National Aeronautics and
Space Administration
John H. Glenn Research Center
Lewis Field
Plum Brook Station
Sandusky, OH 44870

April 27, 2005

Reply to Attn of: QD

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Re: Submittal of Decommissioning Plan for the Plum Brook Reactor, Revision 4

Dear Sir:

Enclosed for your information is Revision 4 to the Decommissioning Plan for the Plum Brook Reactor Facility. The revisions incorporate the revised license conditions related to backfilling of excavations as approved by the NRC by issuance of Amendment 12 to License TR-3 and Amendment 8 to License R-93 by letters dated April 21, 2005. We have determined that the revisions to the Plan can be made without prior NRC approval pursuant to 10 CFR 50.59.

Attachment 1 is our written evaluation demonstrating that the revisions may be implemented without prior NRC approval pursuant to 10 CFR 50.59 and Paragraph 3.A.1 of Facility Licenses TR-3 and R-93. The evaluation, which has been reviewed by our Decommissioning Safety Committee as required by Section 9 of the Decommissioning Plan, includes a detailed description of the changes made to the document.

If additional information is required, please contact me at (419)-621- 3314.

Sincerely,

Timothy J. Polich
Project Manager
PBRF Decommissioning Project

A020

Enclosures (2)
1. Decommissioning Plan, Revision 4
2. Written Evaluation (Attachment 1)

cc:
Q/V. W. Wessel
QD/F. J. Greco
ODH/R. E. Owen
ODH/R. H. Vandegrift
USNRC/P. J. Isaac
USNRC/T. F. Dragoun

SAFETY AND MISSION ASSURANCE DIRECTORATE

Plum Brook Reactor Facility

DECOMMISSIONING PLAN

FOR THE

PLUM BROOK REACTOR FACILITY

Revision 4

Form
AD-01/7
Rev 2

NASA PBRF DECOMMISSIONING PROJECT ROUTING AND APPROVAL SHEET

Document Title: **Decommissioning Plan for the Plum Brook Reactor Facility**

Document Number: Un-Numbered Plan **Revision Number: 4**

ROUTING

	Signature	Date
Procedure Originator		25 Jan 05
Review and Concurrence:		
NASA Project RSO		1/25/05
NASA Project Safety Officer		1-25-05
NASA Project Construction Manager		1-25-05
NASA Project Environmental Manager		25 Jan 2005
NASA Project QA Manager		25 Jan 2005
NASA Senior Project Engineer		25 JAN 05

IMPLEMENTING APPROVAL: _____ / 4/26/05
NASA Decommissioning Project Manager Date

EFFECTIVE DATE: April 26, 2005

Form
AD-01/2
Rev 1

NASA PBRF DECOMMISSIONING PROJECT CHANGE/CANCELLATION RECORD		
DOCUMENT TITLE: Decommissioning Plan for the Plum Brook Reactor Facility	**DOCUMENT NO:** Unnumbered	**REVISION NO:** 4

Revision 0: Initial issue of Procedure

Revision 1: Revision to incorporate responses to the NRC's Request for Additional Information sent during NRC review process.

Revision 2: Revision to incorporate responses to the NRC's Request for Additional Information sent during NRC review process.

Revision 3: General revision of the entire document to correct formatting inconsistencies, spelling, and grammar, errors. Wording was incorporated to clarify that demolition of structures after License Termination is a viable option. Wording from the license and technical specifications was incorporated in discussion of procedure preparation, review, and approval. Clarification was added that indicates that characterization surveys are ongoing, FSS information is a preliminary proposal, and that submittal to and approval by the NRC of a FSS Plan is still required and must be done prior to FSS or backfilling operations. Organization chart and description of responsibilities was updated. Unnecessary wording regarding WEP's, JSA's, and RWP's was replaced with explanations that these processes will be controlled by project approved procedures. Incorporated wording from the License regarding reviews in addition to 10CFR50.59 required for Decommissioning Plan changes allowed without NRC approval.

Revision 4: Revision to incorporate amended wording of License Condition 3.A.4 relating to backfilling of excavations. This is a result of NRC issuance of Amendment 12 to License TR-3 and Amendment 8 to License R-93. Revised wording appears on pages 1-23, 2-1. 2-30, and 2-39.

Form
AD-01/3
Rev 1

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LIST OF ACRONYMS

ALARA	as low as reasonably achievable
AMCG	average member of the critical group
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
DCGL	derived concentration guideline
DOE	U.S. Department of Energy
DQO	Data Quality Objective
FSS	Final Site Survey
GRC	Glenn Research Center
HEPA	high-efficiency particulate air (filter)
JSA	Job Safety Analysis
LBGR	Lower Boundary of the Gray Region
MDA	minimum detectable activity
MUR	Mock Up Reactor
MWD	Mega Watt Day
NASA	National Aeronautics and Space Administration
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
OSHA	Occupational Safety and Health Administration
PBRF	Plum Brook Reactor Facility
PCB	polychlorinated biphenyl
QA	quality assurance
QC	quality control
RCRA	Resource Conservation and Recovery Act
RWP	Radiation Work Permit
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TSCA	Toxic Substances Control Act
U.S. EPA	U.S. Environmental Protection Agency
WBS	Work Breakdown Structure
WEP	Work Execution Package

1. SUMMARY OF PLAN

1.1 Introduction

This decommissioning plan describes the decontamination and dismantlement of the National Aeronautics and Space Administration's (NASA's) Plum Brook Reactor Facility (PBRF). The PBRF consists of a complex of buildings and includes two reactors. The PBRF is located within a fenced area in the northern portion of NASA's Plum Brook Station (Figure 1-1). The Plum Brook Station is located about 6-km (4-mi) south of Sandusky, Ohio, about midway between Cleveland and Toledo, south of Lake Erie, and just north of the Ohio Turnpike.

