concrete piping from the site ditches; and removal of radioactive wastes, such as filters, resins, and slurries). During deferred decontamination, the time not expended on these tasks is offset by the time spent on familiarization of the work force with the facility, removal of locks and barriers installed to secure the facility, and restoration of essential services that were unneeded during the safe storage period. Therefore, it is assumed that the basic work force and time required for deferred decontamination are the same as for DECON.

J.2.5 Estimated Costs of SAFSTOR

The estimated costs of activities required to place and maintain the reference test reactor facility (including the MUR) in safe storage are presented in this section, together with costs for possible deferred decontamination.

J.2.5.1 Cost of Placing the Reference Test Reactor in Safe Storage

The estimated cost of preparations for safe storage is summarized in Table J.2-11. The total cost of preparations is estimated at approximately $6.7 million. The major contributors to the total cost are staff labor, disposal of radioactive materials, and services of specialty contractors (at approximately 58%, 26%, and 11%, respectively). Combined costs of special tools and equipment, miscellaneous supplies, and energy make up an additional 5.3% of the total. Since the reference test reactor is assumed to be federally owned, nuclear insurance and license fees do not contribute to the costs.

TABLE J.2-11. Summary of Estimated Costs of Placing the Reference Test Reactor in Safe Storage

Cost Category	Estimated Costs ($ millions)[a,b]	Percent of Total[c]
Disposal of Radioactive Materials (Radioactive Wastes[d])	1.384	25.9
Staff Labor	3.096	57.9
Energy	0.021	0.4
Special Tools and Equipment	0.196	3.7
Miscellaneous Supplies	0.065	1.2
Specialty Contractors[e]	0.585	10.9
Nuclear Insurance	--[f]	--
License Fees	--[g]	--
Subtotal	5.347	100.0
Contingency (25%)	1.337	
Total, Preparations for Safe Storage	6.684	
Other Possible Costs		
Spent Fuel Shipment	0.204[h]	
Contingency (25%)	0.051	
Total, Other Possible Costs	0.255	

(a) 1981 costs used.
(b) The number of figures shown is for computational accuracy and does not imply precision to the nearest $1000.
(c) Individually rounded to the nearest 0.1%.
(d) Includes both wet solid wastes and dry solid wastes.
(e) Includes selected demolition, security preparations, and environmental monitoring services.
(f) Indemnity fees are currently $100/yr for each license (i.e., the test reactor license and the MUR license) at the reference test facility and are not included in this study since they represent only a small fraction of 1% of the total decommissioning cost.
(g) Because the reference test reactor is assumed to be federally owned these fees are not applicable; however, where applicable for other nuclear R&T reactor facilities, the schedule of fees for license amendments and other approvals required by the license or NRC regulations is given in 10 CFR 170.
(h) Does not include costs for handling at the reactor or costs for handling and storage at the repository (see Section I.2.3.9 of Appendix I).

The cost for the only other requirement anticipated during the preparations for safe storage (i.e., spent fuel shipment) is the same as for DECON--about $255,000, including a 25% contingency (see Section I.2.3.9 of Appendix I for details).

Detailed cost data for the individual cost categories shown in Table J.2-11 are presented and discussed in the following subsections.

Disposal of Radioactive Materials. Wet solid wastes, dry solid wastes, and contaminated concrete pipe and soil in the reference test reactor facility require disposal during preparations for safe storage. Table J.2-12 contains a breakdown of the disposal costs for the dry solid wastes. The wet solid wastes and the contaminated soil and buried concrete pipe are disposed of as in DECON. The total cost of disposal for all of these materials is about $1.4 million and is approximately 26% of the total cost of preparations for safe storage. The disposal cost includes the container, transportation, and burial costs, but does not include the direct labor costs for removing and packaging these materials. Labor costs are discussed later. The costs of offsite disposal for those materials shipped to a low-level waste burial ground are summarized in Table J.2-13.

Staff Labor. The cost of staff labor during preparations for safe storage is shown in Table J.2-14. Approximately 58% of the total preparations cost is due to the staff labor (presented in Table J.2-11). A total staff labor cost of about $3.1 million is estimated for preparing the reference test reactor facility for safe storage. Specialty contractor labor is not included in this total.

Energy. The cost of energy during the preparations for safe storage is presented in Table J.2-15. The use of electricity and natural gas shown in the table is based on data (1978) supplied in Reference 6, adjusted for inflation to mid-1981, and applied to the time frame estimated for SAFSTOR tasks (i.e., about 6-3/4 months). The total cost of energy is about $21,350 and represents less than 0.5% of the total cost of preparations for safe storage.

Special Tools and Equipment. Based on the information presented in Table J.2-1, the estimated cost of special tools and equipment that are

TABLE J.2-12. Costs of Disposal for Dry Solid Wastes

Component	Number of Disposable Containers(a)	Container Costs ($)(b)	Estimated Number Requiring Shielding	Cask Rental Costs ($)(c)	Shipments Shielded/Unshielded(d)	Transportation Costs ($)(e)	Handling Costs ($)(f)	Burial Volumes (m^3)(g)	Burial Costs ($)(h)	Total Disposal Costs ($)(i)
Exhaust Filters(j)	15	450	0	0	2/1	1 076	0	3.2	1 020	2 546
Misc. Dry Waste	105	3 150	28	4 500		2 530	1 340	22	7 180	18 700
Totals	120	3 600	28	4 500	2/1	3 606	1 340	25.2	8 200	21 246

(a) Based on a 5:1 compaction of estimated waste volumes into standard 0.21-m^3 steel drums; estimated on a taskwise assessment of expected dry solid waste generation rates.
(b) Based on Table M.2-1 in Appendix M.
(c) Based on Table M.3-1; assumes a maximum of seven containers per cask and five days per shipment.
(d) Assumes two cask loads per shipment.
(e) Based on Table M.4-4 for overweight shipments (2) and legal-weight shipments (1).
(f) Based on Table M.5-1, cask handling fee.
(g) Includes the disposable container; rounded to next whole m^3.
(h) Based on Table M.5-1; surface dose rates assumed to be 0.21 to 1.00 R/hr for those drums requiring shielding during shipment, <0.2 R/hr for all others; rounded to the next highest $10.
(i) The number of figures shown is for computational accuracy only and does not imply precision to that many significant figures.
(j) Based on Table I.2-11 in Appendix I and Section C.5.5 in Appendix C.

TABLE J.2-13. Summary of Costs of Offsite Disposal for Radioactive Materials While Placing the Reference Test Reactor in Safe Storage

Material Category	Number of Shipments	Burial Volume (m^3)	Disposal Costs ($ millions)
Contaminated			
Concrete Pipe and Soil	163	2779	1.352[(a)]
Radioactive Wastes			
Dry Solid Wastes	3	25	0.021[(b)]
Wet Solid Wastes	2	17	0.011[(c)]
Totals	168	2821	1.384

(a) Based on Table I.2-11 in Appendix I.
(b) Based on Table J.2-12.
(c) Based on Table I.2-12.

required while preparing the reference test reactor for safe storage is presented in Table J.2-16. The estimated total cost of special tools and equipment is approximately $0.2 million and is approximately 4% of the total cost for preparations.

Miscellaneous Supplies. A variety of supplies are used during the preparations for safe storage. These include expendable glass-fiber and HEPA filters, anticontamination clothing, cleaning and contamination control supplies (chemical agents, sweeping compounds, rags, mops, and plastic bags and sheeting), expendable hand tools, cutting and welding supplies (saw blades, torch gas, and welding rods), and decontamination chemicals, as well as office supplies. The estimated cost of these items is given in Table J.2-17. The total estimated cost of miscellaneous supplies during preparations for safe storage of the reference test reactor is about $0.065 million and represents less than 1.5% of the total preparations cost.

Specialty Contractors. The estimated cost of specialty contractors is given in Table J.2-18. The use of specialty contractors while preparing the reference test reactor for safe storage is similar to that discussed for DECON (see Section I.2.3.6 of Appendix I) with only minor modifications and allowances made to account for the shorter time frame.

TABLE J.2-14 Cost of Staff Labor While Preparing the Reference Test Reactor for Safe Storage

Position	Time Relative to Final Reactor Shutdown (year) -2	-1	1	Total Staff Labor Costs[b] ($ thousands)
	Annual Staff Labor Costs ($ thousands)[a,b]			
Management and Support Staff				
Decommissioning Superintendent	44.5	89.1	65.1	198.7
Secretary	22.1	24.2	17.7	64.0
Clerk	0.0	24.2	13.8	38.0
Decommissioning Engineer	38.0	76.0	55.5	160.5
Assistant Decommissiong Engineer	26.2	52.4	29.9	108.5
Radioactive Shipment Specialist	0.0	39.3	22.4	61.7
Procurement Specialist	7.9	39.3	22.4	69.6
Tool Crib Attendant	0.0	0.0	15.9	15.9
Control Room Operator	0.0	0.0	19.7	19.7
Security Supervisor	0.0	0.0	31.9	31.9
Security Shift Supervisor	0.0	0.0	103.8	103.8
Security Patrolman	0.0	0.0	231.7	231.7
Contracts and Accounting Specialist	9.5	47.1	34.4	91.0
Clerk	0.0	24.2	17.7	41.9
Health and Safety Supervisor	29.9	59.8	43.7	133.4
Health Physicist	0.0	23.4	26.8	50.2
Protective Equipment Attendant	0.0	0.0	31.7	31.7
Industial Safety Specialist	10.5	52.4	29.9	92.8
Quality Assurance Supervisor	10.5	52.2	38.1	100.8
Quality Assurance Engineer	14.1	46.9	26.8	87.8
Quality Assurance Technician	0.0	13.9	31.7	45.6
Consultants (Safety Review Committee)	30.0	50.0	50.0	130.0
Subtotal	243.2	714.4	960.6	1918.2
Decommissioning Workers				
Shift Engineer	26.1	104.4	59.5	190.0
Crew Leader	0.0	44.4	126.6	171.0
Utility Operator	0.0	64.2	109.8	174.0
Laborer	0.0	0.0	88.1	88.1
Craft Supervisor	0.0	23.5	53.3	77.0
Craftsman	0.0	80.3	183.0	263.3
Senior Health Physics Technician	0.0	39.3	44.8	84.1
Health Physics Technician	0.0	45.0	85.5	130.5
Subtotal	26.1	401.1	750.8	1178.0
Totals	269.3	1115.5	1711.4	3096.2

(a) Calculated as the product of the data given in Table J.2-4 and the corresponding data given in Table M.1-1 in Appendix M; rounded to next higher $100.
(b) The number of figures shown is for computational accuracy and does not imply precision to the nearest $100.

TABLE J.2-15. Cost of Energy During the Preparations for Safe Storage for the Reference Test Reactor

Energy Form	Estimated Energy Usage Months	Estimated Average Cost/Month ($)[a]	Costs ($)
Electricity	~7	2 000	14 000
Natural Gas	~7	1 050	7 350
Total Energy Cost			21 350

(a) Based on data (1978) supplied in Reference 6 and adjusted for inflation to mid-1981.

The cost of a hauling contractor is not shown in Table J.2-18, but is included in the disposal costs given in Table J.2-13 for disposal of radioactive wastes.

A specialty contractor, who is responsible for security during the safe storage period, begins work during the preparations period, including making a site-security survey, reducing the size of the security area, and procuring and installing the necessary remote-readout security equipment.

The total cost of specialty contractors during preparations for safe storage, excluding the hauling contractor, is $584,700, which is about 11% of the total cost of preparations.

Nuclear Insurance. Indemnity fees are currently $100/yr for each license (i.e., the test reactor license and the MUR license) at the reference test facility and are not included in this study since they represent only a small fraction of 1% of the total decommissioning cost.

Licensing Fees. The fees charged for licensing services performed by the NRC are delineated in 10 CFR 170.[8] The costs of licensing fees for the federally owned reference test reactor are not included in this study since the federal government does not charge itself for these inspections.

Another Possible Cost. One possible additional cost is the shipment of irradiated reactor fuel to a federal reprocessing plant as described previously

TABLE J.2-16. Costs of Special Tools and Equipment Used While Preparing the Reference Test Reactor for Safe Storage

Item	Estimated Number Required[a]	Estimated Costs ($ thousands)
Portable Oxyacetylene Torch	2	2
Nibbler	2	2[b]
Guillotine Pipe Saw	2	8
Power-Operated Reciprocating Hacksaw	2	1.6
Closed-Circuit, High-Resolution Television	As Required	15[c]
Submersible Pump with Disposable Filter	3	6
High-Pressure Water Jet	2	40
Powered Floor Scrubber	2	0.6
9100-kg Capacity Forklift	1	28
Concrete Drill with HEPA-filtered Dust Collecting System	1	2
Concrete Surface Spaller	1	5
Fron-End Loader (highly maneuverable, light-duty)	1	20
Wet-Dry Vacuum Cleaner (HEPA-filtered)	3	9[d]
Portable Ventilation Enclosure, Filtered	3	18[d]
Filtered Exhaust Fan Unit	2	10
Supplied-Air Plastic Suit	50	2.5
Polyurethane Foam Generator	2	10[e]
Paint Sprayer	2	1.6
HEPA Filter	50	10 total
Roughing Filter	50	5 total
Total Cost		196.3

(a) Based on Table J.2-1.
(b) Remote cutting extensions would add to this cost.
(c) Estimated for modifications of existing systems.
(d) Depends on size and capacity.
(e) Depends on capacity of system.

TABLE J.2-17. Costs of Miscellaneous Supplies Used While Preparing the Reference Test Reactor for Safe Storage

Item[a]	Estimated Costs ($ thousands)
Filters[b]	9 000
Office Supplies	3 000
Graphic Reproduction	5 300
Maintenance and Repair	1 300
Service Supplies	6 300
Protective Clothing[c]	20 200
Electronic Repair Parts	4 900
Lab Supplies	3 500
Decon Supplies	7 000
Radiological Waste Supplies	3 500
Unclassified	1 000
Total	65 000

(a) Based on data (1978) in Reference 6; adjusted for inflation to mid-1981 dollars; plus, proportional adjustments based on the two-shift operation for estimated 6-3/4 month time frame, unless otherwise noted.
(b) Exhaust air-handling unit filters are assumed to be changed out once.
(c) Estimated at four changes per day per decommissioning worker. See Reference 7.

for DECON (see Section I.2.3.9 of Appendix I). The estimated total cost of this task is about $255,000, including a 25% contingency.

J.2.5.2 Costs During Safe Storage

The annual costs during the safe storage period for the reference test reactor are developed in this section, including a planned maintenance program for the postulated 100-year storage period.

The organization shown in Figure J.2-2 is postulated to take over the surveillance, maintenance, and security of the reference test facility for the

TABLE J.2-18. Costs of Specialty Contractors While Preparing the Reference Test Reactor for Safe Storage

Specialty Contractor[a]	Cost Estimating Basis ($/unit)	Estimated Costs ($)
Demolition:[b]		
ERB	---[c]	143 300[d]
Site Ditches	---[c]	254 100[d]
CRA	---[c]	40 300[e]
Environmental Monitoring	50 000/yr[f]	35 000
Security	112 000/job[g]	112 000
Total Cost		584 700

(a) Does not include hauling contractor costs, which are shown as "transport costs" in Table J.2-12 and Table I.2-11 (concrete pipe and soil) and Table I.2-12 (wet solid waste) of Appendix I.
(b) Refers to subcontractor task requirements in Figure J.2-1.
(c) Based on data (1978 dollars) in Reference 3, adjusted for inflation to mid-1981 dollars and proportioned to the estimated volumes of soil and concrete pipe at the reference test reactor presented in Appendix C, Section C.4.
(d) Includes a subcontractor's fee based on 8% of the sum of the subcontractor's charge for manpower, equipment, and materials. (Does not include waste management costs.)
(e) Includes $27,000 to remove the CRA roofs (see Appendix L, Section L.3.2.7).
(f) Based on information presented on page G-37 of Reference 1; adjusted upward to account for inflation to 1981 dollars, and pro-rated for the preparations for safe storage time period (~7 months).
(g) Based on information presented as option C in Table H.4-3 in Reference 1 and adjusted upward to account for inflation to 1981 dollars.

duration of the safe storage period. Estimated annual costs of these services are tabulated in Table J.2-19. The costs are based on information derived from Reference 1 and communications with NASA personnel regarding current shutdown costs. The NASA costs are proportionately adjusted in this study to reflect the storage conditions assumed for the reference test reactor that are different from the Plum Brook Reactor Facility (PBRF). The most significant difference is that the reference test reactor is on a generic site, by itself, whereas the PBRF is on a site shared with other U.S. government agencies who

TABLE J.2-19. Estimated Annual Safe Storage Costs for the Reference Test Reactor[a]

SAFSTOR Item	Estimated Annual Cost ($)[b]
Minor Maintenance Repair:	5 600
Custodial (twice per year)	
Grounds and Yard	
Utilities	
Trapping Varmints	
Major Repair[c]	32 000
Offsite Laboratory Work and Equipment Repairs	5 000
Reference Reactor Facility Services:	42 000
Lab Samples (outfall, air, water, health physics	
Surveillance/Monitoring	
EPA Samples and Reporting	
Requirements	
Security	11 500[d]
Subtotal	96 100
Contingency (25%)	24 025
Total, Annual Safe Storage Costs	120 125

(a) These services are assumed to be provided by specialty contractors.
(b) The number of figures shown is for computational accuracy and does not imply precision to the nearest dollar.
(c) Accruing for use; frequency varies depending on type of repair.
(d) See Table J.2-9 for initial costs.

collectively share not only selected maintenance costs for the entire site but also onsite security costs. Since the reference test reactor is assumed to be on a site by itself, security is postulated to be provided by an offsite commercial security agency. Details of security during safe storage were described previously in Section J.2.3.2.

J.2.5.3 Cost of Deferred Decontamination

The costs of accomplishing deferred decontamination are estimated by examining the general cost categories for DECON shown in Table I.2-7 in Appendix I and determining the impact of the differences in accomplishing decontamination after periods of safe storage of several lengths. It is assumed that the management and support staff is the same for deferred decontamination as it is for DECON. However, fewer decommissioning workers are required for deferred decontamination since several tasks were performed previously and the radiation dose rates are lower.

Estimates are given in Table J.2-20 of the volumes of the various types of radioactive materials that are packaged and shipped for burial when decontamination occurs either immediately or 10, 30, 50, and 100 years after reactor shutdown. The volume of radioactive waste from preparations for safe storage are also given. The volume of activated material is assumed to remain constant over the 100-year time span.

The volume of contaminated material is assumed to remain constant through 30 years, but to decrease to 18 m^3 by 50 years and thereafter.

The volume of radioactive waste estimated for deferred decontamination after 10 to 30 years is assumed to be the difference between the volumes for DECON and for preparations for safe storage. This volume is assumed to be reduced to 55 m^3 at 50 years and to 38 m^3 at 100 years.

The large volume of contaminated soil and buried concrete pipe that is removed during DECON is also removed during the preparations for safe storage since the reference radionuclide inventory for soil and concrete pipe on the reference test reactor site presented in Section E.2.2.4 of Appendix E is such that the radioactivity in the soil and concrete piping will continue to be present in quantities beyond unrestricted release levels for more than 100 years.

Estimated costs are given in Table J.2-21 for decontamination of the reference test reactor after 10 to 30, 50, and 100 years of safe storage. The

TABLE J.2-20. Burial Volumes of Radioactive Materials from Decommissioning the Reference Test Reactor

Decommissioning Alternative	Start of Decommissioning (years after shutdown)	Burial Volume (m^3) Activated Material	Contaminated Material	Radioactive Waste	Alternative Total	Decommissioning Total
DECON[a]	0	62	4762	110	4934	4934
Preparations for Safe Storage[b]	0	--	2779	42	2821	--
Deferred Decontamination	10	62	1983	68	2113	4934
	30	62	1983	68	2113	4934
	50	62	18	55	135	2956
	100	62	18	38	118	2939

(a) Based on Table I.2-8 in Appendix I.
(b) Based on Table J.2-13.

TABLE J.2-21. Estimated Costs of Deferred Decontamination for the Reference Test Reactor

Cost Category	DECON[a]	Decontamination Deferred 10 to 30 Years	50 Years	100 Years
	Costs ($ millions)			
Disposal of Radioactive Materials	0.135	0.135	0.135	0.135
Neutron Activated Materials	2.338	0.974	0.009	0.009
Contaminated Materials	0.099	0.064	0.052	0.036
Staff Labor	8.63	6.076	6.076	6.076
Energy	0.076	0.055	0.055	0.055
Special Tools and Equipment	0.361	0.260	0.260	0.260
Miscellaneous Supplies	0.203	0.140	0.140	0.140
Specialty Contractors	0.616	0.107	0.107	0.107
Nuclear Insurance[b]	--	--	--	--
License Fees[c]	0	0	0	0
Subtotal	12.458	7.811	6.834	6.818
Contingency (25%)	3.115	1.953	1.709	1.705
Totals	15.573	9.764	8.543	8.523

(a) From Table I.2-7 in Appendix I.
(b) Indemnity fees are currently $100/yr for each license (i.e., the test reactor license and the MUR license) at the reference test facility and are not included in this study since they represent only a small fraction of 1% of the total decommissioning cost.
(c) Because the reference test reactor is assumed to be federally owned, these fees are not applicable; however, where applicable for other nuclear R&T reactor facilities, the schedule of fees for license amendments and other approvals required by the license or NRC regulations is given in 10 CFR 170.

estimated costs of DECON, taken from Table I.2-7 in Appendix I, are included for comparison. The only cost category that does not decrease with time is the disposal of activated materials.

The total cost of each of the decommissioning alternatives is given in Table J.2-22. The values listed under "total" include the cost during safe storage for the number of years shown. For example, the total given for deferred decontamination starting 10 years after reactor shutdown includes the cost of preparations for safe storage, the cost of safe storage for the period

TABLE J.2-22. Total Costs of Various Decommissioning Alternatives for the Reference Test Reactor

Years After Reactor Shutdown	Decommissioning Costs in Millions of 1981 Dollars				
	DECON	Preparations for Safe Storage	Safe Storage	Deferred Decontamination	Total
0	15.6	--	--	--	15.6
10	--	6.7	1.1	9.8	17.6
30	--	6.7	3.5	9.8	20.0
50	--	6.7	6.0	8.5	21.2
100	--	6.7	12.0	8.5	27.2

from the end of preparations for safe storage until the start of deferred decontamination, and the cost of deferred decontamination.

J.2.6 Estimated Occupational Radiation Doses for SAFSTOR at the Reference Test Reactor

Estimates are made of the external occupational radiation doses that are accumulated by the decommissioning workers while preparing the reference test reactor facility for safe storage and during the safe storage period. The estimates for the preparations are based on a task analysis to determine the man-hours of effort required in radiation zones and the anticipated dose rates associated with each task. The basic assumptions made in developing these estimates are the same as those described for DECON in Section I.2.4 of Appendix I. The estimate of the external occupational radiation doses associated with the surveillance and maintenance workers at the reference test reactor during the safe storage period are based on 8 years of actual safe storage experience for the PBRF.

The total corrected external occupational radiation dose accumulated during preparations for safe storage is estimated to be about 112 man-rem.

J.2.6.1 Estimated External Occupational Radiation Dose Accumulated During Preparations for Safe Storage

Some of the same basic activities that are performed for DECON are also performed for the preparations for safe storage. These include defueling the reactor, comprehensive radiation surveys, weekly radiation surveys, general

cleanup, system draining, and radioactive liquid waste handling. Other tasks, of course, are different, but the external occupational doses of all tasks are based on a 6-hour effective working day in the radiation zone per worker, the source of radiation dose and its intensity, and the percentage of time a given worker spends working on that particular task. Subjective judgement is used for each task analysis regarding the percentage of time spent on that task by individual workers.

The results of the analyses for the preparations for safe storage are presented in Table J.2-23. The dose estimates for each task are corrected for radioactive decay using the half-life of ^{60}Co from reactor shutdown to the time when the task is one-half completed.

J.2.6.2 Estimated Occupational Radiation Dose Accumulated During the Safe Storage Period

Since many of the same ALARA considerations (e.g., fences, locks, withdrawing of ladders from Q&Cs, and other methods for barring access to radiation zones) are exercised in this study for the SAFSTOR alternative for the reference test reactor as were applied at PBRF in 1973, it is assumed that similar conditions prevail and that external radiation exposures for surveillance and maintenance personnel at the reference test reactor during safe storage are at the threshold levels of detection for personnel monitoring devices. Security techniques, administrative procedures, and the physical layout of the reference test reactor, plus the aforementioned ALARA considerations provide the means for controlled entrance, observation, surveillance, and egress from all buildings and areas without deliberately exposing the safe storage personnel to external radiation exposures of reactor origin. This conclusion is based on the negligible external radiation exposures reported for the surveillance, maintenance, and security forces during the past eight years of safe storage of the PBRF.[a]

(a) Based on information supplied by Mr. John E. Ross of Teledyne Isotopes, General Manager of Plum Brook Operations, Sandusky, Ohio.

TABLE J.2-23. Estimated Occupational Radiation Doses for Preparations for Safe Storage of the Reference Test Reactor

Location / Task	Exposure (man-hr)	Dose (man-rem)	Decay Factor[a]	Corrected Dose (man-rem)[b]
Reactor Building/MUR/Primary Containment				
1. Comprehensive Radiation Survey for Total Facility (i.e., all buildings)	528	2.64	0.995	2.627
2. Discharge Fuel (including MUR)	66	0.750	0.988	0.741
3. Prepare and Ship Spent Fuel	2 500	6.0	0.963	5.781
4. General Cleanup and Equipment Inventory (i.e., all buildings)	924	1.848	0.989	1.828
5. Drain, Clean, Dry Quadrants A, B, C and D and Canals E and F	780	7.8	0.98	7.647
6. Drain, Clean, Dry Canal H (SAFSTOR MUR)	300	3.0	0.945	2.835
7. Drain, Clean, Dry Canal G	180	1.8	0.943	1.697
8. Drain and Flush PCWS	72	0.36	0.942	0.339
9. SAFSTOR tasks within the CV and for the RB as given in Tables J.2-5 and J.2-6, respectively (except for Q&C work)	2 016	10.08	0.965	9.726
10. Final Radiation Survey	40	0.2	0.940	0.188
Subtotals[b,c]	7 406	35		34
Hot Laboratory Building				
1. Decontaminate Hot Cells	624	17.119	0.989	16.932
2. Drain, Clean, Dry Canals J and K	360	3.6	0.977	3.516
3. SAFSTOR tasks for the HLB as given in Table J.2-7 (except for hot cells and Canals J and K)	960	9.6	0.970	9.309

TABLE J.2-23. (contd)

Location / Task	Exposure (man-hr)	Dose (man-rem)	Decay Factor[a]	Corrected Dose (man-rem)[b]
4. Final Radiation Survey	40	0.2	0.963	0.193
Subtotals[b,c]	1 984	31		30
Other Contaminated Structures and Areas				
1. Primary Pump House preparatory tasks (see Table J.2-8)	792	7.92	0.939	7.437
• Final Radiation Survey	16	0.16	0.936	0.150
2. Cold Retention Area Remove contaminated concrete (see Table D.2-3) and contaminated soil (see Appendix C, Section C.4)	1 320	6.6	0.972	6.414
• Final Radiation Survey	36	0.018	0.959	0.017
3. Hot Retention Area preparatory tasks (see Table J.2-8)	1 584	4.752	0.941	4.473
• Final Radiation Survey	24	0.072	0.936	0.067
4. Remove and package contaminated piping and soil from ERB and site ditches (see Table C.4-1)	1 338	0.134	0.961	0.129
5. Office and Laboratory Building preparatory tasks (see Table J.2-8)	1 692	0.846	0.955	0.808
• Final Radiation Survey	16	0.008	0.946	0.008
6. Fan House preparatory tasks (see Table J.2-8)	792	3.96	0.934	3.697
• Final Radiation Survey	36	0.180	0.931	0.168
7. Waste Handling Building preparatory tasks (see Table J.2-8)	792	2.376	0.934	2.218
• Final Radiation Survey	36	0.072	0.930	0.067
Subtotals[b,c]	8 474	27		26

TABLE J.2-23. (contd)

Location / Task	Exposure (man-hr)	Dose (man-rem)	Decay Factor[a]	Corrected Dose (man-rem)[b]
Ancillaries				
1. Radwaste Handling and Laundry Operations	1 110	7.38	0.963	7.109
2. SAFSTOR contaminated air systems (see Section J.2.2.3)	660	3.3	0.945	3.118
3. Install Intrusion Alarms	850	1.275	0.937	1.195
4. Routine Radiation Surveys	178	0.534	0.963	0.514
5. Miscellaneous[d]	--	--	--	10.2
Subtotals, Ancillaries[b,c]	2 798	13		22
Totals[b]	20 662	106		112

(a) Based on the half-life of ^{60}Co; calculated at the midpoint of the task times shown in Figure J.2-1.
(b) The number of figures shown is for computational accuracy and does not imply precision to the nearest millirem.
(c) Dose totals are rounded to the nearest whole number.
(d) Consists of an allowance of 10% of the total explicitly estimated task radiation dose to account for any omissions and uncertainties in the analysis.

J.2.6.3 External Occupational Radiation Dose Accumulated During Deferred Decontamination

For this study it is assumed that the occupational radiation dose accumulated by the decommissioning workers is controlled largely by the radiation levels of ^{60}Co throughout the plant. Thus, if a given task performed immediately after shutdown caused a radiation dose of N_0, that same task performed t years later during deferred decontamination would cause a dose of $N(t) = N_0 e^{-\lambda t}$, where λ is the decay constant for ^{60}Co in years.

Since one of the key asumptions made for deferred decontamination is that most of the tasks are performed in approximately the same way as for DECON,

using the same techniques and equipment, the occupational radiation dose accumulated during deferred decontamination can be estimated from the product of the dose that was accumulated during DECON times the decay factor for ^{60}Co over the safe storage period. Several of the job items given in Table I.2-20 in Appendix I for the radiation dose accumulated during DECON, such as reactor defueling and draining and flushing the PCWS, should be subtracted from the total dose before reduction, since they are performed while preparing the facility for safe storage. Several other items, such as unsecuring the facility and re-establishing essential services, should be added. However, for purposes of estimating the occupational dose for deferred decontamination a simple reduction of the DECON dose in proportion to the decay of ^{60}Co over the safe storage period is a reasonable and conservative approach. These estimates are given in Table J.2-24 for decontamination starting 10, 30, 50 and 100 years after reactor shutdown. After 100 years, essentially all of the remaining radioactivity is contained in the activated reactor vessel components, and the occupational radiation exposure associated with deferred decontamination is extremely small.

TABLE J.2-24. Estimated Occupational Radiation Dose from Various Decommissioning Alternatives for the Reference Test Reactor

Years After Reactor Shutdown	Occupational Radiation Dose (man-rem)				
	DECON	Preparations for Safe Storage	Safe Storage[a]	Deferred Decontamination	Totals
0	322	--	--	--	322
10	--	112	0	86	198
30	--	112	0	6	118
50	--	112	0	<1	112
100	--	112	0	<1	112

(a) See Section J.2.6.2 for details.

REFERENCES

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2. H. D. Oak, et al., Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1980.

3. E. S. Murphy, Technology, Safety and Costs of Decommissioning Reference Non-Fuel-Cycle Nuclear Facilities, NUREG/CR-1754, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, February 1981.

4. NASA-PBRF Monthly Reports - 1973 (February through July).

5. NASA-Plumbrook Reactor Facility End Condition Statements. Revised: April 30, 1974.

6. "OPTIONS" Teledyne Isotopes Study (July 1978).

7. C. A. Pelletier and P. G. Voillegue, Potential Benefits of Reducing Occupational Radiation Exposure, AIF/NESP-010, Atomic Industrial Forum, Inc., Washington, D.C., May 1978.

8. U.S. Code of Federal Regulations, Title 10, Part 170, "Fees for Facilities and Materials Licenses and Other Regulatory Services Under the Atomic Energy Act of 1954, As Amended, "Superintendent of Documents, GPO, Washington, D.C., 1979.

APPENDIX K

DETAILS OF ENTOMB

ENTOMB is one of the alternatives considered in this study for the decommissioning of the reference research and test (R&T) reactors. ENTOMB, as defined by the Nuclear Regulatory Commission (NRC), implies that the radioactivity contained within the entombment structure will decay sufficiently during a 100-year entombment period to permit unrestricted release of the property at the end of that time. This requirement necessitates the removal and disposal elsewhere of materials containing long-lived radionuclides. Thus, the highly activated core internals are removed, but slightly activated materials are enclosed within the entombment structure.

Included in this appendix are discussions of the postulated tasks and work schedules, estimated costs, and estimated radiation doses to workers and to the public for the entombment of the reference R&T reactors. Much of the work associated with ENTOMB is the same as is postulated for DECON or SAFSTOR. Thus, the ENTOMB analysis is accomplished primarily by examining those efforts that are different from the DECON or SAFSTOR efforts, and including by reference those efforts discussed in Appendices I and J that apply. The information presented in this appendix is summarized in the ENTOMB portions of Sections 10, 11, and 12 of Volume 1.

K.1 REFERENCE RESEARCH REACTOR

The entire concrete structure of the reference research reactor (the TRIGA reactor shown in Figure B.2-4 of Appendix B) is postulated as the entombment barrier. Both the Reactor Pool (RP) and the Pool Irradiation Facility (PIF) are utilized for storage of the contaminated materials and radioactive wastes generated during decommissioning activities. In order to accommodate all of the radioactive material volumes described in Table I.1-7 of Appendix I, enlargement of the PIF is necessary. This is accomplished by the vertical

addition of 0.6-m-thick concrete walls from the top of the existing facility to the top of the reactor structure. Trusses are built into the new walls to support the entombment cap of the PIF. All equipment, materials, and wastes known to contain unreleasable quantities of radionuclides from all the buildings associated with the reactor are placed within the RP and PIF. Support trusses are placed across the top of the RP, and the entombment structure cap for both the RP and PIF is formed and poured. The cap is 0.6-m-thick concrete bonded to the existing structures and is designed to support the loadings from anticipated usage.

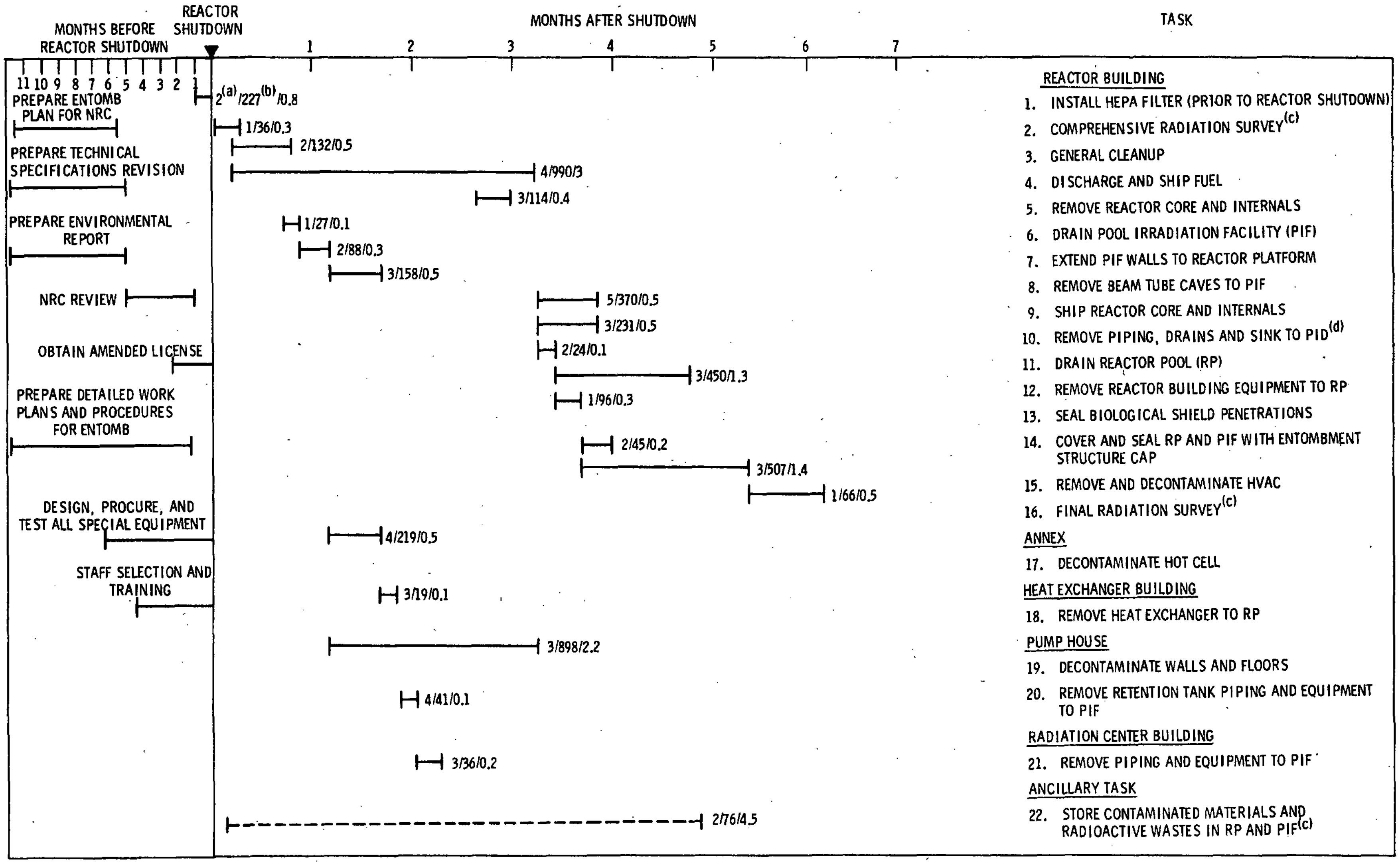
After the removal and transfer of radioactive equipment, piping, drains, etc., from each of the buildings to the entombment structure, each building is decontaminated as necessary to unrestricted release levels.

K.1.1 ENTOMB Tasks, Schedules, and Manpower Needs

Tasks necessary for entombment of the reference research reactor are nearly identical to those required for DECON and preparations for safe storage. Three tasks are unique to ENTOMB and a few tasks have subtle changes with respect to manpower needs and costs.

The task schedule and sequence for ENTOMB is given in Figure K.1-1. Timing in the schedule reflects the need for early clearing of the pool areas within the entombment structure to allow for material storage. Draining of the reactor pool is strategically delayed to provide shielding for personnel while other necessary tasks are performed. The total time duration for accomplishing ENTOMB is 6 months, which is less than for DECON but slightly longer than for SAFSTOR.

The dedicated manpower requirements for ENTOMB at the reference research reactor are given in Table K.1-1. For those tasks which are identical or nearly so, the work crew makeup and shift requirements for ENTOMB and DECON are assumed to be the same. The 22 tasks defined for ENTOMB are one more than necessary for DECON; however, it can be seen from Table K.1-1 that the direct staff needs of ENTOMB are about 4% less than those contained in Table I.1-3 for DECON. This reduction is primarily due to the relief from removal of the biological shield concrete and the reactor vessel for the ENTOMB alternative.



LEGEND:

├──┤ CONTINUOUS OPERATIONS OVER TIME SPAN SHOWN

├- - -┤ INTERMITTENT OPERATION OVER TIME SPAN SHOWN

TOTAL: PERSON MONTHS 37
PERSON HOURS 6470
EXPOSURE HOURS 4850

(a) TASK INFORMATION NUMBERS INDICATE IN SEQUENCE: DIRECT STAFF PER DAY/EXPOSURE HOURS/CALENDAR MONTHS DURATION. WORKERS DEDICATING 15% OR LESS OF THEIR TIME TO THE TASK ARE NOT INCLUDED IN THE DIRECT STAFF PER DAY NUMBER.

(b) THIS NUMBER INCLUDES ALL WORKER EXPOSURE TIME AND IT IS ASSUMED THAT 75% OF THE WORKING DAY IS IN THE RADIATION ZONE.