The PBRF operated from 1961 to 1973. NASA currently has two 10 CFR Part 50 facility licenses to "possess but not operate" two reactors within the Reactor Building (1111) at the PBRF. U.S. Nuclear Regulatory Commission (NRC) license TR-3 is for the 60-megawatt research test reactor, constructed for testing materials to be used in space program applications. NRC license R-93 is for the 100-kilowatt swimming-pool type Mock-Up Reactor (MUR). Upon approval of revision 2 of this Decommissioning Plan, these two licenses were amended on March 20, 2002 to allow decommissioning of the facility. NASA previously terminated a material license for operating the Hot Laboratory (1112). The facility is to be decommissioned, with the end objective being removal and disposal of remaining radioactive materials, release of the 11-ha (27-acre) facility for unrestricted use, and termination of the NRC licenses. The radiological criteria for license termination to allow unrestricted use are set forth in 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination," and in the NRC guidance in Draft Regulatory Guide DG-4006, *Demonstrating Compliance with the Radiological Criteria for License Termination* (NRC 1998a).

1.2 Background

The Plum Brook Station is surrounded by farmlands and low density housing. Approximately 2185-ha (5400-acres) of the Plum Brook Station are enclosed within a 2.1-m (7-ft) high security fence. In addition, individual security fences surround several of the existing facilities and test sites within Plum Brook Station, including the PBRF.

The PBRF consists of a complex of buildings within an 11-ha (27-acre) fenced area in the northern portion of the 2590-ha (6400-acre) Plum Brook Station (Figure 1-1). The purpose of the PBRF was to perform irradiation testing of fueled and unfueled experiments for space program application. The PBRF includes:

- The Reactor Building (1111), which contains a 60-megawatt materials test reactor and a 100-kilowatt swimming-pool type mock-up reactor, both of which have been shut down and defueled.
- A seven-cell Hot Laboratory complex (1112).

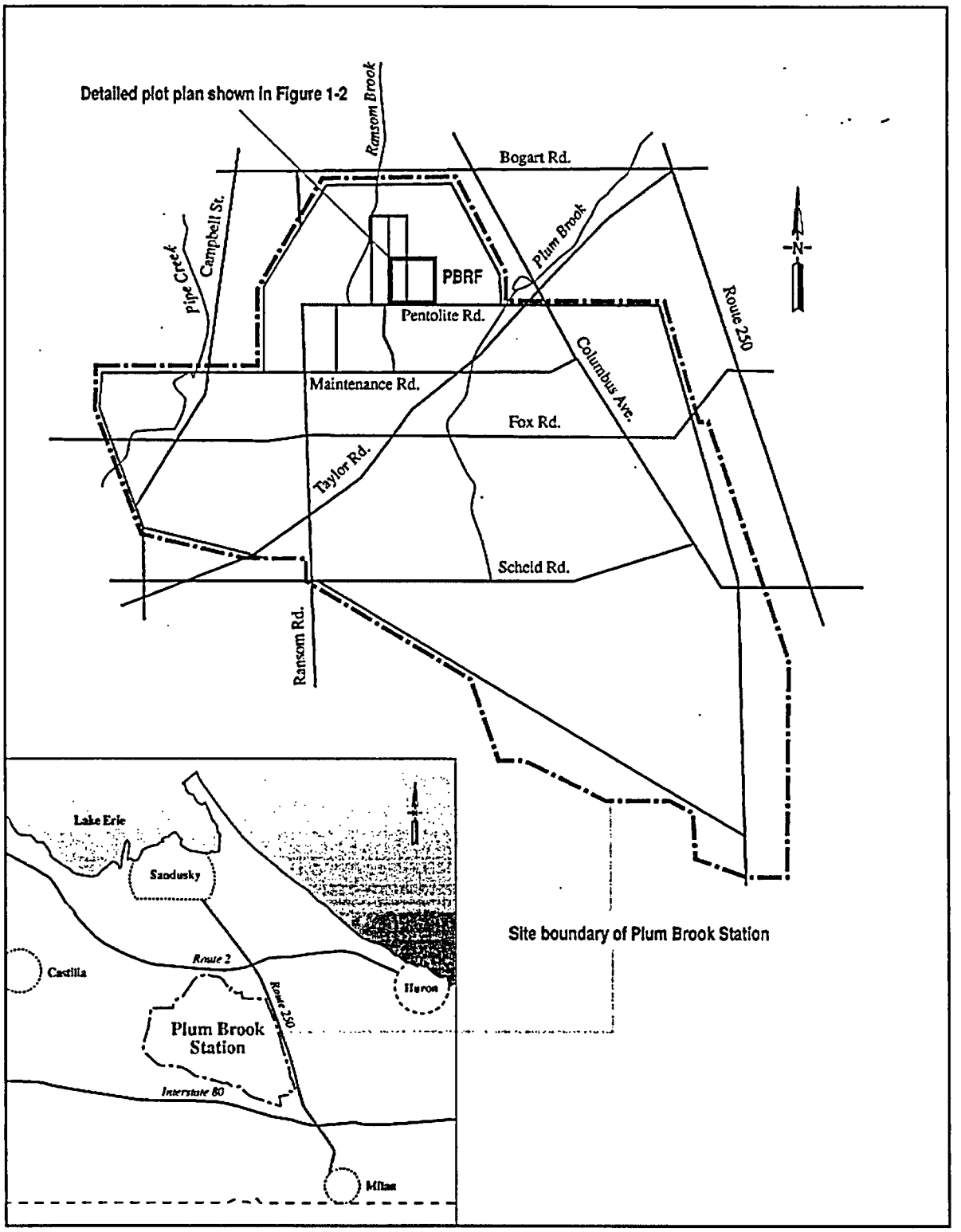

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Figure 1-1. Location of the Plum Brook Reactor Facility at Plum Brook Station (modified from NASA 1980a)

- Reactor and laboratory operations support facilities, which include the Reactor Office and Laboratory Building (1141), the Primary Pump House (1134), the Fan House (1132), the Waste Handling Building (1133), the Hot Retention Area (1155), the Cold Retention Basins (1154), and a hot pipe tunnel.
- Areas of environmental contamination, which include either in-ground or earthen structures or soil that was contaminated as a result of past operations (e.g., spills). These areas are the Emergency Retention Basin, a drainage system, the Water Effluent Monitoring Station (1192), the Pentolite Ditch, and two known past spill areas.
- General support facilities, which include the Reactor Services Equipment Building (1131).