(c) TASK INCLUDES ALL BUILDINGS.

(d) INCLUDES HEAT EXCHANGER BUILDING COMPONENTS.

FIGURE K.1-1. Overall Task Schedule and Sequence for ENTOMB at the Reference Research Reactor

TABLE K.1-1. Dedicated Manpower Requirements for ENTOMB at the Reference Research Reactor

Task	Task Duration (months)	Dedicated Manpower Requirements (man-months)					
		Supervisor	Utility Operator	Laborer	Craftsman	Health Physics Technician	Total
Reactor Building							
1. Install HEPA Filters[a]	0.80	0.12	--[b]	--	1.60	--	1.72
2. Comprehensive Radiation Survey[b]	0.27	--	--	--	--	0.27	0.27
3. General Cleanup	0.50	--	--	1.00	--	--	1.00
4. Discharge and Ship Fuel	3.00	3.0	2.5	0.50	--	1.50	7.50
5. Remove Reactor Core and Internals	0.36	0.16	0.02	0.39	0.25	0.05	0.87
6. Drain Pool Irradiation Facility (PIF)	0.12	0.03	--	0.12	0.05	0.01	0.21
7. Extend PIF Walls to Reactor Platform	0.25	0.09	0.05	0.51	--	0.03	0.68
8. Remove Beam Tub Caves to PIF	0.50	0.15	0.25	0.75	--	0.05	1.20
9. Ship Reactor Core and Internals	0.50	1.00	1.20	--	--	0.60	2.80
10. Remove Piping-Drains and Sink to PIF	0.52	0.22	--	1.00	0.48	0.05	1.75
11. Drain Reactor Pool (RP)	0.07	0.02	--	0.14	--	0.02	0.18
12. Remove RB Equipment to RP	1.30	0.43	0.14	2.41	0.30	0.13	3.41
13. Seal Biological Shield Penetrations	0.30	0.10	--	0.40	0.10	0.05	0.65
14. Cover and Seal RP and PIF with ENTOMBMENT Structure Cap	0.15	0.02	0.03	0.18	0.08	0.02	0.33
15. Remove and Decontaminate HVAC	1.40	--	--	2.70	1.00	0.14	3.84
16. Final Radiation Survey[c]	0.50	--	--	1.40	--	0.50	1.90
Annex							
17. Decontaminate Hot Cell	0.50	--	--	1.40	--	0.26	1.66
Heat Exchanger Building							
18. Remove Heat Exchanger to RP	0.05	0.01	0.02	0.05	0.05	0.01	0.14
Pump House							
19. Decontaminate Walls and Floors	2.20	--	--	6.60	--	0.20	6.80
20. Remove Retention Tank, Piping and Equipment to PIF	0.10	0.04	0.10	0.10	--	0.07	0.31
Radiation Center Building							
21. Remove Piping and Equipment to PIF	0.16	0.03	--	0.16	0.06	0.02	0.27
Ancillary Task[c]							
22. Store Contaminated Materials and Radiactive Waste in RP and PIF	--	--	--	1.00	0.01	0.10	1.11
Totals		5.42	4.31	20.82	3.98	4.07	38.60

(a) Performed before reactor shutdown.
(b) Denotes no manpower.
(c) Includes all buildings.

K.1.2 Cost of ENTOMB for the Reference Research Reactor

A summary of estimated costs of ENTOMB for the reference research reactor is given in Table K.1-2. Costs estimated for entombing the reference research reactor are considerably lower than those estimated for DECON. This is due to a combination of factors affecting ENTOMB, including: a) direct staff reductions discussed in Section K.1.1, b) decreased costs for radioactive waste disposal, and c) shorter time schedule requirements. The reduced schedule requirements significantly impact the overall staff labor costs, particularly the management and support staff and provide the most cost reduction.

K.1.2.1 Waste Disposal Costs

The highly neutron-activated materials are removed from the reference research reactor to meet the criterion for unrestricted release of the entombment structure after 100 years. These materials are found in the reactor internals and are disposed of in the same manner as they are for DECON. All other radioactive materials are placed into the reactor pool and the enlarged pool irradiation facility cavities. The waste disposal costs of $16,610 for off-site burial of the neutron-activated materials only is given in Table I.1-7.

K.1.2.2 Specialty Contractors

Extension of the PIF walls and installation of the entombment cap will require a specialty contractor to install the forms, structural support members, and concrete. It is estimated that approximately 25 m^3 of concrete is required to construct 0.6-m-thick structures for the pool walls and the two entombment caps. The total cost for this task is $8620.

K.1.2.3 Reactor Building

It is postulated that the reactor building, which houses the proposed entombment structure also remains intact, thus allowing the continued use of laboratories, offices, and equipment located therein. It is estimated that the continuing maintenance costs incurred in this decision will be considerably less than the costs for providing a replacement structure for the existing needed space and facilities. These continuing maintenance costs are,

TABLE K.1-2. Summary of Estimated Costs of ENTOMB for the Reference Research Reactor

Cost Category	Estimated Costs ($)[a,b]	Percent of Total
Disposal of Neutron Activated Materials	16 610	3.8
Disposal of Radioactive Wastes[c]	6 800	1.5
Staff Labor	378 890	85.2
Energy	9 290	2.1
Special Tools and Equipment	2 340	0.5
Miscellaneous Supplies	5 210	1.2
Specialty Contractor[d]	8 620	1.9
Nuclear Insurance	2 790	0.7
License Fees	13 950	3.1
Subtotal	444 500	100.0
Contingency (25%)	111 130	
Total, Costs of Entombment[e]	555 630	
Other Possible Costs		
Spent Fuel Shipment	60 980	
Facility Demolition & Site Restoration	20 100[f]	
Subtotal	81 080	
Contingency (25%)	20 270	
Total	101 350	

(a) 1981 costs used.
(b) The number of figures shown is for computational accuracy and does not imply precision to the nearest $10.
(c) Only includes dry solid wastes.
(d) For installation of the entombment structure.
(e) The "total" ENTOMB costs would also include the annual surveillance and maintenance service costs of $6,120 times "x" number of years these services are provided.
(f) Does not include demolition of the Reactor Building and the reactor structure.

therefore, not applied against the ENTOMB costs, even though security benefits are available for the entombment structure due to the presence of the Reactor Building as a protective barrier.

K.1.2.4 Estimated Annual Surveillance and Maintenance Requirements and Costs During the Entombment Period

A 50% reduction in the surveillance and maintenance costs is estimated when ENTOMB rather than SAFSTOR is chosen to decommission the reference research reactor. This is due to the added integrity incorporated into the entombment structure, which is designed and constructed to prevent both accidental and intentional access. As previously described, the presence of the surrounding Reactor Building structure is, in effect, the first line of defense against intrusion. Also, the existing video monitoring system can be used to continually survey the entombed reactor structure.

Although no radioactive material leakage is anticipated from the entombment structure, a limited environmental surveillance program will probably be required to verify this expectation. The program is anticipated to include periodic surveys of the entombment structure, as well as surface and groundwater sample analysis. Both the video monitoring and environmental monitoring programs are expected to continue at a level of 33% of the estimated SAFSTOR costs. The security, maintenance, and surveillance programs are performed by personnel on the reference university staff.

It is postulated that if the amended nuclear facility license is maintained in force for a period of 100 years, the contained radioactivity will have decayed sufficiently to allow unrestricted release of the site. Then, after verification of the contained radioactivity levels, the license is terminated. It should be recognized that there is no fixed number of years for nuclear reactor facilities to be entombed; it depends on the facility-specific radionuclides and how long they take to decay to unrestricted use levels. For the purposes of this study, all ENTOMB time periods given are for illustration only. While the license is in force, the licensing agency, the NRC or its successor, conducts annual inspections required under the terms of the license.

During these inspections it is anticipated that the integrity of the entombment structure is checked and a review is made of the results of of the environmental surveillance program. The annual fee for these inspections is presently established in 10 CFR 170.23 at $650.

An annual fee is also anticipated for nuclear liability insurance. Based upon a 90% credit from operating premiums for the shutdown facility and an additional 25% credit of the remaining premium because of entombment of the radioactive materials, an annual premium of $580 is derived.

The total annual cost of continuing care for ENTOMB is estimated to be about $6,100. Other annual fees assessable to the entombment structure (such as property taxes) are not considered under the scope of this study, because their cost and structure are particularly site-sensitive.

K.1.3 Estimated External Occupational Radiation Doses

Estimates of the external radiation doses received by decommissioning workers during ENTOMB at the reference research reactor are given in Table K.1-3. Many tasks are essentially identical with those of DECON and SAFSTOR and are based on estimates previously developed and have identical labor requirements and radiation dose exposures. However, because of the early removal of the reactor vessel internals in ENTOMB, lower exposures are accumulated for several succeeding tasks which are otherwise identical to the other decommissioning alternatives. The occupational external radiation doses are computed from the estimated exposure hours for each task as shown in Table K.1-3. These values are corrected for reductions in the dose rates due to radioactive decay (based on ^{60}Co) from the time of reactor shutdown to the midpoint of each listed task. The total cumulative radiation dose to decommissioning workers for ENTOMB at the reference research reactor is estimated to be 16.6 man-rem. This total does not include additional doses accumulated elsewhere by waste disposal workers and transportation workers. A discussion of radiation doses to these additional workers is found in Appendix N.

K.2 REFERENCE TEST REACTOR

The entombment structure postulated for the reference test reactor encompasses the below-grade portion of the reactor containment vessel (CV).

TABLE K.1-3. Estimated Occupational Radiation Doses for ENTOMB at the Reference Research Reactor

Task	Supervisors[a] Exposure (man-hr)	Supervisors[a] Dose (man-rem)	Utility Operators Exposure (man-hr)	Utility Operators Dose (man-rem)	Laborers Exposure (man-hr)	Laborers Dose (man-rem)	Craftsmen Exposure (man-hr)	Craftsmen Dose (man-rem)	Health Physics Technicians Exposure (man-hr)	Health Physics Technicians Dose (man-rem)	Task Totals Exposure (man-hr)	Task Totals Dose (man-rem)	Task Totals Decay Factor[b]	Task Totals Corrected Dose (man-rem)
Reactor Building														
1. Install HEPA Filters	16	0.016	--[c]	--	--	--	211	0.211	--	--	227	0.227	1.00	0.227
2. Comprehensive Radiation Survey[d]	--	--	--	--	--	--	--	--	36	0.360	36	0.360	0.989	0.356
3. General Cleanup	--	--	--	--	132	0.132	--	--	--	--	132	0.132	0.989	0.131
4. Discharge & Ship Fuel	396	2.772	304	2.128	66	0.462	--	--	224	1.568	990	6.930	0.973	6.742
5. Remove Reactor Core and Vessel Internals	21	0.884	4	0.016	51	0.204	33	0.132	5	0.020	114	0.456	0.968	0.441
6. Drain Pool Irradiation Facility (PIF)	4	0.008	--	--	16	0.032	6	0.012	1	0.002	27	0.054	0.989	0.053
7. Extend PIF Walls to Reactor Platform	12	0.024	4	0.008	68	0.136	--	--	4	0.008	88	0.176	0.978	0.172
8. Remove Beam Tube Caves	20	0.040	33	0.006	99	0.198	--	--	6	0.012	158	0.316	0.978	0.309
9. Ship Reactor Core & Internals	132	0.462	158	0.553	--	--	--	--	80	0.280	370	1.295	0.957	1.230
10. Remove Piping Drains & Sink to PIF	29	0.029	--	--	132	0.132	63	0.063	7	0.007	231	0.231	0.957	0.221
11. Drain Reactor Pool (RP)	3	0.006	--	--	18	0.036	--	--	3	0.006	24	0.048	0.946	0.046
12. Remove Reactor Building Equipment to RP	57	0.342	18	0.108	318	1.908	40	0.240	17	0.102	450	2.700	0.957	2.556
13. Seal Biological Shield Penetrations	18	0.036	--	--	53	0.106	18	0.036	7	0.014	96	0.192	0.957	0.184
14. Cover & Seal RP & PIF With Entombment Structure Cap	3	0.006	4	0.008	24	0.048	11	0.022	3	0.006	45	0.090	0.946	0.097
15. Remove & Decontaminate HVAC	--	--	--	--	356	0.178	132	0.066	19	0.010	507	0.254	0.946	0.240
16. Final Radiation Survey[d]									66	0.034	66	0.034	0.936	0.032
Annex														
17. Decontaminate Hot Cell	--	--	--	--	185	1.850	--	--	33	0.330	219	2.219	0.978	2.170
Heat Exchanger Building														
18. Remove Heat Exchanger to RP	1	0.001	3	0.003	7	0.007	7	0.007	1	0.001	19	0.019	0.978	0.019
Pump House														
19. Decontaminate Walls and Floor	--	--	--	--	871	0.871	--	--	27	0.027	898	0.898	0.968	0.368
20. Remove Retention Tank, Piping, & Equipment to PIF	6	0.012	13	0.026	13	0.026	9	0.018	--	--	41	0.082	0.978	0.080
Radiation Center Building														
21. Remove Piping & Equipment to PIF	4	0.004	--	--	21	0.021	8	0.008	3	0.003	36	0.036	0.968	0.368
All Buildings														
22. Store Contaminated Materials & Radioactive wastes to RP & PIF	--	--	--	--	61	0.305	2	0.010	13	0.065	76	0.380	0.968	
Subtotals	723		541		2491		540		555		4850			
Ancillaries														
23. Routine Radiation Surveys	--	--	--	--	--	--	--	--	238	--	238	0.060	0.973	0.058
Totals	723		541		2491		540		793		5138			16.635

(a) Includes shift engineers, crew leaders, craft supervisors and senior health physics technicans.
(b) Based on 60-Co half-life, calculated at the midpoint of the task schedule shown in Figure K.1-1.
(c) Indicates that category of decommissioning staff is not involved in the particular task.
(d) Task includes all buildings.

Radioactive materials and equipment are removed from their locations external to the CV and are placed within the quadrants surrounding the reactor pressure vessel and biological shield. The contaminated drains from within the CV are cut, plugged, and capped at the CV wall. After the radioactive materials and equipment are placed within the CV, the concrete floors and other surfaces at the 0 elevation are partially removed to permit installation of forming and structural support for the entombment structure cap. This cap is nominally 0.6 m in thickness, is bonded to the concrete structures forming the quadrants and the CV liner, and is designed to support floor loadings typical of a high-bay warehouse, manufacturing, or maintenance facility.

The above-grade portion of the CV is decontaminated and released for unrestricted use, as is the remainder of the facility external to the CV.

K.2.1 ENTOMB Tasks, Schedules, and Manpower Needs

Most of the tasks required to entomb the reference test reactor are identical to the tasks required for DECON. Three new tasks are added, seven DECON tasks are deleted, and several tasks are reduced in scope, from the schedule shown in Figure I.2-1 of Appendix I, to create the schedule for entombing the reactor CV, as shown in Figure K.2-1. The points in time for performing some of the tasks are shifted relative to the DECON schedule to allow for the placement of the material being entombed.

The task schedules for the Hot Laboratory and the Other Contaminated Structures are identical in content but are also shifted in time relative to the DECON schedules. The overall schedule for the ENTOMB project is shown in Figure K.2-2. While the time distribution of tasks for ENTOMB is different from DECON, the total duration of ENTOMB activities is essentially the same as DECON, about 25 months following reactor shutdown.

The shift schedule and the makeup of the work crews are assumed to be the same for ENTOMB as are given for DECON in Section I.2.2.4 of Appendix I. The elimination of seven cleanup and removal tasks in the CV reduces the total direct staff labor need for ENTOMB by about 12% relative to DECON. However, it is estimated that the additional effort assoicated with the three new tasks

TASK	REACTOR BUILDING/MUR/CONTAINMENT VESSEL: MONTHS AFTER SHUTDOWN (1–24)
1. COMPREHENSIVE RADIATION SURVEY FOR TOTAL FACILITY (i.e., ALL BUILDINGS)	6/528/1
2. DISCHARGE FUEL (INCLUDING MUR)	5/66/0.1
3. PREPARE AND SHIP SPENT FUEL	8/2500/3
4. GENERAL CLEANUP AND EQUIPMENT INVENTORY (i.e., ALL BUILDINGS)	14/924/0.5
5. DRAIN, CLEAN, DRY QUADRANTS A, B, C AND D AND CANALS E AND F	10/780/0.6
6. PLACE RADIOACTIVE MATERIAL IN CV	5/9000/17
7. DRAIN, CLEAN, DRY CANAL H	10/180/0.14
8. REMOVE, PACKAGE, AND SHIP MUR AND ASSOCIATED HARDWARE	10/576/0.45
9. DRAIN, CLEAN, DRY CANAL G	10/180/0.14
10. REMOVE LOOSE EQUIPMENT IN Q&Cs AND DRY ANNULUS	10/936/0.55
11. DRAIN AND FLUSH PCWS	6/72/0.1
12. ISOLATE RV AND ADD DEIONIZED WATER FOR SHIELDING	6/72/0.1
13. REMOVE RV INTERNALS AND SHIP ACTIVATED RV INTERNALS	16/10366/3.75
14. CUT AND CAP CV PENETRATIONS	16/3840/1.5
15. REMOVE FIXED EQUIPMENT OUTSIDE CV	101/1728/1.3
16. REMOVE PCWS PIPING TO PPH	10/2820/2.14
17. REMOVE RB CONTAMINATED CONCRETE	10/1300/1
18. REMOVE MISCELLANEOUS CONTAMINATED DRAINS	15/2520/1.62
19. REMOVE CONTAMINATED HVAC FROM RB/CV	8/960/0.91
20. INSTALL ENTOMBMENT CAP	SUBCONTRACT/7000/4
21. FINAL RADIATION SURVEY	2/50/0.5

ABBREVIATIONS:

CV	CONTAINMENT VESSEL
HVAC	HEATING, VENTILATION, AND AIR CONDITIONING
MUR	MOCK-UP REACTOR
PCWS	PRIMARY COOLING WATER SYSTEM
PPH	PRIMARY PUMP HOUSE
Q&C	QUADRANTS AND CANALS
RB/CV	REACTOR BUILDING / CONTAINMENT VESSEL
RV	REACTOR VESSEL

PERSONS PER DAY/EXPOSURE HOURS/CALENDAR MONTHS

CONTINUOUS OPERATIONS OVER THE TIME SPAN SHOWN

INTERMITTENT OPERATIONS OVER THE TIME SPAN SHOWN

TOTALS		
	PERSON-MONTHS	324
	PERSON-HOURS	56,094
	EXPOSURE-HOURS	46,398

FIGURE K.2-1. Task Schedule and Sequence and Decommissioning Worker Requirements for ENTOMB Activities in the Reactor/Containment Building

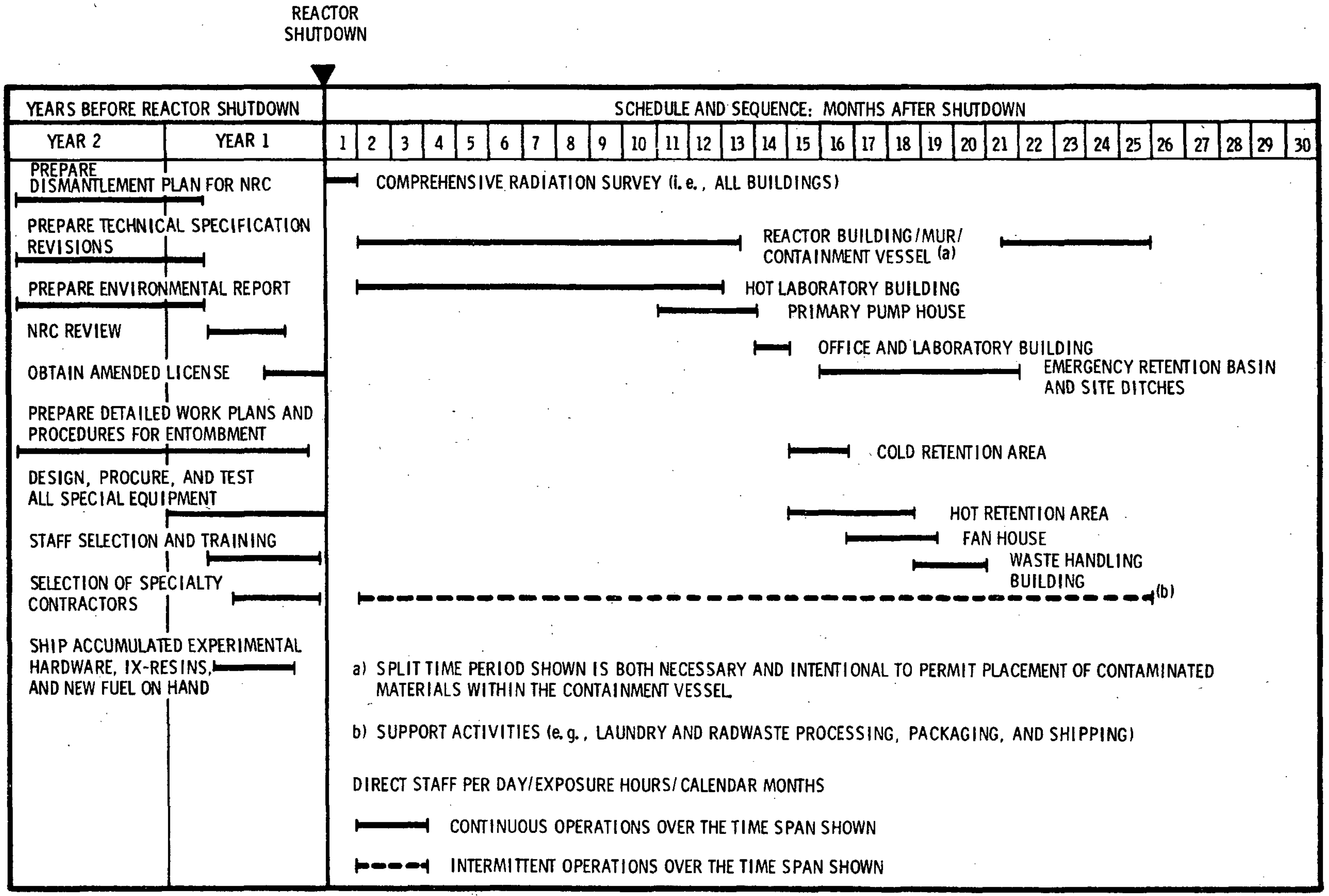


FIGURE K.2-2. Overall ENTOMB Project Schedule

plus placing the contaminated materials within the CV will increase the total ENTOMB direct staff labor by a like amount, resulting in no significant change in staff labor needs.

K.2.2 Cost of ENTOMB for the Reference Test Reactor

The costs estimated for entombing the reference test reactor are very similar to the costs estimated for DECON, differing principally in decreased cost for radioactive waste disposal and increased cost for subcontract work (the entombment cap). The total estimated cost of ENTOMB is given in Table K.2-1, with the discussions of those cost elements of ENTOMB that are different from DECON presented in the subsequent subsections.

K.2.2.1 Waste Disposal Costs

To meet the criterion for unrestricted release of the entombment structure after 100 years, it is necessary to remove the neutron-activated materials from the reference test reactor and from the mock-up reactor, as is done in DECON. The contaminated equipment and material from outside the CV and the contaminated concrete from surfaces external to the CV are placed within the quadrants, thus eliminating the packaging, shipment, and burial costs for those materials. The wet solid radioactive wastes are also placed within the quadrants. The dry solid radioactive wastes are disposed of as in DECON. These wastes are largely combustible material. While the likelihood of a fire occurring in this material within the sealed entombment structure is rather remote, it seems prudent to exclude combustibles.

The costs of offsite disposal for those materials shipped to a low-level waste burial ground are summarized in Table K.2-2.

K.2.2.2 Specialty Contractors

Installation of the entombment cap will require a contractor to install the forms and structural support members and concrete. It is estimated that approximately 445 m^3 of concrete is required to form a cap 0.6-m thick at the elevation level within the CV at a cost of about $\$553/m^3$, for a total cost of about \$246,000. This cost is in addition to the contractor cost identified previously for DECON.

TABLE K.2-1. Estimated Costs of ENTOMB for the Reference Test Reactor

Cost Category		Estimated Costs ($ millions)[a,b]	Percent of Total[c]
Disposal of Radioactive Materials			
Neutron-Activated Materials			
Reference Test Reactor	0.131		
Mock-Up Reactor (MUR)	0.004		
Contaminated Materials	1.352		
Radioactive Wastes[d]	0.087		
Total Disposal Costs		1.574	13.5
Staff Labor		8.63	73.7
Energy		0.076	0.6
Special Tools and Equipment		0.361	3.1
Miscellaneous Supplies		0.202	1.7
Specialty Contractors[e]		0.862	7.4
Nuclear Insurance		--[f]	--
License Fees		--[g]	--
Subtotal		11.706	100.0
Contingency (25%)		2.927	
Total, Costs of Entombment[h]		14.633	
Other Possible Costs			
Spent Fuel Shipment		0.204[i]	
Facility Demolition & Site Restoration		1.783[j]	
Subtotal		1.987	
Contingency (25%)		0.497	
Total, Other Possible Costs		2.484	
Annual Surveillance & Maintenance Costs			
Security		0 - 0.011	
Environmental Monitoring		0.010 - 0.020	
Maintenance		0 - 0.001	
Subtotal		0.010 - 0.032	
Contingency (25%)		0.003 - 0.009	
Total, Annual Costs		0.013 - 0.041	

(a) 1981 costs.
(b) The number of significant figures shown is for computational accuracy and does not imply precision to the nearest $1000.
(c) Individually rounded to the nearest 0.1%.
(d) Includes only dry solid wastes.
(e) Includes selected demolition, explosives, temporary radwaste, environmental monitoring services, and entombment cap installation.
(f) Indemnity fees are currently $100/yr for each license (i.e., the test reactor license and the MUR license) at the reference test facility and are not included in this study since they represent only a small fraction of 1% of the total decommissioning cost.
(g) Because the reference test reactor is assumed to be federally owned, these premiums and fees are not applicable; however, where applicable for other nuclear R&T reactor facilities, the schedule of fees for license amendments and other approvals required by the license or NRC regulations is given in 10 CFR 170.
(h) The "total" ENTOMB costs would also include the annual surveillance and maintenance service costs of about $41,000 (maximum) times "x" number of years that these services are provided.
(i) Does not include costs for handling at the reactor or costs for handling and storage at the repository.
(j) This cost includes demolition of structures external to the Containment Building.

TABLE K.2-2. Estimated Costs of ENTOMB for Disposal of Radioactive Materials from the Reference Test Reactor

Material Category	Number of Shipments	Burial Volume (m^3)	Disposal Costs ($ million)
Neutron-Activated			
Test Reactor	15	56	0.131
Mock-Up Reactor	1	6	0.004
Contaminated			
Concrete Pipe and Soil	163	2779	1.352
Radioactive Waste			
Dry Solid Waste	12	93	0.087
Totals	191	2934	1.574

K.2.2.3 Facility Demolition

It is postulated that all structures external to the CV are removed following decontamination, leaving the CV intact with the below-grade portion entombed and the above-grade portion decontaminated and released for unrestricted use. As a result, the demolition cost for the Reactor Building and CV given in Table L.3-3 of Appendix L is expected to be reduced by approximately two-thirds, for a net cost of about $260,000. Thus, the total cost for demolition of the decontaminated structures external to the CV and for onsite restoration work is estimated to be about $1.78 million, without contingency.

K.2.2.4 Estimated Annual Surveillance and Maintenance Requirements and Costs During the Entombment Period

One of the principal reasons for considering ENTOMB rather than SAFSTOR is that consolidating and entombing the residual radioactivity greatly reduces the requirements and costs for surveillance and maintenance. The entombment structure is designed to withstand the ravages of nature for at least 100 years without losing its containment integrity. It will assuredly prevent any accidental access and make intentional access to the stored radioactive materials

very difficult. Since, the above-grade portion of the CV is postulated to be utilized as a high-bay work area for unrestricted activities, the security system provided for the above-grade facility will be sufficient to prevent any entrance attempts into the below-grade portion of the structure. Should the above-grade portion of the CV be abandoned and/or demolished, an electronic surveillance system for security of the entombment structure would have to be installed.

The high structural integrity of and the lack of any driving forces within the entombment structure make release of any radioactivity from the structure very unlikely. However, a limited environmental surveillance program will be necessary to demonstrate that no leakage of radioactivity is occurring. This surveillance would be limited to periodic surveys of the entombment cap surface and to a continuing groundwater-sampling program in the vicinity of the entombment structure. It is anticipated that surveys and samples would be taken on a quarterly basis for the first 5 years, on a semi-annual basis for the next 10 years, and on an annual basis thereafter until the license is terminated.

Both the security (if needed) and the surveillance programs will be performed by specialty contractors. The estimated annual costs for these programs are given in Table K.2-3.

TABLE K.2-3. Estimated Annual Costs of Security and Surveillance for the Entombed Reference Test Reactor

	Installation Cost ($)	Annual Operating Cost ($)
Security Program (If needed)	25 000	11 000
Environmental Monitoring Program	Already in Place	
0 - 5 Years		20 000
5 - 15 Years		15 000
>15 Years		10 000

It is postulated that if the amended nuclear facility license is maintained in force for a period of 100 years, the contained radioactivity will have decayed to unrestricted release levels and the license can be terminated with no further action. It should be recognized that there is no fixed number of years for nuclear reactor facilities to be entombed; it depends on the facility-specific radionuclides and how long they take to decay to unrestricted use levels. For the purposes of this study, all ENTOMB time periods given are for illustration only. During the licensed period, the licensing agency, the NRC or its successor, conducts annual inspections of the entombment structure and reviews the results of the environmental surveillance program. The annual fee for these inspections is presently established in 10 CFR 170.23 at $650. However, since the licensee is another government agency, this fee is not charged.

Similarly, as a government-owned facility, the usual requirements for nuclear liability insurance do not apply.

Maintenance costs for the entombment structure are anticipated to range from zero (when the above-ground portion of the CV is being utilized and maintained) to an average of $1,000 per year (if the above-ground portion of the CV is removed and a water-tight surface is maintained on the entombment cap).

The total annual cost for surveillance and maintenance of the entombment structure will vary depending on the need for a security system and on the level of environmental surveillance required. This cost is estimated to be about $32,000 initially ($21,000 without security) and to decrease to about $21,000 ($10,000 without security) at times greater than 15 years after entombment, not including contingency.

K.2.3 Estimated External Occupational Radiation Doses

Estimates of the external radiation doses received by decommissioning workers during the entombment of the reference test reactor are based on the estimates developed for DECON, with modifications as appropriate to reflect differences in tasks. As discussed in Section K.2.2, the significant changes in the tasks for ENTOMB as compared with DECON are in the CV, where three new

tasks are added, seven tasks are deleted, and several tasks are reduced in scope. The occupational external radiation doses are computed from the estimated exposure hours for each task, as shown in Figure K.2-1. These values appear in Table K.2-4, and are corrected for reductions in the dose rates due to radioactive decay (based on ^{60}Co) from the time of reactor shutdown to midpoint of each individual task. The radiation doses accumulated during the decontamination of facilities external to the CV given in Table I.2-21 of Appendix I are assumed to be the same for ENTOMB as for DECON, except for small adjustments in the decay corrections and are listed with no task breakdowns. The total cumulative radiation does to decommissioning workers for ENTOMB is estimated to be 425 man-rem, not including doses to transport workers and disposal-site workers. These latter categories of worker doses are discussed in Appendix N.

TABLE K.2-4. Estimated Occupational Radiation Doses for ENTOMB of the Reference Test Reactor

Location/Task	Exposure (man-hr)	Dose (man-rem)	Decay Factor[a]	Corrected Dose (man-rem)[b]
Reactor Building/MUR/Primary Containment				
1. Comprehensive Radiation Survey for Total Facility (i.e., all buildings)	528	2.64	0.995	2.626
2. Discharge Fuel (including MUR)	66	0.75	0.988	0.741
3. Prepare and Ship Spent Fuel	2 500	6.0	0.963	5.781
4. General Cleanup and Equipment Inventory (i.e., all buildings)	924	1.848	0.986	1.823
5. Drain, Clean, Dry Quadrants A, B, C and D and Canals E and F	780	7.8	0.980	7.648
6. Place Radioactive Material in CV	9 000	90.0	0.886	79.778
7. Drain, Clean, Dry Canal H	180	1.8	0.946	1.702
8. Remove, Package, and Ship MUR and Associated Hardware	576	10.0	0.944	9.436
9. Drain, Clean Dry Canal G	180	1.8	0.940	1.693
10. Remove Loose Equipment in Q&Cs and Dry Annulus	936	9.36	0.937	8.774
11. Drain and Flush PCWS	72	0.36	0.934	0.336
12. Isolate RV and Add Deionized Water for Shielding	72	0.072	0.932	0.067
13. Remove RV Internals and Ship Activated RV Internals	10 336	51.83	0.914	47.349
14. Cut and Cap CV Penetrations	3 840	38.40	0.886	34.092
15. Remove Fixed Equipment Outside CV	1 728	1.728	0.865	1.495
16. Remove PCWS Piping to PPH	2 820	5.64	0.839	4.733
17. Remove RB Contaminated Concrete	1 300	1.30	0.825	1.069
18. Remove Miscellaneous Contaminated Drains	2 520	5.04	0.896	4.516
19. Remove Contaminated HVAC from RB/CV	960	2.4	0.810	1.943
20. Install Containment Cap	7 000	35.0	--	35.0
Final Radiation Survey	50	0.025	0.801	0.020
Subtotals[b,c]	46 398	239		250
Hot Laboratory Building Total	10 616			60
Other Contaminated Structures & Areas	22 821			71
Ancillaries	3 796			44
ENTOMB Total	83 631			425

(a) Based on ^{60}Co half-life, calculated at the midpoint of the task times shown in Figure K.2-1.
(b) The number of significant figures shown is for computational accuracy and does not imply precision to the nearest millirem.
(c) Dose totals are rounded to the nearest whole numbers.

APPENDIX L

DEMOLITION AND SITE RESTORATION DETAILS

Once all radioactive materials in a research or test reactor are removed, the Nuclear Regulatory Commission (NRC) is requested to terminate the reactor owner's license and release the site for unrestricted use. Following license termination, the owner decides whether the remaining onsite structures are to be demolished or left standing. (Section 4.2 of Volume 1 contains a discussion of some of the considerations in making this decision.)

Although the NRC has no jurisdiction over removal of uncontaminated facility structures and restoration of the sites, development of demolition and site restoration costs for the reference R&T facilities are presented in this appendix for completeness. To ensure realism, a demolition contractor was retained to estimate demolition costs, just as though he were bidding on the jobs.[1] While this approach is adequate for the reference research reactor, it is not nearly as simple and straightforward for the reference test reactor. Because of the physical layout (i.e., significant quantities of buried, contaminated piping and tanks) at the reference test reactor facility, a modified demolition approach is used. It is assumed that the demolition contractor participates in the planning and preparation for the decommissioning of the reference test reactor from the earliest stages to: 1) take advantage of his expertise; 2) coordinate with the decommissioning crews the removal of significant quantities of buried, contaminated, and potentially contaminated materials; and 3) effect cost savings through the use of existing equipment and access points to facilitate demolition.

L.1 ASSUMPTIONS FOR DEVELOPMENT OF COST ESTIMATES

The analyses of the effort and costs involved in demolishing the reference R&T reactor facilities and restoring the sites are based on the following assumptions:

1. All above-ground structures on the plant site are demolished and removed.

2. Building structures are demolished down to 1 m below grade; holes are drilled in the sub-basement floors for drainage[a]; the empty below-grade volumes are filled to within 1 m of the grade level with concrete rubble; and the last meter is backfilled with topsoil.
3. The demolition contractor has salvage rights (e.g., scrap metal and abandoned piping), with a resulting fair profit going to the contractor. The total estimated cost reflects this profit where applicable (e.g., the water tower of the reference test reactor).
4. Cranes at R&T reactor facilities are left in a safe, useable, and uncontaminated condition for use by the demolition contractor.
5. The reference research reactor site is free of all radioactive contamination except for small amounts of pipe bound in concrete. However, without considerably more information about potentially contaminated buried pipes, tanks, soil, etc., it is recognized that significantly more time for the demolition and site restoration of the reference test reactor site may be required than is estimated in this section.
6. Excess rubble and other debris are disposed of at a landfill 16 km from the site.

L.2 FACTORS CONSIDERED IN ESTIMATION OF DEMOLITION COSTS

Cost estimates for demolition of the reference R&T reactor plants are influenced by several factors, including demolition methods, problem areas in cost estimation, salvage considerations, disposal of debris, labor rates, and equipment requirements. These factors are discussed in the following subsections.

L.2.1 Demolition Methods

Removal of site structures is accomplished using conventional demolition methods, including explosives.

(a) For the reference test reactor, it is assumed that no drilling for percolation is done on the stainless steel shell that constitutes the bottom portion of the reactor containment vessel, since it is 7.6 m below grade.

In general, building demolition proceeds from the top down after machinery and equipment are removed. Where cranes are in place, they are used for moving such things as machinery, equipment, piping, steel stairways, and for placing and removing drills.

In dealing with the massive portions of the structures, heavy concrete walls and slabs are drilled from the top, wherever possible, rather than from the sides, because it is easier and cheaper. Heavier drills (track jacks) can often be used; and once a pattern has been established, it is easier to avoid the rebar. Drilling from the top minimizes working off staging as well as hand labor. For demolition of entire walls, holes are staggered to get better fragmentation.

Wherever possible, the surrounding structure is used to shield rock-fly occurring from explosives.

L.2.2 Problem Areas in Estimation of Demolition Costs

The estimates presented in this appendix result from comparisons of the reference R&T structures with industrial-type structures that have been demolished. In addition, judgment factors are applied, based on experience, for the massiveness, grade of concrete, and reinforcing steel of the structures.

Unlike power reactors, the reference test reactor made use of outdoor buried tanks and piping for the storage and/or transfer of contaminated liquids. While this practice was beneficial during the operating lifetime of the test facility, it poses some problems during decommissioning. For example, soil (some of which may be contaminated) must be excavated so that contaminated tanks and piping can be identified and removed. This activity calls for a close working relationship between an excavation/demolition contractor and the licensee. Such a relationship is assumed to be established early in the decommissioning planning phase of the reference test reactor to take advantage of the demolition contractor's expertise. It should be recognized that realistically there still will be unavoidable situations that make any demolition and/or site restoration estimates quite risky until all such areas are clearly identified and evaluated in the detail appropriate to each specific situation.

L.2.3 Salvage Considerations

Salvage is ordinarily used in the wrecking industry as a contingency/profit item to compensate for such inherent risks as unanticipated demolition difficulty, human error, and variation in employee productivity. Where salvage is easily separable from demolition (i.e., substantial mechanical or electrical equipment salvage), it is given separate consideration in estimating costs. The value of structural salvage is generally accounted for by reducing the unit demolition cost rather than by a separate itemized account. The costs of handling, loading, and shipping salvaged scrap are deducted from the published market price for scrap steel, with the balance credited to the facility owner. Thus, the wrecking contractor recovers his costs, including the added expense of preserving the highest salvage value, and ensures that the owner receives a fair allowance for scrap.

L.2.4 Transport of Demolition Debris

Loading and hauling demolition debris generally costs more than loading and hauling dirt or gravel. Demolition trucks are not loaded or dumped as fast as dirt and gravel trucks because of their longer, higher-sided trailer and the uneven sizes of the materials they haul. Hauling costs in the wrecking industry reflect these conditions.

L.2.5 Labor Rates

The cost estimates for demolition of the reference R&T reactors are based on the labor rates in Seattle, Washington, in May 1981, as given in Table L.2-1. The payroll costs include base pay, FICA, unemployment insurance, union contributions, state industrial insurance, bodily injury insurance, property damage insurance, and medical insurance. They do not include overhead or profit.

L.2.6 Equipment Requirements for Demolition

Because of the variety of operations involved in the demolition of the reference R&T reactors, a number of equipment types and sizes are required.

Table L.2-2 lists the major equipment items for each of the reference facilities, together with the use and required number of each item.

TABLE L.2-1. Labor Rates for Demolition

Occupation	Base Pay ($/hr)	Payroll Cost ($/hr)
Teamsters (38-m^3 trucks)	15.94	6.71
Teamsters (9-m^3 trucks)	15.50	6.65
Laborers	12.93	6.85
Operating Engineers - Operator	16.50	8.19
Operating Engineers - Oiler	15.70	8.01

TABLE L.2-2. Equipment Requirements for Demolition of the Reference R&T Reactors Facilities

Item	Use	Estimated Number Required	
		Test Reactor	Research Reactor
90-Ton Crane	Lifts and Breaking Ball for Concrete Demolition	1	
65-Ton Crane	Lifts and Breaking Ball for Concrete Demolition	2	1
50-Ton Hydraulic Crane	Dismantle Roof Structures	1	
45-Ton Cable Crane	Lifts and Breaking Ball for Concrete Demolition	1	
1-m^3 Backhoe	Clean up and Load Debris	1	
2-m^3 Backhoe	Wrecking and Cleanup	2	
977 Track Loader	Clean up and Load Debris	2	2
9-m^3 Dump Trucks	Haul Concrete Rubble	4	4
Demolition Trailers with Tractors	Haul Debris	3	3
Oxyacetylene Cutting Outfits	Cut Steel and Other Metal	20	6
Clamshell	Pick up Debris	2	1
Track Drills	Drill Holes for Explosives	8	2
Air Hammers	Demolish Concrete	2	
Hydraulic Hammers	Demolish Concrete	2	1
Hydraulic Grapples with Shears	Wrecking and Sorting	3	1
Compressors	Power Pneumatic Tools	As Needed	
Service Truck	Service Equipment	1	1
Breaking Balls	Demolish Concrete	As Needed	
Rigging	Move Equipment and Salvage	As Needed	

L.3 DEMOLITION AND SITE RESTORATION COSTS

The estimated costs of demolition and site restoration for the reference R&T reactors after decommissioning by DECON are summarized in this section. These costs include labor, equipment, supplies, overhead, and profit for each estimate. They do not include extraordinary insurance premiums, bonding, or state sales tax. Details of the cost estimates for the reference R&T facilities are given in the following subsections.

L.3.1 Research Reactor Demolition and Site Restoration Costs

A summary of estimated costs of demolition and site restoration for the reference research reactor is given in Table L.3-1.

TABLE L.3-1. Summary of Estimated Costs of Demolition and Site Restoration for the Reference Research Reactor[a]

Activity	Estimated Cost ($)
Structure Demolition:	
Reactor Building	99 900
Reactor Structure	76 750
Radiation Center	11 500
HX Building	5 100
Pump House	1 500
Site Restoration	2 000[b]
Subtotal	196 750
Contingency (25%)	49 188
Total	245 938

(a) Estimates given in June 1981 dollars.
(b) Assumes no backfill is required and includes grading and seeding of 530 m^2.

A summary of unit costs and estimated waste quantities of structural materials for the demolition of the reference research facility is given in Table L.3-2. Demolition activities for each structure are described in the following subsections.

TABLE L.3-2. Summary of Estimated Quantities of Waste Material and Unit Costs for the Demolition of the Reference Research Facility

Structure	Waste Material: Structural Steel (Mg)	Rebar (Mg)	Concrete (m^3)	Estimated Unit Cost For Concrete ($/$m^3$)
Reactor Building	16.8	41.9	509	196
Reactor Structure	2.7	38.6	235	327
Radiation Center Annex	0.2	0.6	8	196
HX Building	1.5	2.1	26	196
Pump House	0.2	0.6	8	196
Totals	21.4	83.8	786	

L.3.1.1 Reference Research Reactor Structure and Building

The Reactor Building is a Seismic Category 3 structure and its demolition is accomplished by the use of a 59-Mg cable crane with 36-m boom and a 2.7-Mg wrecking ball. Its removal is scheduled to take place after the internally contained reactor structure demolition is complete. This allows the use of the Reactor Building bridge crane for demolition activities on the reactor structure as well as the use of the building shell for shielding and material containment from reactor structure explosive demolition activities.

A track drill is utilized to drill holes in the concrete reactor structure for explosive charge placement. The overhead crane is used to move this drill, as necessary, during explosive demolition activities. First, vertical holes are drilled 8 m deep into the concrete from the top of the reactor through the base structure and into the footings. The track drill is reset on the floor. Several holes are drilled into the walls and footings of the pool irradiation facility. The drill track is removed and the charges are loaded and fired.

Before proceeding with the Reactor Building demolition, structural severance must be made between the Annex and the adjoining Radiation Center Building. Once this is done, the south wall and roof of the Reactor Building are demolished with the wrecking ball. This eliminates the possiblity of the southerly portion of

the Reactor Building falling toward the Radiation Center. The remainder of the Reactor Building is then systematically demolished, the rubble loaded onto trucks with a tractor loader, and hauled to a disposal site.

L.3.1.2 Annex

The Annex is structurally attached to both the Reactor Building and the Radiation Center Building. Using the 59-Mg cable crane, the track drill is moved to the roof of the Annex above the hot cell. Blasting holes are vertically drilled into the concrete hot cell structure, including penetration into the underlying foundation. Explosive charges are used to demolish the hot cell. Mats are used to protect the common wall to the Radiation Center Building. The necessity for severing the Annex from the Radiation Center Building is described in Subsection L.3.1.1. This is accomplished by shoring up the Annex roof and cutting a 1.2-m gap next to the Radiation Center Building on the roof and walls. A tractor loader or hydraulic backhoe with grapple is used to take apart the remaining structure and load the debris.

L.3.1.3 Heat Exchanger Building

Demolition of the steel-framed Heat Exchanger Building is accomplished by use of the crane and wrecking ball, backhoe, and grapple. The floor is broken as it is loaded by the tractor loader or backhoe; if necessary, the wrecking ball is used.

L.3.1.4 Pump House

The small steel-framed Pump House is demolished by means of the tractor loader.

L.3.2 Test Reactor Demolition and Site Restoration Costs

A summary of estimated costs for demolition and site restoration of the reference test reactor is given in Table L.3-3. Estimated waste quantities of structural materials in the reference test facility structures are summarized in Table L.3-4. The quantity estimates are based on a limited number of architectural drawings and photographs and on verbal descriptions. Demolition activities for each structure are described in the following subsections,

TABLE L.3-3. Summary of Estimated Costs of Demolition and Site Restoration for the Reference Test Reactor[a]

Activity	Estimated Cost ($)
Structure Demolition	
Reactor Building and Primary Containment Vessel[b]	788 000
Hot Laboratory Building	670 400
Primary Pump House	162 000
Fan House (includes 30.5-m-high stack)	42 450
Waste Handling Building	33 550
Office and Laboratory Building	86 450
Hot Retention Area	82 750
Cold Retention Area	43 800
Emergency Retention Basin	-0-
Utility Tunnels	61 000
Cooling Tower (wood)	60 000
Water Tower (steel)	50 000
Auxiliary Structures[c]	17 975
Service Equipment Building	110 000
Site Restoration	99 750
Subtotal	2 308 125
Contingency (25%)	577 031
Total	2 885 156

(a) Estimates given in June 1981 dollars.
(b) Includes the Mockup Reactor area.
(c) See Subsection L.3.2.12 for details.

together with a cost breakdown for each structure. In all cases, the cost estimates are subject to a judgement factor as to the difficulty of the work; however, they are believed to represent a reasonable "ballpark" budget figure.

TABLE L.3-4. Estimated Quantities of Structural Material in the Reference Test Reactor Facility

Structure	Concrete (m^3)	Rebar (Mg)	Structure Steel (Mg)
Reactor Building[a]	2554	353	542
Hot Laboratory Building	3395	276	45
Primary Pump House	1193	83	54
Fan House[b]	272	19	16
Waste Handling Building	191	13	20
Service Equipment Building	482	33	94
Office and Laboratory Building	382	26	110
Hot Retention Area	325	22.5	-0-
Cold Retention Basins	365	12.6	41
Cooling Tower	53	5	9
Water Tower	76	5	108
Utility Tunnels	347	48	93
Auxiliary Structures[c]	57	4	12
Totals	9792	900.1	1144

(a) Includes the Reactor Containment Vessel and the Mockup Reactor.
(b) Includes the 30.5-m-high stack.
(c) See Subsection L.3.1.12 for details.

L.3.2.1 Reactor Building and Primary Containment Vessel

Before the polar crane is removed, it is used to remove any equipment and material in the way of the subsequent operations.

A track jack is used to drill vertical holes on all accessible quadrant and canal (Q&C) walls and other interior walls from the first-floor elevation, including drilling down through the upper part of the biological shield from the top, as access permits. The walls of the biological shield are drilled both from scaffolding and from the sub-basement floor. The drilling progresses under the reactor tank above the lead shield. The subpile room is drilled to 0.15 m below its base. Grating, rails, and steel stairs are removed when they

are no longer needed. The track jack is removed and explosive charges are loaded and fired to sever the components. After each blast, the rubble is picked up and the exposed rebar is cut and removed.

Sequentially, all Q&C walls, interior walls, and the remaining core area are charged with explosives and fired. The containment vessel (CV) dome is cut into sections and dropped inside for subsequent removal. The CV urethane foam insulation cover is removed by scraping along cut lines. If the insulation on the underside interferes with burning, it is jetted-off from below with a fire hose or scraped off, while working from the vantage level of the circular crane. This work is coordinated with the total Reactor Building (RB) demolition project to ensure that all of the explosives work inside the RB is completed while the CV steel liner walls are still in place to stop rock fly. The CV steel liner walls are then cut in strips and lifted out. The walls are cut as far down into the subbasement as possible. The remainder of the CV walls below this elevation are abandoned in place.

After the heavy vertical interior walls of the RB have been drilled, loaded, and shot (with the shell of the building in place to provide rock-fly shielding), the RB is demolished using crane and ball, augmented by oxyacetylene torch cutting of the structural steel and/or by using a hydraulic backhoe with grapple. Cleanup is accomplished with a crane with a 0.76- to 1.15-m^3 clamshell and/or by the hydraulic backhoe with grapple, either of which can separate the steel rebar and other unsuitable fill material. A large amount of concrete lies below demolition grade and is abandoned in place. Excess concrete rubble can be used for fill under 1 m below grade.

The summary cost estimate for this task follows:

• 2554 m^3 reinforced steel @ \$261.55/m^3, including 353 Mg contained rebar	\$668,000
• 542 Mg structural steel @ \$221.56/mg	\$120,000
Total	\$788,000

L.3.2.2 Hot Laboratory Building

The existing Hot Laboratory Building (HLB) cranes are used to position drilling machinery and internal structures and cover slabs are used to support the drilling machinery. The cranes are removed after the drilling is done and the drilling equipment is lifted out.

After the final radiation survey, fragmentation blasting and balling begin. Offset drilling patterns are used to shatter the concrete. Selected footings are drilled and shot while the building shell is still intact to provide cover and to expedite final cleanup. As much interior blasting as possible is done before the outside shell of the HLB is removed. Outside walls and concrete slab roofs are broken by crane and ball. For maximum economy, the resulting clean rubble is left in place below the 1 m level as fill material. Any excess is loaded out with tractor loader and dump trucks.

The summary cost estimate for this task follows:

• 3395 m^3 reinforced concrete @ $196/$m^3$, including 275 Mg contained rebar	$665,400
• 45 Mg structural steel @ $111/Mg	$ 5,000
Total	$670,400

L.3.2.3 Primary Pump House

The existing structure and cover slabs are used to support the drilling equipment. After the final radiation survey, fragmentation blasting and balling begin. Offset drilling patterns are used to shatter the concrete for loading out. The inside walls are shot while the building shell is still intact to provide cover and to expedite final cleanup. The outside walls and concrete slab roofs are broken by crane. For maximum economy, the clean concrete rubble is left in place below the -1-m level as fill material. The excess rubble is loaded out with a tractor loader.

The summary cost estimate for this task follows:

• 1193 m^3 reinforced concrete @ $130.78/$m^3$, including 83 Mg contained rebar	$156,000
• 54 Mg structural steel @ $111.11/Mg	$ 6,000
Total	$162,000

L.3.2.4 Fan House and Stack

The steel frame structure of the Fan House is torn apart with a crawler backhoe equipped with a grapple. This method is augmented with oxyacetylene torch cutting equipment as required. The debris and scrap metal are removed with the grapple. A crane and ball are used to break up the deionizer room concrete, Fan House floor, and basement walls.

A crane is used to position a chocker about two-thirds of the way up the 30.5-m-high stack. With the crane holding the stack, the base bolts are cut, and the stack is lifted and lowered to the ground. The stack is assumed to require disposal as radioactively contaminated debris. Explosives are used to shatter the stack base and foundation.

The summary cost estimate for this task follows:

• $272\ m^3$ reinforced concrete @ $\$130.80/m^3$, including 19 Mg contained rebar		\$35,600
• $108\ m^2$ floors and roof @ $\$48.61/m^2$[(a)]		\$ 5,250
• Stack		\$ 1,600
	Total	\$42,450

L.3.2.5 Waste Handling Building, Office and Laboratory Building, and Service Equipment Building

Procedures for demolishing the Waste Handling Building, the Office and Laboratory Building, and the Service Equipment Building are similar to those used for the Fan House. A crawler backhoe equipped with a grapple is the most effective way of removing the steel and frame structures. This is augmented as necessary with oxyacetylene torch cutting equipment. A crane and ball are used to break the concrete floors, inside walls, and outside walls down to 1 m below grade. In addition, explosives are used as necessary to break the foundation walls below grade while the structures themselves are still in place to contain rock fly.

(a) The unit cost can vary widely depending on the construction of the building.

The summary cost estimates for these buildings follow:

1. <u>Waste Handling Building</u>

• 191 m^3 reinforced concrete @ \$130.80/m^3, including 13 Mg contained rebar	\$25,000
• 132 m^2 mill construction (6.1-m-high highbay) @ \$64.59/$m^{2(a)}$; 20 Mg structural steel	\$ 8,550
Total	\$33,550

2. <u>Office and Laboratory Building</u>

• 382 m^3 reinforced concrete @ \$130.80/m^3, including 26 Mg contained rebar	\$50,000
• 752 m^2 floors and roof @ \$48.47/$m^{2(a)}$; 110 Mg structural steel	\$36,450
Total	\$86,450

3. <u>Service Equipment Building</u>

• 482 m^3 reinforced concrete @ \$130.80/m^3	\$63,000
• 971 m^2 floors and roof @ \$48.40/$m^{2(a)}$; 94 Mg structural steel	\$47,000
Total	\$110,000

L.3.2.6 Hot Retention Area

After all of the contaminated piping and valves are removed from the Hot Retention Area (HRA) concrete enclosure, the eight double-walled tanks are isolated. Because large contaminated areas are involved (both inside and outside the tanks), the demolition work is postulated to be coordinated closely with the total decommissioning plan for the HRA.

The plan for this task consists of two parts because of the physical layout of the HRA. First, access to the eight tanks within the HRA concrete enclosure is accomplished by removing soil on the south end of the HRA enclosure

(a) The unit cost can vary widely depending on the construction of the building.

and digging a wide ramp down to the enclosure wall. The south wall is opened to make the tanks accessible to the decommissioning crews so that the tanks can be sectioned and removed one at a time. Then, the liner is removed. The area is decontaminated as necessary, a final radiation survey is performed, and demolition of the concrete enclosure is accomplished. Second, the four smaller contaminated hold tanks just north of the HRA are unearthed and removed for disposal as radioactive debris. Because of their size the smaller tanks are postulated to be used as containers for other contaminated waste material. A hydraulic hammer mounted on a backhoe is used to break the south wall into rubble. This procedure allows the concrete enclosure to be easily converted to a confinement mode by simply sealing the south end of the enclosure, should this be desired at any time during the tanks or lining removal procedures. The tanks and linings are sectioned and removed. The sump pumps and the sumps enclosures are removed as contaminated debris. Hot spots in concrete are removed and a final radiation survey is performed.

Demolition continues with the pulverizing of the concrete enclosure by use of a crane and ball. The columns are broken and left for fill material. The concrete floor is broken for percolation and left in place. The perimeter walls are broken to 1 m below grade, rebar is burned off, and the resulting rubble is left for fill material. Rubble and earth are used as backfill and the area is covered with topsoil and reseeded.

A backhoe is used to expose the four holdup tanks buried north of the HRA enclosure. The contaminated piping is disconnected from the tanks and the tanks are capped or plugged and the piping is removed. Excavation around the tanks is completed as necessary and each tank is lifted out and positioned on a truck trailer. This task is similar to removing tanks at a gasoline service station, but requires special handling. There is no salvage value involved because the tanks are contaminated.

The summary cost estimate for this task follows:

• 325 m^3 reinforced concrete @ \$196.20 m^3, including 22.5 Mg contained rebar	\$63,750
• 573 m^3 soil (uncontaminated) remove and backfill @ \$5.23/ m^3	\$ 3,000
• Remove four holdup tanks @ \$4,000/ea.	\$16,000
Total	\$82,750

L.3.2.7 Cold Retention Area

The roof of each of the two Cold Retention Area (CRA) basins is removed. They are cut in sections, lifted out with a crane, and disposed of by the demolition contractor. The pump and plastic liner (including the sump area) in each basin is assumed to be contaminated. They are removed and packaged for burial by the decommissioning crews. Hot spots in concrete are removed and a final radiation survey is performed.

The sloping walls of the basin(s) are broken from the bottom up. The vertical sides are broken into rubble with a crane and ball and the rubble is pushed into the bottom of the pit. The pit is backfilled with rubble and soil. Topsoil is added and the area is reseeded.

The summary cost estimate for this task follows:

• 96 m^3 concrete @ \$65.40/m^3 (concrete to 1 m below grade), including 12.6 Mg contained rebar	\$ 6,300
• 269 m^3 concrete @ \$39.25/m^3 (concrete below grade)	\$10,500
• 557 m^2 roofing material @ \$48.47/m^2, including 41 Mg structural steel	\$27,000
Total	\$43,800

L.3.2.8 Emergency Retention Basin

A 27.4-m-wide approach is excavated on the north side toward the basin. Without disturbing the contaminated soil inside the basin, a cut is made into the top of the berm and on through to the basin.

A backhoe or telescoping boom grader equipped with a wide grading bucket and/or a mold board is used to pull the contaminated soil off the sloping sides (as necessary) and off the bottom of the basin (0.23 m deep) into piles for subsequent loading in burial packages. Loading is accomplished either directly into burial packages, or if stricter control is needed, an enclosed hopper is added to the loadout technique. A final radiation survey is performed after all contaminated material is packaged and all buried piping is disposed of.

It is estimated that approximately 1070 m^3 of contaminated soil is removed and packaged for disposal. Enough uncontaminated soil (about 9000 m^3) is left in the berm surrounding the basin to provide adequate fill material for regrading the entire Emergency Retention Basin site.

The summary cost estimate for this task is $20,000 for handling the uncontaminated fill and reseeding the area. The cost of scraping, loading, and disposal of the contaminated soil is estimated separately for each decommissioning alternative.

L.3.2.9 Cooling Tower

The wooden Cooling Tower (CT) is disassembled piecemeal with a crane and clamshell or smashed down with a ball. The small amount of steel scrap and pipe can be segregated during disassembly or removed from the ashes. The wood rubble is burned onsite (after appropriate permits are obtained and safety precautions are taken).

Burning is done in the early stages of demolition to benefit from the fire protection provided by the water pressure of the nearby Water Tower. In addition, the CT is assumed to be gone when the Water Tower is pulled over (see Subsection L.3.2.10). A controlled fire is maintained in the CT retention basin, fed by the clamshell.

After cleaning out the ash and metal, the uncontaminated basin and foundation concrete are broken into rubble and disposed of onsite. The summary estimate for this task is $60,000.

L.3.2.10 Water Tower

While it is possible to cut the water tanks and tower in sections and lower them down (as would be necessary in closer quarters), it is postulated instead that the tower is pulled over. This is accomplished safely in the following manner:

1. A 31.75-mm steel cable choker is secured around the central pipe stem at the platform just below the upper tank. The cables and rigging are raised by using a block and tackle. A 31.75-mm steel choker that will reach to the ground is attached using a shackle. This cable is attached to a winch line of equal or greater size on a D-8 size tractor. Using an unmanned D-8 tractor, a snatch block is anchored in line with the direction of fall but beyond reach of the tower. The winch line is run through the snatch block and the pulling tractor is positioned beyond the reach of the tower and to the side. Workers are cleared of the tower before any tension is put on the line.
2. A 2.4-m section is cut out of the leg in the direction the tower is to fall. The pipe leg is cut just above the base guy wires and again about 2.4-m above that.
3. Near the ground, a 0.3-m notch is cut out of the front (fall) side of the center column almost back to the center point. On the back side, a line is cut almost halfway around.
4. The anchor bolts of the back leg are cut (opposite the direction of fall) so that the base plate and guy wires will lift up with the column. In addition, the back leg is cracked with a sledge to ensure that it is free.
5. The area is cleared, particularly in the direction of fall and beyond the initial impact point.
6. Tension is applied with the winch line and the tower is pulled over.

A considerable amount of cutting is done after the tower is felled. The uncontaminated steel is salvagable for scrap. The tower footings are drilled, charged with explosives, and fired to fragment the concrete.

The summary cost estimate for this task is \$50,000 based on the work involved plus the responsibility. The total amount of steel and concrete are estimated at 108 Mg and 76 m^3 (including 5 Mg contained rebar), respectively.

L.3.2.11 Demolition of the Utility Tunnels

Soil is excavated to provide access to the utility tunnels using a backhoe. Concrete slabs are cut and removed. The estimated length of each tunnel, together with the top surface area of each tunnel, is given in Table L.3-5. Tunnel walls are broken down about 0.9 m. Uncontaminated piping is removed and the bottom of each tunnel is broken to provide for percolation.

TABLE L.3-5. Utility Tunnels Length and Area Parameters Used for the Demolition Estimate

Utility Tunnel	Estimated Tunnel Length (m)	Estimated Top Surface Area (m^2)
Reactor Office to Hot Laboratory	50	53
Reactor to Assembly and Test Building (to fence line)	31	116
Reactor to Service Equipment Building	67	292
Service Equipment Building to Cooling Tower	21	102
Cooling Tower to Water Tank	41	97
Total	210	760

The estimate for structural materials follows:

- Concrete slab removal:
  $760\ m^2 \times 0.3048\ m =$ 232 m^3
- Walls down 0.9 m:
  $210\ m \times 2 \times 0.9\ m \times 0.3048\ m =$ 115 m^3

  Total 347 m^3

The summary cost estimate for this task follows:

• 347 m^3 concrete @ $131.40/$m^3$, including 48 Mg contained rebar	$45,600
• 93 Mg piping and structural steel @ $165.00/Mg	$15,400
Total	$61,000

L.3.2.12 Auxiliary Structures

The auxiliary structures are demolished by conventional means either with a tractor loader or backhoe with grapple, with concrete broken up by use of a crane and wrecking ball. The cost breakdown and estimates for structural materials follow:

• Security Building:		
97 m^2 floor and room @ $16.15 m^2		$1,600
• Gas Services Building:		
93 m^2 floor and roof @ $16.15/$m^2$		$1,500
• Compressor Building:		
390 m^2 floor and roof @ $16.15/$m^2$		$6,300
• Weather Tower:		
9 m^2 (lump sum)		$ 500
• Effluent Monitoring Station:		
Building 12 m^2 @ $16.15/$m^2$	$ 200	$2,200
Effluent channel and basin 122 m^2 @ $16.15/$m^2$	$2000	
• Electrical Substation:		
115 m^3 floor and roof @ $16.15/$m^2$		$1,875
• Sludge Settling Basins:		$4,000
61 m^3 concrete @ $65.60/$m^3$		
	Total	$17,975

L.3.2.13 Site Restoration

All of the subgrade portions of the reference test reactor structures are backfilled. Concrete rubble from the demolition of the structures is used as necessary to backfill up to a level 1 m below grade. To restore the site, the top meter is backfilled with topsoil and seeded with native ground cover; costs are given in Table L.3-6.

TABLE L.3-6. Estimated Cost of Site Restoration for the Reference Test Facility

Activity	Material Required (m^3)	Area (m^2)	Unit Cost ($\$/m^3$)	Cost ($)
Loading and Hauling	10 500		5.50	57 750
Placing, Grading, and Seeding		21 000	2.00	42 000
Total				99 750

REFERENCE

1. J. M. McFarland, Report on Representative Cost Estimates for Demolition of Structures at a Reference Research Reactor Site and at a Reference Test Reactor Site, PNL-3892, Prepared for Pacific Northwest Laboratory by McFarland Wrecking Corporation, Seattle, Washington, 98108, June 1981.

APPENDIX M

COST ESTIMATING BASES

The cost information developed in this study is based on unit cost data presented in this appendix. Categories for which basic cost data are given include: labor, waste packaging, transportation, waste disposal, and special equipment. The data presented are all late-1980 or early-1981 costs.

The unit cost data presented can be used to develop cost estimates for other decommissioning projects. However, to ensure the applicability of the estimate to any specific situation, the data should be carefully examined and adjusted as necessary.

M.1. LABOR COSTS

Labor cost data for typical decomissioning staff positions are given in Table M.1-1. The 1978 data base used in earlier LWR decommissioning studies[1,2] has been adjusted by a factor of 1.19, based on building trades labor cost trends reported in the Handy-Whitman Index.[3] The base pay rates in Table M.1-1 are increased by 70% for nonunion employees and by 50% for union employees to account for owner costs such as fringe benefits, taxes, and insurance.

Labor costs shown in Table M.1-1 are representative of average labor costs rather than labor costs for a particular decommissioning project at a given location. A recent decommissioning costs study[4] estimates that regional labor costs can deviate by as much as 17% from the national average. Costs at individual locations might deviate even more. In addition, the owner cost will depend on the values used to estimate fringe benefits, taxes, insurance, and other owner overhead expenses.

M.2. RADIOACTIVE WASTE PACKAGING COSTS

The costs of packaging radioactive waste material prior to shipment to a shallow-land burial site include the shipping container cost, the cost of additional shielding provided by overpacks and casks, and the cost of a

TABLE M.1-1. Decommissioning Labor Cost Data[a]

Position	Base Pay ($/yr)	Assumed Overhead Rate (%)	Cost ($/yr)	Reference
Management & Support Staff				
Decommissioning Superintendent	52 000	70	89 100	b
Decommissioning Engineer	44 700	70	76 000	b
Asst. Decommissioning Engineer	30 800	70	52 400	b
Secretary	16 100	50	24 200	c
Clerk	16 100	50	24 200	c
Health & Safety Supervisor	35 200	70	59 800	c
Health Physicist	27 600	70	46 900	c
Industrial Safety Specialist	30 800	70	52 400	c
Radioactive Shipment Specialist	23 100	70	39 300	c
Procurement Specialist	23 100	70	39 300	c
Contracts & Accounting Specialist	27 700	70	47 100	b
Accounting Clerk	16 100	50	24 200	c
Security Supervisor	32 900	70	55 900	c
Security Shift Supervisor	21 400	70	36 400	c
Security Patrolman	16 900	50	25 400	d
Quality Assurance Supervisor	30 700	70	52 200	c
Quality Assurance Engineer	27 600	70	46 900	c
Quality Assurance Technician	18 500	50	27 800	c
Control Room Operator	23 000	50	34 500	c
Tool Crib Attendant	18 500	50	27 800	e
Protective Equipment Attendant	18 500	50	27 800	c
Consultant	100 000	--	100 000	c
Decommissioning Workers				
Shift Engineer	30 700	70	52 200	c
Craft Supervisor	27 600	70	46 900	c
Craftsman	21 400	50	32 100	e
Crew Leader	26 100	70	44 400	e
Utility Operator	21 400	50	32 100	e
Laborer	20 600	50	30 900	d
Health Physics Technician	20 000	50	30 000	e
Senior Health Physicist	26 200	50	39 300	c

(a) Adjusted to early 1981 (see Section M.1 for details).
(b) U.S. Dept. of Labor, Bureau of Labor Statistics, Bulletin March 1975.
(c) Study estimate.
(d) R. S. Means Co., Building Construction Cost Data - 1975, 33rd Edition.
(e) Hanford Atomic Metal Trades Council Pay Scales.

solidifying or dewatering agent for radioactive liquids or wet wastes. These costs are discussed in the following subsections.

M.2.1 Shipping Container Costs

The shipping containers assumed to be used for packaging radioactive materials for disposal are listed in Table M.2-1, together with a brief description, the displaced burial volume, the particular application, and the unit cost, for each type of container. Because of increases in labor and material costs, some container costs have increased significantly since 1978. Suppliers and users of these containers were consulted to obtain 1981 cost information.

M.2.2 Overpack and Cask Charges

Some packaged wastes with high surface dose rates require transport to a burial site in reusable overpacks or shielded casks. In general, it is more economical to rent such containers than to purchase them, especially the larger ones or those used infrequently or for a short time period. The overpacks and casks assumed for transportation of high activity or high surface dose rate decommissioning wastes are listed in Table M.2-2, together with physical characteristics and estimated rental charges.

M.2.3. Solidifying Agent Costs

The solidifying agents assumed to be used for packaging of wet solid and liquid wastes are listed in Table M.2-3, together with their respective costs.

In this study, it is assumed that all concentrator bottoms, drain-decontamination solutions, and ion-exchange resins are mixed with a solidifying agent and poured into standard steel drums. The assumed solidifying agent is cement blended with varying amounts of sodium silicate and proprietary leach inhibitors. The agent is added at the rate of 498 kg/m^3 of bead resin. The cost of this solidifying agent per container (0.21-m^3-capacity) is estimated at $14.

M.3. TRANSPORTATION COSTS

Most radioactive wastes from decomissioning operations are assumed to be transported to a disposal site by exclusive-use truck. Transportation costs

TABLE M.2-1. Unit Costs of Shipping Containers for Packaging Radioactive Materials

Description	Burial Volume (m^3)	Applications	Estimated Unit Cost ($)[a]
Steel Cask Liner 0.63 m O.D. x 1.02 m high, 150/kg empty	0.33	Shallow-Land Burial of Activated Reactor Components	500
Fiberglassed Plywood Box 1.2 m x 1.2 m x 2.4 m, 175 kg empty	3.64	Low Specific Activity Materials, Piping, Concrete, Contaminated Components	400
Fiberglassed Plywood Box Specially Fabricated	Variable	Low Specific Activity Materials Over-sized or Extra Heavy	$\sim 40/m^2$ of surface
Steel Cask Liner 1.38 m O.D. x 1.9 m high, 680 kg empty	2.84	Solidified Wet Solid Wastes	2000
Standard Steel Drum 0.21 m^3, 23 kg empty	0.21	Compacted Dry Solid Wastes	30
Stainless Steel Canister 0.76 m O.D. x 3.05 m high	1.38	Alternate Deep Geologic Disposal of Highly Activated Reactor Components	6000
Small Steel Drum 0.11 m^3, 18 kg empty	0.11	Compacted Dry Solid Wastes	20
Polyethylene Drum Liner	(b)	Low Specific Activity Materials	1

(a) Adjusted to early 1981.
(b) Included in outer steel drum, no added burial volume.

TABLE M.2-2. Rental Charges for Reusable Casks and Overpacks

Description	Empty Weight (kg)	Daily Rental ($)
1.24 m O.D. x 1.56 m high 150-mm Pb thickness (B3 cask)	9 300	225
1.63 m O.D. x 2.34 m high 100-mm Pb thickness (NP 100x4)	16 330	300
1.95 m O.D. x 1.04 m high 50-mm Pb thickness (7D-3L cask)	7 000	225
2.44 m x 2.44 m x 6.10 m double-walled steel with fire-resistant insulation (Super Tiger)	6 800	300

TABLE M.2-3. Solidifying Agent Costs

Item	Estimated Unit Cost ($)
Cement (45-kg bag)	6/bag
Diatomaceous Earth (23-kg bag)	12/bag

are based on the published rates of a carrier licensed to transport radioactive materials.[5] To compute transportation costs, the following assumptions are made:

- One-way shipping distance is 800 km.[a]

(a) This assumed shipping distance is consistent with that used in previous decommissioning studies in this series. Because of burial ground closures during the past 3 years, the average shipping distance from waste generation sites to disposal sites has significantly increased. However, to enable comparisons with the results of earlier decommissioning studies, the 800 km shipping distance is retained in this study. Regional waste disposal compacts, currently under consideration, may result in the opening of additional shallow-land burial sites and the consequent reduction in the average shipping distance from waste generation sites to disposal sites.

- Shipments not requiring casks or overpacks are separate one-way shipments destined for west of the Mississippi River (the highest rate category). Cask or overpack shipments are continuous excursion round-trips.
- A fuel surcharge is levied at a rate of 18%.[b]
- Where applicable, overweight charges are computed at the rate for the state of Washington.

In general, oversize shipments are used sparingly in this study because the large majority of waste shipments are packaged in containers that can be shipped as legal-size shipments.

A trend that could add significantly to future nuclear transportation costs is the requirement by state and local governments for permits in advance of each radioactive material shipment through their jurisdiction. A major carrier plans to charge its customers $25, plus the cost of the permit, for each such permit required.[6] In future, these permit charges could be substantial for long-distance shipments. No permit charges are included in the transportation cost estimates of this study.

The rate schedule for truck shipments of legal size and weight that forms the basis for transportation costs in this study is shown in Table M.3-1. The gross vehicle weight (GVW) for legal weight shipments by truck is assumed to be less than 21.32 Mg. The maximum allowed GVW is assumed to be 38.55 Mg.[5] Overweight charges by states vary widely. The additional charges assumed in this study to be levied by the carrier and the state for overweight shipments are shown in Table M.3-2.

Example shipping costs, calculated for several different payloads and for one-way and round-trip shipments, are shown in Table M.3-3. For a one-way, 800-km shipment, the base charge is that shown in Column 2 of Table M.3-1. To this charge must be added the 18% fuel surcharge and any applicable overweight charges shown in Table M.3-2.

(b) The fuel surcharge rate is subject to change as fuel prices increase or decrease. The 18% rate was in effect as of February 12, 1981.

TABLE M.3-1. Transportation Rates for Legal-Size and -Weight Shipments (effective August 15, 1980)[a,b,c]

Kilometers One-Way (Not Over)	Rate in Cents Per Kilometer Column 1[d]	Column 2[e]	Column 3[f]	Kilometers One-Way (Not Over)	Rate in Cents per Kilometer Column 1[d]	Column 2[e]	Column 3[f]
160	233	244	168	1200	86	103	71
200	214	226	155	1280	82	100	71
240	196	209	143	1360	81	99	71
280	179	192	133	1440	80	98	71
320	155	169	121	1520	79	97	71
360	147	162	115	1600	77	95	71
400	141	156	108	1760	77	94	71
440	134	150	101	1920	77	94	71
480	128	144	96	2080	77	93	71
520	125	141	91	2240	77	92	71
560	121	137	88	2400	77	92	71
600	116	132	84	2560	77	91	71
640	111	128	82	2720	77	91	71
680	108	124	80	2880	77	90	71
720	102	119	78	3040	77	89	71
760	100	117	76	3200	77	89	71
800	96	114	75	3360	77	88	71
880	94	111	73	3520	77	88	71
960	92	109	71	3680	77	87	71
1040	89	106	71	3840	77	86	71
1120	87	104	71	4000 & Beyond	77	86	71

(a) Reproduced from the published rates of a carrier[5] licensed to transport radioactive materials.
(b) Effective August 15, 1980.
(c) Rates do not include a fuel surcharge, which amounted to 18% of the base rate as of February 13, 1981.
(d) Column 1 rates applicable to one-way shipments having a destination east of the Mississippi River.
(e) Column 2 rates applicable to one-way shipments having a destination west of the Mississippi River.
(f) Column 3 rates apply to continuous excursion moves in which a subsequent shipment is made available to the carrier within 24 hours after arrival at the point of loading or unloading.

TABLE M.3-2. Additional Charges when Gross Vehicle Weight Exceeds 21.32 Mg[a,b]

Gross Vehicle Weight (Mg)	State Surcharge ($)	Carrier Surcharge ($)	Total Overweight Surcharge ($)
21.32 to 23.12	10 + 0.031/km	0.131/km	10 + 0.162/km
23.13 to 25.84	10 + 0.062/km	0.131/km	10 + 0.193/km
25.85 to 28.56	10 + 0.093/km	0.131/km	10 + 0.224/km
28.57 to 31.28	10 + 0.155/km	0.131/km	10 + 0.286/km
31.29 to 34.00	10 + 0.218/km	0.131/km	10 + 0.349/km
34.01 to 36.72	10 + 0.280/km	0.131/km	10 + 0.411/km
36.73 to 38.55	10 + 0.373/km	0.131/km	10 + 0.504/km

(a) State surcharge is based on rates for the state of Washington.
(b) Carrier surcharge is based on the published rates[5] of a carrier licensed to transport radioactive materials.

TABLE M.3-3. Example Shipping Costs of Truck Shipments

Status	Number of Drivers	Payload (Mg)	GVW (Mg)	Cost ($)
Legal weight, one-way[a]	1	8.61	21.31	1,076
Overweight, one-way[a]	1	19.95	32.65	1,365
Overweight, one-way[a]	1	25.85	38.55	1,489
Overweight, round-trip[b]	2	19.95	32.65	2,133
Overweight, round-trip[b]	2	25.85	38.55	2,381

(a) 800-km distance.
(b) Shipments involving casks or overpacks, with overweight charges applicable both directions. Charges computed on the basis of two 800-km trips.

Casks and overpacks are assumed to be picked up loaded at the site of decommissioning operations, delivered to the disposal site to be unloaded, and then returned to the decommissioning site. Thus, each 1600 km round-trip consists of two 800-km one-way moves, with charges based on the continuous-excursion rates shown in Column 3 of Table M.3-1. From the reference rate schedule, the

basic charge for the round trip is $1200. With the additional 18% fuel surcharge, this is increased to $1416. Applicable overweight charges must also be added. To ensure rapid turnaround on these shipments and to minimize cask rental charges, a second driver is assumed to be used, costing an additional $0.093/km.[5]

M.4. WASTE DISPOSAL COSTS

A basic assumption of this study is that all radioactive wastes from decommissioning operations can be disposed of by burial at a commercial shallow-land burial facility. Disposal requirements for highly radioactive and long-lived components from decommissioning operations are not yet defined. A requirement for deep geologic disposal of these materials would certainly increase the cost of disposal. Since a deep geologic disposal facility does not now exist, interim storage of wastes destined for geologic disposal might be required. The costs of interim disposal are also not well-defined, but are assumed to be comparable to those for shallow-land burial.

The shallow-land burial costs used in this study are based on a November 1980 price list from U.S. Ecology, Inc.,[7] which operates burial sites at Richland, Washington, and Beatty, Nevada. These prices are comparable to those charged by Chem-Nuclear Services, Inc.,[8] at their Barnwell, South Carolina, disposal site. Burial ground charges are shown in Table M.4-1.

M.5. EQUIPMENT COSTS

Equipment costs from the 1978 data base have been reviewed and updated as appropriate to reflect 1981 costs. Costs of construction-type items (hoists, cranes, lifts, etc.) are based on costs shown in the 1981 catalog of building construction costs published by R. S. Means Company.[9] Equipment costs are shown in Table M.5-1.