Table 1-1 identifies the major facilities (all are radiologically contaminated), areas of environmental contamination outside of buildings, and support facilities (both contaminated and uncontaminated). Figure 1-2 is a plot plan of the PBRF showing the facilities that compose it.

Table 1-1. Facilities and Areas Composing the PBRF*

Major Facilities	Environmental Contamination	Support Facilities
• Reactor Building (1111) – Reactor tank and internal components – Reactor primary cooling water system and primary cooling shutdown system – Reactor biological shield – Reactor quadrants and canals and pump-out, recirculation, and drain systems – Reactor building rooms – Hot drains, sumps, pumps, and valves – Mock Up Reactor (MUR) • Hot Laboratory (1112) – Hot Dry Storage Area – Hot cells – Rooms	• Emergency Retention Basin • Drainage System • Water Effluent Monitoring Station (1192) • Pentolite Ditch • Areas of contaminated pavement (includes spill areas)	Contaminated: • Reactor Office and Laboratory Building (1141) • Primary Pump House (1134) • Fan House (1132) • Waste Handling Building (1133) • Hot Retention Area (1155) • Cold Retention Basins (1154) • Hot pipe tunnel Uncontaminated: • Cold pipe tunnel • Reactor Services Equipment Building (1131) • Reactor Gas Services Building (1135) • Reactor Compressor Building (1136) • Reactor Substation (1161) • Reactor Security Building (1191) • Reactor water tower (1151) • Reactor sludge basins (1153) • Reactor precipitator (1157) • Reactor Cyrogenic and Gas Supply Farm and Building (1195 and 9837) • Reactor Gas Storage Structure (1196)

* Refer to Figure 1-2 for location of facilities.

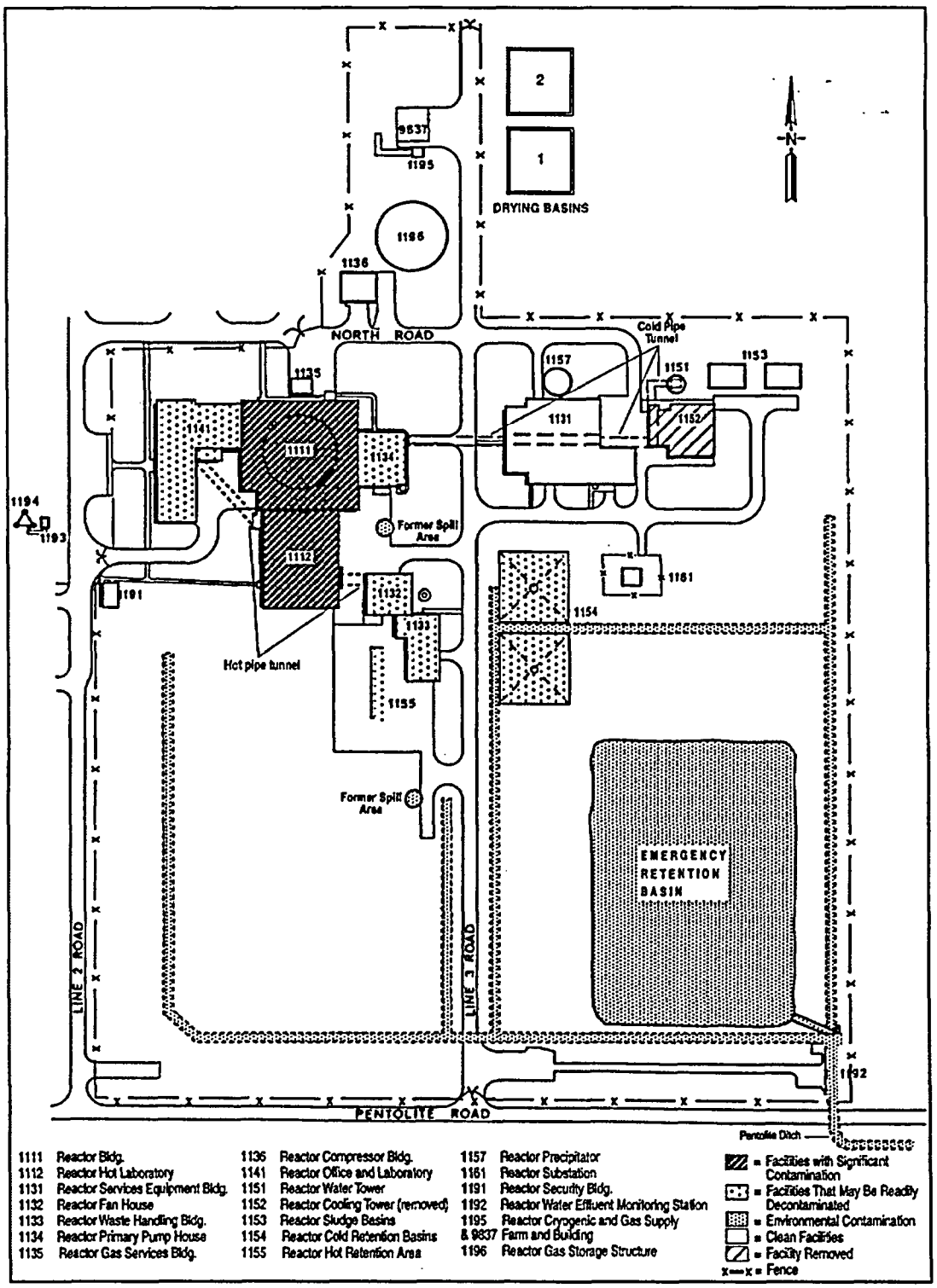


Figure 1-2. Plot Plan of Plum Brook Reactor Facility (modified from NASA 1980b)