TABLE M.4-1. Commercial Shallow-Land Burial Charges[a,b]

I. DISPOSAL CHARGES, NON-TRU WASTE

A. Steel Drums, Wood Boxes

Container Surface Dose Rate (R/hr)[c]	Price Per Unit Volume ($/$m^3$)
0.00 to 0.20	307.20
0.201 to 1.00	335.45
1.01 to 2.00	376.05
2.01 to 5.00	459.05
5.01 to 10.00	542.00
10.01 to 20.00	702.65
20.01 to 40.00	870.40
40.01 to 60.00	1332.95
60.01 to 80.00	1601.30
80.01 to 100.00	1765.50
>100	by request

B. Disposable Liners

Container Surface Dose Rate (R/hr)(c)	Surcharge per Liner ($)	Price Per Unit Volume ($/m3)
0.01 to 0.20	None	307.20
0.201 to 1.00	119.00	307.20
1.01 to 2.00	292.00	307.20
2.01 to 5.00	411.00	307.20
5.01 to 10.00	594.00	307.20
10.01 to 20.00	758.00	307.20
20.01 to 40.00	941.00	307.20
40.01 to 60.00	1116.00	307.20
60.01 to 80.00	1288.00	307.20
80.01 to 100.00	1463.00	307.20
>100	by request	by request

TABLE M.4-1. (contd)

II. SURCHARGES

A.	State of Washington Surcharge:	$\$10.60/m^3$
B.	Curie Surcharge (per load):	
	Less than 100 curies	No charge
	101 to 300 curies	$660.00
	301 to License Limits (i.e., 50,000 Ci)	$660.00 + $0.09/Ci
C.	Handling Surcharge:	
	0 to 4.54 Mg	No charge
	>4.54 Mg	$87.50 + $0.044/kg over 4.54 Mg
	Special Equipment	By special quotation
D.	Cask Handling Fee	$335.00 per cask

(a) Reproduced from the published rates(7) of a licensed burial ground operator.
(b) Prices effective November 17, 1980.
(c) Maximum reading at container surface, irrespective of physical size or configuration.

TABLE M.5-1. Costs of Special Tools and Equipment

Item	Estimated Unit Cost ($ thousands)
Underwater Plasma-Arc Torch	20
Underwater Oxyacetylene Torch	5
Portable Plasma-Arc Torch	20
Portable Oxyacetylene Torch	1
Guillotine Pipe Saw	4
Power-Operated Reciprocating Hacksaw	0.8
Nibbler	1[a]
Closed Circuit TV System	15[b]
Submersible Pump with Disposable Filter	2
High-Pressure Water Jet	20
Powered Floor Scrubber	0.3
Wet-Dry Vaccum Cleaner (HEPA Filtered)	1-5[c]
Supplied-Air Plastic Suit	0.05
Power-Operated Mobile Manlift	40
9100-kg Mobile Hydraulic Crane	28
9100-kg Forklift	28
Concrete Drill with HEPA Filtered Dust Collection System	2
Concrete Surface Spaller	5
Front-End Loader (Light Duty)	20
Portable Filtered Ventilation Enclosure	2-10[c]
Filtered-Exhaust Fan Unit	5
Polyurethane Foam Generator	5
Paint Sprayer	0.5-1[d]
HEPA Filter	0.2
Roughing Filter	0.1

(a) Remote operating extensions would add to this cost.
(b) Estimated for modifications of existing systems.
(c) Depends on size and capacity.
(d) Depends on capacity of system.

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APPENDIX N

PUBLIC RADIOLOGICAL SAFETY ASSESSMENT DETAILS

The purpose of this appendix is to quantify the parameters and define the methodology for estimating the public radiological safety impacts of decommissioning the reference research and test (R&T) reactors. The impacts of decommissioning on the safety of the public are principally related to the hazards associated with the atmospheric release of radioactive materials during decommissioning, both from planned tasks and from accidents. For each of three separate decommissioning alternatives analyzed, atmospheric releases of radioactivity are calculated for the decommissioning tasks assumed to be performed at the reference R&T reactors. Details for estimating the public radiation exposure resulting from waste shipments are also included in this appendix. The scenarios are analyzed using realistically maximized parameters. These parameters are carefully chosen to cover a broad spectrum of conditions. Atmospheric release mechanisms are quantified using either measured release factors or information about known physical or chemical behaviors under the postulated conditions. The estimated routine radionuclide releases to the atmosphere are tabulated in the chronological order of the decommissioning tasks for each building at each reactor for each decommissioning alternative. While detailed probabilistic analysis of the postulated accident scenarios is not within the scope of this study, selected generic operational accident categories are considered.

The following sections contain detailed discussions of the technical approach and public safety details for DECON, preparations for safe storage, ENTOMB, and waste transportation. A summary of this information is given in Sections 12.2 and 12.4 of Chapter 12 (Volume 1).

N.1 TECHNICAL APPROACH

To estimate the public safety impacts of the release of radioactivity to the atmosphere from decommissioning tasks, the following basic assumptions are made:

1. The facility has fulfilled existing criteria for the containment of radionuclides for accidents and natural phenomena during its operational lifetime.

2. To ensure proper air flow for the decommissioning workers, and to protect against uncontrolled atmospheric releases of radioactivity, HEPA filters are assumed to be installed in the research reactor. These filters are tested in-place on a regular basis. The measured particle collection efficiency of these HEPA filters is 99.95%.[1] Airborne releases of radioactivity are assumed to take place at ground level for atmospheric dispersion calculations, and are assumed to pass through a single HEPA filter system with an assumed transmission factor of 5×10^{-4}.[1]

3. The containment integrity of the buildings at the reference R&T reactors is assumed to be breached only after the radioactive contamination levels are at the accepted levels for unrestricted use.

4. In areas with high levels of radioactive contamination, a temporarily installed "greenhouse," or contamination control envelope, is assumed to be used. The contamination control envelope is assumed to have a HEPA filter with a transmission factor of 5×10^{-4} to reduce the airborne radionuclide concentrations in the R&T reactor buildings from selected decommissioning tasks.

5. The leakage rate from a contamination control envelope is assumed to be 10% for all tasks involving its use. This assumed leakage is used as a maximized value to account for routine ruptures or failures of the temporary structures.

6. The average airflow rates assumed in this study are: 3000 m^3/hr for all contamination control envelopes, and 1.3×10^4 m^3/hr for all other buildings at the R&T reactors.

7. Unneeded hazardous chemicals and equipment are removed after the reference R&T reactors are shut down. Decontamination agents such as phosphoric acid, ethylenediaminetetraacetic acid (EDTA), oxalic acid, and citric acid are available in the plant. Unneeded ion exchange resins and resin beds are removed.

8. The reference radionuclide inventories referenced in this appendix refer to the inventories defined in Appendix E. These inventories characterize activated structural materials, as well as surface contamination.

9. All chemical or manual decontamination tasks are assumed to remove 90% of the surface contamination.

10. The airborne concentrations of dust or liquid droplets are assumed to be 1×10^{-2} g/m^3, equal to the concentrations observed at the Elk River reactor decommissioning.[2,3] For tasks involving blasting or explosions, the airborne concentrations are assumed to be a factor of 10 higher, or 1×10^{-1} g/m^3.[4]

11. All radioactive wastes shipped offsite are assumed to be shipped in accordance with Department of Transportation (DOT) regulations. Spent reactor fuel is assumed to be shipped 2400 km by truck to a reprocessor, and radioactive wastes are assumed to be shipped 800 km by truck to a shallow-land burial ground.

12. Radiation doses to the maximum-exposed individual and to the population residing within 80 km of the reference site are calculated using the environmental data and assumptions discussed in Appendix F of this study. These methods are consistent with the methods outlined in Regulatory Guide 1.109.[5]

13. For exhumation of soils in the Emergency Retention Basin at the reference test reactor, atmospheric releases are calculated by using a mechanical mixing resuspension analysis defined for waste exhumation operations at a generic low-level waste burial ground.[6] A constant resuspension rate of 1×10^{-7} sec^{-1} is assumed.[7] It is further assumed that only 10% of the waste/soil mixture is of a correct physical form and particle size to permit airborne transport.[8-10] Because of the wet nature of the soil at the reference site, the airborne concentration is assumed to be further reduced by a factor of 10.

Other assumptions relating to specific decommissioning tasks are made and discussed where they apply to the analysis of public radiation exposure.

N.2 PUBLIC SAFETY ASSESSMENT FOR DECON

The first option considered in the safety evaluation of decommissioning the reference R&T reactors is DECON. This alternative results in removal of the radioactively contaminated portions of the buildings, the equipment, and the sites so that unrestricted use by the public can be permitted.

The assessment of public safety during DECON includes a consideration of both routine tasks and postulated accidents. These tasks and accidents can generate considerable amounts of airborne radioactivity, primarily in the form of solid particles and/or suspended liquid droplets. Airborne contamination control measures are assumed to be used where they are considered necessary. The use of other appropriate controls is also considered. The inventories of radioactive contamination in the reference R&T reactors are discussed in detail in Appendix E, and are summarized in Table N.2-1.

In the following sections, the atmospheric releases from DECON are described for both routine tasks and postulated accidents. The atmospheric releases are estimated by determining the realistic maximum atmospheric release for each task or accident, and using this value whenever the task or accident is encountered, even for areas containing lower contamination levels.

N.2.1 Routine DECON Tasks

A complete discussion of the tasks for DECON at the reference R&T reactors is contained in Appendix I. The chronological sequence of DECON tasks for the reference research reactor is listed in Figure I.1-1; and for the reference test reactor in Figures I.2-1 through I.2-3. These tasks and their time requirements are listed for each building or area at the reference R&T reactors. The following subsections contain discussions of the methods used for calculating atmospheric releases of radioactivity and the public radiation doses resulting from DECON of the reference R&T reactors. Information about the routine atmospheric releases is presented in the chronological order of the DECON tasks at each reactor.

N.2.1.1 Methods for Calculating Atmospheric Releases of Radioactivity

Reference DECON tasks are defined and analyzed to determine generic airborne release mechanisms. A summary of the reference DECON tasks considered in this study is shown in Table N.2-2. Each task considered is discussed below:

TABLE N.2-1. Summary of Radioactivity Levels at the Reference R&T Reactors

Reactor Inventory/Location	Reference Radionuclide Inventory Table No.[a]	Radioactivity Levels, C_s (Ci/m^2), C_m (Ci/m^3)
Research Reactor		
Activated Vessel Internals	E.1-5 and E.1-6	$C_m = 7.5 \times 10^3$ and $C_m = 2.9 \times 10^2$
Activated Concrete	E.1-7	$C_m = 2.4 \times 10^{-3}$
Contaminated Piping	E.1-5[b]	$C_s = 7.0 \times 10^{-5}$
Hot Cell	E.1-8	$C_s = 9.6 \times 10^{-3}$
Test Reactor		
Activated Vessel Internals		
• Stainless Steel	E.2-1	$C_m = 5.3 \times 10^3$
• Aluminum	E.2-2	$C_m = 3.5 \times 10^3$
Activated Bio-Shield		
• Concrete	E.2-3	$C_m = 1.1 \times 10^0$
Contaminated Piping		
• PCWS	E.2-1[b]	$C_s = 1.4 \times 10^{-2}$
Hot Cells	E.2-10	$C_s = 4.8 \times 10^{-2}$
Contaminated Soil	E.2-11[c]	2.8×10^2 pCi/g[c]

(a) These are the table numbers in Appendix E that contain the reference radionuclide inventories.
(b) Internal corrosion product deposition is calculated in Appendix E for ^{60}Co; the internal surface contamination mixture is assumed to have the same radionuclide mixture as that shown for activated stainless steel.
(c) Soil contamination is shown in Table E.2-11 in units of pCi/g.

Radiation Survey. Radiation surveys are performed at the start of decommissioning, prior to termination of the nuclear license, and at other times during decommissioning to determine the residual levels of radioactive contamination. Radiation surveys involve crews of health physics technicians who walk

TABLE N.2-2. Safety Analysis Summary of the Reference DECON Tasks Considered in this Study

Task	Operation Description	Routine Airborne Release	Routine Atmospheric Release Calculation Summary	Postulated Hazardous Situations
Radiation Survey	Crews of Health Physics Technicians Walk Through the Plant to Locate and Characterize Residual Radioactive Contamination	Airborne Dust is Removed from Floor Surfaces	Resuspension Analysis: (Resuspension Factor) x (Surface Radioactive Concentration) x (Air Volume Covering Floor) x (Building Transmission Factor) = (Atmospheric Release)	• None Analyzed
Surface Cleaning				
1. Sweeping	Small Quantities of Loose Contamination on Floors are Removed by Manual Sweeping with Sweeping Compounds	Airborne Dust is Removed from Floor Surfaces During Sweeping	Resuspension Analysis: (Resuspension Factor) x (Surface Radioactivity Concentration) x (Air Volume Covering Floor) x (Building Transmission Factor) = (Atmospheric Release)	• Fire in Contaminated Sweeping Compound
2. Vacuuming	Small Quantities of Loose Contamination on Surfaces are Removed by Manual Vacuuming Procedures	Airborne Dust is Removed from Floor Surfaces During Vacuuming	Resuspension Analysis: (Resuspension Factor) x (Surface Radioactivity Concentration) x (Air Volume Through Vacuum Cleaner) x (Vacuum HEPA Transmission Factor) x (Building Transmission Factor) = (Atmospheric Release)	• Vacuum Bag Rupture
3. Water-Jet	Hand-Held High-Pressure Water Spray is Used to Remove Loose and Soluble Surface Contamination	Airborne Particles are Caused by High-Pressure Water Sprays	Entrainment of Contaminated Water: (Surface Contamination Removal Fraction) x (Surface Radioactive Concentration) x (Water-Jet Cleaning Rate) ÷ (Liquid Flow Rate) ÷ (Solution Density) x (Airborne Concentration) x (Building Air Flow Rate) x (Water-Jet Use Time) x (Building Transmission Factor) = (Atmospheric Release)	• Spilling Contaminated Liquids • Loss of Services
4. Hand Washing	Limited Areas Are Cleaned with Mops or Sponges	Liquids are Made Airborne as Sprays	Not Calculated	• Spilling Contaminated Liquids
Piping or Equipment Removal & Segmentation				
1. Cutting in Air	Piping and Equipment is Segmented for Offsite Disposal	Vaporized Metal is Made Airborne (Contamination Control Envelope)	Contaminated Piping or Equipment: (Length of Cut) x (Width of Cut) x (Surface Radioactivity Concentration) x (Building Transmission Factor) = (Atmospheric Release) Activated Piping or Equipment: Same as Above x (Material Thickness) ÷ (Material Density)	• Loss of Services • Oxyacetylene Explosion • Local Loss of Contamination Control

TABLE N.2-2. (Contd)

Task	Operation Description	Routine Airborne Release	Routine Atmospheric Release Calculation Summary	Postulated Hazardous Situations
2. Cutting Under Water	Piping and Equipment is Segmented for Offsite Disposal	Liquid Sprays are Made Airborne (Contamination Control Envelope)	Entrainment of Contaminated Water: (Water Radioactivity Concentration) x (Airborne Radioactivity Concentration) x (Control Envelope Airflow Rate) x (Cutting Time) x ((Control Envelope Leak Rate) x (Building HEPA Filter Transmission Factor) ÷ (Water Density) = (Atmospheric Release)	• Loss of Services • Oxyacetylene Explosion • Local Loss of Contamination Control
Concrete Removal 1. Drilling	Holes are Drilled for Insertion of Concrete Spallers or Explosive Charges	Concrete Dust is Made Airborne (Contamination Control Envelope)	Concrete Dust Generation: (Number of Holes) x (Surface Area per Hole) x (Surface Radioactivity Concentration) x (Airborne Concentration) x (Control Envelope Air Flow Rate) x (Drilling Time) x (Control Envelope Leak Rate) x (Building Filter Transmission Factor) ÷ (Volume of Holes Drilled) ÷ (Density of Concrete) = (Atmospheric Release)	• Loss of Services • Equipment Failure • Local Loss of Contamination Control
2. Spalling	Hydraulically Controlled Concrete Spallers	Concrete Dust is Made Airborne (Contamination Control Envelope)	Not Calculated	• Loss of Services • Loss of Contamination Control
3. Packaging	LNG-Powered Front-End Loader	Concrete Dust is Made Airborne (Contamination Control Envelope)	Not Calculated	• LNG Explosion
Onsite Waste Exhumation 1. Exhumation/Packaging	Crews Use Front-end Loaders, Dozers, Loading Bins, Lift Trucks, and Other Equipment As Necessary to Exhume and Package Wastes	Dust Generated During Exhumation and Packaging Operations in Open Air	Waste Dust Generation: (Resuspension Rate) x (Total Mixing Time) x (Volume of Waste) x (Specific Activity) = (Atmospheric Release)	• None analyzed

through the reference plants and who are assumed to disturb settled radioactivity. The general housekeeping practices followed prior to and during decommissioning determine the amount of loose surface contamination, which varies with location within the reference R&T reactors.

The relationship between the amount of material in the air above a surface and the contamination level of that surface has been studied experimentally.[11] This relationship is shown by Equation N.1.

$$S_f = \frac{C_a}{C_s} \tag{N.1}$$

where:

S_f • the resuspension factor, m^{-1}

C_a • the concentration of airborne radioactivity at a given location inside a building, Ci/m^3

C_s • the concentration of radioactivity on building surfaces, Ci/m^2.

Measured values of S_f vary between 1×10^{-5} and 2×10^{-3} m^{-1}, depending upon experimental conditions and measurement methods.[11-13]

The concentration of airborne radioactivity at a given location inside a building is calculated by rearranging Equation N.1 as shown in Equation N.2.

$$C_a = S_f \, C_s \tag{N.2}$$

where the terms S_f, C_s and C_a are the same as defined for Equation N.1.

The total radioactivity made airborne in a building is calculated by using an assumed average resuspension factor associated with the radiation survey of 1×10^{-4} m^{-1}, the surface radionuclide concentrations in the building, and the total volume of air contaminated during the radiation survey, as shown in Equation N.3.

$$Q_s = (1 \times 10^{-4}\ m^{-1})\ C_s\ V_s \tag{N.3}$$

where:

Q_s • the total airborne radioactivity inside a building, Ci

C_s • the surface radioactivity concentration on building surfaces, Ci/m^2

V_s • the volume of air covering the surface involved in the radiation survey (assumed to be 1 m deep), m^3.

The airborne radioactivity generation rate during the radiation survey is found by dividing the airborne radioactivity, Q_s, by the time required to perform the survey. The total radioactivity released to the atmosphere is found by multiplying the airborne radioactivity generated in each building by the HEPA filter transmission factor.

Surface Cleaning. The surface cleaning operations considered are sweeping, vacuuming, and water-jet sprays.

1. Sweeping. Loose floor contamination is assumed to be removed by annual sweeping operations with the use of sweeping compounds. The amount of airborne radioactivity generated during sweeping is determined by using a resuspension factor analysis similar to the one discussed for the radiation survey. The amount of radioactivity made airborne is highly variable and is characterized by the properties of the loose contamination and the sweeping method. Sweeping compounds are designed to lower the resuspension by binding small particles of surface contamination to larger particles of sweeping compound.

   Data from experimental studies of resuspension during sweeping are limited. One study suggests a resuspension factor of $1.9 \times 10^{-4}\ m^{-1}$.[12] For this study, a conservatively large resuspension factor of $5 \times 10^{-4}\ m^{-1}$ is used.

   This factor is 5 times larger than the resuspension factor assumed for the radiation survey. The resuspended radioactivity is assumed to be confined in the lower meter of air in the room in which the sweeping occurs. Thus, the airborne radioactivity in a building is calculated using the volume of

air disturbed by sweeping (assumed to be the volume 1 m deep above the surface area swept), the surface radioactivity concentration, and the assumed resuspension factor. The total atmospheric radioactivity released from sweeping is found by multiplying the airborne radioactivity generated in each building by the HEPA filter transmission factor.

2. Vacuuming. Vacuuming can be used as an alternative to manual sweeping for removal of loose surface contamination. Thus, certain areas of the building considered in this study are designated for vacuuming. The vacuum exhaust is assumed to be fitted with a 99.95%-efficient HEPA filter system. The airborne radioactivity resulting from vacuuming in a building is calculated by finding the product of the surface contamination level, the total area vacuumed, the assumed surface contamination removal fracton for vacuuming (0.5), and the vacuum system HEPA filter transmission factor. The total radioactivity released to the atmosphere is found by multiplying the airborne radioactivity in each building from vacuuming by the building HEPA transmission factor.

3. Water-Jet Sprays. The water-jet is a hand-held high-pressure water spray designed to remove loose surface contamination. The operating water pressure is about 3.1×10^4 kPa. The principal mechanism for the generation of airborne radioactivity from the water-jet is the suspension of liquid droplets containing contamination removed from surfaces. The water spray produces fine droplets smaller than 300 μm in diameter (about the size of the droplets found in fogs or mists). Larger droplets are broken into smaller droplets as they impact on the contaminated surfaces. Thus, there is the potential for a significant formation of small droplets with a considerable variation in the total quantity of radioactivity contained in the droplets.

   Direct data are not available to define the quantity of droplets formed. A conservative estimate is made by assuming that a sufficient quantity of droplets is generated to maintain an airborne liquid concentration of 1×10^{-2} g/m^3 with vigorous mixing in air. This is the maximum mass concentration that can be found in air velocities less than 0.046 m/s.(13)

Measurements have shown that there is fairly constant distribution of 10- to 20-μm diameter particles in a water-jet spray at a concentration of about 1×10^{-2} g/m^3, including the gross entrainment of larger particles.(14) Since particles smaller than 10μm are in the respirable-size range, they are potentially hazardous. The quantity of radioactivity found in these airborne droplets is influenced by factors such as the quantity of radioactivity on the surface, the solubility of the surface radioactivity in the liquid, and the contact between the liquid and the surface.

Operating parameters of the water-jet vary with the requirements of the surface decontamination effort. For this study, a water-flow rate of 2.3×10^{-2} m^3/min and a cleaning rate of 0.77 m^2/min are assumed. The water-jet is assumed to remove 90% of the surface contamination.

The concentration of radioactivity in the liquid resulting from use of the water-jet is defined by Equation N.4.

$$C_v = 0.9\ C_s \left(\frac{0.77\ \mathrm{m^2/min}}{2.3 \times 10^{-2}\ \mathrm{m^3/min}} \right) \quad \text{(N.4)}$$

where:

- C_v • the concentration of radioactivity in the liquid resulting from use of the water-jet, Ci/m^3
- 0.9 • the fraction of the surface contamination removed by the water spray
- C_s • the radioactivity concentration on the building surfaces, Ci/m^2
- 0.77 • the water-jet cleaning rate, m^2/min
- 2.3×10^{-2} • the water flow rate for the water-jet, m^3/min.

The airborne radioactivity generated in a building resulting from use of the water-jet is calculated from the liquid radioactivity concentration using Equation N.5.

$$Q_h = \frac{C_v}{\rho} (1 \times 10^{-2}\ \mathrm{g/m^3})\ F_v\ t \quad \text{(N.5)}$$

where:

Q_h • the airborne radioactivity in the building resulting from water-jet operations, Ci

C_v • liquid radioactivity concentration, Ci/m^3

ρ • the density of the water-jet solution, g/m^3

1×10^{-2} • the airborne concentration, g/m^3

F_v • the building ventilation flow rate, m^3/hr

t • the time spent using the water-jet, hr.

The atmospheric release of radioactivity from water-jet sprays is calculated by multiplying the airborne radioactivity in the building by the building HEPA filter transmission factor.

Piping or Equipment Removal. Contaminated or activated piping and equipment are assumed to be removed and segmented for disposal during DECON. For this study, three cutting methods are assumed to be used: plasma-arc torch, oxyacetylene torch, and power hacksaw. A summary of the cutting parameters for these methods is listed in Table N.2-3.

TABLE N.2-3. Cutting Parameters for Piping and Equipment Removal

Cutting Method	Cut (kerf) Width, (m)	Airborne Release
Plasma Torch	3.1×10^{-3} to 6.4×10^{-3}	Condensed Metals, Gases, Smoke Particles
Oxyacetylene Torch	6.4×10^{-3}	Condensed Metals, Vapors
Power Hacksaw	6.4×10^{-3}	Metal Fragments, Vapors

In this study, cutting tasks are assumed both in air and under a water cover.

1. In Air. For piping or equipment containing surface contamination, the airborne release in the building from cutting tasks in air is calculated using Equation N.6.

$$Q_c = L\ K\ C_s \qquad (N.6)$$

where:

Q_c • the airborne radioactivity from cutting contaminated pipe or equipment, Ci

L • the length of the cut, m

K • the width of the cut (kerf), m

C_s • the surface radioactivity concentration, Ci/m^2.

Equation N.6 is based on the conservative assumption that all of the surface radioactive contamination associated with the cut width is vaporized and made airborne during cutting operations.

The airborne release associated with cutting neutron-activated piping or equipment is calculated using Equation N.7.

$$Q_c' = L\ K\ T\ C_m \tag{N.7}$$

where:

Q_c' • the airborne radioactivity from cutting neutron-activated material in air, Ci

L, K • are defined previously for Equation N.6

T • the thickness of the material being cut, m

C_m • the concentration of radioactivity in the material being cut, Ci/m^3.

Again, all material in the kerf width is conservatively assumed to be vaporized and is assumed to be airborne in the building. Equations N.6 and N.7 are used for all cutting methods in air, assuming a maximum kerf width of 6.4×10^{-3} m for the plasma-arc torch, oxyacetylene torch, and hacksaw.

The amount of radioactivity released to the atmosphere is calculated by multiplying either Q_c or Q_c by the building HEPA filter transmission factor.

2. <u>Under water</u>. Highly activated piping or equipment is assumed to be segmented by a remote, underwater cutting method, using the plasma-arc torch. The heat and agitation associated with this cutting method are assumed to

drive some of the contaminated cover water into the air. In some cases, a contamination envelope is assumed to be used to mitigate the airborne release. The total contamination suspended in the water is assumed to be the maximum value calculated for all underwater cutting tasks; thus, the airborne releases are maximized.

The air in the vicinity of the cutting tasks is assumed to become filled with radioactive water vapor to a concentration of 1×10^{-2} g/m^3 during underwater cutting tasks. Cutting parameters are assumed to be the same as those described for in-air piping or equipment removal. Equation N.8 is used to calculate the quantity of airborne radioactivity released from underwater cutting tasks.

$$Q_c'' = \frac{C_v}{\rho} \left(1 \times 10^{-2} \text{ g/m}^3\right) F_v' \, t \, F_c \qquad \text{(N.8)}$$

where:

Q_c'' • the quantity of airborne radioactivity transmitted in the building airflow, Ci

C_v • the maximum concentration of radioactive material in the cover water, Ci/m^3

ρ • the density of the cover water and radioactivity mixture, g/m^3

1×10^{-2} • the airborne concentration in the contamination control envelope, g/m^3

F_v' • the ventilation flow rate, m^3/hr

t • the duration of the cutting operation, hr

F_c • the leak fraction of the contamination control envelope: if used, the fraction is 0.1; if not used, the fraction is 1.0.

The total release to the environment from underwater cutting tasks is calculated by multiplying the quantity of radioactive contamination entering the building airflow by the transmission factor of the building HEPA filter system.

Contaminated Concrete Removal. Structural concrete that cannot be decontaminated to unrestricted-use levels by surface cleaning methods must be mechanically removed. For this study, the concrete spaller is assumed to be used to remove these contaminated concrete areas. Airborne contamination control is assumed to be provided by the use of a vacuum exhaust system during this dusty task. Holes are drilled into the concrete surfaces for insertion of the concrete spaller. To calculate the airborne radioactivity in a building associated with the dust generated during drilling, the volume of concrete for each hole, the number of holes drilled, the radioactive contamination level, and the effectiveness of the vacuum system must be determined. Each hole is assumed to be 5.1×10^{-2} m deep and remove a contaminated surface area of 5.0×10^{-2} m^2.

For structural concrete with surface contamination only, the total radioactivity associated with drilling is found by multiplying the number of holes drilled by the surface area per hole by the surface contamination level. The surface contamination is assumed to be uniformly mixed in volume of dust associated with drilling the holes. A vacuum exhaust system is used to collect the dust generated during drilling. This sytem is assumed to have an airflow of 1000 m^3/hr and reduce the air concentration of dust by a factor of 0.3. The quantity of airborne radioactivity generated by drilling is given by Equation N.9.

$$Q_d = \frac{N_h \ (7.8 \times 10^{-5} \ m^2/\text{hole}) \ C_s}{V_h \ \rho} F'_v \ t(0.3) \ 1 \times 10^{-1} \ g/m^3 \qquad (N.9)$$

where:

Q_d • the quantity of radioactivity made airborne during drilling, Ci
N_h • the number of holes drilled
7.8×10^{-5} • the surface area removed per hole, m^2/hole
C_s • the concentration of surface radioactivity, Ci/m^2
V_h • the total volume of all holes drilled, m^3
ρ • the density of concrete, 2.5×10^3 kg/m^3
F'_v • the vacuum system airflow rate, 1.0×10^3 m^3/hr
t • the duration of the drilling operation, hr

0.3 • airborne cleanup factor

1×10^{-1} • the airborne concentration in the vacuum system, g/m^3.

The radioactivity released to the atmosphere is found by multiplying the quantity of radioactivity generated by the leakage from the control envelope (if one is used) and by the building HEPA filter transmission factor.

After the holes are drilled, the concrete spaller is inserted into the holes and the concrete surface is scarfed. The concrete fractures and spalls in large pieces with a relatively small amount of dust generated. Water sprays are assumed to be used to control the airborne dust during the concrete spalling. Thus, the airborne release from concrete spalling is assumed to be insignificant when compared to the drilling operations and is not considered further in this study.

Waste Exhumation. Areas of contaminated soil are assumed to be encountered in the Emergency Retention Basin (ERB) at the reference test reactor. Atmospheric releases are found for waste exhumation using the mechanical mixing resuspension analysis defined for low-level waste exhumation operations for decommissioning a low-level waste burial ground.[6] The airborne radioactivity resulting from mechanical resuspension is calculated by determining the product of the mechanical mixing resuspension rate (1×10^{-7} s^{-1}), the volume of waste disturbed (m^3), the total disturbance time (s^{-1}), and the specific activity of the waste (Ci/m^3). The time estimates are for waste exhumation as defined in Appendix I. The contaminated soil is assumed to be in a wet form that reduces the resuspension rate by a factor of 10. It is also further assumed that only 10% of the waste is of the proper physical form and particle size to permit airborne transport.

N.2.1.2 Atmospheric Releases from DECON Tasks

Atmospheric releases of radioactivity resulting from DECON tasks at the reference R&T reactors are calculated based on the general methods described in Section N.2.1.1. The operational data for the research and test reactors are shown in Tables N.2-4 and N.2-5, respectively. The buildings, areas, and tasks listed in these tables are further discussed for DECON in Appendix I. Table N.2-4 contains the necessary safety analysis data, estimates of operational

TABLE N.2-4. Public Safety Analysis Data for DECON at the Reference Research Reactor

Building/Task	General Method	Operational Data				Comments	Operational Time (hr)	Reference[a] Radionuclide Inventory	Contamination Levels, C_s (Ci/m^2), C_m (Ci/m^3)
Reactor Building									
1. Install HEPA Filters	N/A[b]								
2. Comprehensive Radiation Survey	Direct Survey Methods	Surface Area (m^2)	Air Volume V (m^3)						
		3.1×10^3	3.1×10^3			Health physics technicians tour facility, 3 complete surveys	3.6×10^1	Table E.1-5[c]	$C_s = 2.4 \times 10^{-5}$
3. General Cleanup	Sweeping	3.1×10^3	3.1×10^3			All floor surfaces are assumed to be swept and vacuumed	4.4×10^1	Table E.1-5[c]	$C_s = 2.4 \times 10^{-5}$
	Vacuuming	3.1×10^3	3.1×10^3				2.2×10^1	Table E.1-5[c]	$C_s = 2.4 \times 10^{-5}$
4. Discharge and Ship Spent Fuel	N/A								
5. Remove Beam Tube Caves	N/A								
6. Drain Pool Irradiation Facility	N/A								
7. Remove Reactor Core and Vessel Internals		Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)	Volume of Liquid V_L, (m^3)				
- Contaminated	Plasma Arc Torch	7.4×10^0	6.4×10^{-3}	7.8×10^{-1}	1.9×10^1	Remote cutting operations, under water cover	9.5×10^0	Table E.1-5	$C_s = 7.0 \times 10^{-3}$
- Activated	Plasma Arc Torch	1.6×10^1	6.4×10^{-3}	1.6×10^0	1.9×10^1		1.0×10^1	Table E.1-5	50% $C_m = 7.5 \times 10^3$
								Table E.1-6	50% $C_m = 2.9 \times 10^2$
8. Drain Reactor Pool	N/A								
9. Remove Reactor Vessel	Plasma Arc Torch	Cut Length L, (m)	Kerf Width K, (m)	Material Thickness (m)	Cut Rate (m/hr)				
		5.2×10^0	6.45×10^{-3}	6.4×10^{-3}	6.5×10^{-1}	Remotely cut in air for shipment as one piece in a cask	8.0×10^0	Table E.1-5	$C_m = 9.8 \times 10^0$
10. Ship Reactor Core, Vessel and Internals	N/A								
11. Remove Activated Concrete	Drilling/ Spalling	Number of Holes, N_H	Hole Surface Area (m^2)	Hole Volume H (m^3)					
		1.1×10^3	8.5×10^{-2}	4.4×10^{-3}		4×10^{-6} m^3 per hole, 200 holes/day, surface area = 9.1×10^1 m^2	1.6×10^2	Table E.1-7	$C_m = 2.4 \times 10^{-3}$
12. Remove Reactor Building Equipment	Power Hacksaw	Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)					
		1.6×10^1	6.4×10^{-3}	1.4×10^0		Power hacksaw is used to segment equipment into sections for offsite transport	1.2×10^1	Table E.1-5[d]	$C_s = 7.0 \times 10^{-5}$

TABLE N.2-4. (Contd)

Building/Task	General Method	Operational Data			Comments	Operational Time (hr)	Reference[a] Radionuclide Inventory	Contamination Levels, $C_s(Ci/m^2)$, $C_m(Ci/m^3)$
Reactor Building								
13. Remove Piping, Drains, and Sinks	Power Hacksaw	5.0×10^1	6.4×10^{-3}	1.4×10^0	Small valves are left in pipe length, large valves are separated for packaging	3.5×10^1	Table E.1-5[d]	$C_s = 7.0 \times 10^{-5}$
14. Remove HVAC and Electrical Systems								
- HVAC	Oxyacetylene Torch	1.7×10^2	6.4×10^{-3}	1.9×10^1	Requires scaffolding	9.1×10^0	Table E.1-5[c]	$C_s = 2.4 \times 10^{-5}$
- Electrical	Oxyacetylene Torch	4.0×10^0	6.4×10^{-3}	6.7×10^0		6.0×10^{-1}	Table E.1-5[c]	$C_s = 2.4 \times 10^{-5}$
15. Final Radiation Survey[b]	N/A							
Annex Building								
16. Decontaminate Hot Cell								
		Number of Holes, NH	Hole Surface Area (m^2)	Hole Volume $V_H(m^3)$				
- Contaminated Concrete Removal	Drilling/ Spalling	1.7×10^2	1.3×10^{-2}	6.7×10^{-4}	4×10^{-6} m^3 per hole, 200 hole/day Surface Area = 1.4×10^1 m^2	2.4×10^1	Table E.1-8	$C_s = 9.6 \times 10^{-3}$
Heat Exchanger Building								
		Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)				
17. Remove Heat Exchanger	Oxyacetylene Torch	3.1×10^0	6.4×10^{-3}	1.6×10^0	Segment and cap ends for offsite shipment	2.0×10^0	Table E.1-5[d]	$C_s = 7.0 \times 10^{-5}$
Pump House								
		Cleaning Rate m^2/hr)	Surface Area $A_S(m^2)$	Liquid Volume (m^3)				
18. Decontaminate Walls and Floor	Water Jet	4.6×10^1	1.4×10^2	4.3×10^{-1}	All interior surfaces, walls, and floor	3.0×10^0	Table E.1-5[c]	$C_s = 4.8 \times 10^{-4}$
		Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)				
19. Remove Retention Tank, Piping and Equipment	Oxyacetylene Torch	2.3×10^1	6.4×10^{-3}	1.5×10^0	Segment and cap ends for offsite shipment	1.5×10^1	Table E.1-5[d]	$C_s = 7.0 \times 10^{-5}$
Radiation Center Building								
20. Remove Piping and Equipment	Oxyacetylene Torch	7.0×10^0	6.4×10^{-3}	1.5×10^0	Segment for packaging	5.0×10^0	Table E.1-5[d]	$C_s = 7.0 \times 10^{-5}$
All Buildings								
21. Package and Ship Contaminated Materials and Radioactive Wastes	N/A							

(a) The reference radionuclide inventories refer to the tables of radionuclides shown in Appendix E.
(b) N/A = Not applicable.
(c) Deposited surface contamination is assumed to have the same radionuclide mixture as that shown in Table E.1-5 for activated stainless steel.
(d) Internal corrosion product deposition is calculated in Appendix E for ^{60}Co; the internal surface contamination is assumed to have the same radionuclide mixture as that shown for activated stainless steel.

TABLE N.2-5. Public Safety Analysis Data for DECON at the Reference Test Reactor

Building/Task	General Method	Operational Data				Comments	Operational Time (hr)[a]	Reference Radionuclide Inventory[b]	Contamination Levels, C_s(Ci/m^2), C_m(Ci/m^3)
Reactor Building/MUR/ Containment Vessel		Surface Area (m^2)	Air Volume V (m^3)						
1. Comprehensive Radi-Survey	Direct Survey Methods	7.3×10^3	7.3×10^3			Health physics technicians tour facility, 3 complete surveys	8.8×10^1	Table E.2-1[d]	$C_s = 4.8 \times 10^{-4}$
2. Discharge Fuel (Including MUR)	N/A[c]					Routine fuel handling task	N/A	N/A	N/A
3. Prepare and Ship Spent Fuel	N/A					After on-site storage; covered by written procedures (see Section I.7.9.1 of Appendix I	N/A	N/A	N/A
4. General Cleanup and Equipment Inventory (i.e., All Buildings)	Sweeping/ Vacuuming	2.4×10^3 2.4×10^3	2.4×10^3 2.4×10^3			All floor surfaces are assumed to be swept and vacuumed	3.4×10^1 1.7×10^1	Table E.2-1[d] Table E.2-1[d]	$C_s = 4.8 \times 10^{-4}$ $C_s = 4.8 \times 10^{-4}$
		Cleaning Rate (m^2/hr)	Surface Area A_s (m^2)	Liquid Volume (m^3)					
5. Drain, Clean, Dry Quadrants A, B, C, and D and Canals E and F	Water-Jet and Underwater Vacuum	4.6×10^1	1.9×10^3	5.8×10^0		All surfaces are water-jet cleaned simultaneously with draining operations; vacuum used for debris collection before final draining	4.1×10^1	Table E.2-1[d]	$C_s = 9.6 \times 10^{-4}$
6. Drain, Clean, Dry Canal H	Water-Jet and Underwater Vacuum	4.6×10^1	1.9×10^2	5.8×10^{-1}		All surfaces are water-jet cleaned simultaneously with draining operations; vacuum used for debris collection before final draining	4.1×10^0	Table E.2-1[d]	$C_s = 1.9 \times 10^{-3}$
		Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)					
7. Remove, Package, and Ship MUR and Associated Hardware	Plasma-Arc	8.0×10^0	6.4×10^{-3}	1.5×10^0		Most connections are flanged and bolted, little cutting required	5.0×10^0	Table E.2-1[d]	$C_s = 9.6 \times 10^{-3}$
		Cleaning Rate (m^2/hr)	Surface Area A_s (m^2)	Liquid Volume (m^3)					
8. Drain, Clean, Dry Canal G	Water-Jet and Underwater Vacuum	4.6×10^1	2.2×10^2	6.7×10^{-1}		After draining, all surfaces are water-jet cleaned	4.8×10^0	Table E.2-1[d]	$C_s = 9.6 \times 10^{-4}$
9. Remove Loose Equipment in Q and C's and Dry Annulus	N/A					General cleanup task, no release calculated	N/A	N/A	N/A
10. Drain and Flush PCWS	N/A					General cleanup task, no release calculated	N/A	N/A	N/A
11. Isolate RV and Add Required Water for Shielding	N/A					General cleanup task, no release calculated	N/A	N/A	N/A
		Cut Length L, (m)	Kerf Width K, (m)	Material Thickness (m)	Cut Rate (m/hr)				
12. Remove RV Internals and Ship Activated RV Internals	Plasma-Arc Torch - Stainless Steel - Aluminum	1.9×10^2 1.8×10^1	6.4×10^{-3} 6.4×10^{-3}	4.8×10^{-2} 6.6×10^{-2}	2.0×10^1 2.0×10^1	Unbolting, shearing, and remote cutting operations in 5.6 m^3 of water using a contamination control envelope	9.5×10^0 9.0×10^{-1}	Table E.2-1 Table E.2-2	$C_m = 5.3 \times 10^3$ $C_m = 3.5 \times 10^3$
13. Remove RV and Ship RV Segments	Plasma-Arc Torch - Stainless Steel - Aluminum	2.7×10^2 2.3×10^1	6.4×10^{-3} 6.4×10^{-3}	1.9×10^{-2} 1.9×10^{-2}	4.8×10^0 4.8×10^0	Drained and cut in a contamination control envelope	5.6×10^1 4.8×10^0	Table E.2-5 Table E.2-6	$C_m = 3.0 \times 10^1$ $C_m = 2.0 \times 10^1$

TABLE N.2-5. (Contd)

Building/Task	General Method	Operational Data			Comments	Operational Time (hr)[a]	Reference Radionuclide Inventory[b]	Contamination Levels, C_s(Ci/m^2), C_m(Ci/m^3)
Reactor Building/MUR/ Containment Vessel								
		Number of Holes, N_H	Hole Surface Area (m^2)	Hole V_H(m^3)				
14. Remove Bio-Shield Concrete	Drilling/Spalling	2.5×10^2	2.0×10^{-2}	1.0×10^{-3}	4×10^{-6} m^3 per hole, 200 holes/day, Surface Area = 21 m^2	1.0×10^1	Table E.2-3	$C_m = 1.1 \times 10^0$
		Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)				
15. Remove Fixed Equipment in CV (except HVAC)	Oxyacetylene Torch	2.7×10^1	6.4×10^{-3}	1.0×10^0	Unbolting and direct cutting operations	2.7×10^1	Table E.2-1[e]	$C_s = 7.0 \times 10^{-4}$
16. Remove Fixed Equipment Outside CV	Oxyacetylene Torch	3.0×10^0	6.4×10^{-3}	1.0×10^0	Unbolting and direct cutting operations	3.0×10^0	Table E.2-1[e]	$C_s = 7.0 \times 10^{-5}$
17. Remove Quadrant Piping	Plasma-Arc	2.6×10^2	6.4×10^{-3}	1.2×10^1	Direct cutting in air	2.2×10^1	Table E.2-1[e]	$C_s = 7.0 \times 10^{-4}$
18. Segment and Remove Sub-Pile Room	Oxyacetylene Torch	4.0×10^1	6.4×10^{-3}	8.0×10^0	Direct cutting in air	5.0×10^0	Table E.2-1[d]	$C_s = 3.5 \times 10^{-3}$
19. Remove Lead Shield from Below Reactor Cavity	Oxyacetylene Torch	3.0×10^1	6.4×10^{-3}	8.0×10^0	Direct cutting in air	3.8×10^0	Table E.2-1[d]	$C_s = 1.4 \times 10^{-2}$
20. Remove Pipes from Bio-Shield	Explosives for Concrete Plasma-Arc	7.6×10^1	6.4×10^{-3}	1.0×10^1	After concrete is removed, direct pipe cutting in air	7.6×10^0	Table E.2-1[e]	$C_s = 1.4 \times 10^{-2}$
21. Remove PCWs Piping from	Plasma-Arc	3.0×10^1	6.4×10^{-3}	1.0×10^1	Direct cutting in air	3.0×10^0	Table E.2-1[e]	$C_s = 1.4 \times 10^{-2}$
		Number of Holes, N_H	Hole Surface Area, (m^2)	Hole Volume (m^3)				
22. Remove RB/CV Contaminated Concrete	Drilling/Spalling	7.1×10^4	5.7×10^0	2.8×10^{-1}	4×10^{-6} m^3 per hole, 200 holes/day, Surface Area = 6.0×10^3 m^2	2.1×10^3	Table E.2-1[d]	$C_s = 1.2 \times 10^{-3}$
		Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)				
23. Remove Q and C, and misc. contaminated drains	Oxyacetylene Torch	3.2×10^2	6.4×10^{-3}	1.0×10^1	After concrete is removed, direct cutting in air	3.2×10^1	Table E.2-1[e]	$C_s = 7.0 \times 10^{-3}$
24. Remove Contaminated HVAC from RB/CV	Nibbler and Oxyacetylene Torch	1.4×10^2	6.4×10^{-3}	5.4×10^0	Direct cutting in air	2.6×10^1	Table E.2-1[e]	$C_s = 7.0 \times 10^{-4}$
25. Final Radiation Survey	N/A				Assumed to be at acceptable levels--no calculations	N/A	N/A	N/A
Hot Laboratory Building								
		Cleaning Rate (m^2/hr)	Surface Area A_s, (m^2)	Liquid Volume (m^3)				
1. Decontaminate Hot Cells	Water-Jet	4.6×10^1	5.8×10^2	1.8×10^0	The water-jet is used on all equipment and cell surfaces	1.3×10^1	Table E.2-10[f]	$C_s = 4.8 \times 10^{-2}$
		Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)				
2. Remove and Package Hot Cell Equipment and Piping	Oxyacetylene Torch	1.9×10^3	6.4×10^{-3}	1.2×10^1	Direct cutting in air	1.6×10^2	Table E.2-10[f]	$C_s = 7.0 \times 10^{-4}$
3. Remove and Package Hot Cell SS Cladding	Plasma Arc Torch	4.7×10^3	6.4×10^{-3}	2.5×10^0	Stainless steel cladding is removed by a portable plasma arc torch	1.9×10^2	Table E.2-10[f]	$C_s = 4.8 \times 10^{-3}$
		Number of Holes, N_H	Hole Surface Area, (m^2)	Hole Volume (m^3)				
4. Remove Contaminated Concrete from Hot Cells	Drilling/Spalling	6.9×10^3	2.8×10^{-2}	2.8×10^{-2}	4×10^{-6} m^2 per hole, 200 holes/day, Surface Area = 2.3×10^2 m^2	2.1×10^2	Table E.2-10[f]	$C_s = 4.8 \times 10^{-4}$
		Surface Area (m^2)	Air Volume (m^3)					
5. Decontaminate the HLB (including cranes)	Vacuuming[h]	3.4×10^3	3.4×10^3		All building surfaces (including cranes) are decontaminated; vacuuming is shown in calculations	2.4×10^1	Table E.2-10[f]	$C_s = 4.8 \times 10^{-4}$

TABLE N.2-5. (Contd)

Building/Task	General Method	Operational Data			Comments	Operational Time (hr)[a]	Reference[b] Radionuclide Inventory	Contamination Levels, $C_s(Ci/m^2)$, $C_m(Ci/m^3)$
Hot Laboratory Building								
6. Drain, Clean, Dry, Canals J and K	Water-Jet	Cleaning Rate (m^2/hr) 4.6×10^1	Surface Area A_s, (m^2) 5.8×10^2	Liquid Volume (m^3) 1.8×10^0	After draining, all surfaces are water-jet cleaned	1.3×10^1	Table E.2-1	$C_s = 2.4 \times 10^{-3}$
7. Remove Loose Equipment	N/A				General cleanup task, No release calculated	N/A	N/A	N/A
8. Remove Fixed and Permanent Equipment (except HVAC), including the Hot Pipe Tunnel	Oxyacetylene Torch	Cut Length L, (m) 1.3×10^1	Kerf Width K, (m) 6.4×10^{-3}	Cut Rate (m/hr) 1.0×10^1	Unbolting and direct cutting operations	1.3×10^0	Table E.2-10[g]	$C_s = 4.8 \times 10^{-3}$
9. Remove SS Cladding from Decon. Room 23	Plasma Arc Torch	5.8×10^1	6.4×10^{-3}	2.5×10^0	Stainless steel cladding is removed by a portable plasma arc torch	2.3×10^1	Table E.2-10[f]	$C_s = 4.8 \times 10^{-4}$
10. Remove Hot Cell Windows		N/A			Unbolting - no calculations	N/A	N/A	N/A
11. Remove and Package HLB Contaminated Concrete	Drilling/Spalling	Number of Holes, N_H 1.4×10^4	Hole Surface Area, (m^2) 5.6×10^{-2}	Hole Vacuum (m^3) 5.7×10^{-2}	4×10^{-6} m^2 per hole 200 holes per day, Surface Area = 1.2×10^3 m^2	4.2×10^2	Table E.2-10[g]	$C_s = 4.8 \times 10^{-4}$
12. Remove and Package Contaminated HVAC from HLB	Oxyacetylene Torch	Cut Length L, (m) 2.8×10^2	Kerf Width K, (m) 6.4×10^{-3}	Cut Rate (m/hr) 5.4×10^0	Direct cutting in air	5.1×10^1	Table E.2-10[g]	$C_s = 7.0 \times 10^{-4}$
13. Final Radiation Survey	N/A				Assumed to be at acceptable levels--no calculation	N/A	N/A	N/A
Other Contaminated Buildings and Areas								
1. Radiation Survey	Direct Survey Methods	Surface Area, (m^2) 3.3×10^3	Air Volume V, (m^3) 3.3×10^3		Health physics technicians tour facility	4.0×10^1	Table E.2-1[d]	$C_s = 4.8 \times 10^{-4}$
2. Primary Pump House: - Remove Fixed equipment (except HVAC)	Oxyacetylene Torch	Cut Length L, (m) 4.2×10^1	Kerf Width K, (m) 6.4×10^{-3}	Cut Rate (m/hr) 1.0×10^1	Unbolting and direct cutting operations	4.2×10^0	Table E.2-1[e]	$C_s = 7.0 \times 10^{-4}$
- Decontaminate PPH and Remove Contaminated Concrete	Water-Jet	Cleaning Rate (m^2/hr) 4.6×10^1	Surface Area A_s, (m^2) 1.8×10^2	Liquid Volume (m^3) 5.5×10^{-1}	All surfaces are cleaned with the water-jet	3.9×10^0	Table E.2-1[d]	$C_s = 9.6 \times 10^{-3}$
	Drilling/Shalling	Number of Holes N_H 2.1×10^3	Hole Surface Area, (m^2) 8.4×10^{-3}	Hole Volume (m^3) 8.5×10^{-3}	4×10^{-6} m^3 per hole, 200 holes/day, Surface Area = 1.8×10^2	6.4×10^1	Table E.2-1[e]	$C_s = 9.6 \times 10^{-4}$
- Remove and Package contaminated HVAC	Oxyacetylene Torch	Cut Length L, (m) 2.7×10^2	Kerf Width K, (m) 6.4×10^{-3}	Cut Rate (m/hr) 5.4×10^0	Direct cutting in air	5.0×10^1	Table E.2-1[e]	$C_s = 7.0 \times 10^{-4}$

TABLE N.2-5. (Contd)

Building/Task	General Method	Operational Data			Comments	Operational Time (hr)[a]	Reference[b] Radionuclide Inventory	Contamination Levels, $C_s(Ci/m^2)$, $C_m(Ci/m^3)$
Other Contaminated Buildings and Areas								
3. Office and Laboratory Building (OLB):								
- Remove Contaminated Hoods and Sinks	Oxyacetylene Torch	1.3×10^2	6.4×10^{-3}	2.5×10^0	Direct cutting in air	5.3×10^1	Table E.2-1[d]	$C_s = 4.8 \times 10^{-4}$
		Number of Holes, N_H	Hole Surface Area, (m^2)	Hole Volume (m^3)				
- Remove Contaminated Concrete	Drilling/Spalling	6.0×10^2	2.4×10^{-3}	2.4×10^{-3}	4×10^{-6} m^3 per hole, 200 holes/day, Surface Area = 5.1×10^1	1.8×10^1	Table E.2-1[d]	$C_s = 4.8 \times 10^{-4}$
4. Emergency Retention Basin (ERB) and Site Ditches:								
- Drain the ERB	N/A				No release is calculated for Draining Operations	N/A	N/A	N/A
- Remove and Package Contaminated Piping and Soil	Piping				Tongue and groove piping is separated - no calculation	N/A	N/A	N/A
		Volume of Soil, (m^3)						
	Soil Exhumation	1.6×10^3			Airborne releases are based on methods discussed for mechanical mixing tasks in NUREG/CR-0570 (Appendix I)	6.6×10^1	Table E.2-11[i]	2.8×10^2 pCi/g[i]
5. Cold Retention Area (CRA)								
		Number of Holes, N_H	Hole Surface Area, (m^2)	Hole Volume $(m\ )^3$				
- Remove Contaminated Concrete and Soil	Drilling/Spalling	1.4×10^4	5.6×10^{-2}	5.6×10^{-2}	4×10^{-6} m^3 per hole, 200 holes/day, Surface Area = 1.2×10^3	4.2×10^2	Table E.2-1[d]	$C_s = 9.6 \times 10^{-4}$
		Volume of Soil, (m^3)						
	Soil Exhumation	4.1×10^2			Airborne releases are based on methods discussed for mechanical mixing tasks in NUREG/CR-0570 (Appendix I)	1.7×10^1	Table E.2-11[i]	2.8×10^2 pCi/g[i]
6. Hot Retention Area (HRA)		Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)				
- Remove and Package Contaminated Piping	Oxyacetylene Torch	1.5×10^2	6.4×10^{-3}	1.0×10^1	Direct cutting in air	1.5×10^1	Table E.2-1[e]	$C = 7.0 \times 10^{-4}$
- Provide Tank Access to 8 HRA Tanks	N/A				Remove uncontaminated soil - No release calculations	N/A	N/A	N/A
- Remove and Package HRA Tanks, Floor Plates, and Partitions	Oxyacetylene Torch	3.0×10^3	6.4×10^{-3}	2.5×10^0	Tanks are removed before sectioning; cutting is done in air	1.2×10^3	Table E.2-1[e]	$C_s = 9.6 \times 10^{-4}$
- Uncover and Prepare HRA Tanks 9-12 for Shipment	N/A				Remove uncontaminated soil - No release calculations	N/A	N/A	N/A
		Number of Holes, N_H	Hole Surface Area, (m^2)	Hole Volume $(m\ )^3$				
- Remove Contaminated Concrete	Drilling/Spalling	2.5×10^3	1.7×10^{-2}	1.0×10^{-2}	4×10^{-6} m^3 per hole, 200 holes/day, Surface area = 2.1×10^2	7.5×10^1	Table E.2-1[d]	$C_s = 9.6 \times 10^{-4}$

TABLE N.2-5. (Contd)

Building/Task	General Method	Operational Data			Comments	Operational Time (hr)[a]	Reference Radionuclide Inventory[b]	Contamination Levels, C_s(Ci/m^2), C_m(Ci/m^3)
7. Fan House								
- Remove Contaminated Concrete	Drilling/Spalling	7.2×10^2	3.9×10^{-3}	2.9×10^{-3}	4×10^{-6} m^3 per hole, 200 holes/day, Surface Area = 6.2×10^1 m^2	2.2×10^1	Table E.2-1[d]	$C_s = 9.6 \times 10^{-4}$
		Cut Length L, (m)	Kerf Width K, (m)	Cut Rate (m/hr)				
- Remove Fixed Equipment	Oxyacetelyene Torch	1.8×10^1	6.4×10^{-3}	2.5×10^0	Unbolting and direct cutting operations	7.2×10^0	Table E.2-1[e]	$C_s = 7.0 \times 10^{-4}$
- Raze Stack, Segment, and Package	Oxyacetylene Torch	1.9×10^2	6.4×10^{-3}	2.5×10^0	Stack is lowered by a crane and sectioned in air	7.6×10^1	Table E.2-1[e]	$C_s = 4.8 \times 10^{-4}$
8. Waste Handling Building (WHB)								
- Remove Fixed Equipment, Including Evaporator and HVAC	Oxyacetylene Torch	1.3×10^1	6.4×10^{-3}	1.0×10^0	Unbolting and direct cutting operations	1.3×10^0	Table E.2-1[e]	$C_s = 7.0 \times 10^{-4}$
		Number of Holes, N_H	Hole Surface Area, (m^2)	Hole Volume (m^3)				
- Remove Contaminated Concrete	Drilling/Spalling	1.9×10^3	1.3×10^{-2}	7.6×10^{-3}	4×10^{-6} m^3 per hole, 200 holes/day, Surface Area = 1.6×10^2 m^2	5.7×10^1	Table E.2-1[d]	$C_s = 9.6 \times 10^{-4}$

(a) Operational time does not include set-up or clean-up time.
(b) The reference radionuclide inventories refer to the tables of radionuclides shown in Appendix E.
(c) N/A = Not applicable.
(d) Deposited surface contamination (C_s) is assumed to have the same mixture of radionuclides as shown in Table E.2-1 for activated stainless steel.
(e) Internal corrosion product deposition is calculated in Appendix E for ^{60}Co; the internal surface contamination mixture is assumed to have the same radionuclide mixture as that shown for activated stainless steel.
(f) Deposited surface contamination (C_s) is assumed to have the same mixture of radionuclides as shown in Table E.2-10 for the hot cells.
(h) Several decontamination methods are used including hand operations, water-jet sprays, and sweeping. Vacuuming is selected as the reference method for airborne release analysis.
(g) Internal deposition is calculated in Appendix E for ^{60}Co; the internal surface contamination mixture in the hot laboratory is assumed to be the same as that shown for the hot cells in Table E.2-10.
(h) Several decontamination methods are used including hand operations, water-jet sprays, and sweeping. Vacuuming is selected as the reference method for airborne release analysis.
(i) Soil contamination is shown in Table E.2-11 in units of pCi/g.

times, and contamination levels for each building and area at the reference research reactor. Table N.2-5 contains similar necessary data for the buildings and areas at the reference test reactor.

The calculated atmospheric releases for the research and test reactors are shown in Tables N.2-6 and N.2-7, respectively. These tables contain the calculated airborne radioactivity generation rates, the total airborne radioactivity generated, and the total release to the atmosphere for each DECON task considered. These atmospheric releases are used in radiation dose calculations to estimate the impacts of DECON at the reference R&T reactors, as discussed in the following section.

N.2.1.3 Public Radiation Doses from DECON Tasks

Radiation doses to the public from atmospheric releases of radioactivity during routine DECON tasks at the R&T reactors are calculated using the radiation dose models discussed in Appendix F. The first-year doses and the fifty-year committed dose-equivalents to the maximum-exposed individual and to the population residing within 80 km of the reference sites are calculated for both the reference research and test reactors. Each of the atmospheric releases is assumed to be a chronic release; that is, one that occurs at a uniform rate for a period of 1 year. A uniform chronic release is assumed so that direct comparisons of the impact of DECON tasks can be made.

Radiation doses from atmospheric releases of radioactivity are calculated for direct exposure, inhalation, and ingestion radiation exposure pathways. Radiation doses from air submersion are not calculated since they have been shown to be insignificant in previous decommissioning studies.[15-17] For inhalation, radionuclides are assumed to have soluble clearance times (as defined in Appendix F) for the organs of reference, except for the lung where insoluble clearance times are assumed. For this study, the organs of reference for which radiation dose is calculated include: total body, bone, lung, and GI-tract (lower large intestine).

Radiation doses to the maximum-exposed individual are shown for the reference research reactor in Table N.2-8, and for the reference test reactor in Table N.2-9. Only those tasks in buildings or areas that result in an atmospheric

TABLE N.2-6. Calculated Atmospheric Releases for DECON at the Reference Research Reactor

Building/Task[a]	General Method	Operational Time (hr)	Reference[b] Radionuclide Inventory	Airborne Generation Rate Inside Building (Ci/hr)	Total Airborne Radioactivity Inside Building (Ci)	Total[c] Atmospheric Release (Ci/yr)
Reactor Building						
2. Comprehensive Radiation Survey	Direct Survey Methods	3.6×10^1	Table E.1-5	2.1×10^{-7}	7.4×10^{-6}	1.0×10^{-10}
3. General Cleanup	Sweeping	4.4×10^1	Table E.1-5	8.4×10^{-7}	3.7×10^{-5}	1.9×10^{-8}
	Vacuuming	2.2×10^1	Table E.1-5	8.6×10^{-7}	1.9×10^{-5}	9.3×10^{-9}
7. Remove Reactor Core and Vessel Internals						
- Contaminated	Plasma Arc Torch	9.5×10^0	Table E.1-5	2.2×10^{-9}	2.1×10^{-8}	1.1×10^{11}
- Activated	Plasma Arc Torch	1.0×10^1	Table E.1-5	2.6×10^{-5}	2.6×10^{-4}	1.3×10^7
			Table E.1-6	1.0×10^{-6}	1.0×10^{-5}	5.1×10^{-9}
9. Remove Reactor Vessel	Plasma Arc Torch	8.0×10^0	Table E.1-6	2.6×10^{-4}	2.1×10^{-3}	1.0×10^{-6}
11. Remove Activated Concrete	Drilling/Spalling	1.6×10^2	Table E.1-7	7.6×10^{-7}	1.2×10^{-4}	6.1×10^{-8}
12. Remove Reactor Building Equipment	Power Hacksaw	1.2×10^1	Table E.1-5	6.0×10^{-7}	7.2×10^{-6}	3.6×10^{-9}
13. Remove Piping, Drains, and Sinks	Power Hacksaw	3.5×10^1	Table E.1-5	6.0×10^{-7}	2.2×10^{-5}	1.1×10^{-8}
14. Remove HVAC and Electrical Systems						
- HVAC	Oxyacetylene Torch	9.1×10^0	Table E.1-5	2.9×10^{-6}	2.6×10^{-5}	1.3×10^{-8}
- Electrical	Oxyacetylene Torch	6.0×10^1	Table E.1-5	1.0×10^{-6}	6.1×10^{-7}	3.1×10^{-10}
Annex Building						
16. Decontaminate Hot Cell						
- Contaminated Concrete Removal	Drilling/Spalling	2.4×10^1	Table E.1-8	2.2×10^{-6}	5.4×10^{-5}	2.7×10^{-8}
Heat Exchanger Building						
17. Remove Heat Exhanger	Oxyacetylene Torch	2.0×10^0	Table E.1-5	6.9×10^{-7}	1.4×10^{-6}	6.9×10^{-10}
Pump House						
18. Decon. Walls and Floor	Water-Jet	3.0×10^0	Table E.1-5	1.9×10^{-6}	5.6×10^{-6}	2.8×10^{-9}
19. Remove Retention Piping and Equipment	Oxyacetylene Torch	1.5×10^1	Table E.1-5	4.5×10^{-7}	1.0×10^{-5}	5.2×10^{-9}
Radiation Center Building						
20. Remove Piping and Equipment	Oxyacetylene Torch	5.0×10^0	Table E.1-5	4.5×10^{-7}	3.1×10^{-6}	1.6×10^{-9}

(a) Only those tasks from Table N.2-4 that result in an atmospheric release are included in this table.
(b) The table numbers listed are shown in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.2-7. Calculated Atmospheric Releases for DECON at the Reference Test Reactor

Building/Task[a]	General Method	Operational Time (hr)	Reference[b] Radionuclide Inventory	Airborne Generation Rate Inside Building (Ci/hr)	Total Airborne Radioactivity Inside Building (Ci)	Building[c] Transmission Factor	Total[d] Atmospheric Release (Ci/yr)
Reactor Building/MUR/ Containment Vessel							
1. Comprehensive Radiation Survey	Direct Survey Methods	8.8×10^{1}	Table E.2-1	4.0×10^{-6}	3.5×10^{-4}	5×10^{-4}	1.8×10^{-7}
4. General Cleanup and Equipment Inventory (i.e., All Buildings)	- Sweeping	3.4×10^{1}	Table E.2-1	1.5×10^{-5}	3.5×10^{-4}	5×10^{-4}	2.9×10^{-7}
	- Vacuuming	1.7×10^{1}	Table E.2-1	1.7×10^{-5}	5.8×10^{-4}	5×10^{-4}	1.4×10^{-7}
5. Drain, Clean, Dry Quadrants A, B, C, and D and Canals E and F	Water-Jet and Underwater Vacuum	4.1×10^{1}	Table E.2-1	3.8×10^{-6}	1.5×10^{-4}	5×10^{-4}	7.7×10^{-8}
6. Drain, Clean, Dry Canal H	Water-Jet and Underwater Vacuum	4.1×10^{0}	Table E.2-1	7.3×10^{-6}	3.0×10^{-5}	5×10^{-4}	1.5×10^{-8}
7. Remove, Package, and Ship MUR and Associated Hardware	Oxyacetylene Torch	5.0×10^{0}	Table E.2-1	9.8×10^{-5}	4.9×10^{-4}	5×10^{-5}	2.5×10^{-8}
8. Drain, Clean, Dry Canal G	Water-Jet and Underwater Vacuum	4.8×10^{0}	Table E.2-1	3.8×10^{-6}	1.8×10^{-5}	5×10^{-4}	9.0×10^{-9}
12. Remove RV Internals and Ship Activated RV Internals	Plasma Arc Torch						
	- Stainless Steel	9.5×10^{0}	Table E.2-1	5.5×10^{-4}	5.2×10^{-3}	5×10^{-5}	2.6×10^{-7}
	- Aluminum	9.0×10^{-1}	Table E.2-2	4.2×10^{-6}	3.8×10^{-6}	5×10^{-5}	1.9×10^{-10}
13. Remove RV and Ship RV Segments	Plasma Arc Torch						
	- Stainless Steel	5.6×12^{1}	Table E.2-1	1.8×10^{-2}	9.8×10^{-1}	5×10^{-5}	4.9×10^{-5}
	- Aluminum	4.8×10^{0}	Table E.2-2	1.2×10^{-2}	5.6×10^{-2}	5×10^{-5}	2.8×10^{-6}
14. Remove Bio-Shield Concrete	Drilling/Spalling	1.0×10^{1}	Table E.2-3	1.7×10^{-6}	1.7×10^{-5}	5×10^{-4}	8.6×10^{-9}
15. Remove Fixed Equipment in CV (Except HVAC)	Oxyacetylene Torch	2.7×10^{1}	Table E.2-1	4.5×10^{-6}	1.2×10^{-4}	5×10^{-4}	6.0×10^{-8}
16. Remove Fixed Equipment Outside CV	Oxyacetylene Torch	3.0×10^{0}	Table E.2-1	4.5×10^{-7}	1.3×10^{-6}	5×10^{-4}	6.7×10^{-10}
17. Remove Quadrant Piping	Oxyacetylene Torch	2.2×10^{1}	Table E.2-1	5.3×10^{-5}	1.2×10^{-3}	5×10^{-4}	5.8×10^{-7}
18. Segment and Remove Sub-Pile Room	Oxyacetylene Torch	5.0×10^{0}	Table E.2-1	1.8×10^{-4}	9.0×10^{-4}	5×10^{-4}	4.5×10^{-7}
19. Remove Lead Shield from Below Reactor Cavity	Oxyacetylene Torch	3.8×10^{0}	Table E.2-1	7.1×10^{-5}	2.7×10^{-3}	5×10^{-4}	1.3×10^{-6}
20. Remove Pipes from Bio-Shield	Oxyacetylene Torch	7.6×10^{0}	Table E.2-1	9.0×10^{-4}	6.8×10^{-3}	5×10^{-4}	3.4×10^{-6}
21. Remove PCWS Piping from Bio-Shield	Oxyacetylene Torch	3.0×10^{0}	Table E.2-1	9.0×10^{-5}	2.7×10^{-3}	5×10^{-4}	1.3×10^{-6}
22. Remove RB/CV Contaminated Concrete	Drilling/Spalling	2.1×10^{3}	Table E.2-1	6.2×10^{-5}	8.0×10^{-2}	5×10^{-5}	4.0×10^{-6}
23. Remove Q and C, and Misc. Contaminated Drains	Oxyacetylene Torch	3.2×10^{1}	Table E.2-1	4.5×10^{-4}	1.4×10^{-2}	5×10^{-4}	7.2×10^{-6}
24. Remove Contaminated HVAC from RB/CV	Oxyacetylene Torch	2.6×10^{1}	Table E.2-1	2.4×10^{-5}	6.3×10^{-4}	5×10^{-4}	3.1×10^{-7}

TABLE N.2-7. (Contd)

Building/Task[a]	General Method	Operational Time (hr)	Reference[b] Radionuclide Inventory	Airborne Generation Rate Inside Building (Ci/hr)	Total Airborne Radioactivity Inside Building (Ci)	Building[c] Transmission Factor	Total[d] Atmospheric Release (Ci/yr)
Hot Laboratory Building							
1. Decontaminate Hot Cells	Water-Jet	1.3×10^{1}	Table E.2-10	1.9×10^{-4}	2.4×10^{-3}	5×10^{-4}	1.2×10^{-6}
2. Remove and Package Hot Cell Equipment and Piping	Oxyacetylene Torch	1.6×10^{2}	Table E.2-10	5.3×10^{-5}	8.5×10^{-3}	5×10^{-5}	4.3×10^{-7}
3. Remove and Package Hot Cell SS Cladding	Oxyacetylene Torch	1.9×10^{2}	Table E.2-10	7.6×10^{-4}	1.4×10^{-1}	5×10^{-5}	7.0×10^{-6}
4. Remove Contaminated Concrete from Hot Cells	Drilling/Spalling	2.1×10^{2}	Table E.2-10	7.5×10^{-8}	1.6×10^{-5}	5×10^{-5}	7.9×10^{-10}
5. Decon. the HLB (Including the Cranes)	Vacuuming	2.4×10^{1}	Table E.2-10	1.7×10^{-5}	4.1×10^{-4}	5×10^{-4}	2.0×10^{-7}
6. Drain, Clean, Dry Canals J & K	Water-Jet	1.3×10^{1}	Table E.2-1	9.4×10^{-6}	1.2×10^{-4}	5×10^{-4}	6.1×10^{-8}
8. Remove Fixed and Permanent Equipment (Except HVAC), Including the Hot Pipe Tunnel	Oxyacetylene Torch	1.3×10^{0}	Table E.2-10	3.1×10^{-4}	4.0×10^{-4}	5×10^{-4}	2.0×10^{-7}
9. Remove SS Cladding from Decon. Room 23	Plasma Arc Torch	2.3×10^{1}	Table E.2-10	7.8×10^{-6}	1.8×10^{-4}	5×10^{-4}	8.9×10^{-8}
11. Remove and Package HLB Contaminated Concrete	Drilling/Spalling	4.2×10^{2}	Table E.2-10	7.5×10^{-8}	3.1×10^{-5}	5×10^{-4}	1.5×10^{-8}
12. Remove and Package Contaminated HVAC from HLB	Oxyacetylene Torch	5.1×10^{-2}	Table E.2-10	2.5×10^{-8}	1.2×10^{-5}	5×10^{-4}	6.3×10^{-8}
Other Contaminated Structures							
1. Radiation Survey	Direct Survey Methods	4.0×10^{1}	Table E.2-1	4.0×10^{-6}	1.6×10^{-4}	5×10^{-4}	7.9×10^{-8}
2. Primary Pump House							
- Remove Fixed Equipment (Except HVAC)	Oxyacetylene Torch	4.2×10^{0}	Table E.2-1	4.5×10^{-6}	1.9×10^{-4}	5×10^{-4}	2.2×10^{-9}
- Decon. PPH and Remove Contaminated Concrete	Water-Jet	3.9×10^{0}	Table E.2-1	3.8×10^{-6}	1.5×10^{-4}	5×10^{-4}	7.3×10^{-9}
	Drilling/Spalling	6.4×10^{1}	Table E.2-1	1.5×10^{-7}	9.5×10^{-6}	5×10^{-4}	4.7×10^{-9}
- Remove and Package Contaminated HVAC	Oxyacetylene Torch	5.0×10^{1}	Table E.2-1	2.4×10^{-5}	1.2×10^{-3}	5×10^{-4}	6.0×10^{-7}

TABLE N.2-7. (Contd)

Building/Task[a]	General Method	Operational Time (hr)	Reference[b] Radionuclide Inventory	Airborne Generation Rate Inside Building (Ci/hr)	Total Airborne Radioactivity Inside Building (Ci)	Building[c] Transmission Factor	Total[d] Atmospheric Release (Ci/yr)
Other Contaminated Structures							
3. Office and Laboratory Building (OLB):							
- Remove Contaminated Hoods and Sinks	Oxyacetylene Torch	5.3×10^{1}	Table E.2-1	3.1×10^{-6}	4.0×10^{-4}	5×10^{-4}	2.0×10^{-7}
- Remove Contaminated Concrete	Drilling/Spalling	1.8×10^{1}	Table E.2-1	7.5×10^{-8}	1.1×10^{-6}	5×10^{-4}	6.7×10^{-10}
4. Emergency Retention Basin (ERB) and Site Ditches		1		-8	-6	-4	-10
- Remove and Package Contaminated Piping and Soil	Soil Exhumation	6.6×10^{1}	Table E.2-11	2.9×10^{-6}	1.9×10^{-4}	1.0	1.9×10^{-4}
5. Cold Retention Area (CRA)							
- Remove Contaminated Concrete and Soil	Drilling/Spalling	4.2×10^{2}	Table E.2-1	1.5×10^{-7}	6.3×10^{-5}	5×10^{-4}	3.1×10^{-8}
	Soil Exhumation	1.7×10^{1}	Table E.2	7.4×10^{-7}	1.3×10^{-5}	1.0	1.3×10^{-5}
6. Hot Retention Area (HRA)							
- Remove and Package Contaminated Piping	Oxyacetylene Torch	1.5×10^{1}	Table E.2-1	4.5×10^{-5}	6.7×10^{-4}	5×10^{-4}	3.4×10^{-7}
- Remove and Package HRA Tanks, Floor Plates, and Partitions	Oxyacetylene Torch	1.2×10^{3}	Table E.2-1	1.5×10^{-5}	1.8×10^{-2}	5×10^{-4}	9.2×10^{-6}
- Remove Contaminated Concrete	Drilling/Spalling	7.5×10^{1}	Table E.2-1	2.6×10^{-7}	1.9×10^{-5}	5×10^{-4}	9.6×10^{-9}
7. Fan House							
- Remove Contaminated Concrete	Drilling/Spalling	2.2×10^{1}	Table E.2-1	2.0×10^{-7}	4.4×10^{-6}	5×10^{-4}	2.2×10^{-9}
- Remove Fixed Equipment	Oxyacetylene Torch	7.2×10^{0}	Table E.2-1	1.1×10^{-5}	8.1×10^{-5}	5×10^{-4}	4.0×10^{-8}
- Raze Stack, Segment, and Package	Oxyacetylene Torch	7.6×10^{1}	Table E.2-1	7.7×10^{-6}	5.8×10^{-4}	1×10^{-1}	5.8×10^{-5}
8. Waste Handling Building							
- Remove Fixed Equipment, Including Evaporator and HVAC	Oxyacetylene Torch	1.3×10^{0}	Table E.2-1	4.5×10^{-5}	5.8×10^{-5}	5×10^{-4}	2.9×10^{-8}
- Remove Contaminated Concrete	Drilling/Spalling	5.7×10^{1}	Table E.2-1	2.6×10^{-7}	1.5×10^{-5}	5×10^{-4}	7.3×10^{-9}

(a) Only those tasks that result in an atmospheric release are included in this table.
(b) The reference radionuclide inventories refer to the tables of radionuclides shown in Appendix E.
(c) Normal building HEPA filters have a transmission factor of 5×10^{-4}. Tasks performed outdoors have a transmission factor of 1.0.
(d) For comparison, dose calculations are made based on annual releases.