Reactor Building (1111)

The Reactor Building is a 46-m × 49-m (150-ft × 160-ft) flat-roofed, four-story building (two basement levels, main level, and a second story level). Elevation and plan views of the Reactor Building are shown in Figures 1-3 and 1-4. The reactor and quadrants are enclosed within a 30-m (100-ft) diameter, 1.9-cm (¾-in) thick steel containment vessel (Figures 1-3 and 1-4), extending from 17-m (56-ft) below grade to 16-m (53-ft) above grade. The reactor tank, a carbon steel vessel with an internal stainless steel cladding, is 2.7-m (9-ft) in diameter and 9.4-m (31-ft) high and is encased in a concrete biological shield varying in thickness up to 2.7-m (9-ft). A cut-away view of the reactor tank and biological shield is shown in Figure 1-5. The top of the reactor tank is near grade level (Figure 1-3). The reactor core consists of uranium/aluminum alloy fuel elements clad with aluminum alloy, arranged in a 3 × 9 lattice with five fueled cadmium control rods in the center row of the lattice (Figure 1-6). Forty-four beryllium reflector pieces surround the fuel eccentrically along with two cadmium/beryllium regulating rods and three shim safety rods (cadmium/beryllium). This 81-cm × 86-cm (32-in × 34-in) array is housed in a core box made of three aluminum alloy side plates, one beryllium side plate, and aluminum alloy top and bottom grids (Figure 1-7). A lockalloy (beryllium/aluminum alloy) flow divider plate is also part of the core box. Three concentric stainless steel thermal shields protect the reactor tank near the reactor core, and two concentric thermal shields are below the reactor core. The reactor was light-water cooled and moderated with a primary beryllium reflector and secondary water reflector. Experiments were 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. Various hydraulic rabbit and instrument thimble assemblies are also present inside the reactor tank.

The reactor tank and concrete biological shield are surrounded by four quadrants, three of which (A, C, and D) could be flooded with water for additional biological shielding. Quadrant B was a dry area. The floors of these quadrants are located approximately 8-m (25-ft) below grade. A system of canals was used to transfer materials or fuel assemblies to and from the reactor tank, the fuel storage area, and the adjacent Hot Laboratory (1112). The layout of the transfer and storage canal system is shown in Figure 1-4. Canal D contained the underwater beam room. Quadrants A and C connect to Canal E inside the perimeter of the containment vessel. A door between Canals E and F permitted transfer of material to and from the containment vessel. Canal G was used for storing spent fuel and contains the spent fuel storage racks. Canal H contains the mock-up reactor. (Canals J and K are located in the Hot Laboratory.)

The quadrant and canal recirculation system recirculated water from Quadrants A, C, and D through two filter units and two mixed resin deionizers in the Fan House. The quadrant and canal pump-out system was used to pump water from Quadrants A, C, and D and Canals E through K into the Cold Retention Basins for storage. There were also options for routing quadrant and canal water to and from the Hot Retention Area.

The Hot Drain System consists of the drain collection systems for all wastewater drainage that originated directly or indirectly from a radioactively contaminated area. The system is made up of 12-collection sumps (located in the Fan House, Waste Handling Building, Reactor Office and