TABLE N.2-8. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine DECON Tasks at the Reference Research Reactor

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (rem) Total Body	Bone	Lungs	GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total Body	Bone	Lungs	GI-LLI
Reactor Building										
2. Comprehensive Radiation Survey	Table E.1-5	1.0×10^{-10}	3.2×10^{-14}	3.0×10^{-14}	1.2×10^{-13}	3.2×10^{-14}	3.2×10^{-14}	3.2×10^{-14}	3.6×10^{-13}	3.2×10^{-14}
3. General Cleanup	Table E.1-5	1.9×10^{-8}	6.1×10^{-12}	5.7×10^{-12}	2.3×10^{-11}	6.1×10^{-12}	6.1×10^{-12}	6.1×10^{-12}	6.8×10^{-11}	6.1×10^{-12}
	Table E.1-5	9.3×10^{-9}	3.0×10^{-12}	2.8×10^{-12}	1.1×10^{-11}	3.0×10^{-12}	3.0×10^{-12}	3.0×10^{-12}	3.4×10^{-11}	3.0×10^{-12}
7. Remove Reactor Core and Vessel Internals	Table E.1-5	1.1×10^{-8}	3.5×10^{-12}	3.3×10^{-12}	1.3×10^{-11}	3.5×10^{-12}	3.5×10^{-12}	3.5×10^{-12}	4.0×10^{-11}	3.5×10^{-12}
- Contaminated	Table E.1-5	1.3×10^{-7}	4.2×10^{-11}	3.9×10^{-11}	1.6×10^{-10}	4.2×10^{-11}	4.2×10^{-11}	4.2×10^{-11}	4.7×10^{-10}	4.2×10^{-11}
- Activated	Table E.1-6	5.1×10^{-9}	4.0×10^{-13}	3.5×10^{-13}	9.2×10^{-13}	2.4×10^{-13}	6.6×10^{-13}	6.5×10^{-13}	1.7×10^{-12}	2.4×10^{-13}
9. Remove Reactor Vessel	Table E.1-6	1.0×10^{-6}	7.9×10^{-11}	6.9×10^{-11}	1.8×10^{-10}	4.8×10^{-11}	1.3×10^{-10}	1.2×10^{-10}	3.4×10^{-10}	4.8×10^{-11}
11. Remove Activated Concrete	Table E.1-7	6.1×10^{-8}	7.3×10^{-13}	9.2×10^{-13}	3.7×10^{-12}	7.2×10^{-13}	9.8×10^{-13}	2.6×10^{-12}	1.2×10^{-11}	7.9×10^{-13}
12. Remove Reactor Building Equipment	Table E.1-5	3.6×10^{-9}	1.2×10^{-12}	1.1×10^{-12}	4.3×10^{-12}	1.2×10^{-12}	1.2×10^{-12}	1.2×10^{-12}	1.3×10^{-11}	1.2×10^{-12}
13. Remove Piping, Drains, and Sinks	Table E.1-5	1.1×10^{-8}	3.5×10^{-12}	3.3×10^{-12}	1.3×10^{-11}	3.5×10^{-12}	3.5×10^{-12}	3.5×10^{-12}	4.0×10^{-11}	3.5×10^{-12}
14. Remove HVAC and Electrical Systems										
- HVAC	Table E.1-5	1.3×10^{-8}	4.2×10^{-12}	3.9×10^{-12}	1.6×10^{-11}	4.2×10^{-12}	4.2×10^{-12}	4.2×10^{-12}	4.7×10^{-11}	4.2×10^{-12}
- Electrical	Table E.1-5	3.1×10^{-10}	9.9×10^{-14}	9.3×10^{-14}	3.7×10^{-13}	9.9×10^{-14}	9.9×10^{-14}	9.9×10^{-14}	1.1×10^{-12}	9.9×10^{-14}
Reactor Building Totals			1.2×10^{-10}	1.3×10^{-10}	4.1×10^{-10}	1.1×10^{-10}	1.9×10^{-10}	1.8×10^{-10}	1.0×10^{-9}	1.1×10^{-10}
Annex Building										
16. Decontaminate Hot Cell										
- Contaminated Concrete Removal	Table E.1-8	6.9×10^{-8}	2.2×10^{-11}	2.1×10^{-11}	8.3×10^{-11}	2.2×10^{-12}	2.2×10^{-10}	2.2×10^{-10}	2.5×10^{-10}	2.2×10^{-12}
Heat Exchanger Building										
17. Remove Heat Exchanger	Table E.1-5	6.9×10^{-10}	2.2×10^{-13}	2.1×10^{-13}	8.3×10^{-13}	2.2×10^{-13}	2.2×10^{-13}	2.2×10^{-13}	2.5×10^{-12}	2.2×10^{-13}
Pump House										
18. Decon. Walls and Floor	Table E.1-5	2.8×10^{-9}	9.0×10^{-13}	8.4×10^{-13}	3.4×10^{-12}	9.0×10^{-13}	9.0×10^{-13}	9.0×10^{-13}	1.1×10^{-11}	9.0×10^{-13}
19. Remove Retention Tank, Piping, and Equipment	Table E.1-5	5.2×10^{-9}	1.7×10^{-12}	1.6×10^{-12}	6.2×10^{-12}	1.7×10^{-12}	1.7×10^{-12}	1.7×10^{-12}	1.9×10^{-11}	1.7×10^{-12}
Radiation Center Building										
20. Remove Piping and Equipment	Table E.1-5	1.6×10^{-9}	5.1×10^{-13}	4.8×10^{-13}	1.9×10^{-12}	5.1×10^{-13}	5.1×10^{-13}	5.1×10^{-13}	5.8×10^{-12}	5.1×10^{-13}
Test Reactor Totals			1.6×10^{-10}	1.6×10^{-10}	4.8×10^{-10}	1.2×10^{-10}	3.0×10^{-10}	8.7×10^{-10}	1.3×10^{-9}	1.2×10^{-10}

(a) Only those tasks that result in an atmospheric release are shown in this table.
(b) The reference radionuclide inventories refer to the tables of radionuclides shown in Appendix E

TABLE N.2-9. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine DECON Tasks at the Reference Test Reactor

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
			Total Body	Bone	Lungs	GI-LLI	Total Body	Bone	Lungs	GI-LLI
Reactor Building/MUR/ Containment Vessel										
1. Comprehensive Radiation Survey	Table E.2-1	1.8×10^{-7}	5.8×10^{-11}	5.4×10^{-11}	2.2×10^{-10}	5.8×10^{-11}	5.8×10^{-11}	5.8×10^{-11}	6.5×10^{-10}	5.8×10^{-11}
4. General Cleaning and Equipment Inventory (i.e., All Buildings)	Table E.2-1 Table E.2-1	2.9×10^{-7} 1.4×10^{-7}	6.7×10^{-9} 4.5×10^{-11}	6.3×10^{-9} 4.2×10^{-11}	2.5×10^{-8} 1.7×10^{-10}	6.7×10^{-9} 4.5×10^{-11}	6.7×10^{-9} 4.5×10^{-11}	6.7×10^{-9} 4.5×10^{-11}	7.5×10^{-8} 5.0×10^{-10}	6.7×10^{-9} 4.5×10^{-11}
5. Drain, Clean, Dry Quadrants A, B, C, and D and Canals E and F	Table E.2-1	7.7×10^{-7}	2.5×10^{-11}	2.3×10^{-11}	9.2×10^{-10}	2.5×10^{-11}	2.5×10^{-11}	2.5×10^{-11}	2.8×10^{-10}	2.5×10^{-11}
6. Drain, Clean, Dry Canal H	Table E.2-1	1.5×10^{-7}	4.8×10^{-11}	4.5×10^{-11}	1.8×10^{-10}	4.8×10^{-11}	4.8×10^{-11}	4.8×10^{-11}	5.4×10^{-10}	4.8×10^{-11}
7. Remove, Package, and Ship MUR and Associated Hardware	Table E.2-1	2.5×10^{-8}	8.0×10^{-12}	7.5×10^{-12}	3.0×10^{-11}	8.0×10^{-12}	8.0×10^{-12}	8.0×10^{-12}	9.0×10^{-11}	8.0×10^{-12}
8. Drain, Clean, Dry Canal G	Table E.2-1	9.0×10^{-9}	2.9×10^{-12}	2.7×10^{-12}	1.1×10^{-11}	2.9×10^{-12}	2.9×10^{-12}	2.9×10^{-12}	3.2×10^{-11}	2.9×10^{-12}
12. Remove RV Internals and Ship Activated RV Internals	Table E.2-1 Table E.2-2	2.6×10^{-7} 1.9×10^{-10}	8.3×10^{-11} 1.5×10^{-14}	7.8×10^{-11} 1.3×10^{-14}	3.1×10^{-12} 3.4×10^{-14}	8.3×10^{-11} 9.1×10^{-15}	8.3×10^{-11} 2.5×10^{-14}	8.3×10^{-11} 2.4×10^{-14}	9.4×10^{-10} 6.5×10^{-14}	8.3×10^{-11} 9.1×10^{-15}
13. Remove RV and Ship RV Segments	Table E.2-1 Table E.2-2	4.9×10^{-5} 2.8×10^{-6}	1.6×10^{-8} 2.2×10^{-10}	1.5×10^{-8} 1.9×10^{-10}	5.9×10^{-8} 5.0×10^{-10}	1.6×10^{-8} 1.3×10^{-10}	1.6×10^{-8} 3.6×10^{-10}	1.6×10^{-8} 3.5×10^{-10}	1.8×10^{-7} 9.5×10^{-10}	1.6×10^{-8} 1.3×10^{-10}
14. Remove Bio-Shield Concrete	Table E.2-3	8.6×10^{-9}	1.0×10^{-13}	1.4×10^{-13}	5.2×10^{-13}	1.0×10^{-13}	1.4×10^{-13}	3.7×10^{-13}	1.6×10^{-12}	1.1×10^{-13}
15. Remove Fixed Equipment in CV (Except HVAC)	Table E.2-1	6.0×10^{-8}	1.9×10^{-11}	1.8×10^{-11}	7.2×10^{-11}	1.9×10^{-11}	1.9×10^{-11}	1.9×10^{-11}	2.2×10^{-10}	1.9×10^{-11}
16. Remove Fixed Equipment Outside CV	Table E.2-1	6.7×10^{-10}	2.1×10^{-13}	2.0×10^{-13}	8.0×10^{-13}	2.1×10^{-13}	2.1×10^{-13}	2.1×10^{-13}	2.4×10^{-12}	2.1×10^{-13}
17. Remove Quadrant Piping	Table E.2-1	5.8×10^{-7}	1.9×10^{-10}	1.7×10^{-10}	7.0×10^{-10}	1.9×10^{-10}	1.9×10^{-10}	1.9×10^{-10}	2.1×10^{-9}	1.9×10^{-10}
18. Segment and Remove Sub-Pile Room	Table E.2-1	4.5×10^{-7}	1.4×10^{-10}	1.4×10^{-10}	5.4×10^{-10}	1.4×10^{-10}	1.4×10^{-10}	1.4×10^{-10}	1.6×10^{-9}	1.4×10^{-10}
19. Remove Lead Shield from Below Reactor Cavity	Table E.2-1	1.3×10^{-6}	4.2×10^{-10}	3.9×10^{-10}	1.6×10^{-9}	4.2×10^{-10}	4.2×10^{-10}	4.2×10^{-10}	4.7×10^{-9}	4.2×10^{-10}
20. Remove Pipes from Bio-Shield	Table E.2-1	3.4×10^{-6}	1.1×10^{-9}	1.0×10^{-9}	4.1×10^{-9}	1.1×10^{-9}	1.1×10^{-9}	1.1×10^{-9}	1.2×10^{-8}	1.1×10^{-9}
21. Remove PCWC Piping from Bio-Shield	Table E.2-1	1.3×10^{-6}	4.2×10^{-10}	3.9×10^{-10}	1.6×10^{-9}	4.2×10^{-10}	4.2×10^{-10}	4.2×10^{-10}	4.7×10^{-9}	4.2×10^{-10}
22. Remove RB/CV Contaminated Concrete	Table E.2-1	4.0×10^{-6}	1.3×10^{-9}	1.2×10^{-9}	4.8×10^{-9}	1.3×10^{-9}	1.3×10^{-9}	1.3×10^{-9}	1.4×10^{-8}	1.3×10^{-9}

TABLE N.2-9. (Contd)

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (rem) Total Body	Bone	Lungs	GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total Body	Bone	Lungs	GI-LLI
Reactor Building/MUR/Containment Vessel										
23. Remove Q and C, and Misc. Contaminated Drains	Table E.2-1	7.2×10^{-6}	2.3×10^{-9}	2.2×10^{-9}	8.6×10^{-9}	2.3×10^{-9}	2.3×10^{-9}	2.3×10^{-9}	2.6×10^{-8}	2.3×10^{-9}
24. Remove Contaminated HVAC from RB/CV	Table E.2-1	3.1×10^{-7}	9.9×10^{-11}	9.3×10^{-11}	3.7×10^{-10}	9.9×10^{-11}	9.9×10^{-11}	9.9×10^{-11}	1.1×10^{-9}	9.9×10^{-11}
RV/MUR/CV Totals			3.0×10^{-8}	2.7×10^{-8}	1.1×10^{-7}	3.0×10^{-8}	3.0×10^{-8}	3.0×10^{-8}	3.2×10^{-7}	3.0×10^{-8}
Hot Laboratory Building										
1. Decontaminate Hot Cells	Table E.2-10	1.2×10^{-6}	5.3×10^{-10}	1.4×10^{-9}	2.3×10^{-9}	4.2×10^{-10}	4.6×10^{-9}	3.0×10^{-8}	9.0×10^{-9}	4.2×10^{-10}
2. Remove and Package Hot Cell Equipment and Piping	Table E.2-10	4.3×10^{-7}	1.9×10^{-10}	5.2×10^{-10}	8.2×10^{-10}	1.5×10^{-10}	1.6×10^{-9}	1.1×10^{-8}	3.2×10^{-9}	1.5×10^{-10}
3. Remove and Package Hot Cell SS Cladding	Table E.2-10	7.0×10^{-6}	3.1×10^{-9}	8.4×10^{-9}	1.3×10^{-8}	2.4×10^{-9}	2.7×10^{-8}	1.8×10^{-7}	5.3×10^{-8}	2.4×10^{-9}
4. Remove Contaminated Concrete from Hot Cells	Table E.2-10	7.9×10^{-10}	3.5×10^{-13}	9.5×10^{-13}	1.5×10^{-12}	2.8×10^{-13}	3.0×10^{-12}	2.0×10^{-11}	5.9×10^{-12}	2.8×10^{-13}
5. Decon. the HLB (Including the Cranes)	Table E.2-10	2.0×10^{-7}	8.8×10^{-11}	2.4×10^{-10}	3.8×10^{-10}	7.0×10^{-11}	7.6×10^{-10}	5.0×10^{-9}	1.5×10^{-9}	7.0×10^{-11}
6. Drain, Clean, Dry Canals J and K	Table E.2-1	6.1×10^{-8}	2.0×10^{-11}	1.8×10^{-11}	7.3×10^{-11}	2.0×10^{-11}	2.0×10^{-11}	2.0×10^{-11}	2.2×10^{-10}	2.0×10^{-11}
8. Remove Fixed and Permanent Equipment (Except HVAC), Including the Hot Pipe Tunnel	Table E.2-10	2.0×10^{-7}	8.8×10^{-11}	2.4×10^{-10}	3.8×10^{-10}	7.0×10^{-11}	7.6×10^{-10}	5.0×10^{-9}	1.5×10^{-9}	7.0×10^{-11}
9. Remove SS Cladding from Decon. Room 23	Table E.2-10	8.9×10^{-8}	3.9×10^{-11}	1.1×10^{-10}	1.7×10^{-10}	3.1×10^{-11}	3.4×10^{-10}	2.2×10^{-9}	6.7×10^{-10}	3.1×10^{-11}
11. Remove and Package HLB Contaminated Concrete	Table E.2-10	1.5×10^{-8}	6.6×10^{-12}	1.8×10^{-11}	2.8×10^{-11}	5.2×10^{-12}	5.7×10^{-11}	3.8×10^{-10}	1.1×10^{-10}	5.2×10^{-12}
12. Remove and Package Contaminated HVAC from HLB	Table E.2-10	6.3×10^{-8}	2.8×10^{-11}	7.6×10^{-10}	1.2×10^{-10}	2.2×10^{-11}	2.4×10^{-10}	1.6×10^{-9}	4.7×10^{-10}	2.2×10^{-11}
Hot Lab. Totals			4.3×10^{-9}	1.2×10^{-8}	1.8×10^{-8}	3.4×10^{-9}	3.8×10^{-8}	2.5×10^{-7}	7.4×10^{-8}	3.4×10^{-9}
All Other Buildings and Areas										
1. Radiation Survey	Table E.2-1	7.9×10^{-8}	2.5×10^{-11}	2.4×10^{-11}	9.5×10^{-11}	2.5×10^{-11}	2.5×10^{-11}	2.5×10^{-11}	2.8×10^{-10}	2.5×10^{-11}
2. Primary Pump House										
- Remove Fixed Equipment (Except HVAC)	Table E.2-1	2.2×10^{-9}	7.0×10^{-13}	6.6×10^{-13}	2.6×10^{-12}	7.0×10^{-13}	7.0×10^{-13}	7.0×10^{-13}	7.9×10^{-12}	7.0×10^{-13}
- Decon. PPH and Remove Contaminated Concrete	Table E.2-01	7.3×10^{-8}	2.3×10^{-11}	2.2×10^{-11}	8.8×10^{-11}	2.2×10^{-11}	2.2×10^{-11}	2.2×10^{-11}	2.6×10^{-10}	2.2×10^{-11}
	Table E.2-1	4.7×10^{-9}	1.5×10^{-12}	1.4×10^{-12}	5.6×10^{-12}	1.5×10^{-12}	1.5×10^{-12}	1.5×10^{-12}	1.7×10^{-11}	1.5×10^{-12}
- Remove and Package Contaminated HVAC	Table E.2-1	6.0×10^{-7}	1.9×10^{-10}	1.8×10^{-10}	7.2×10^{-10}	1.9×10^{-10}	1.9×10^{-10}	1.9×10^{-10}	2.2×10^{-9}	1.9×10^{-10}

TABLE N.2-9. (Contd)

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (rem) Total Body	Bone	Lungs	GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total Body	Bone	Lungs	GI-LLI
All Other Buildings and Areas										
3. Office and Laboratory Building (OLB):										
- Remove Contaminated Hoods and Sinks	Table E.2-1	2.0×10^{-7}	6.4×10^{-11}	6.0×10^{-11}	2.4×10^{-10}	6.4×10^{-11}	6.4×10^{-11}	6.4×10^{-11}	7.2×10^{-10}	6.4×10^{-11}
- Remove Contaminated Concrete	Table E.2-1	6.7×10^{-10}	2.1×10^{-13}	2.0×10^{-13}	8.0×10^{-13}	2.1×10^{-13}	2.1×10^{-13}	2.1×10^{-13}	2.4×10^{-12}	2.1×10^{-13}
4. Emergency Retention Basin (ERB) and Site Ditches										
- Remove and Package Contaminated Piping and Soil	Table E.2-11	1.9×10^{-4}	1.2×10^{-7}	1.3×10^{-7}	2.8×10^{-7}	8.6×10^{-8}	2.3×10^{-7}	1.6×10^{-7}	8.7×10^{-7}	8.6×10^{-8}
5. Cold Retention Area (CRA)	Table E.2-1	3.1×10^{-8}	9.9×10^{-12}	9.3×10^{-12}	3.7×10^{-11}	9.9×10^{-12}	9.9×10^{-12}	9.9×10^{-12}	1.1×10^{-10}	9.9×10^{-12}
- Remove Contaminated Concrete and Soil	Table E.2-1	3.1×10^{-8}	9.9×10^{-12}	9.3×10^{-12}	3.7×10^{-11}	9.9×10^{-12}	9.9×10^{-12}	9.9×10^{-12}	1.1×10^{-10}	9.9×10^{-12}
	Table E.2-11	1.3×10^{-5}	8.3×10^{-9}	9.1×10^{-9}	2.0×10^{-8}	5.8×10^{-9}	1.6×10^{-8}	5.2×10^{-8}	6.0×10^{-8}	5.9×10^{-9}
6. Hot Retention Area (HRA)										
- Remove and Package Contaminated Piping	Table E.2-1	3.4×10^{-7}	1.1×10^{-10}	1.0×10^{-10}	4.1×10^{-10}	1.1×10^{-10}	1.1×10^{-10}	1.1×10^{-10}	1.2×10^{-9}	1.1×10^{-10}
- Remove and Package MRA Tanks, Floor Plates, and Partitions	Table E.2-1	9.2×10^{-6}	2.9×10^{-9}	2.8×10^{-9}	1.1×10^{-8}	2.9×10^{-9}	2.9×10^{-9}	2.9×10^{-9}	3.3×10^{-8}	2.9×10^{-9}
- Remove Contaminated Concrete	Table E.2-1	9.6×10^{-9}	3.1×10^{-12}	2.9×10^{-12}	1.2×10^{-11}	3.1×10^{-12}	3.1×10^{-12}	3.1×10^{-12}	3.5×10^{-11}	3.1×10^{-12}
7. Fan House										
- Remove Contaminated Concrete	Table E.2-1	2.2×10^{-9}	7.0×10^{-13}	6.6×10^{-13}	2.6×10^{-12}	7.0×10^{-13}	7.0×10^{-13}	7.0×10^{-13}	7.9×10^{-12}	7.0×10^{-13}
- Remove Fixed Equipment	Table E.2-1	4.0×10^{-8}	1.3×10^{-11}	1.2×10^{-11}	4.8×10^{-11}	1.3×10^{-11}	1.3×10^{-11}	1.3×10^{-11}	1.4×10^{-10}	1.3×10^{-11}
- Raze Stack, Segment, and Package	Table E.2-1	5.8×10^{-5}	1.9×10^{-8}	1.7×10^{-8}	7.0×10^{-8}	1.9×10^{-8}	1.9×10^{-8}	1.9×10^{-8}	2.1×10^{-7}	1.9×10^{-8}
8. Waste Handling Building										
- Remove Fixed Equipment, Including Evaporator and HVAC	Table E.2-1	2.9×10^{-8}	9.3×10^{-12}	8.7×10^{-12}	3.5×10^{-11}	9.3×10^{-12}	9.3×10^{-12}	9.3×10^{-12}	1.0×10^{-10}	9.3×10^{-12}
- Remove Contaminated Concrete	Table E.2-1	7.3×10^{-9}	2.3×10^{-12}	2.2×10^{-12}	8.8×10^{-12}	2.3×10^{-12}	2.3×10^{-12}	2.3×10^{-12}	2.6×10^{-11}	2.3×10^{-12}
All Other Buildings and Areas Totals			$15. \times 10^{-7}$	1.6×10^{-7}	3.8×10^{-7}	1.1×10^{-7}	2.7×10^{-7}	8.3×10^{-7}	1.2×10^{-6}	1.1×10^{-7}
Test Reactor Totals			1.8×10^{-7}	2.0×10^{-7}	5.1×10^{-7}	1.4×10^{-7}	3.4×10^{-7}	1.1×10^{-6}	1.6×10^{-6}	1.4×10^{-7}

(a) Only those tasks that result in an atmospheric release are shown in this table.
(b) The reference radionuclide inventories refer to the tables shown in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

release of radioactivity are included in these tables. Population doses resulting from the atmospheric releases during DECON are shown in Table N.2-10 for the research reactor and in Table N.2-11 for the test reactor. The atmospheric dispersion ($\overline{X}/Q'$) values used for the routine release dose calculations for both reactor sites are 7.5×10^{-6} sec/m^3 for the maximum-exposed individual and 2.4×10^{-9} sec/m^3 for the population. A total of 1.4×10^6 people is assumed to reside within an 80-km radius of the research reactor site, and a total of 3.5×10^6 people is assumed to reside within an 80-km radius of the test reactor. More information on the reference sites is found in Appendix A, and more information on the radiation dose models is found in Appendix F.

N.2.2 Postulated Accidents During DECON

During DECON, situations may arise that lead to the accidental atmospheric release of radioactivity. Accident scenarios and calculated airborne radioactivity generation rates are listed in Table N.2-12 for the reference research reactor and in Table N.2-13 for the reference test reactor. Accident scenarios are based on the technologies used during DECON tasks. A general estimate of the frequency of occurrence of the level of the atmospheric release associated with each postulated accident is also given. The frequency of occurrence is arrived at by considering not only the probability of the accident but also the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar or greater magnitude per year is greater than 10^{-2}, as "medium" if between 10^{-2} and 10^{-5}, and as "low" if less than 10^{-5}.

While it is beyond the scope of this study to try to evaluate every potential accident for decommissioning reference R&T reactors, operational, natural phenomena, and indirect man-related events are discussed in the following subsections. Multiple failure-event accidents are not considered; that is, for each accident, only one failure event is analyzed.

N.2.2.1 Postulated Operational Accidents

Generic accidental atmospheric releases from the incidents such as equipment failure, human error, or service conditions during routine DECON tasks are discussed in this section. The accident analyses are based on the assumed technologies for DECON tasks discussed in Appendix I.

TABLE N.2-10. Radiation Doses to the Population from Atmospheric Releases During Routine DECON Tasks at the Reference Research Reactor

Building/Task[a]	Reference Radionuclide Inventory[b]	Total Atmospheric Release (Ci/hr)[c]	First-Year Dose (man-rem)[d]				Fifty-Year Committed Dose Equivalent (man-rem)[d]			
			Total Body	Bone	Lungs	GI-LLI	Total Body	Bone	Lungs	GI-LLI
Reactor Building										
2. Comprehensive Radiation Survey	Table E.1-5	1.0×10^{-10}	7.2×10^{-12}	6.7×10^{-12}	4.7×10^{-11}	7.8×10^{-12}	7.3×10^{-12}	7.2×10^{-12}	1.6×10^{-10}	7.8×10^{-12}
3. General Cleanup	Table E.1-5	1.9×10^{-8}	1.4×10^{-9}	1.3×10^{-9}	8.9×10^{-9}	1.5×10^{-9}	1.4×10^{-9}	1.4×10^{-9}	3.0×10^{-8}	1.5×10^{-9}
	Table E.1-5	9.3×10^{-9}	6.7×10^{-10}	6.2×10^{-10}	4.4×10^{-9}	7.2×10^{-10}	6.9×10^{-10}	6.7×10^{-10}	1.5×10^{-8}	7.2×10^{-10}
7. Remove Reactor Core	Table E.1-5	1.1×10^{-9}	7.9×10^{-10}	7.45×10^{10}	5.2×10^{-9}	8.6×10^{-10}	8.2×10^{-10}	7.9×10^{-10}	1.8×10^{-8}	8.6×10^{-10}
- Contaminated	Table E.1-5	1.3×10^{-9}	9.4×10^{-10}	8.7×10^{-10}	6.1×10^{-9}	1.0×10^{-10}	9.6×10^{-10}	9.4×10^{-10}	2.1×10^{-8}	1.0×10^{-10}
- Activated	Table E.1-6	5.1×10^{-9}	1.1×10^{-10}	9.2×10^{-11}	3.7×10^{-10}	6.1×10^{-11}	1.8×10^{-10}	1.6×10^{-10}	7.1×10^{-10}	6.1×10^{-11}
9. Remove Reactor Vessel	Table E.1-6	1.0×10^{-6}	2.1×10^{-8}	1.8×10^{-8}	7.3×10^{-8}	1.2×10^{-8}	3.6×10^{-8}	3.2×10^{-8}	1.4×10^{-7}	1.2×10^{-8}
11. Remove Activated Concrete	Table E.1-7	6.1×10^{-8}	1.8×10^{-10}	2.4×10^{-10}	1.5×10^{-9}	2.0×10^{-10}	2.4×10^{-10}	7.3×10^{-10}	5.1×10^{-9}	2.0×10^{-10}
12. Remove Reactor Building Equipment	Table E.1-5	3.6×10^{-9}	2.6×10^{-10}	2.4×10^{-10}	1.7×10^{-9}	2.8×10^{-10}	2.6×10^{-10}	2.6×10^{-10}	5.8×10^{-9}	2.8×10^{-10}
13. Remove Piping, Drains, and Sinks	Table E.1-5	1.1×10^{-9}	7.9×10^{-10}	7.4×10^{-10}	5.2×10^{-9}	8.6×10^{-10}	8.1×10^{-10}	7.9×10^{-10}	1.8×10^{-9}	8.6×10^{-10}
14. Remove HVAC and Electrical Systems										
- HVAC	Table E.1-5	1.3×10^{-10}	9.4×10^{-10}	8.7×10^{-9}	6.1×10^{-9}	1.0×10^{-10}	9.6×10^{-10}	9.4×10^{-8}	2.1×10^{-9}	1.0×10
- Electrical	Table E.1-5	3.1×10^{-10}	2.2×10^{-11}	2.1×10^{-11}	1.5×10^{-10}	2.4×10^{-11}	2.3×10^{-11}	2.2×10^{-11}	5.0×10^{-10}	2.4×10^{-11}
Reactor Building Totals			3.5×10^{-8}	3.1×10^{-8}	1.6×10^{-7}	2.7×10^{-8}	5.0×10^{-8}	4.6×10^{-8}	4.5×10^{-7}	2.7×10^{-8}
Annex Building										
16. Decontaminate Hot Cell										
- Contaminated Concrete Removal	Table E.1-8	2.7×10^{-8}	3.5×10^{-9}	1.2×10^{-8}	2.3×10^{-8}	2.3×10^{-9}	3.0×10^{-8}	2.5×10^{-7}	9.2×10^{-8}	2.3×1^{-9}
Heat Exchanger Building										
17. Remove Heat Exchanger	Table E.1-5	6.9×10^{-10}	5.0×10^{-11}	4.6×10^{-11}	3.2×10^{-10}	5.4×10^{-11}	5.2×10^{-11}	5.0×10^{-11}	1.1×10^{-9}	5.4×10^{-11}
Pump House										
18. Decon. Walls and Floor	Table E.1-5	2.8×10^{-9}	2.0×10^{-10}	1.9×10^{-10}	1.3×10^{-9}	2.2×10^{-10}	2.1×10^{-10}	2.0×10^{-10}	4.5×10^{-9}	2.2×10^{-10}
19. Remove Retention Tank, Piping, and Equipment	Table E.1-5	5.2×10^{-9}	3.7×10^{-10}	3.5×10^{-10}	2.4×10^{-9}	4.1×10^{-10}	3.8×10^{-10}	3.7×10^{-10}	8.3×10^{-9}	4.1×10^{-10}
Radiation Center Building										
20. Remove Piping and Equipment	Table E.1-5	1.6×10^{-9}	1.2×10^{-10}	1.1×10^{-10}	7.5×10^{-10}	1.2×10^{-10}	1.2×10^{-10}	1.2×10^{-10}	2.6×10^{-9}	1.2×10^{-10}
Research Reactor Totals			3.9×10^{-8}	4.4×10^{-8}	1.9×10^{-7}	3.0×10^{-8}	8.1×10^{-8}	3.0×10^{-7}	5.6×10^{-7}	3.0×10^{-8}

(a) Only those tasks that result in an atmospheric release are included in this table.
(b) The reference radionuclide inventories refer to the tables of radionuclides shown in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.
(d) For the research reactor, the total population within 80-km of the reference site is 1.4×10^6.

TABLE N.2-11. Radiation Doses to the Population from Atmospheric Releases During Routine DECON Tasks at the Reference Test Reactor

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (man-rem)[d]				Fifty-Year Committed Dose Equivalent (man-rem)[d]			
			Total Body	Bone	Lungs	GI-LLI	Total Body	Bone	Lungs	GI-LLI
Reactor Building/MUR/ Containment Vessel										
1. Comprehensive Radiation Survey	Table E.2-1	1.8×10^{-7}	3.2×10^{-8}	3.1×10^{-8}	2.2×10^{-7}	3.4×10^{-8}	3.2×10^{-8}	3.2×10^{-8}	7.0×10^{-7}	3.4×10^{-8}
4. General Cleanup and Equipment Inventory (i.e., All Buildings)	Table E.2-1	2.9×10^{-7}	5.2×10^{-8}	4.9×10^{-8}	3.5×10^{-7}	5.5×10^{-8}	5.2×10^{-8}	5.2×10^{-8}	1.1×10^{-6}	5.5×10^{-8}
	Table E.2-1	1.4×10^{-7}	2.5×10^{-7}	2.4×10^{-8}	1.7×10^{-7}	2.7×10^{-8}	2.5×10^{-7}	2.5×10^{-7}	5.5×10^{-7}	2.7×10^{-8}
5. Drain, Clean, Dry Quadrants A, B, C, and D and Canals E and F	Table E.2-1	7.7×10^{-8}	1.4×10^{-8}	1.3×10^{-8}	9.2×10^{-8}	1.5×10^{-8}	1.4×10^{-8}	1.4×10^{-8}	3.0×10^{-7}	1.5×10^{-8}
6. Drain, Clean, Dry Canal H	Table E.2-1	1.5×10^{-8}	2.7×10^{-9}	2.6×10^{-9}	1.8×10^{-8}	2.8×10^{-9}	2.7×10^{-9}	2.7×10^{-9}	5.8×10^{-8}	2.8×10^{-9}
7. Remove, Package, and Ship MUR and Associated Hardware	Table E.2-1	2.5×10^{-8}	4.5×10^{-9}	4.2×10^{-9}	3.0×10^{-8}	4.8×10^{-9}	4.5×10^{-9}	4.5×10^{-9}	9.8×10^{-8}	4.8×10^{-9}
8. Drain, Clean, Dry Canal G	Table E.2-1	9.0×10^{-9}	1.6×10^{-9}	1.5×10^{-9}	1.1×10^{-8}	1.7×10^{-9}	1.6×10^{-9}	1.6×10^{-9}	3.5×10^{-8}	1.7×10^{-9}
12. Remove RV Internals and Ship Activated RV Internals	Table E.2-1	2.6×10^{-7}	4.7×10^{-8}	4.4×10^{-8}	3.1×10^{-7}	4.9×10^{-8}	4.7×10^{-8}	4.7×10^{-8}	1.0×10^{-6}	4.9×10^{-8}
	Table E.2-2	1.9×10^{-10}	9.9×10^{-12}	8.4×10^{-12}	3.4×10^{-11}	5.7×10^{-12}	1.7×10^{-11}	1.5×10^{-11}	6.8×10^{-11}	5.7×10^{-12}
13. Remove RV and Ship RV Segments	Table E.2-1	4.9×10^{-7}	8.8×10^{-8}	8.3×10^{-8}	5.9×10^{-7}	9.3×10^{-8}	8.8×10^{-8}	8.8×10^{-8}	1.9×10^{-6}	9.3×10^{-8}
	Table E.2-2	2.8×10^{-6}	1.5×10^{-7}	1.2×10^{-7}	5.0×10^{-7}	8.4×10^{-8}	2.5×10^{-7}	2.2×10^{-7}	1.0×10^{-6}	8.4×10^{-8}
14. Remove Bio-Shield Concrete	Table E.2-3	8.6×10^{-9}	2.5×10^{-11}	3.4×10^{-11}	2.1×10^{-10}	2.8×10^{-11}	3.4×10^{-11}	1.0×10^{-10}	7.0×10^{-10}	2.8×10^{-11}
15. Remove Fixed Equipment in CV (Except HVAC)	Table E.2-1	6.0×10^{-8}	1.1×10^{-8}	1.0×10^{-8}	7.2×10^{-8}	1.1×10^{-8}	1.1×10^{-8}	1.1×10^{-8}	2.3×10^{-7}	1.1×10^{-8}
16. Remove Fixed Equipment Outside CV	Table E.2-1	6.7×10^{-10}	1.2×10^{-10}	1.1×10^{-10}	8.0×10^{-10}	1.3×10^{-10}	1.2×10^{-10}	1.2×10^{-10}	2.6×10^{-9}	1.3×10^{-10}
17. Remove Quadrant Piping	Table E.2-1	5.8×10^{-7}	1.0×10^{-7}	9.9×10^{-8}	7.0×10^{-7}	1.1×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	2.3×10^{-6}	1.1×10^{-7}
18. Segment and Remove Sub-Pile Room	Table E.2-1	4.5×10^{-7}	8.1×10^{-8}	7.6×10^{-8}	5.4×10^{-7}	8.6×10^{-8}	8.1×10^{-8}	8.1×10^{-8}	1.8×10^{-6}	8.6×10^{-8}
19. Remove Lead Shield from Below Reactor Cavity	Table E.2-1	1.3×10^{-6}	2.3×10^{-7}	2.2×10^{-7}	1.6×10^{-6}	2.5×10^{-7}	2.3×10^{-7}	2.3×10^{-7}	5.1×10^{-6}	2.5×10^{-7}
20. Remove Pipes from Bio-Shield	Table E.2-1	3.4×10^{-6}	7.2×10^{-8}	6.8×10^{-8}	4.8×10^{-7}	7.6×10^{-8}	7.2×10^{-8}	7.2×10^{-8}	1.6×10^{-6}	7.6×10^{-8}
21. Remove PCWC Piping from Bio-Shield	Table E.2-1	1.3×10^{-6}	2.3×10^{-7}	2.2×10^{-7}	1.6×10^{-6}	2.5×10^{-7}	2.3×10^{-6}	2.3×10^{-6}	5.1×10^{-6}	2.5×10^{-6}

TABLE N.2-11. (Contd)

Building/Task[a]	Reference Radionuclide Inventory[b]	Total Atmospheric Release (Ci/yr)[c]	First-Year Dose (man-rem)[d] Total Body	Bone	Lungs	GI-LLI	Fifty-Year Committed Dose Equivalent (man-rem)[d] Total Body	Bone	Lungs	GI-LLI
Reactor Building/MUR/Containment Vessel										
22. Remove RB/CV Contaminated Concrete	Table E.2-1	4.0×10^{-6}	7.2×10^{-7}	6.8×10^{-7}	4.8×10^{-6}	7.6×10^{-7}	7.2×10^{-7}	7.2×10^{-7}	1.6×10^{-6}	7.6×10^{-7}
23. Remove Q and C, and Misc. Contaminated Drains	Table E.2-1	7.2×10^{-6}	1.3×10^{-6}	1.2×10^{-6}	8.6×10^{-6}	1.4×10^{-6}	1.3×10^{-6}	1.3×10^{-6}	2.8×10^{-5}	1.4×10^{-6}
24. Remove Contaminated HVAC from RB/CV	Table E.2-1	3.1×10^{-7}	5.6×10^{-8}	5.3×10^{-8}	3.7×10^{-7}	5.9×10^{-8}	5.6×10^{-8}	5.6×10^{-8}	1.2×10^{-6}	5.9×10^{-8}
RB/MUR/CV Totals			1.2×10^{-5}	1.1×10^{-5}	8.0×10^{-5}	1.3×10^{-5}	1.4×10^{-5}	1.4×10^{-5}	2.4×10^{-4}	1.5×10^{-5}
Hot Laboratory Building										
1. Decontaminate Hot Cells	Table E.2-10	1.2×10^{-6}	3.7×10^{-7}	1.3×10^{-6}	2.5×10^{-6}	2.5×10^{-7}	3.2×10^{-6}	2.8×10^{-5}	1.0×10^{-5}	2.5×10^{-7}
2. Remove and Package Hot Cell Equipment and Piping	Table E.2-10	4.3×10^{-7}	1.3×10^{-7}	4.7×10^{-7}	9.0×10^{-7}	9.0×10^{-8}	1.2×10^{-6}	9.9×10^{-6}	3.6×10^{-6}	9.0×10^{-8}
3. Remove and Package Hot Cell SS Cladding	Table E.2-10	7.0×10^{-6}	2.2×10^{-6}	7.7×10^{-6}	1.5×10^{-5}	1.5×10^{-6}	1.9×10^{-5}	1.6×10^{-4}	5.8×10^{-5}	1.5×10^{-6}
4. Remove Contaminated Concrete from Hot Cells	Table E.2-10	7.9×10^{-10}	2.4×10^{-10}	8.7×10^{-10}	1.7×10^{-9}	1.7×10^{-10}	2.1×10^{-9}	1.8×10^{-8}	6.6×10^{-9}	1.7×10^{-10}
5. DECON the HLB (Including the cranes)	Table E.2-10	2.0×10^{-7}	6.2×10^{-8}	2.2×10^{-7}	4.2×10^{-7}	4.2×10^{-8}	5.4×10^{-7}	4.6×10^{-6}	1.7×10^{-6}	4.2×10^{-8}
6. Drain, Clean, Dry Canals J and K	Table E.2-1	6.1×10^{-8}	1.1×10^{-8}	1.0×10^{-8}	7.3×10^{-8}	1.2×10^{-8}	1.1×10^{-8}	1.1×10^{-8}	2.4×10^{-7}	1.2×10^{-8}
8. Remove Fixed and Permanent Equipment (Except HVAC), Including the Hot Pipe Tunnel	Table E.2-10	2.0×10^{-7}	6.2×10^{-8}	2.2×10^{-7}	4.2×10^{-7}	4.2×10^{-8}	5.4×10^{-7}	4.6×10^{-6}	1.7×10^{-6}	4.2×10^{-8}
9. Remove SS Cladding from Decon. Room 23	Table E.2-10	8.9×10^{-8}	2.8×10^{-8}	9.8×10^{-8}	1.9×10^{-7}	1.9×10^{-8}	1.9×10^{-7}	2.0×10^{-6}	7.4×10^{-7}	1.9×10^{-8}
11. Remove and Package HLB Contaminated Concrete	Table E.2-10	1.5×10^{-8}	4.6×10^{-9}	1.6×10^{-8}	3.2×10^{-8}	3.2×10^{-9}	4.0×10^{-8}	3.4×10^{-7}	1.2×10^{-7}	3.2×10^{-9}
12. Remove and Package Contaminated HVAC from HLR	Table E.2-10	6.3×10^{-7}	2.0×10^{-7}	6.9×10^{-7}	1.3×10^{-6}	1.3×10^{-7}	1.7×10^{-6}	1.4×10^{-5}	5.2×10^{-6}	1.3×10^{-7}
Hot Lab. Totals			3.1×10^{-6}	1.1×10^{-5}	2.1×10^{-5}	2.1×10^{-6}	2.6×10^{-5}	2.2×10^{-4}	8.1×10^{-5}	2.1×10^{-6}
All Other Buildings and Areas										
1. Radiation Survey	Table E.2-1	7.9×10^{-8}	1.3×10^{-8}	1.3×10^{-8}	9.4×10^{-8}	1.5×10^{-8}	1.3×10^{-8}	1.3×10^{-8}	3.0×10^{-7}	1.5×10^{-8}
2. Primary Pump House										
- Remove Fixed Equipment (Except HVAC)	Table E.2-1	2.2×10^{-9}	3.8×10^{-10}	3.4×10^{-10}	2.6×10^{-10}	4.2×10^{-10}	3.8×10^{-10}	3.8×10^{-10}	8.6×10^{-9}	4.2×10^{-10}
- Decon. PPH and Remove Contaminated Concrete	Table E.2-1	7.3×10^{-8}	1.2×10^{-8}	1.2×10^{-8}	8.8×10^{-8}	1.4×10^{-8}	1.2×10^{-8}	1.2×10^{-8}	2.8×10^{-7}	1.4×10^{-8}
	Table E.2-1	4.7×10^{-9}	8.1×10^{-10}	8.0×10^{-10}	5.6×10^{-9}	8.9×10^{-10}	8.1×10^{-10}	8.1×10^{-10}	1.8×10^{-9}	8.9×10^{-10}
- Remove and Package Contaminated HVAC	Table E.2-1	6.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	7.2×10^{-7}	1.1×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	2.3×10^{-6}	1.1×10^{-7}

TABLE N.2-11. (Contd)

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (man-rem)[d] Total Body	Bone	Lungs	GI-LLI	Fifty-Year Committed Dose Equivalent (man-rem)[d] Total Body	Bone	Lungs	GI-LLI
All Other Buildings and Areas										
3. Office and Laboratory Building (OLB):										
- Remove Contaminated Hoods and Sinks	Table E.2-1	2.0×10^{-7}	3.4×10^{-8}	3.4×10^{-8}	2.4×10^{-7}	3.8×10^{-8}	3.4×10^{-8}	3.4×10^{-8}	7.8×10^{-7}	3.8×10^{-8}
- Remove Contaminated Concrete	Table E.2-1	6.7×10^{-10}	1.2×10^{-10}	1.1×10^{-10}	8.0×10^{-10}	1.3×10^{-10}	1.1×10^{-10}	1.1×10^{-10}	2.6×10^{-9}	1.3×10^{-10}
4. Emergency Retention Basin (ERB) and Site Ditches										
- Remove and Package Contaminated Piping and Soil	Table E.2-11	1.9×10^{-4}	7.8×10^{-5}	8.9×10^{-5}	2.8×10^{-4}	5.1×10^{-5}	1.6×10^{-4}	2.7×10^{-4}	9.3×10^{-4}	5.1×10^{-5}
5. Cold Retention Area (CRA)										
- Remove Contaminated Concrete and Soil	Table E.2-1	3.1×10^{-8}	5.3×10^{-9}	5.3×10^{-9}	3.7×10^{-8}	5.9×10^{-9}	5.3×10^{-9}	5.3×10^{-9}	1.2×10^{-7}	5.9×10^{-9}
	Table E.2-11	1.3×10^{-5}	5.3×10^{-6}	6.1×10^{-6}	2.0×10^{-5}	3.5×10^{-6}	1.1×10^{-5}	1.8×10^{-5}	6.4×10^{-5}	3.5×10^{-6}
6. Hot Retention Area (HRA)										
- Remove and Package Contaminated Piping	Table E.2-1	3.4×10^{-7}	5.8×10^{-8}	5.8×10^{-8}	4.1×10^{-7}	6.5×10^{-8}	5.8×10^{-8}	5.8×10^{-8}	1.3×10^{-6}	6.5×10^{-8}
- Remove and Package HRA Tanks, Floor Plates, and Partitions	Table E.2-1	9.2×10^{-6}	1.6×10^{-6}	1.6×10^{-6}	1.1×10^{-5}	1.8×10^{-6}	1.6×10^{-6}	1.6×10^{-6}	3.6×10^{-5}	1.8×10^{-6}
- Remove Contaminated Concrete	Table E.2-1	9.6×10^{-9}	1.6×10^{-9}	1.6×10^{-9}	1.2×10^{-8}	1.8×10^{-9}	1.6×10^{-9}	1.6×10^{-9}	3.7×10^{-8}	1.8×10^{-9}
7. Fan House										
- Remove Contaminated Concrete	Table E.2-1	2.2×10^{-9}	3.8×10^{-10}	3.7×10^{-10}	2.6×10^{-9}	4.2×10^{-10}	3.8×10^{-10}	3.8×10^{-10}	8.65×10^{-9}	4.2×10^{-10}
- Remove Fixed Equipment	Table E.2-1	4.0×10^{-8}	6.9×10^{-9}	6.8×10^{-9}	4.8×10^{-8}	7.6×10^{-9}	6.9×10^{-9}	6.9×10^{-9}	1.6×10^{-7}	7.6×10^{-9}
- Raze Stack, Segment and Package	Table E.2-1	5.8×10^{-5}	1.0×10^{-5}	9.9×10^{-6}	7.0×10^{-5}	1.1×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	2.3×10^{-4}	1.1×10^{-5}
8. Waste Handling Building										
- Remove Fixed Equipment, Including Evaporator and HVAC	Table E.2-1	2.9×10^{-8}	5.0×10^{-9}	4.9×10^{-9}	3.5×10^{-8}	5.5×10^{-9}	5.0×10^{-9}	5.0×10^{-9}	1.1×10^{-7}	5.5×10^{-9}
- Remove Contaminated Concrete	Table E.2-1	7.3×10^{-9}	1.2×10^{-9}	1.2×10^{-9}	8.8×10^{-9}	1.4×10^{-9}	1.2×10^{-9}	1.2×10^{-9}	2.8×10^{-8}	1.4×10^{-9}
All Other Buildings and Areas Totals			9.5×10^{-5}	9.8×10^{-5}	3.8×10^{-4}	6.8×10^{-5}	1.8×10^{-4}	3.0×10^{-4}	1.3×10^{-3}	6.8×10^{-5}
Test Reactor Totals			1.1×10^{-4}	1.2×10^{-4}	4.8×10^{-4}	8.3×10^{-5}	2.2×10^{-4}	5.3×10^{-4}	1.6×10^{-3}	8.5×10^{-5}

(a) Only those tasks that result in an atmospheric release are included in this table.
(b) The reference radionuclide inventories refer to the tables of radionuclides shown in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.
(d) For the test reactor, the total population within 80-km of the reference site is 3.5×10^6.