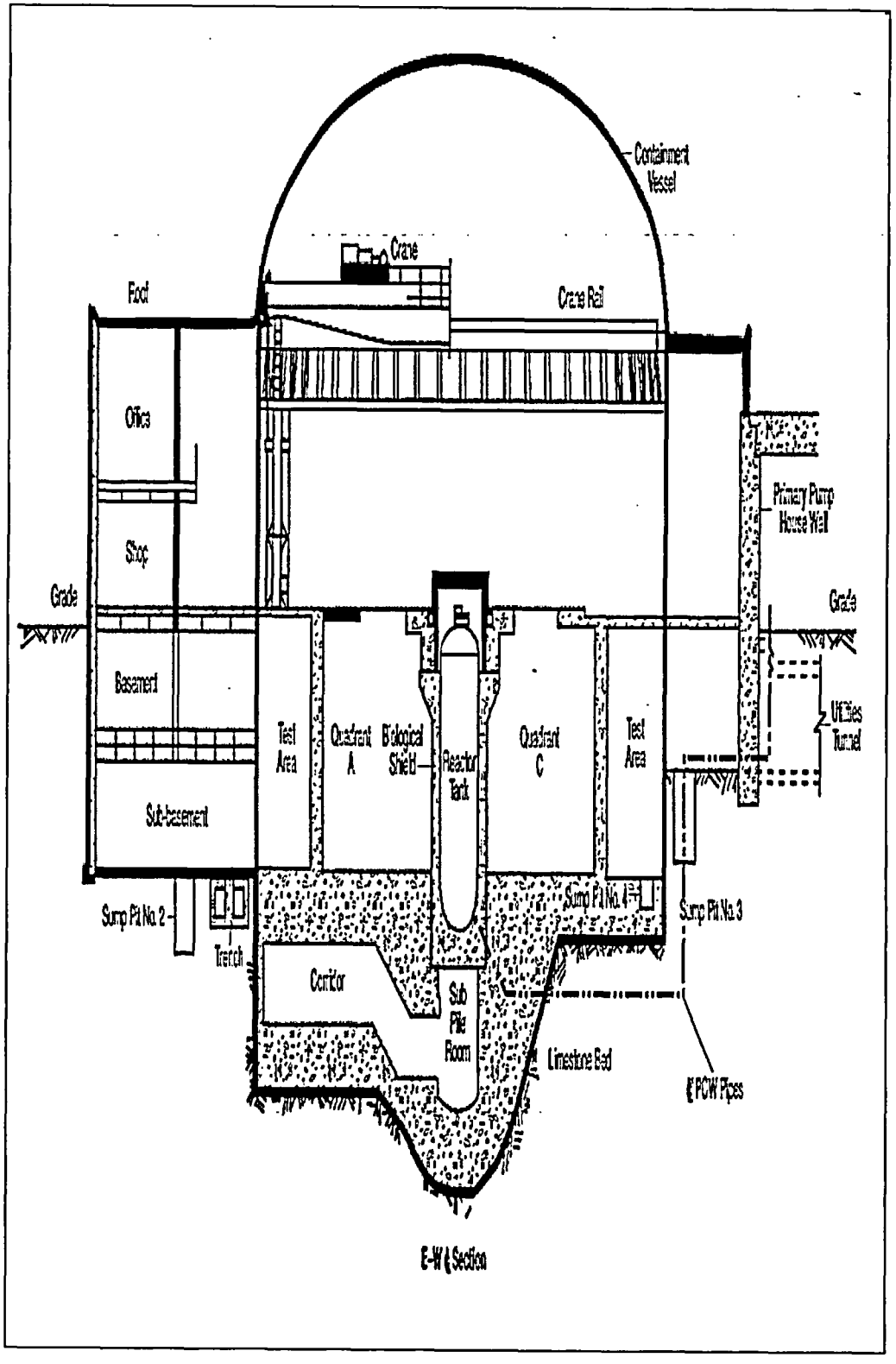


Figure 1-3. Elevation View of Reactor Building (1111) (modified from NASA 1980b)

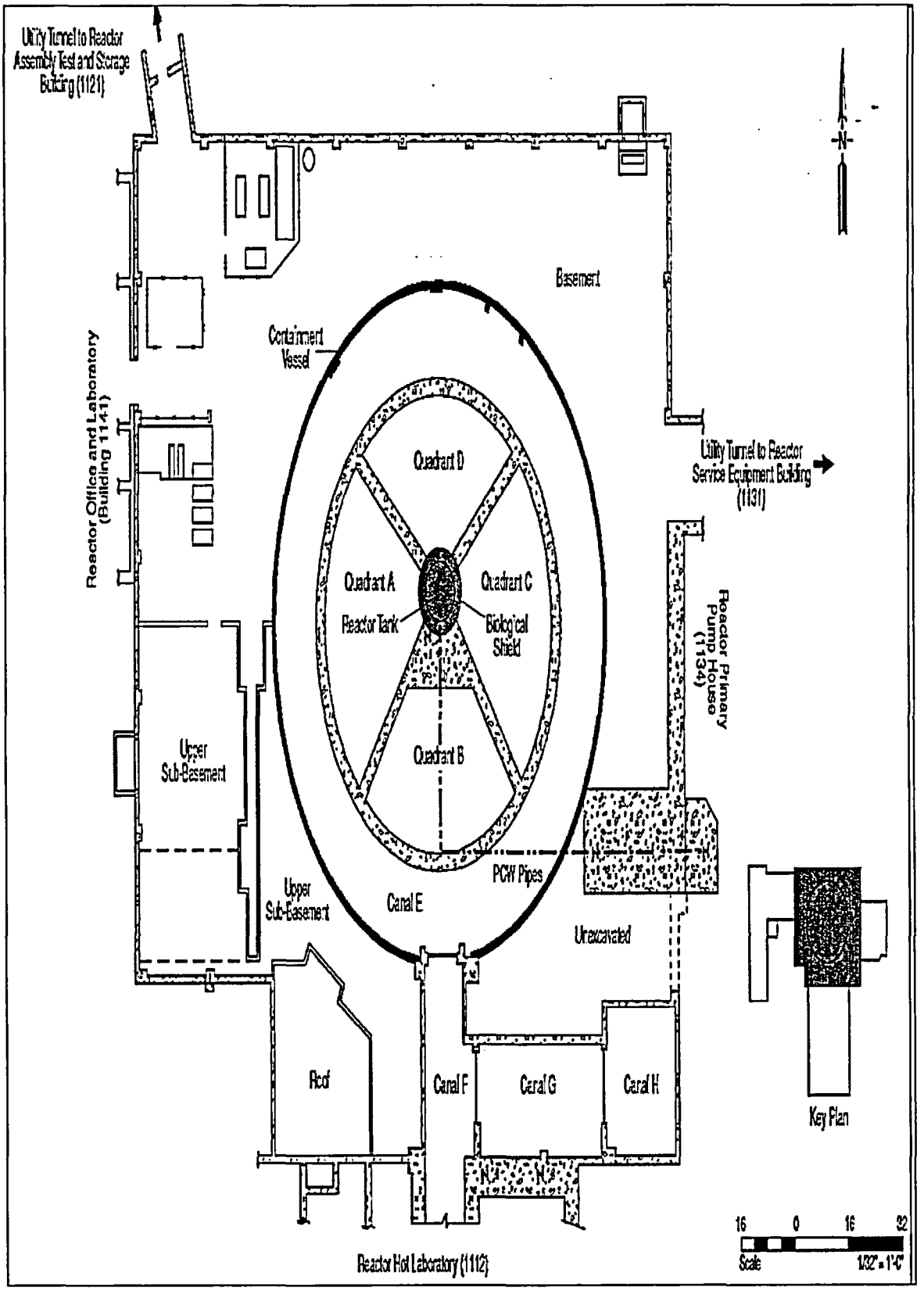


Figure 1-4. Plan View of the Reactor Building (1111) (modified from NASA 1969

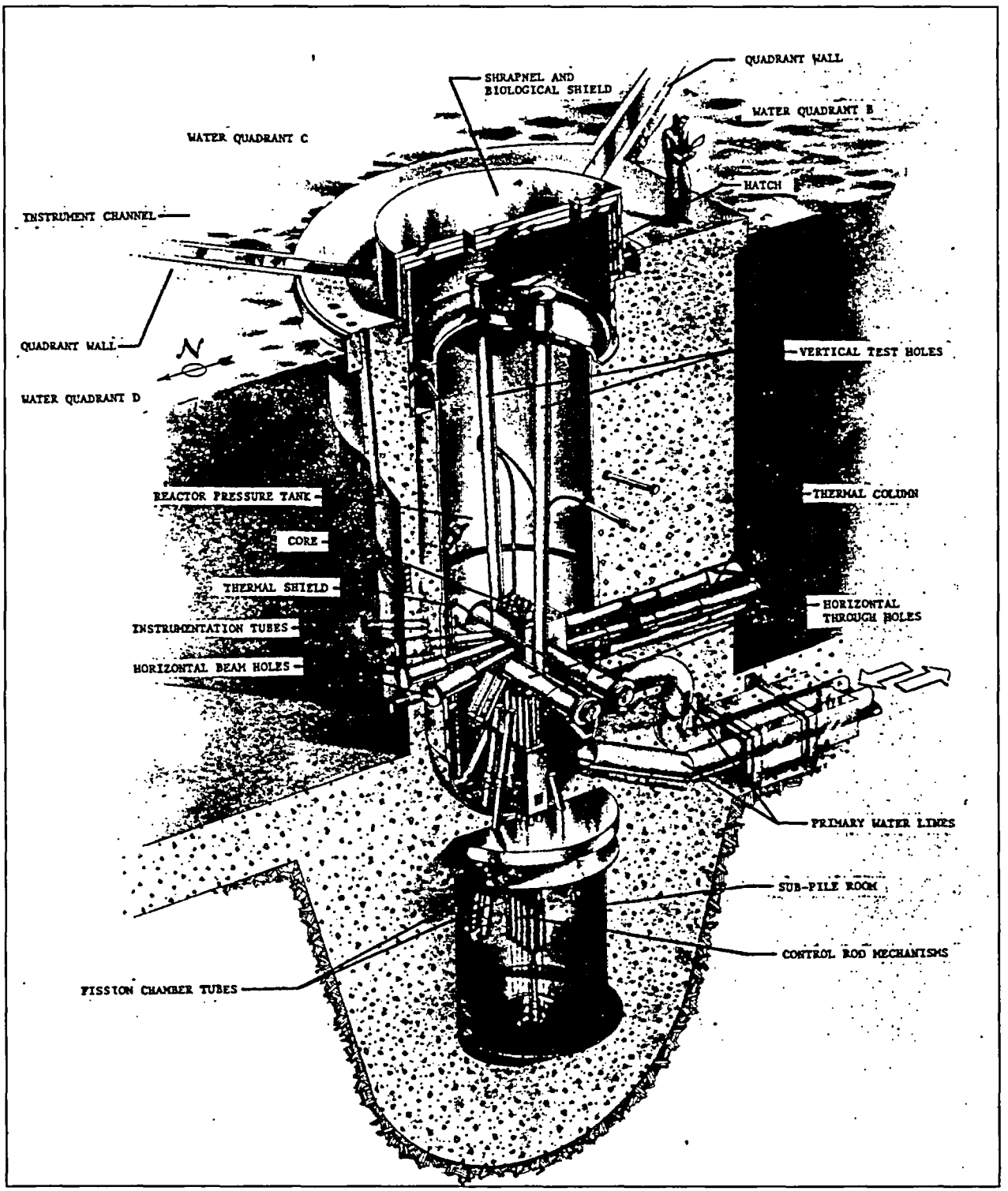


Figure 1-5. Cut-away View of Reactor Tank and Biological Shield (modified from NASA 1980b)

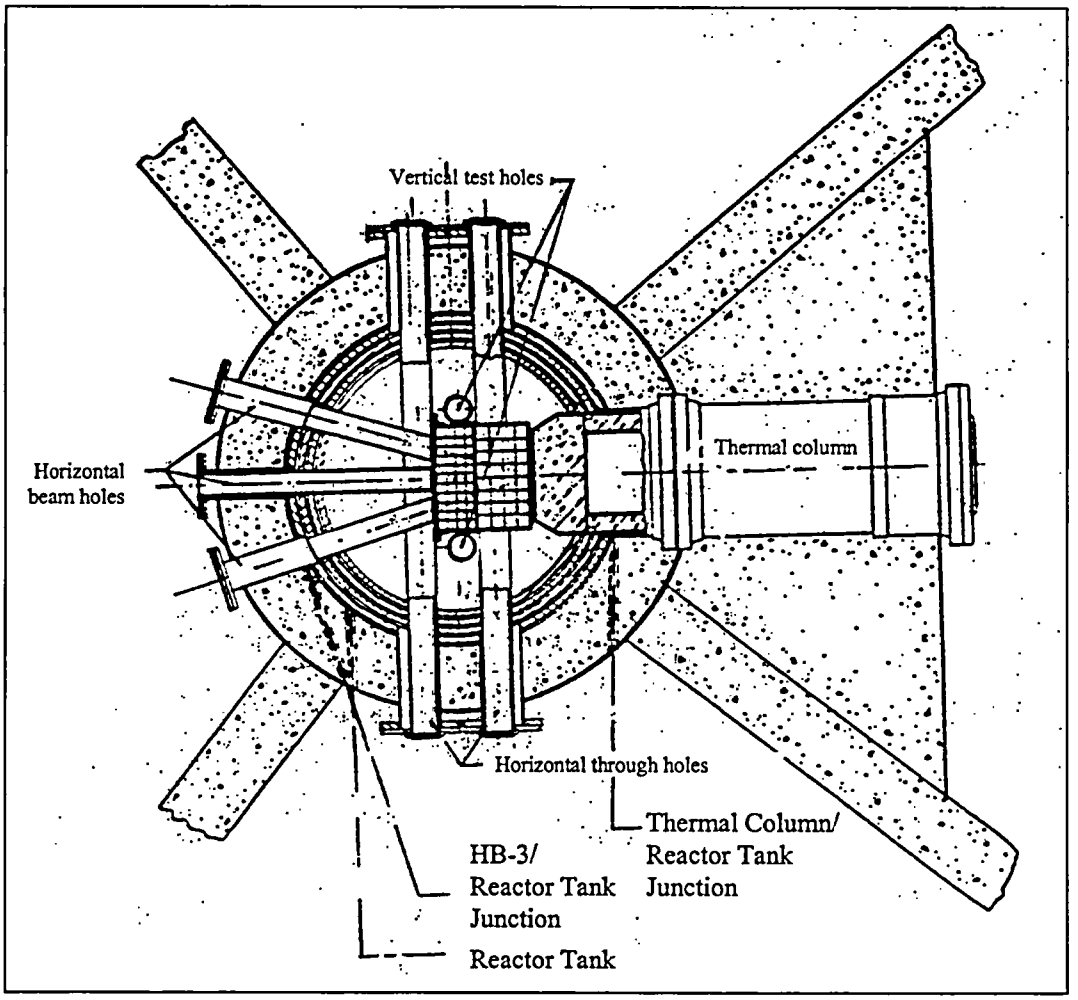


Figure 1-6. Horizontal Cross Section of Concrete Biological Shield and Reactor Core (modified from Teledyne Isotopes 1987)