TABLE N.2-12. Postulated Accidental Atmospheric Releases at the Reference Research Reactor

Accident	Comments	Frequency of[a] Occurrence	Radionuclide[b] Inventory	Total Airborne Radioactivity Inside Building (Ci)	Contamination[c] Control System Transmission Factor	Total[d] Atmospheric Release (Ci/hr)
Oxyacetylene Explosion	Explosion during reactor vessel removal - in air cutting	Medium	Table E.1-6	5.2×10^{-2}	1.0	5.2×10^{-2}
HEPA Filter Failure[e]	HEPA ducts fail in place - all building air leaks to atmosphere	Low	Table E.1-5 Table E.1-6	2.6×10^{-4} 1.0×10^{-5}	1.0 1.0	2.6×10^{-4} 1.0×10^{-5}
LPG Explosion	LPG Leaked from a front-end loader	Low	Table E.1-7	1.4×10^{-5}	1.0	1.4×10^{-5}
Vacuum Filter-Bag Rupture[e,f]	Rupture of a full filter-bag during surface cleaning	Medium	Table E.1-5	3.7×10^{-3}	5.0×10^{-4}	1.8×10^{-6}
Accidental Cutting of Activated Al - in Air[e]	Remote cutting accident for 1 hour - in air	High	Table E.1-6	5.8×10^{-4}	5.0×10^{-4}	2.9×10^{-7}
Contaminated Sweeping[e,f] Compound Fire	A fire consumes 0.5 m^3 of used sweeping compound	Medium	Table E.1-5	3.6×10^{-6}	5.0×10^{-4}	1.9×10^{-9}
Combustible Waste Fire[e,f]	1 m^3 of combustible waste is consumed by fire	High	Table E.1-5	1.8×10^{-6}	5.0×10^{-4}	9.0×10^{-10}
Loss of Services	Loss of water supply, electrical power, or air flow	--[g]	--	--	--	-
Natural Phenomena	Floods, earthquakes, tornadoes, or high winds	--[g]	--	--	--	--
Aircraft Crashes	Comparison to routine operation probability	--[g]	--	--	--	--
Man-Caused Events	Detailed analysis is beyond the scope of this study	--[g]	--	--	--	--

(a) The frequency of occurrence considers not only the probability of the accident, but also the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar magnitude is $>10^{-2}$ per year, as "medium" if between 10^{-2} and 10^{-5}, and as "low" if $<10^{-5}$.
(b) These numbers refer to the tables of radionuclides shown in Appendix E.
(c) The HEPA system has an assumed transmission factor of 5×10^{-4}.
(d) For comparison, all accidental releases are assumed to occur in a 1-hr period.
(e) The accident shown applies to both DECON and ENTOMB.
(f) The accident shown applies to both DECON and preparations for safe storage.
(g) Dashes indicate that no calculations were attempted for loss of services, natural phenomena, aircraft crashes, and person-caused events.

TABLE N.2-13. Postulated Accidental Atmospheric Releases at the Reference Test Reactor

Accident	Comments	Frequency of Occurrence[a]	Radionuclide Inventory[b]	Total Airborne Radioactivity Inside Building (Ci)	Contamination Control System Transmission Factor[c]	Total Atmospheric Release (Ci/hr)[d]
Oxyacetylene Explosion	Explosion during RV segmenting in air	Medium	Table E.2-1	5.6×10^{-2}	1.0	5.6×10^{-2}
LPG Explosion[e]	LPG leaked from a front-end loader	Low	Table E.2-1	6.5×10^{-3}	1.0	6.5×10^{-3}
HEPA Filter Failure[e]	HEPA ducts fail in place - all building air leaks to atmosphere	Low	Table E.2-1 Table E.2-2	4.9×10^{-5} 2.8×10^{-6}	1.0 1.0	5.2×10^{-4} 3.8×10^{-6}
Accidental Cutting of Activated Stainless Steel in Air[e]	Remote cutting accident for 1 hour in air	High	Table E.2-1	1.8×10^{-1}	5×10^{-4}	8.8×10^{-5}
Vacuum Filter-Bag Rupture[e,f]	Rupture of a full filter-bag during surface cleaning	Medium	Table E.2-1	5.8×10^{-2}	5×10^{-4}	2.9×10^{-5}
Contaminated Sweeping Compound Fire[e,f]	A fire consumes 0.5 m^3 of used sweeping compound	High	Table E.2-1	7.2×10^{-5}	5×10^{-4}	3.6×10^{-8}
Combustible Waste Fire[e,f]	1 m^3 of waste is consumed by fire	High	Table E.2-1	1.5×10^{-5}	5×10^{-4}	1.8×10^{-8}
Loss of Services	Loss of water supply, electrical power, or air flow	--[g]	--	--	--	--
Natural Phenomena	Floods, earthquakes, tornadoes, or high winds	--[g]	--	--	--	--
Aircraft Crashes	Comparison to routine operation probability	--[g]	--	--	--	--
Man-Caused Events	Detailed analysis is beyond the scope of this study	--[g]	--	--	--	-

(a) The frequency of occurrence considers not only the probability of the accident, but also the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar magnitude is $>10^{-2}$ per year, as "medium" if between 10^{-2} and 10^{-5}, and as "low" if $<10^{-5}$.
(b) These numbers refer to the tables of radionuclides shown in Appendix E.
(c) The HEPA system has an assumed transmission factor of 5×10^{-4}.
(d) For comparison, all accidental releases are assumed to occur in a 1-hr period.
(e) The accident shown applies to both DECON and ENTOMB.
(f) The accident shown applies to both DECON and preparations for safe storage.
(g) Dashes indicate that no calculations were attempted for loss of services, natural phenomena, aircraft crashes, and person-caused events.

Explosion of LPG Leaked from a Front-End Loader. The use of a liquified propane gas (LPG)-powered front-end loader for loading concrete rubble and moving light equipment is assumed in this study, since this method is consistent with current industrial methods. It is anticipated that by the time large-scale decommissioning operations are required, an alternative method with a safer fuel supply will be available and used instead of the LPG-powered front-end loader.

Flammability limits for LPG range from a mixture of 2.1 volume % to a mixture of 9.5 volume %.[18,19] Information on the rate of pressure rise from explosions of various mixtures of propane in air is given in Table N.2-14.

TABLE N.2-14. LPG Explosion Information[a]

Concentration in Air (volume %)	Pre-Detonation Pressure (kPa)	Rate of Pressure Rise (kPa/sec)
3	510	8 600
5	660	17 000
7	330	240

(a) Summarized from information in Reference 20.

An accidental leak of LPG is assumed to occur during the loading of concrete rubble in the reactor building at the research reactor and in the containment vessel at the test reactor. A realistic leak of LPG of 4.5 kg is assumed to occur in these areas requiring 110 m^3 of air to produce a flammable mixture. The total volume of the research reactor building is about 3.6×10^3 m^3, and the volume of the top portion of the test reactor containment vessel is about 6.4×10^3 m^3.

The LPG explosion is assumed to occur after the LPG-air mixture is drawn into the building ventilation system ductwork. The force of the explosion is assumed to cause total failure of the HEPA filters and ventilation blowers. Loss of the blowers would aerodynamically isolate the building from the atmosphere. Remedial action, such as sealing off vent outlets, would further limit the atmospheric release.

The mass of materials that can be deposited on HEPA filters without causing serious operational problems, such as an excessive pressure drop, varies considerably with the filter construction and the particle size of the deposited materials.[21,22] The amount of deposited material per filter assumed for this study is 2.3 kg. If the pre-filters and filters in both filter banks rupture simultaneously (with 50 filters per bank), and assuming that the ductwork does not rupture, about 14 kg of material could be released from the building to the atmosphere.

Radioactive material deposited on the HEPA filters is assumed to be made airborne by the force of the LPG explosion. For the research reactor, the explosion is assumed to occur at the end of Task 11, removal of contaminated concrete. If 14 kg of material are made airborne from the filters, about 1.3×10^{-5} Ci of activated concrete are made airborne. For the test reactor, the explosion is assumed to occur at the end of Task 14, removal of bioshield concrete. Again, 14 kg of material are assumed to be made airborne by the explosion from the filters. This material could contain about 6.2×10^{-3} Ci of activated concrete.

If the volumes of the reactor building and containment vessel are further assumed to fill with dust to a level of 0.1 g/m^3 by the explosion,[4] additional releases of 3.5×10^{-7} Ci of activated aluminum from the research reactor and 2.8×10^{-4} Ci of activated concrete from the test reactor are calculated. The total atmospheric release for this accident is calculated to be 1.4×10^{-5} Ci of activated concrete for the research reactor, and 6.5×10^{-3} of activated concrete for the test reactor. An explosion of leaked LPG resulting in this magnitude of release is estimated to have a "low" frequency of occurrence.

Accidental Cutting of Activated Reactor Components in Air

This accident is assumed to occur during the reactor vessel segmentation tasks for both the research and test reactors. For the research reactor, 0.65 m of activated aluminum are assumed to be remotely cut in 1 hour. The contamination level is assumed to be 22 Ci/m^3, and the total airborne radioactivity is calculated to be 5.8×10^{-4} Ci. For the test reactor, 4.8 m of activated stainless steel are assumed to be cut, with a contamination level of 300 Ci/m^3. The

total airborne radioactivity is calculated to be 1.8×10^{-1} Ci. Assuming that the building HEPA filters remain intact, the atmospheric releases are calculated to be 2.9×10^{-7} Ci for the research reactor and 8.8×10^{-5} Ci for the test reactor. These accidents are estimated to have a "high" frequency of occurrence.

Vacuum Filter-Bag Rupture. Sharp objects, such as metal shards, could rupture a filter-bag during surface cleaning operations involving the use of a vacuum cleaner. When the filter-bag is ruptured, all of the collected material in the bag is assumed to become airborne in the building, because of the mechanical and aerodynamic forces of the 1.4-m^3/min vacuum cleaner air flow. To maximize the calculation of the atmospheric release, the bag rupture is assumed to occur at the time just prior to bag change (i.e., when the filter bag is full).

For the research reactor, a total of 3.1×10^3 m^2 of floor surfaces are vacuumed. With an average contamination level of 2.4×10^{-5} Ci/m^2, and with a removal fraction of 0.5, a total of 3.7×10^{-2} Ci of material is removed. If 10 bag changes are assumed during the entire task, a total of 3.7×10^{-3} Ci is released in the building by this accident. For the test reactor, 4.8×10^3 m^2 are vacuumed, and a total of 5.8×10^{-1} Ci is made airborne by the accident in the building. The total release from each building, assuming that the building HEPA filters remain intact, is 1.8×10^{-8} Ci from the research reactor and 2.9×10^{-5} Ci from the test reactor. Both releases are assumed to be characterized by the activated stainless steel radionuclide inventory. An accident of this release magnitude and frequency is estimated to have a "medium" frequency of occurrence.

Oxyacetylene Explosion. Oxyacetylene cutting torches are assumed to be used for removing and segmenting various activated stainless steel components during DECON. Acetylene gas has a flammability range of 2.5 to 80 volume % mixed in air.[23,24] Violent explosions can occur when acetylene and oxygen are incorrectly mixed. The degree of explosive violence depends on how closely the gas mixture approximates the ratio for complete combustion. Some maximum pressures and pressure rise rates from three acetylene/air mixtures in the explosive range are shown in Table N.2-15.[20]

TABLE N.2-15. Example Explosive Data for Acetylene/Air Mixtures

Percent of Acetylene in Mixture	Maximum Explosive Pressure (kPa)	Maximum Rate of Explosive Pressure Rise (kPa/sec)
5	540	17 000
13	1 000	8 300
20	690	9 700

Oxyacetylene explosions can occur from such causes as flow reversals, nozzle obstructions, and flashbacks (a flare going back up the gas hose).[25,26] For the research reactor, this accident is assumed to occur during Task 9, and for the test reactor it is assumed to occur during Task 13. Both tasks are for segmenting the reactor vessel in air. The airborne material generated as a result of the explosion will fill 3.6×10^3 m^3 in the research reactor and 6.4×10^3 m^3 in the test reactor with a dust loading of 0.1 g/m^3.[4] The estimated airborne radioactivity in each building is 1.3×10^{-3} Ci of activated aluminum for the research reactor, and 2.4×10^{-3} Ci of activated stainless steel for the test reactor.

This explosion is further assumed to release material on the HEPA filters. If all 14 kg of material on the filters are released (see the LPG explosion discussion), then 5.0×10^{-2} Ci of activated aluminum are released from the research reactor and 5.2×10^{-2} Ci of activated stainless steel are released from the test reactor. The total release is found by summing the releases from the building dust and the HEPA filter material. For the research reactor, the total release is 5.2×10^{-2} Ci of activated aluminum, and for the test reactor it is 5.6×10^{-2} Ci of activated stainless steel. An oxyacetylene explosion that results in a release of this magnitude is estimated to have a "medium" frequency of occurrence.

HEPA Filter Failure. For this accident, the building HEPA filters are assumed to fail, resulting in a direct release to the atmosphere. For the research reactor, the filters are assumed to fail during Task 7, the removal of the activated reactor core and vessel internals. The total release is

found in Table N.2-6 to be 2.6 x 10^{-4} Ci of activated stainless steel and 1.0 x 10^{-5}Ci of activated aluminum. For the test reactor, the accident is assumed to occur during task 12, the removal of activated reactor vessel internals. The total release is found in Table N.2-7 to be 5.2 x 10^{-4} Ci of activated stainless steel and 3.8 x 10^{-6} Ci of activated aluminum. An accidental failure of HEPA filters that results in a release of this magnitude is estimated to have a "low" frequency of occurrence.

Contaminated Sweeping Compound Fire. Sweeping compound is composed of sawdust treated with oil or other additives to enhance the collection of loose surface contamination. A fire is postulated to occur in used sweeping compound containing radioactivity removed from floor surfaces. It is assumed that an average of 2.5 x 10^{-4} m^3 of sweeping compound is used for each m^2 of surface swept. It is also assumed that the density of the sweeping compound is about the same as light wood, or about 6.0 x 10^5 g/m^3.[27] The total floor surfaces that are swept are 3.1 x 10^3 m^2 for the research reactor and 2.4 x 10^3 m^2 for the test reactor. The contamination levels on the floor surfaces at these reactors are: 2.4 x 10^{-5} Ci/m^2 at the research reactor, and 4.8 x 10^{-4} Ci/m^2 at the test reactor. These contamination levels are assumed to be associated with the activated stainless steel inventory discussed in Appendix E.

Assuming that half of the surface contamination is removed by sweeping, that 0.5 m^3 of used sweeping compound is involved in the fire, and that the release fraction during the fire is similar to that measured from burning waste (1.5 x 10^{-4}),[28] the total airborne radioactivity is calculated to be 3.6 x 10^{-6} Ci for the research reactor and 7.2 x 10^{-5} Ci for the test reactor. Assuming that the building HEPA filters remain intact during this accident, the total atmospheric releases are: 1.9 x 10^{-9} Ci for the research reactor and 3.6 x 10^{-8} Ci for the test reactor. A fire in combustible sweeping compound that results in a release of this magnitude is estimated to have a "medium" frequency of occurrence.

Combustible Waste Fire. Absorbent materials, such as rags or paper wipes, are assumed to be used for a variety of purposes during DECON and discarded after use. Materials that have come into contact with contaminated surfaces

hold small quantities of radionuclides. Anti-contamination clothing (coveralls, caps, hoods, or shoe covers) become contaminated through use. Plastics are generally considered combustible, although many kinds do not burn readily or are self-extinguishing.

It is postulated that contaminated combustible waste containing low levels of radioactivity, ignites and burns in a working area. Large amounts of these wastes are not likely to accumulate in one place, so it is assumed that 1 m^3 containing 1.2×10^{-2} Ci at the research reactor and 2.4×10^{-1} Ci at the test reactor is involved in a fire. These contamination levels are assumed to be associated with the activated stainless steel inventory discussed in Appendix E. Assuming the fractional release measured for burning waste of 1.5×10^{-4},[28] the total amount of airborne radioactivity is calculated to be 9.0×10^{-10} Ci for the research reactor and 1.8×10^{-8} Ci for the test reactor. A waste fire resulting in a release of this magnitude is estimated to have a "high" frequency of occurrence.

Loss of Services. Loss of services, such as water supply, electrical power, or air flow, is immediately apparent to the workers performing the varied DECON tasks. In the case of manual tasks, the operator would discontinue the task immediately upon loss of services. Most remotely controlled tasks would be monitored frequently enough to almost immediately detect the loss of a service. Since the loss of services constitutes a lesser magnitude release than other postulated accidents, no further analysis is attempted in this study.

N.2.2.2 Natural Phenomena

The designs of structures, systems, and components are based on withstanding the most severe natural phenomena recorded for the site, plus appropriate margins to account for uncertainties in the recorded data. It is assumed that the structural integrity of the reactors is preserved during their operational lifetimes. It is further assumed that their structural integrity is preserved during decommissioning as long as required for safety. It is estimated that, while natural phenomena can cause severe damage to the reference R&T reactors potentially resulting in severe radiological public safety impacts, they are low probability events. Also, the impact of these events should be less than the impacts

calculated in the FSAR design-basis accidents for the operating reactors. For this reason, no detailed analysis of the impacts of these phenomena is attempted beyond the brief discussions that follow.

Floods. The FSAR for each reactor considers runoff floods, surges, seiches, wave action, tsunamis, wind-generated wave action, and combinations of these phenomena. Also, a probable maximum flood is proposed. The reactors are designed to be safe from damage by such a flood or other potential flooding.

Earthquakes. It is estimated that the probability of an earthquake that would occur and cause damage to the R&T reactors is 10^{-7} per year.(29) At this probability, earthquake-induced accidents are extremely unlikely.

Tornadoes. The probability of a large tornado striking the site is on the average less than 5×10^{-6} per year.(29) At this probability, tornado-induced effects during decommissioning are extremely unlikely, and they are not considered further in this study.

High Winds. The design-basis wind velocity for all Category I and Category II structures is 168 km/hr at 9.1 m above nominal ground elevation. For this study, it is assumed that DECON tasks do not reduce the ability of these structures to withstand high winds, and no releases are anticipated.

N.2.2.3 Aircraft Crashes

Each facility site has its own potential risk from aircraft crashes, based on its proximity to airports and air-traffic lanes. However, current information leads to the conclusion that the risk of damage from aircraft crashes is low for most sites.(30) Moreover, the risk is not escalated by DECON operations. The aircraft-crash probability for most plants is very low (10^{-6} to 10^{-8} per year) and no further analysis is attempted.

N.2.2.4 Person-Caused Events

Accidental airborne releases of radioactivity resulting from person-caused events could cover a wide spectrum of magnitudes, ranging from releases induced by casual trespassers to releases induced by armed terrorists. A detailed analysis of person-caused events during decommissioning is beyond the scope of this study, since an analysis of human motives is difficult to conduct. Thus, no airborne release calculations are made.

N.2.2.5 Radiation Doses from Postulated DECON Accidents

Radiation doses to the maximum-exposed individual from accidental releases of radionuclides during DECON are calculated using the radiation dose models discussed in Appendix F. The first-year doses and the fifty-year committed dose equivalents to the maximum-exposed individual from inhalation for the accidents shown in Tables N.2-12 and N.2-13 are shown in Tables N.2-16 and N.2-17 for the R&T reactors. Each accidental release is assumed to occur for 1 hour so that a comparison of the accidents can be made. The atmospheric dispersion ($\overline{X}/Q'$) value used for the accidents is 6.5×10^{-4} sec/m^3.

N.3 SAFSTOR

The second decommissioning alternative considered in this study for the reference R&T reactors is SAFSTOR. This alternative is divided into three phases: preparations for safe storage, maintenance during safe storage, and deferred decontamination. Preparations for safe storage includes tasks designed to immobilize all remaining radioactivity and to deactivate all unneeded equipment. The safe storage period permits decay of all short-lived radioactivity, thus reducing the occupational radiation doses associated with deferred decontamination. After a finite period of safe storage, deferred decontamination is assumed to be required for the termination of the nuclear facility license. Details of the tasks and requirements for SAFSTOR are found in Appendix J for both the R&T reactors.

In this section, the atmospheric releases of radioactivity and the resulting radiation doses to the public from both routine tasks and postulated accidents during preparations for safe storage are presented. This analysis uses the same methods described in Section N.2 for DECON.

N.3.1 Routine Tasks for Preparations for Safe Storage

The task schedules and sequence of events for preparations for safe storage are found in Figure J.1-1 for the research reactor and in Figure J.2-1 for the test reactor. Several of the tasks identified for preparations for safe storage in these figures are identical to those analyzed previously in Section N.2 for DECON. In those cases where the tasks for DECON and preparations for safe

TABLE N.2-16. Calculated Radiation Doses to the Maximum-Exposed Individual from Accidental Atmospheric Releases at the Reference Research Reactor

Accident	Reference[a] Radionuclide Inventory	Total[b] Atmospheric Release (Ci/hr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
			Total Body	Bone	Lungs	GI-LLI	Total Body	Bone	Lungs	GI-LLI
Oxyacetylene Explosion	Table E.1-6	5.2×10^{-2}	4.4×10^{-5}	3.6×10^{-5}	1.2×10^{-3}	3.2×10^{-5}	6.3×10^{-5}	6.6×10^{-5}	1.6×10^{-3}	3.2×10^{-5}
HEPA Filter Failure[c]	Table E.1-5	2.6×10^{-4}	1.4×10^{-7}	2.5×10^{-7}	7.3×10^{-7}	7.3×10^{-7}	1.4×10^{-4}	8.9×10^{-8}	7.8×10^{-7}	7.3×10^{-7}
	Table E.1-6	1.0×10^{-5}	8.4×10^{-9}	6.9×10^{-9}	2.4×10^{-7}	6.1×10^{-9}	1.2×10^{-8}	1.3×10^{-8}	3.1×10^{-7}	6.1×10^{-9}
LPG Explosion[c]	Table E.1-5	1.4×10^{-5}	7.6×10^{-9}	1.3×10^{-9}	3.9×10^{-9}	3.9×10^{-8}	7.7×10^{-9}	4.8×10^{-9}	4.2×10^{-8}	3.9×10^{-8}
Vacuum Filter-Bag Rupture[c,d]	Table E.1-6	1.8×10^{-6}	1.5×10^{-9}	1.2×10^{-9}	4.3×10^{-8}	1.1×10^{-9}	2.2×10^{-9}	2.3×10^{-9}	5.6×10^{-8}	1.1×10^{-9}
Accidental Cutting of Activated Al-in Air[c]	Table E.1-6	2.9×10^{-7}	2.4×10^{-10}	2.0×10^{-10}	6.9×10^{-4}	1.8×10^{-10}	3.5×10^{-10}	3.6×10^{-10}	9.1×10^{-9}	1.8×10^{-10}
Contaminated Sweeping Compound Fire[c,d]	Table E.1-5	1.9×10^{-9}	1.0×10^{-12}	1.8×10^{-13}	5.3×10^{-12}	5.3×10^{-12}	1.0×10^{-12}	6.5×10^{-13}	5.7×10^{-12}	5.3×10^{-12}
Combustible Waste Fire[c,d]	Table E.1-5	9.0×10^{-10}	4.8×10^{-13}	8.6×10^{-14}	1.5×10^{-10}	2.5×10^{-12}	4.9×10^{-13}	3.1×10^{-13}	3.2×10^{-10}	2.5×10^{-12}

(a) These numbers refer to the tables of radionuclides shown in Appendix E.
(b) For comparison, all accidental releases are assumed to occur in a 1-hr period.
(c) The accident shown applies to both DECON and ENTOMB.
(d) The accident shown applies to both DECON and preparations for safe storage.

TABLE N.2-17. Calculated Radiation Doses to the Maximum-Exposed Individual from Accidental Atmospheric Releases at the Reference Test Reactor

Accident	Reference[a] Radionuclide Inventory	Total[b] Atmospheric Release (Ci/hr)	First-Year Dose (rem) Total Body	First-Year Dose (rem) Bone	First-Year Dose (rem) Lungs	First-Year Dose (rem) GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total Body	Fifty-Year Committed Dose Equivalent (rem) Bone	Fifty-Year Committed Dose Equivalent (rem) Lungs	Fifty-Year Committed Dose Equivalent (rem) GI-LLI
Oxyacetylene Explosion	Table E.2-1	5.6×10^{-2}	3.0×10^{-5}	5.4×10^{-6}	1.6×10^{-4}	1.6×10^{-4}	5.3×10^{-4}	1.9×10^{-5}	1.7×10^{-4}	1.6×10^{-4}
LPG Explosion[c]	Table E.2-1	6.5×10^{-3}	3.1×10^{-6}	6.2×10^{-7}	1.8×10^{-5}	1.8×10^{-5}	3.6×10^{-6}	2.2×10^{-6}	2.0×10^{-5}	1.8×10^{-5}
HEPA Filter Failure[c]	Table E.2-1	5.2×10^{-4}	2.8×10^{-7}	5.0×10^{-8}	1.5×10^{-6}	1.5×10^{-6}	2.9×10^{-7}	1.8×10^{-7}	1.6×10^{-6}	1.5×10^{-6}
	Table E.2-2	3.8×10^{-6}	3.2×10^{-9}	2.6×10^{-9}	9.1×10^{-8}	2.3×10^{-9}	4.6×10^{-9}	4.9×10^{-9}	1.2×10^{-7}	2.3×10^{-9}
Accidental Cutting of Activated Stainless Steel[c]	Table E.2-1	8.8×10^{-5}	4.8×10^{-8}	8.4×10^{-9}	2.5×10^{-7}	2.5×10^{-7}	8.4×10^{-7}	3.0×10^{-8}	2.6×10^{-7}	2.5×10^{-8}
Vacuum Filter-Bag Rupture[c,d]	Table E.2-1	2.9×10^{-5}	1.6×10^{-8}	2.8×10^{-9}	8.1×10^{-8}	8.1×10^{-8}	2.8×10^{-7}	9.9×10^{-9}	8.7×10^{-8}	8.1×10^{-8}
Contaminated Sweeping Compound Fire[c,d]	Table E.2-1	3.6×10^{-8}	1.9×10^{-11}	3.5×10^{-12}	1.0×10^{-10}	1.0×10^{-10}	3.4×10^{-10}	1.2×10^{-11}	1.1×10^{-10}	1.0×10^{-10}
Combustible Waste Fire[c,d]	Table E.2-1	1.8×10^{-8}	9.7×10^{-12}	1.7×10^{-12}	5.0×10^{-11}	5.0×10^{-11}	1.7×10^{-10}	6.1×10^{-12}	5.4×10^{-11}	5.0×10^{-11}

(a) These numbers refer to the tables of radionuclides shown in Appendix E.
(b) For comparison, all accidental releases are assumed to occur in a 1-hr period.
(c) The accident shown applies to both DECON and SAFSTOR.
(d) The accident shown applies to both DECON and preparations for safe storage.

storage are identical, the atmospheric releases and resulting radiation doses are assumed to be identical to those reported in Section N.2.

The following subsections contain discussions of the methods used to calculate the atmospheric releases and the resulting public radiation doses during preparations for safe storage at the R&T reactors. The data for those tasks that are unique to preparations for safe storage are presented. Summed doses for all tasks similar to DECON are also given.

N.3.1.1 Methods and Data for Calculating Atmospheric Releases

The tasks for preparations for safe storage are analyzed using the same airborne release mechanisms discussed for DECON in Section N.2.1.1. Specific tasks that may be unique to preparations for safe storage, including some manual decontamination methods, sealing building areas, and isolating equipment, are not anticipated to result in an atmospheric release; thus, they are not analyzed further in this section.

N.3.1.2 Atmospheric Releases from Preparations for Safe Storage

For the research reactor, no tasks that are unique to preparations for safe storage are defined in Figure J.1-1. For this study, all of the tasks that are similar to tasks during DECON are assumed to result in identical atmospheric releases. A summary of the releases (by building and radionuclide inventory) at the reference research reactor is shown in Table N.3-1.

For the test reactor, the tasks identified for preparations for safe storage are shown in Figure J.2-1. The majority of the tasks identified are not anticipated to result in a significant atmospheric release. Thus, only those tasks that are similar to DECON tasks are included in the atmospheric release analysis. The releases are assumed to be identical to those calculated for DECON. A summary of the releases at the reference test reactor is shown in Table N.3-2.

TABLE N.3-1. Calculated Atmospheric Releases During Preparations for Safe Storage at the Reference Research Reactor

Building/Task[a]	Reference[b] Radionuclide Inventory	Average Airborne Generation Rate Inside Building (Ci/hr)	Total Airborne Radioactivity Inside Building (Ci)	Total[c] Atmospheric Release (Ci/yr)
Reactor Building				
Tasks 1-14	Table E.1-5	4.8×10^{-6}	8.9×10^{-5}	4.1×10^{-8}
Annex Building				
Tasks 15	Table E.1-8	2.2×10^{-6}	5.4×10^{-5}	2.7×10^{-8}
Heat Exchanger Building				
Task 16	Table E.1-5	6.9×10^{-7}	1.4×10^{-6}	6.9×10^{-10}
Pump House				
Tasks 17 & 18	Table E.1-5	2.4×10^{-6}	1.6×10^{-6}	8.0×10^{-9}
Radiation Center Building				
Tasks 19 & 20	Table E.1-5	4.5×10^{-7}	3.1×10^{-6}	1.6×10^{-9}

(a) No tasks are identified that are significantly different than those previously identified for DECON. The tasks identified for preparations for safe storage are shown in Figure J.1-1 of Appendix J.
(b) The table numbers listed are shown in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.3-2. Calculated Atmospheric Releases During Preparations for Safe Storage at the Reference Test Reactor

Building[a]	Reference[b] Radionuclide Inventory	Average Airborne Generation Rate Inside Building (Ci/hr)	Total Airborne Radioactivity Inside Building (Ci)	Total[c] Atmospheric Release (Ci/yr)
Reactor Building/MUR/CV	Table E.2-1	1.5×10^{-4}	2.0×10^{-3}	7.4×10^{-7}
Hot Laboratory Building	Table E.2-10	1.9×10^{-4}	2.4×10^{-3}	1.2×10^{-6}
	Table E.2-1	9.4×10^{-6}	1.2×10^{-4}	6.1×10^{-8}
Other Contaminated Buildings	Table E.2-1	1.1×10^{-4}	1.0×10^{-3}	4.9×10^{-7}
	Table E.2-11	3.6×10^{-6}	2.0×10^{-4}	2.0×10^{-4}

(a) No tasks are identified that are significantly different than those previously identified for DECON. The specific tasks identified for preparations for safe storage are shown in Figure J.2-1 of Appendix J.
(b) The table numbers shown are listed in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

N.3.1.3 Public Radiation Doses from Routine Tasks During Preparations for Safe Storage

First-year doses and fifty-year committed dose equivalents to both the maximum-exposed individual and the population residing within 80 km of the R&T reactor sites are calculated using the dose models discussed in Appendix F. Each routine atmospheric release is assumed to occur at a constant rate for 1 year so direct comparisons of the public radiation doses from the various buildings at each reactor can be made. The organs of reference for which radiation doses are calculated include: total body, bone, lung, and GI tract (lower large intestine).

For the research reactor, radiation doses to the maximum-exposed individual are listed in Table N.3-3 and doses to the public are listed in Table N.3-4. For the test reactor, radiation doses to the maximum-exposed individual are listed in Table N.3-5, and doses to the population are listed in Table N.3-6. These tables show only the total for the tasks in each building for each radionuclide inventory, since all of the tasks are assumed to be identical to the tasks previously considered for DECON.

N.3.2 Postulated Accidents During Preparations for Safe Storage

A comparison of the postulated accidents for DECON and preparations for safe storage is presented in Table N.3-7. A general estimate of the frequency of occurrence of the level of atmospheric release associated with each postulated accident is also given. The frequency of occurrence is based on both the probability of the accident and on the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar or greater magnitude per year is greater than 10^{-2}, as "medium" if between 10^{-2} and 10^{-5}, and as "low" if less than 10^{-5}. Only accidents with a calculated atmospheric release are compared in Table N.3-7.

No accidents are postulated for preparations for safe storage that are different from the accidents that are postulated for DECON. None of the tasks defined for preparations for safe storage involve removal of the activated steel or concrete, thus the accidents defined for DECON involving these materials do not apply. The calculated radiation doses to the maximum-exposed individual

TABLE N.3-3. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine Preparations for Safe Storage Tasks at the Reference Research Reactor

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
			Total Body	Bone	Lungs	GI-LLI	Total Body	Bone	Lungs	GI-LLI
Reactor Building										
Tasks 1-14	Table E.1-5	4.1×10^{-8}	1.3×10^{-11}	1.2×10^{-11}	4.9×10^{-11}	1.3×10^{-11}	1.3×10^{-11}	1.3×10^{-11}	1.5×10^{-10}	1.3×10^{-11}
Annex Building										
Task 15	Table E.1-8	2.7×10^{-8}	1.2×10^{-11}	3.2×10^{-11}	5.1×10^{-11}	9.4×10^{-12}	1.0×10^{-10}	6.8×10^{-10}	2.0×10^{-10}	9.4×10^{-12}
Heat Exchanger Building										
Task 16	Table E.1-5	6.9×10^{-10}	2.2×10^{-13}	2.1×10^{-13}	8.3×10^{-13}	2.2×10^{-13}	2.2×10^{-13}	2.2×10^{-13}	2.5×10^{-12}	2.2×10^{-13}
Pump House										
Tasks 17 & 18	Table E.1-5	8.0×10^{-9}	2.6×10^{-12}	2.4×10^{-12}	90.6×10^{12}	2.6×10^{-12}	2.6×10^{-12}	2.6×10^{-12}	2.9×10^{-11}	2.6×10^{-12}
Radiation Center Building										
Tasks 19 & 20	Table E.1-5	1.6×10^{-9}	5.1×10^{-13}	4.8×10^{-13}	1.9×10^{-12}	5.1×10^{-13}	5.1×10^{-13}	5.1×10^{-13}	5.8×10^{-12}	5.1×10^{-13}
Totals			2.8×10^{-11}	4.7×10^{-11}	1.1×10^{-10}	2.6×10^{-11}	1.2×10^{-10}	7.0×10^{-10}	3.8×10^{-10}	2.6×10^{-11}

(a) No tasks are identified that are significantly different than those defined for DECON. The specific tasks identified for preparations for safe storage are shown in Figure J.1-1.
(b) The table numbers shown are listed in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.3-4. Radiation Doses to the Population from Atmospheric Releases During Routine Preparations for Safe Storage Tasks at the Reference Research Reactor

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (man-rem)				Fifty-Year Committed Dose Equivalent (man-rem)			
			Total Body	Bone	Lungs	GI-LLI	Total Body	Bone	Lungs	GI-LLI
Reactor Building										
Tasks 1-14	Table E.1-5	4.1×10^{-8}	3.0×10^{-9}	2.8×10^{-9}	1.9×10^{-8}	3.2×10^{-9}	3.0×10^{-9}	3.0×10^{-9}	6.6×10^{-8}	3.2×10^{-9}
Annex Building										
Task 15	Table E.1-8	2.7×10^{-8}	3.5×10^{-9}	1.2×10^{-8}	2.3×10^{-8}	2.3×10^{-9}	3.0×10^{-8}	2.5×10^{-7}	9.2×10^{-8}	2.3×10^{-9}
Heat Exchanger Building										
Task 16	Table E.1-5	6.9×10^{-10}	5.0×10^{-11}	4.6×10^{-11}	3.2×10^{-10}	5.4×10^{-11}	5.0×10^{-11}	5.0×10^{-11}	1.1×10^{-9}	5.4×10^{-11}
Pump House										
Tasks 17 & 18	Table E.1-5	8.0×10^{-9}	5.8×10^{-10}	5.4×10^{-10}	3.8×10^{-9}	6.2×10^{-10}	5.8×10^{-10}	5.8×10^{-10}	1.3×10^{-8}	6.2×10^{-10}
Radiation Center Building										
Tasks 19 & 20	Table E.1-5	1.6×10^{-9}	1.2×10^{-10}	1.1×10^{-10}	7.5×10^{-10}	1.2×10^{-10}	1.2×10^{-10}	1.2×10^{-10}	2.6×10^{-9}	1.2×10^{-10}
Totals			7.2×10^{-9}	1.6×10^{-8}	4.7×10^{-8}	6.3×10^{-9}	3.4×10^{-8}	2.5×10^{-7}	1.8×10^{-7}	6.3×10^{-9}

(a) No tasks are identified that are significantly different than those defined for DECON. The specific tasks identified for preparations for safe storage are shown in Figure J.1-1.
(b) The table numbers shown are listed in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.3-5. Calculated Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine Preparations for Safe Storage Tasks at the Reference Test Reactor

Building[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
			Total Body	Bone	Lungs	GI-LLI	Total Body	Bone	Lungs	GI-LLI
Reactor Building/MUR/CV	Table E.2-1	7.4×10^{-7}	2.4×10^{-10}	2.2×10^{-10}	8.9×10^{-10}	2.4×10^{-10}	2.4×10^{-10}	2.4×10^{-10}	2.7×10^{-9}	2.4×10^{-10}
Hot Laboratory Building	Table E.2-10	1.2×10^{-6}	5.3×10^{-10}	1.4×10^{-9}	2.3×10^{-9}	4.2×10^{-10}	4.6×10^{-9}	3.0×10^{-8}	9.0×10^{-9}	4.2×10^{-10}
	Table E.2-1	6.1×10^{-8}	2.0×10^{-11}	1.8×10^{-11}	7.3×10^{-11}	2.0×10^{-11}	2.0×10^{-11}	2.0×10^{-11}	2.2×10^{-10}	2.0×10^{-11}
Other Contaminated Buildings	Table E.2-1	4.9×10^{-7}	1.6×10^{-10}	1.5×10^{-10}	5.9×10^{-10}	1.6×10^{-10}	1.6×10^{-10}	1.6×10^{-10}	1.8×10^{-9}	1.6×10^{-10}
	Table E.2-11	2.0×10^{-4}	1.3×10^{-7}	1.4×10^{-7}	3.0×10^{-7}	9.0×10^{-8}	2.4×10^{-7}	8.0×10^{-7}	9.2×10^{-7}	9.0×10^{-8}
Totals			1.3×10^{-7}	1.4×10^{-7}	3.0×10^{-7}	9.1×10^{-8}	2.4×10^{-7}	8.3×10^{-7}	9.3×10^{-7}	9.1×10^{-8}

(a) No tasks are identified that are significantly different than those defined for DECON. The specific tasks identified for preparations for safe storage are shown in Figure J.2-1 in Appendix J.
(b) The table numbers shown are listed in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.3-6. Calculated Radiation Doses to the Population from Atmospheric Releases During Routine Preparations for Safe Storage Tasks at the Reference Test Reactor

Building[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (man-rem)				Fifty-Year Committed Dose Equivalent (man-rem)			
			Total Body	Bone	Lungs	GI-LLI	Total Body	Bone	Lungs	GI-LLI
Reactor Building/ MUR/CV	Table E.2-1	7.4×10^{-7}	1.3×10^{-7}	1.3×10^{-7}	8.9×10^{-7}	1.4×10^{-7}	1.3×10^{-7}	1.3×10^{-7}	2.9×10^{-6}	1.4×10^{-7}
Hot Laboratory Building	Table E.2-10	1.2×10^{-6}	3.7×10^{-7}	1.3×10^{-6}	2.5×10^{-6}	2.5×10^{-7}	3.2×10^{-6}	2.8×10^{-5}	1.0×10^{-5}	2.5×10^{-7}
	Table E.2-1	6.1×10^{-8}	1.1×10^{-8}	1.0×10^{-8}	7.3×10^{-8}	1.2×10^{-8}	1.1×10^{-8}	1.1×10^{-8}	1.9×10^{-7}	9.3×10^{-8}
Other Contaminated Buildings	Table E.2-1	4.9×10^{-7}	8.8×10^{-8}	8.3×10^{-8}	5.9×10^{-7}	9.3×10^{-8}	8.8×10^{-8}	8.8×10^{-8}	1.9×10^{-6}	9.3×10^{-8}
	Table E.2-11	2.0×10^{-4}	8.2×10^{-5}	9.5×10^{-5}	3.0×10^{-4}	5.4×10^{-5}	1.7×10^{-4}	2.8×10^{-4}	9.8×10^{-4}	5.4×10^{-5}
Totals			8.3×10^{-7}	9.6×10^{-7}	3.0×10^{-4}	5.5×10^{-5}	1.7×10^{-4}	3.1×10^{-4}	1.0×10^{-3}	5.5×10^{-5}

(a) No tasks are identified that are significantly different than those defined for DECON. The specific tasks identified for preparations for safe storage are shown in Figure J.2-1.
(b) The table numbers shown are listed in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.3-7. Comparison of Postulated Accidents for DECON and Preparations for Safe Storage

Accident	Comments	Frequency of Occurrence[(a)]	Reference Radionuclide Inventory[(b)]	Analysis for: DECON	Analysis for: Safe Storage
LPG Explosion	4.5 kg of LPG leaked during concrete loading	Low	E.1-5	✓	--
Oxyacetylene Explosion	Explosion During Reactor Vessel Removal	Medium	E.1-5 E.1-6	✓	--
Vacuum Filter-Bag Rupture	Rupture of a Full Filter Bag During Surface Cleaning	Medium	E.1-5	✓	✓
HEPA Filter Failure	During Reactor Vessel Removal	Low	E.1-5 E.1-6	✓	--
Accidental Cutting of Activated Material	In-Air Remote Operations	High	E.1-5 E.1-6	✓	--
Contaminated Sweeping Compound Fire	A Fire in 0.5 m^3 of Used Compound	Medium	E.1-5	✓	✓
Combustible Waste Fire	A Fire in 1 m^3 of Waste	High	E.1-5	✓	✓

(a) The frequency of occurrence considers not only the probability of the accident, but also the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar or greater magnitude per year is $>10^{-2}$, as "medium" if between 10^{-2} and 10^{-5}, and as "low" if $<10^{-5}$.

(b) The table numbers listed are shown in Appendix E.

from the accidents considered for preparations for safe storage are the same as the doses calculated for DECON, and are shown by footnotes in Table N.2-16 for the research reactor and in Table N.2-17 for the test reactor.

N.3.3 Continuing Care During Safe Storage

Radioactive contamination during the safe storage period is assumed to be fixed firmly in place and is not readily available for airborne release. The tasks during safe storage involve surveillance, maintenance, and security with no operational components or equipment except for automatic monitoring or security systems. Thus, no routine airborne releases are anticipated during safe storage of the reference R&T reactors.

Only low probability accidents with causes external to the plants can result in an accidental release of radioactive materials during safe storage.

Such causes include earthquakes or intrusion into the facilities. The combination of the low probability of these events and the immobility of the radionuclide inventories within the reference R&T reactors will reduce the impact of accidents during safe storage. Thus, no analysis of postulated safe storage accidents is attempted in this study.

N.3.4 Deferred Decontamination

Deferred decontamination is assumed to be required after a period of safe storage to permit termination of the nuclear facility license. Since a period of safe storage will permit radioactive decay of the short-lived activation products and fission products, the inventories within the reference R&T reactors will be greatly reduced. The time dependence of the reference radionuclide inventories is discussed in Appendix E. Thus, if the same procedures are used, the atmospheric releases during deferred decontamination should be less than those during DECON. The public radiation doses from either routine deferred decontamination tasks or postulated accidents should also be less than the radiation doses calculated for DECON. Therefore, public safety during deferred decontamination is not further analyzed. The radiation doses are assumed to be equal to or less than the radiation doses from DECON, depending on the length of the safe storage period.

N.4 ENTOMB

The third decommissioning alternative considered in this study is ENTOMB. This alternative results in removal of the radioactively contaminated portions of the buildings and equipment and utilizes onsite storage of waste by converting the reactor vessel and portions of the containment buildings at the R&T reactor into waste storage facilities. Onsite storage reduces the volume of waste shipped offsite for disposal, but requires a continuation of the amended nuclear license. Details of the tasks and requirements of entombment are discussed in Appendix K.

The inventories of radioactive contamination used in this public safety evaluation of entombment tasks are the shutdown inventories discussed in Appendix E. Shutdown inventories are used to maximize the calculated atmospheric releases.

In the following sections, the atmospheric releases of radioactivity from ENTOMB are described for both routine tasks and postulated accidents. The atmospheric release analysis uses the same calculational methods described in Section N.2.1.1 for DECON.

N.4.1 Routine ENTOMB Tasks

The chronological sequence of ENTOMB tasks are listed in Tables K.1-1 and K.2-1 in Appendix K. ENTOMB tasks are listed with their operational time requirements for each building or area at the reference R&T reactors in these tables. The majority of the tasks defined for ENTOMB are identical to the tasks defined for DECON. The major difference is the chronological sequence of events. Therefore, no new analysis of the atmospheric releases from ENTOMB tasks at the R&T reactors is attempted. Furthermore, the atmospheric releases and resulting radiation doses to the public are assumed to be identical with those calculated for DECON (see Section N.2).

The following sections contain brief discussions of the methods used to calculate atmospheric releases and the resulting radiation doses to the public during ENTOMB tasks at the reference R&T reactors.

N.4.1.1 Methods for Calculating Atmospheric Releases of Radioactivity from ENTOMB Tasks

The tasks defined for ENTOMB are analyzed using the same airborne release mechanisms discussed for DECON in Section N.2.1.1. A summary of the analysis methods is shown in Table N.2-2. Specific tasks that may be unique to ENTOMB, including some manual decontamination methods, sealing building areas, and securing the building against intrusion, are not anticipated to result in an atmospheric release; thus, they are not analyzed further in this section.

N.4.1.2 Atmospheric Releases from ENTOMB Tasks

For the research reactor, none of the tasks defined in Table K.1-1 are unique to ENTOMB and require an analysis beyond that performed for DECON. For this study, all of the tasks that are similar to DECON are assumed to result in identical atmospheric releases. A summary of the releases (by building and radionuclide inventory) at the reference research reactor is shown in Table N.4-1.

TABLE N.4-1. Calculated Atmospheric Releases During ENTOMB Tasks at the Reference Research Reactor

Building/Task[a]	Reference[b] Radionuclide Inventory	Average Airborne Generation Rate Inside Building (Ci/hr)	Total Airborne Radioactivity Inside Building (Ci)	Total[c] Atmospheric Release (Ci/yr)
Reactor Building				
Tasks 1-16	Table E.1-5	3.2×10^{-5}	3.5×10^{-4}	1.8×10^{-7}
	Table E.1-6	1.0×10^{-6}	1.0×10^{-5}	5.1×10^{-9}
Annex Building				
Task 17	Table E.1-8	2.2×10^{-6}	5.4×10^{-5}	2.7×10^{-8}
Heat Exchanger Building				
Task 18	Table E.1-5	6.9×10^{-7}	1.4×10^{-6}	6.9×10^{-10}
Pump House				
Tasks 19 & 20	Table E.1-5	2.4×10^{-6}	1.6×10^{-6}	8.0×10^{-9}
	Table E.2-11	3.6×10^{-6}	2.0×10^{-4}	2.0×10^{-4}
Radiation Center Buildings				
Tasks 21 & 22	Table E.1-5	4.5×10^{-7}	3.1×10^{-6}	1.6×10^{-9}

(a) No tasks are identified that are significantly different than those previously identified for DECON. The tasks identified for ENTOMB are shown in Figure K.1-1 in Appendix K.
(b) The table numbers listed are shown in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

For the test reactor, the tasks identified for ENTOMB are shown in Table K.2-1. Again, only those tasks that are similar to DECON tasks are assumed to result in atmospheric releases. A summary of the atmospheric releases calculated for ENTOMB tasks at the reference test reactor is shown in Table N.4-2.

N.4.1.3 Public Radiation Doses from Routine ENTOMB Tasks

First-year doses and fifty-year committed dose equivalents to both the maximum-exposed individual and the population residing within 80 km of the R&T reactor sites are calculated using the dose models discussed in Appendix F. Each routine atmospheric release is assumed to occur at a constant rate for 1 year so direct comparisons of the public radiation doses from the various buildings at each reactor can be made. The organs of reference for which radiation doses are calculated include: total body, bone, and GI-tract (lower-large intestine).

TABLE N.4-2. Calculated Atmospheric Releases During ENTOMB Tasks at the Reference Test Reactor

Building[a]	Reference[b] Radionuclide Inventory	Average Airborne Generation Rate Inside Building (Ci/hr)	Total Airborne Radioactivity Inside Building (Ci)	Total[c] Atmospheric Release (Ci/yr)
Reactor Building/MUR/CV	Table E.2-1	7.8×10^{-4}	8.8×10^{-2}	5.3×10^{-6}
	Table E.2-2	4.2×10^{-6}	3.8×10^{-6}	1.9×10^{-10}
Hot Laboratory Building	Table E.2-10	1.9×10^{-4}	2.4×10^{-3}	1.2×10^{-6}
	Table E.2-1	9.4×10^{-6}	1.2×10^{-4}	6.1×10^{-8}
Other Contaminated Buildings	Table E.2-1	1.1×10^{-4}	1.0×10^{-3}	4.9×10^{-7}
	Table E.2-11	3.6×10^{-6}	2.0×10^{-4}	2.0×10^{-4}

(a) No tasks are identified that are significantly different than those previously identified for DECON. The ENTOMB tasks considered are shown in Figure K.1-1 in Appendix K.
(b) The table numbers listed are shown in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

For the research reactors, radiation doses to the maximum-exposed individual are listed in Table N.4-3, and doses to the public are listed in Table N.4-4. For the test reactor, radiation doses to the maximum-exposed individual are listed in Table N.4-5, and doses to the population are listed in Table N.4-6. These tables show only the totals for the tasks in each building for each radionuclide inventory, since all of the tasks are assumed to be identical to the tasks previously considered for DECON.

N.4.2 Postulated Accidents During ENTOMB

A comparison of the postulated accidents for DECON and ENTOMB is presented in Table N.4-7. A general estimate of the level of atmospheric release associated with each postulated accident is also given. The frequency of occurrence is based on both the probability of the accident and on the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar or greater magnitude per year is greater than 10^{-2}, as "medium" if between 10^{-2} and 10^{-5}, and as "low" if less than 10^{-5}. Only accidents with a calculated atmospheric release are compared in Table N.4-7.

TABLE N.4-3. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine ENTOMB Tasks at the Reference Research Reactor

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (rem) Total Body	First-Year Dose (rem) Bone	First-Year Dose (rem) Lungs	First-Year Dose (rem) GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total Body	Fifty-Year Committed Dose Equivalent (rem) Bone	Fifty-Year Committed Dose Equivalent (rem) Lungs	Fifty-Year Committed Dose Equivalent (rem) GI-LLI
Reactor Building										
Tasks 1 - 16	Table E.1-5	1.8×10^{-7}	5.8×10^{-11}	5.4×10^{-11}	2.2×10^{-10}	5.8×10^{-11}	5.8×10^{-11}	5.8×10^{-11}	6.5×10^{-10}	5.8×10^{-11}
	Table E.1-6	5.1×10^{-9}	4.0×10^{-13}	3.5×10^{-13}	9.2×10^{-13}	2.4×10^{-13}	6.6×10^{-13}	6.1×10^{-13}	1.7×10^{-12}	2.4×10^{-13}
Annex Building										
Task 17	Table E.1-8	2.7×10^{-8}	1.2×10^{-11}	3.2×10^{-11}	5.1×10^{-11}	9.4×10^{-12}	1.0×10^{-10}	6.8×10^{-10}	2.0×10^{-10}	9.4×10^{-12}
Heat Exchanger Building										
Task 18	Table E.1-5	6.9×10^{-10}	2.2×10^{-13}	2.1×10^{-13}	8.3×10^{-13}	2.2×10^{-13}	2.2×10^{-13}	2.2×10^{-13}	2.5×10^{-12}	2.2×10^{-13}
Pump House										
Tasks 19 & 20	Table E.1-5	8.0×10^{-9}	2.6×10^{-12}	2.4×10^{-12}	9.6×10^{-12}	2.6×10^{-12}	2.6×10^{-12}	2.6×10^{-12}	2.9×10^{-11}	2.6×10^{-12}
Radiation Center Building										
Tasks 21 & 22	Table E.1-5	1.6×10^{-9}	5.1×10^{-13}	4.8×10^{-13}	1.9×10^{-12}	5.1×10^{-13}	5.1×10^{-13}	5.1×10^{-13}	5.8×10^{12}	5.1×10^{-13}
Totals			7.4×10^{-11}	8.9×10^{-11}	2.8×10^{-10}	7.1×10^{-11}	1.6×10^{-10}	7.4×10^{-10}	8.9×10^{-10}	7.1×10^{-11}

(a) No tasks are identified that are significantly different than those defined for DECON. The specific tasks identified for ENTOMB are shown in Figure K.1-1 in Appendix K.
(b) The table numbers shown are listed in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.4-4. Radiation Doses to the Population from Atmospheric Releases During Routine ENTOMB Tasks at the Reference Research Reactor

Building/Task[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (man-rem) Total Body	Bone	Lungs	GI-LLI	Fifty-Year Committed Dose Equivalent (man-rem) Total Body	Bone	Lungs	GI-LLI
Reactor Building										
Tasks 1 - 16	Table E.1-5	1.8×10^{-7}	1.3×10^{-8}	1.2×10^{-8}	8.5×10^{-8}	1.4×10^{-8}	1.3×10^{-8}	1.3×10^{-8}	2.9×10^{-7}	1.4×10^{-8}
	Table E.1-6	5.1×10^{-9}	1.1×10^{-10}	9.2×10^{-11}	3.7×10^{-10}	6.1×10^{-11}	1.8×10^{-10}	1.6×10^{-10}	7.1×10^{-10}	6.1×10^{-11}
Annex Building										
Task 17	Table E.1-8	2.7×10^{-8}	3.5×10^{-9}	1.2×10^{-8}	2.3×10^{-8}	2.3×10^{-9}	3.0×10^{-8}	2.5×10^{-7}	9.2×10^{-8}	2.3×10^{-9}
Heat Exchanger Building										
Task 18	Table E.1-5	6.9×10^{-10}	5.0×10^{-11}	4.6×10^{-11}	3.2×10^{-10}	5.4×10^{-11}	5.0×10^{-11}	5.0×10^{-11}	1.1×10^{-9}	5.4×10^{-11}
Pump House										
Tasks 19 & 20	Table E.1-5	8.0×10^{-9}	5.8×10^{-10}	5.4×10^{-10}	3.8×10^{-9}	6.2×10^{-10}	5.8×10^{-10}	5.2×10^{-10}	1.3×10^{-8}	6.2×10^{-10}
Radiation Center Building										
Tasks 21 & 22	Table E.1-5	1.6×10^{-9}	1.2×10^{-10}	1.1×10^{-10}	7.5×10^{-10}	1.2×10^{-10}	1.2×10^{-10}	4.2×10^{-10}	2.6×10^{-9}	1.2×10^{-10}
Totals			1.7×10^{-8}	2.5×10^{-8}	1.1×10^{-7}	1.7×10^{-8}	4.4×10^{-8}	2.6×10^{-7}	$4.0 \times 10^{-}$	1.7×10^{-8}

(a) No tasks are identified that are significantly different than those defined for DECON. The specific tasks identified for ENTOMB are shown in Figure K.1-1 in Appendix K.
(b) The table numbers shown are listed in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.4-5. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine ENTOMB Tasks at the Reference Test Reactor

Building[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
			Total Body	Bone	Lungs	GI-LLI	Total Body	Bone	Lungs	GI-LLI
Reactor Building/ MUR/CV	Table E.2-1	56.3×10^{-6}	1.7×10^{-9}	1.6×10^{-9}	6.4×10^{-9}	1.7×10^{-9}	1.7×10^{-9}	1.7×10^{-9}	2.4×10^{-8}	1.7×10^{-9}
	Table E.2-2	1.9×10^{-10}	1.5×10^{-14}	1.3×10^{-14}	3.4×10^{-14}	9.1×10^{-15}	2.5×10^{-14}	2.3×10^{-14}	6.5×10^{-14}	9.1×10^{-15}
Hot Laboratory Building	Table E.2-10	1.2×10^{-6}	5.3×10^{-10}	1.4×10^{-9}	2.3×10^{-9}	4.2×10^{-10}	4.6×10^{-9}	3.0×10^{-8}	9.0×10^{-9}	4.2×10^{-10}
	Table E.2-1	6.1×10^{-8}	2.0×10^{-11}	1.8×10^{-11}	7.3×10^{-11}	2.0×10^{-11}	2.0×10^{-11}	2.0×10^{-11}	2.2×10^{-10}	2.0×10^{-11}
Other Contaminated Buildings	Table E.2-1	4.9×10^{-7}	1.6×10^{-10}	1.5×10^{-10}	5.9×10^{-10}	1.6×10^{-10}	1.6×10^{-10}	1.6×10^{-10}	1.8×10^{-9}	1.6×10^{-10}
	Table E.2-11	2.0×10^{-4}	1.3×10^{-7}	1.4×10^{-7}	3.0×10^{-7}	9.0×10^{-8}	2.4×10^{-7}	8.0×10^{-7}	9.2×10^{-7}	9.0×10^{-8}
Totals			1.3×10^{-7}	1.4×10^{-7}	3.1×10^{-7}	9.2×10^{-8}	2.5×10^{-7}	8.3×10^{-7}	9.6×10^{-7}	9.2×10^{-8}

(a) No tasks are identified that are significantly different than those defined for DECON. The specific tasks identified for ENTOMB are shown in Figure K.1-1 in Appendix K.
(b) The table numbers shown are listed in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.4-6. Radiation Doses to the Population from Atmospheric Releases During Routine ENTOMB Tasks at the Reference Test Reactor

Building[a]	Reference[b] Radionuclide Inventory	Total[c] Atmospheric Release (Ci/yr)	First-Year Dose (man-rem) Total Body	Bone	Lungs	GI-LLI	Fifty-Year Committed Dose Equivalent (man-rem) Total Body	Bone	Lungs	GI-LLI
Reactor Building/ MUR/CV	Table E.2-1	5.3×10^{-6}	9.5×10^{-7}	9.0×10^{-7}	6.4×10^{-6}	1.0×10^{-6}	9.5×10^{-7}	9.5×10^{-7}	2.1×10^{-5}	1.0×10^{-6}
	Table E.2-2	1.9×10^{-10}	9.9×10^{-12}	8.4×10^{-12}	3.4×10^{-11}	5.7×10^{-12}	1.7×10^{-11}	1.5×10^{-11}	6.8×10^{-11}	5.7×10^{-12}
Hot Laboratory Building	Table E.2-10	1.2×10^{-6}	3.7×10^{-7}	1.3×10^{-6}	2.5×10^{-6}	2.5×10^{-7}	3.2×10^{-6}	2.8×10^{-5}	1.0×10^{-5}	2.5×10^{-7}
	Table E.2-1	6.1×10^{-8}	1.1×10^{-8}	1.0×10^{-8}	7.3×10^{-8}	1.2×10^{-8}	1.1×10^{-8}	1.1×10^{-8}	2.4×10^{-7}	1.2×10^{-8}
Other Contaminated Buildings	Table E.2-1	4.9×10^{-7}	8.8×10^{-8}	8.3×10^{-8}	5.9×10^{-7}	9.3×10^{-8}	8.8×10^{-8}	8.8×10^{-8}	1.9×10^{-6}	9.3×10^{-8}
	Table E.2-11	2.0×10^{-4}	8.2×10^{-5}	9.5×10^{-5}	3.0×10^{-4}	5.4×10^{-5}	1.7×10^{-4}	2.89×10^{-4}	9.8×10	5.4×10^{-4}
Totals			8.3×10^{-5}	9.7×10^{-5}	3.1×10^{-4}	5.5×10^{-5}	1.7×10^{-4}	3.1×10^{-4}	1.0×10^{-3}	5.5×10^{-5}

(a) No tasks are identified that are significantly different than those defined for DECON. The specific tasks identified for ENTOMB are shown in Figure K.1-1 in Appendix K.
(b) The table numbers shown are listed in Appendix E.
(c) For comparison, dose calculations are made based on annual releases.

TABLE N.4-7. Comparison of Postulated Accidents for DECON and ENTOMB

Accident	Comments	Frequency of Occurrence[a]	Reference Radionuclide Inventory[b]	Analysis for: DECON	Analysis for: Safe Storage
LPG Explosion	4.5 kg of LPG leaked during concrete loading	Low	E.1-5	✓	✓
Oxyacetylene Explosion	Explosion During Reactor Vessel Removal	Medium	E.1-5 E.1-6	✓	--
Vacuum Filter-Bag Rupture	Rupture of a Full Filter Bag During Surface Cleaning	Medium	E.1-5	✓	✓
HEPA Filter Failure	During Reactor Vessel Removal	Low	E.1-5 E.1-6	✓	✓
Accidental Cutting of Activated Material	In-Air Remote Operations	High	E.1-5 E.1-6	✓	✓
Contaminated Sweeping Compound Fire	A Fire in 0.5 m^3 of Used Compound	Medium	E.1-5	✓	✓
Combustible Waste Fire	A Fire in 1 m^3 of Waste	High	E.1-5	✓	✓

(a) The frequency of occurrence considers not only the probability of the accident, but also the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar or greater magnitude per year is $>10^{-2}$, as "medium" if between 10^{-2} and 10^{-5}, and as "low" if $<10^{-5}$.
(b) The table numbers listed are in Appendix E.

No accidents are postulated for ENTOMB that are different from the accidents that are postulated for DECON. The calculated radiation doses to the maximum-exposed individual from the accidents considered for ENTOMB are the same as the doses calculated for DECON, and are shown by footnotes in Table N.2-16 for the research reactor and in Table N.2-17 for the test reactor.

RADIATION DOSES FROM RADIOACTIVE WASTE TRANSPORT FOR DECOMMISSIONING THE REFERENCE R&T REACTORS

During decommissioning of the reference R&T reactors, radioactive waste materials are packaged and shipped offsite for burial. These wastes are assumed to be shipped to a commercial low-level waste burial facility located about 800 km from the site. All wastes are assumed to be shipped by truck. The methods used to estimate radiation doses to transportation workers and to

members of the public from waste shipments are given in Reference 31. Radiation doses received by workers unloading the radioactive materials at the repository or disposal site are not considered in this study since they are assumed to occur at separate licensed facilities.

N.5.1 Technical Approach

The following assumptions are made about truck shipments of radioactive materials:

1. Each shipment of waste contains enough material to result in the maximum exposure rates allowed by regulations. Department of Transportation (DOT) regulations set the following exposure limits:
   - 1000 mR/hr at 1 m from the external surface of any package transported in a closed vehicle
   - 200 mR/hr at the external surface of the vehicle
   - 10 mR/hr at any point 2 m from the vehicle
   - 2 mR/hr at any normally occupied position in the vehicle.[32]
2. For each truck shipment of radioactive waste, two truck drivers spend 12 hours inside the cab and 1 hour outside the cab at a distance of 2 m from the cargo.
3. For each truck shipment of radioactive waste, two garagemen each spend 10 minutes at an average distance of 2 m from the payload.
4. For each truck shipment of radioactive waste, the maximum-exposed individual is located 30 m from the route. His dose rate is calculated to be 1.2×10^{-7} rem per truck shipment.
5. The population density along the transport corridors is 120 persons/km^2.
6. All shipments maintain an average speed of 65 km/hr; thus, the cumulative dose to the public is 2.3×10^{-6} man-rem/km.

7. The $\overline{X}/Q'$ value used for calculating the radiation dose to the maximum-exposed individual from transportation accidents is 3 x 10^{-2} sec/m^3, as discussed in detail in Appendix N of Reference 33. This individual is assumed to be located 100 m downwind from the accident.

Using these assumptions, the following sections contain discussions of the radiation doses during DECON. Both routine and accident conditions are discussed.

N.5.2 Doses From Routine Offsite Transport of Wastes

The numbers of radioactive waste shipments for DECON are found in Appendix I to be 4 for the research reactor and 310 for the test reactor. For preparations for safe storage, the numbers of shipments are 1 for the research reactor and 168 for the test reactor (see Appendix J). For ENTOMB, 2 shipments are required for the reference research reactor and 191 shipments are required for the reference test reactor. Using these numbers of shipments, and the assumptions listed in the technical approach, radiation doses to transport workers and to the public are calculated and listed in Tables N.5-1 and N.5-2 for the research and test reactors, respectively.

N.5.3 Transportation Accidents

Transportation accidents have a wide range of severities. Most accidents occur at low vehicle speeds and have relatively minor consequences. In general, as speed increases, accident severity also increases. However, accident severity is not a function of vehicle speed only. Other factors (e.g., the type of accident, the kind of equipment involved, and the location of the accident) can have an important bearing on accident severity.

Furthermore, damage to a package in a transportation accident is not directly related to accident severity. In a series of accidents of the same severity, or in a single accident involving a number of packages, damage to packages may vary from none to extensive. In relatively minor accidents, serious damage to packages can occur from impacts on sharp objects or from being struck by other cargo. Conversely, even in very severe accidents, damage to packages may be minimal.

TABLE N.5-1. Calculated Radiation Doses from Routine Radioactive Waste Transport at the Reference Research Reactor

Alternative/Group	Radiation Dose per Shipment (man-rem)[a]	Number of Shipments[b]	Total Population Dose per Group (man-rem)[c]
DECON			
Truck Drivers	6.7×10^{-2}	4	2.7×10^{-1}
Garagemen	3.3×10^{-3}	4	1.3×10^{-2}
Total Worker Dose			2.8×10^{-1}
Onlookers	5.0×10^{-3}	4	2.0×10^{-2}
General Public	1.8×10^{-3}	4	7.2×10^{-3}
Total Public Dose			2.7×10^{-2}
Preparations for Safe Storage			
Truck Drivers	6.7×10^{-2}	1	6.7×10^{-2}
Garagemen	3.3×10^{-3}	1	3.3×10^{-3}
Total Worker Dose			7.0×10^{-2}
Onlookers	5.0×10^{-3}	1	5.0×10^{-3}
General Public	1.8×10^{-3}	1	1.8×10^{-3}
Total Public Dose			6.8×10^{-3}
ENTOMB			
Truck Drivers	6.7×10^{-2}	2	1.3×10^{-1}
Garagemen	3.3×10^{-3}	2	6.6×10^{-3}
Total Worker Dose			1.4×10^{-1}
Onlookers	5.0×10^{-3}	2	1.0×10^{-2}
General Public	1.8×10^{-3}	2	3.6×10^{-3}
Total Public Dose			1.4×10^{-2}

(a) Based on one-way trips of 800 km.
(b) Based on the waste disposal requirements discussed in Appendices I, J, and K.
(c) All doses are rounded to two significant figures.

TABLE N.5-2. Calculated Radiation Doses from Routine Radioactive Waste Transport at the Reference Test Reactor

Alternative/Group	Radiation Dose per Shipment (man-rem)[a]	Number of Shipments[b]	Total Population Dose per Group (man-rem)[c]
DECON			
Truck Drivers	6.7×10^{-2}	310	2.1×10^{1}
Garagemen	3.3×10^{-3}	310	1.0×10^{0}
Total Worker Dose			2.2×10^{1}
Onlookers	5.0×10^{-3}	310	1.6×10^{0}
General Public	1.8×10^{-3}	310	5.6×10^{-1}
Total Public Dose			2.2×10^{0}
Preparations for Safe Storage			
Truck Drivers	6.7×10^{-2}	168	1.1×10^{1}
Garagemen	3.3×10^{-3}	168	5.5×10^{-1}
Total Worker Dose			1.2×10^{1}
Onlookers	5.0×10^{-3}	168	8.4×10^{-1}
General Public	1.8×10^{-3}	168	3.0×10^{-1}
Total Public Dose			1.1×10^{-1}
ENTOMB			
Truck Drivers	6.7×10^{-2}	191	1.3×10^{1}
Garagemen	3.3×10^{-3}	191	6.3×10^{0}
Total Worker Dose			1.9×10^{1}
Onlookers	5.0×10^{-3}	191	9.6×10^{-1}
General Public	1.8×10^{-3}	191	3.4×10^{-1}
Total Public Dose			1.3×10^{0}

(a) Based on one-way trips of 800 km.
(b) Based on the waste disposal requirements discussed in Appendices I, J, and K.
(c) All doses are rounded to two significant figures.

The probabilities of truck accidents in this study are based on accident data supplied in Reference 31. Accidents are classified into five severity categories as functions of vehicle speed and fire duration. The five categories and their associated probabilities for truck accidents are shown in Table N.5-3.

TABLE N.5-3. Transportation Accident Severity Categories[a]

Severity	Vehicle Speed (km/hr)	Fire Duration (hr)	Probability per Vehicle-km Truck
Minor	0-50	0	2×10^{-7}
	0-50	<0.5	4×10^{-9}
	50-80	0	6×10^{-7}
Moderate	0-50	0.5-1.0	3×10^{-11}
	50-80	<0.5	6×10^{-9}
	80-110	0	2×10^{-7}
	80-110	<0.5	3×10^{-9}
Severe	0-50	>1.0	3×10^{-12}
	50-80	>1.0	6×10^{-12}
	50-80	0.5-1.0	4×10^{-12}
	80-110	0.5-1.0	4×10^{-12}
	>110	0	6×10^{-11}
	>110	<0.5	6×10^{-11}

(a) Summarized from material in Reference 31.

The majority of the wastes shipped from the reference R&T reactors will require Type A containers. Some of the wastes will be compacted combustible materials. Accident descriptions involving Type A containers are given in Table N.5-4 for minor accidents and in Table N.5-5 for severe accidents. Type A waste packages containing activition products are limited to 3 Ci per package.[34] Since most combustible wastes are from manual operations in low-radioactivity areas, the quantity per container is assumed to equal about 2.5×10^{-3} Ci at the research reactor and 5.0×10^{-2} Ci at the test reactor. These quantities are based on 0.21 m^3 of waste per container at the same specific activities of waste as those defined for the combustible waste fire in Section N.2.2.1. Both containers are assumed to contain the radionuclides

TABLE N.5-4. Minor Accidents for Type A Packages During Truck Transportation

Accident Description	Sequence of Events	Atmospheric Release
Truck Collision or Overturn Involves Waste Containers	1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. Type A packages may rupture, but no fire occurs. 4. Accident is reported to local and federal officials. 5. Packages recovered.	None
Truck Collision or Overturn and a Minor Fire (1/2 Hour or Less) Involves 1 Type A Waste Container	1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. One Type A waste container is involved in a fire with 1/2 hour or less duration. 4. Accident is reported to local and federal officials. 5. Packages recovered.	Release fraction of 5×10^{-4} of the contents of one waste package is assumed.[28]

TABLE N.5-5. Severe Accidents for Type A Packages During Truck Transportation

Accident Description	Sequence of Events	Atmospheric Release
Truck Collision or Overturn and a Major Fire (1 Hour or Longer) Involving 40 Type A Waste Containers.	1. Collision or overturn accident occurs. 2. Truck leaves roadway at high speed and overturns 3. Type A packages rupture and 40 waste containers are involved in a fire. 4. Accident is reported to local and federal officials. 5. Packages recovered.	Release fraction of 5×10^{-4} of the contents of 40 waste packages is assumed.[28]

defined for activated stainless steel in Appendix E. For the minor accident, one container is assumed to burn and release 5×10^{-4} of the radioactivity in the container.[28] The resulting releases are: 1×10^{-6} Ci for research reactor wastes and 2.5×10^{-5} for test reactor wastes. The minor transportation accident is estimated to have a "low" frequency of occurrence.

For the severe accidents, the contents of 40 containers (each containing the amounts of waste defined for the minor accident) are assumed to burn. The total estimated releases are: 5.2×10^{-5} Ci for the research reactor and 1.0×10^{-3} Ci for the test reactor. The severe accident is also estimated to have a "low" frequency of occurrence. A complete listing of the doses from these accidents is shown in Table N.5-6 for the research reactor and in Table N.5-7 for the test reactor.

Although activated core components are assumed to be shipped in a shielded Type B container for the research reactor, no accidents are postulated. This is because only one shipment is required and the material is of solid form that would not be easily made airborne during accident conditions.

TABLE N.5-6. Radiation Doses to the Maximum-Exposed Individual for Releases from Selected Transportation Accidents with Research Reactor Decommissioning Wastes

Accident	Total Atmospheric Release (Ci/hr)[a]	First-Year Dose (rem) Total Body	First-Year Dose (rem) Bone	First-Year Dose (rem) Lung	First-Year Dose (rem) GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total Body	Fifty-Year Committed Dose Equivalent (rem) Bone	Fifty-Year Committed Dose Equivalent (rem) Lung	Fifty-Year Committed Dose Equivalent (rem) GI-LLI
Minor Transportation Accident	1.3×10^{-6}	3.2×10^{-8}	5.7×10^{-9}	1.0×10^{-5}	1.7×10^{-7}	3.2×10^{-8}	2.1×10^{-8}	2.1×10^{-5}	1.7×10^{-7}
Severe Transportation Accident	5.2×10^{-5}	1.3×10^{-6}	2.3×10^{-7}	4.1×10^{-4}	6.8×10^{-6}	1.3×10^{-6}	8.3×10^{-7}	8.3×10^{-7}	6.8×10^{-6}

(a) All releases are assumed to occur in a 1-hour period for comparison purposes.

TABLE N.5-7. Radiation Doses to the Maximum-Exposed Individual for Releases from Selected Transportation Accidents with Test Reactor Decommissioning Wastes

Accident	Total Atmospheric Release (Ci/hr)[a]	First-Year Dose (rem) Total Body	First-Year Dose (rem) Bone	First-Year Dose (rem) Lung	First-Year Dose (rem) GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total Body	Fifty-Year Committed Dose Equivalent (rem) Bone	Fifty-Year Committed Dose Equivalent (rem) Lung	Fifty-Year Committed Dose Equivalent (rem) GI-LLI
Minor Transportation Accident	2.5×10^{-5}	1.2×10^{-7}	2.2×10^{-8}	3.8×10^{-5}	6.5×10^{-7}	1.2×10^{-7}	8.0×10^{-8}	8.0×10^{-5}	6.5×10^{-7}
Severe Transportation Accident	1.0×10^{-3}	2.5×10^{-5}	4.4×10^{-6}	7.8×10^{-3}	1.3×10^{-4}	2.5×10^{-5}	1.6×10^{-5}	1.6×10^{-2}	1.3×10^{-4}

(a) All releases are assumed to occur in a 1-hour period for comparison purposes.

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Safety and Cost Information is developed for the conceptual decommissioning of two representative licensed nuclear research and test reactors. Three decommissioning alternatives are studied to obtain comparisons between costs (in 1981 dollars), occupational radiation doses, potential radiation dose to the public, and other safety impacts. The alternatives considered are: DECON (immediate decontamination), SAFSTOR (safe storage followed by deferred decontamination), and ENTOMB (entombment). The study results are presented in two volumes. Volume 1 (Main Report) contains the results in summary form. Volume 2 (Appendices) contains the detailed data that support the results given in Volume 1, including unit-component data.

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