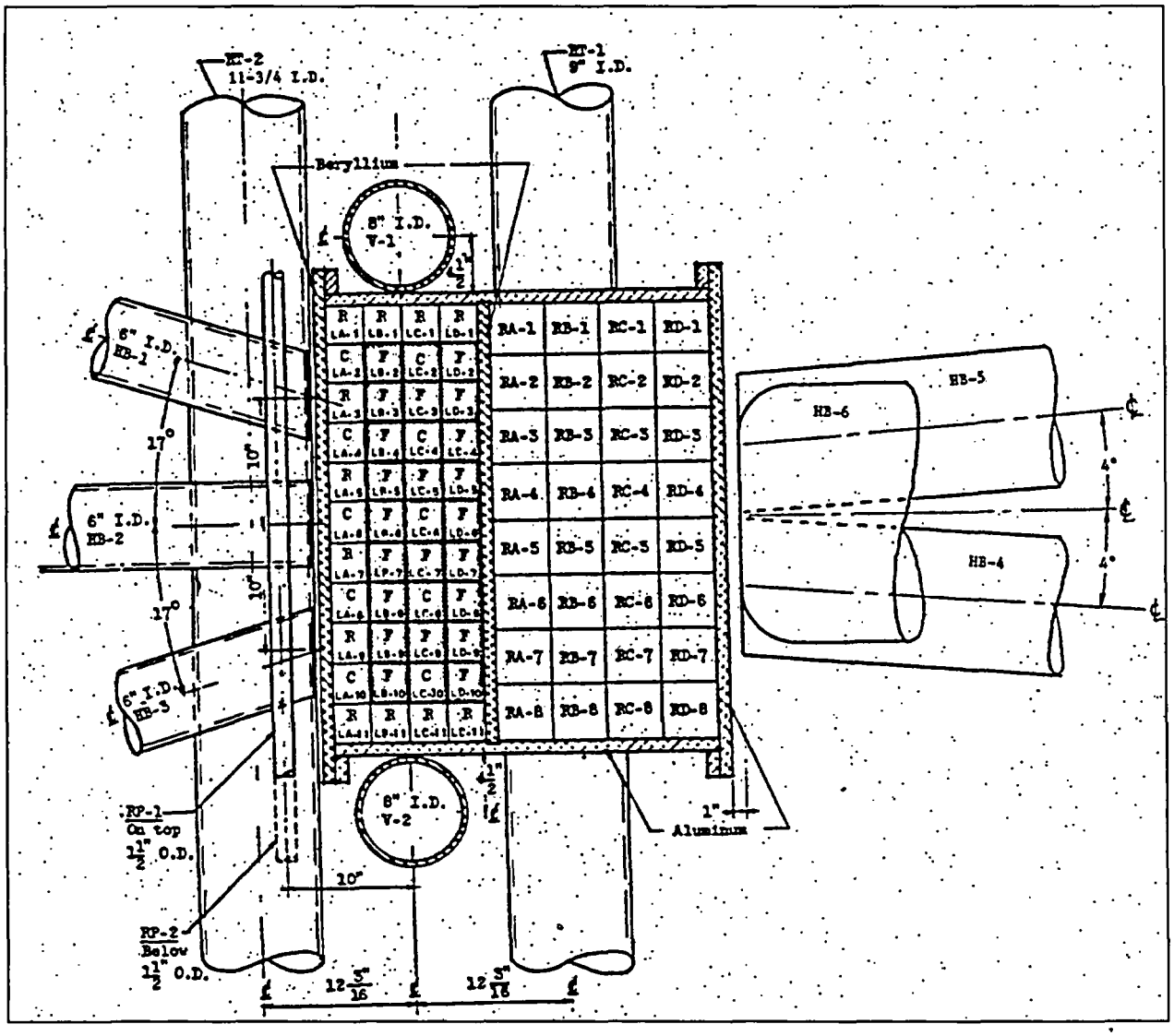


Figure 1-7. Horizontal Cross Section of Reactor Core (modified from NASA 1980b)

Laboratory Building, Primary Pump House, Hot Laboratory, Reactor Building, and inside the containment vessel) along with associated pumps and valves. Pumps were used to pump liquids that had collected in the sumps to the Hot Retention Area.

The MUR is a 100-KW swimming pool type reactor setup to simulate the main Plum Brook Reactor. It was operated essentially as a zero power critical assembly in Canal H to obtain fuel element and experiment calibration data. It was cooled by convection. The MUR was secured and placed in a safe storage condition at the same time the entire PBRF was shutdown and placed in a "possess but do not operate" mode. Figure 1-8 shows the MUR facility layout.

Primary cooling water piping extends from the reactor tank to the Primary Pump House (1134) (adjacent to the east side of the Reactor Building). The piping route is shown in Figures 1-3 and 1-4. The primary cooling water piping was used to remove the heat from the reactor core during operations and transfer the heat to the secondary cooling loop. The primary cooling water piping includes piping in the reactor tank, three primary pumps, two main heat exchangers, a shutdown cooling loop, interconnecting piping, and auxiliary systems such as the bypass cleanup system and the degassifier located in the Primary Pump House. The 61-cm (24-in.) diameter primary system supply and return lines are embedded in concrete (i.e., surrounded by concrete).

The secondary coolant system consists of a single loop system that received waste heat from a pair of primary to secondary heat exchangers and discharged it through a cooling tower. The system then returned the water back to the heat exchangers. These heat exchangers are located at grade level within one of the vault type rooms of the Primary Pump House. The 24-inch diameter secondary cooling water piping passes into the Reactor Building at the –15 foot level, and remains exposed for the remainder of its run. This provides ready access to the system pumps and valves. The piping leaves the Reactor Building and proceeds down the length of the Cold Pipe Tunnel connecting the Reactor Building to the Reactor Services Equipment Building (SEB). It continues on through the SEB and another length of tunnel until it enters the foundation of the cooling tower. The redwood tower was dismantled in the early 1980's due to its potential as a fire hazard. From this point a 24-inch return line retraces the same path through the Cold Pipe Tunnel to the Reactor Building and the Heat Exchanger.

The Reactor Building also houses work space used to set up experiment assemblies, a personnel decontamination facility, a change room, and a control room for remote operation of experiment rigs. The control room and offices are on a mezzanine extending along the north and west walls. Basement areas are accessible by a stairway. The basement areas of the Reactor Building are connected to the Reactor Office and Laboratory Building, Hot Laboratory, and Reactor Services Equipment Building through underground tunnels.

Hot Laboratory (1112)

The Hot Laboratory is a 32-m × 41-m (104-ft × 136-ft), two-story (basement and main level), concrete building attached to the south side of the Reactor Building. Figure 1-9 presents a plan view and Figure 1-10 presents an elevation view of the Hot Laboratory. The layout of the seven hot cells is shown in Figure 1-9. Behind hot cells 1 and 2, next to the Reactor Building, is a heavily shielded 12-m × 23-m (40-ft × 74-ft) hot handling room, (Room 17) having 188-cm (74-in.) thick concrete walls. In the hot handling room, materials were remotely transferred from

Canals J and K to either the hot cells or the Hot Dry Storage Area (Room 19). The hot handling room also contains the off-gas cleanup system (Room 19A). The Hot Dry Storage Area is in the basement area next to Canal J. It is a shielded room used to store irradiated reactor internal components (e.g., beryllium pieces); experiment components (e.g., cadmium sections); and tools (e.g., core fuel element handling tools and end box cut off equipment). The top of the area is covered with removable concrete shielding slabs. A shielded observation window is located in the south wall of the Reactor Building for viewing the Hot Dry Storage Area. A 3-m × 3-m (10-ft × 10-ft) lead-filled sliding door separates the hot handling room from the rest of the hot work area. Rooms in the Hot Laboratory include work areas, an office, storage and repair shop areas, and a decontamination room.

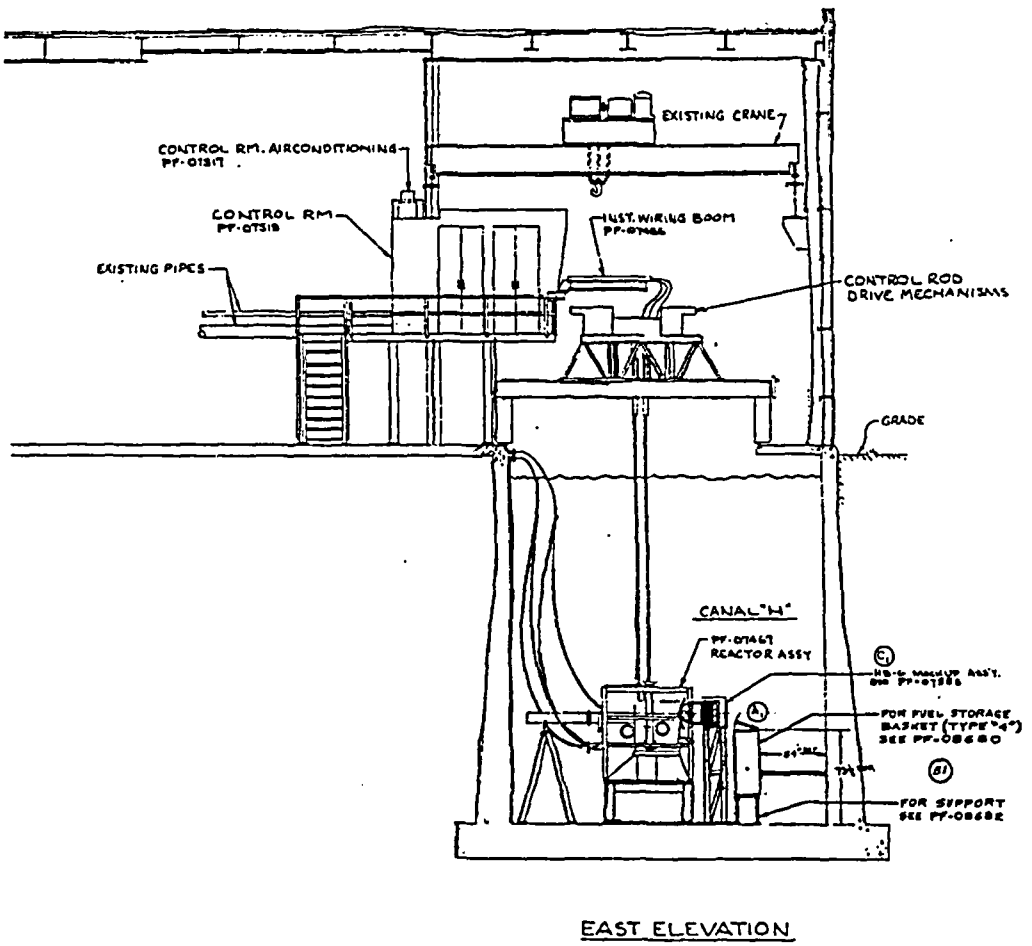


Figure 1-8. Horizontal Cross Section of Mock Up Reactor

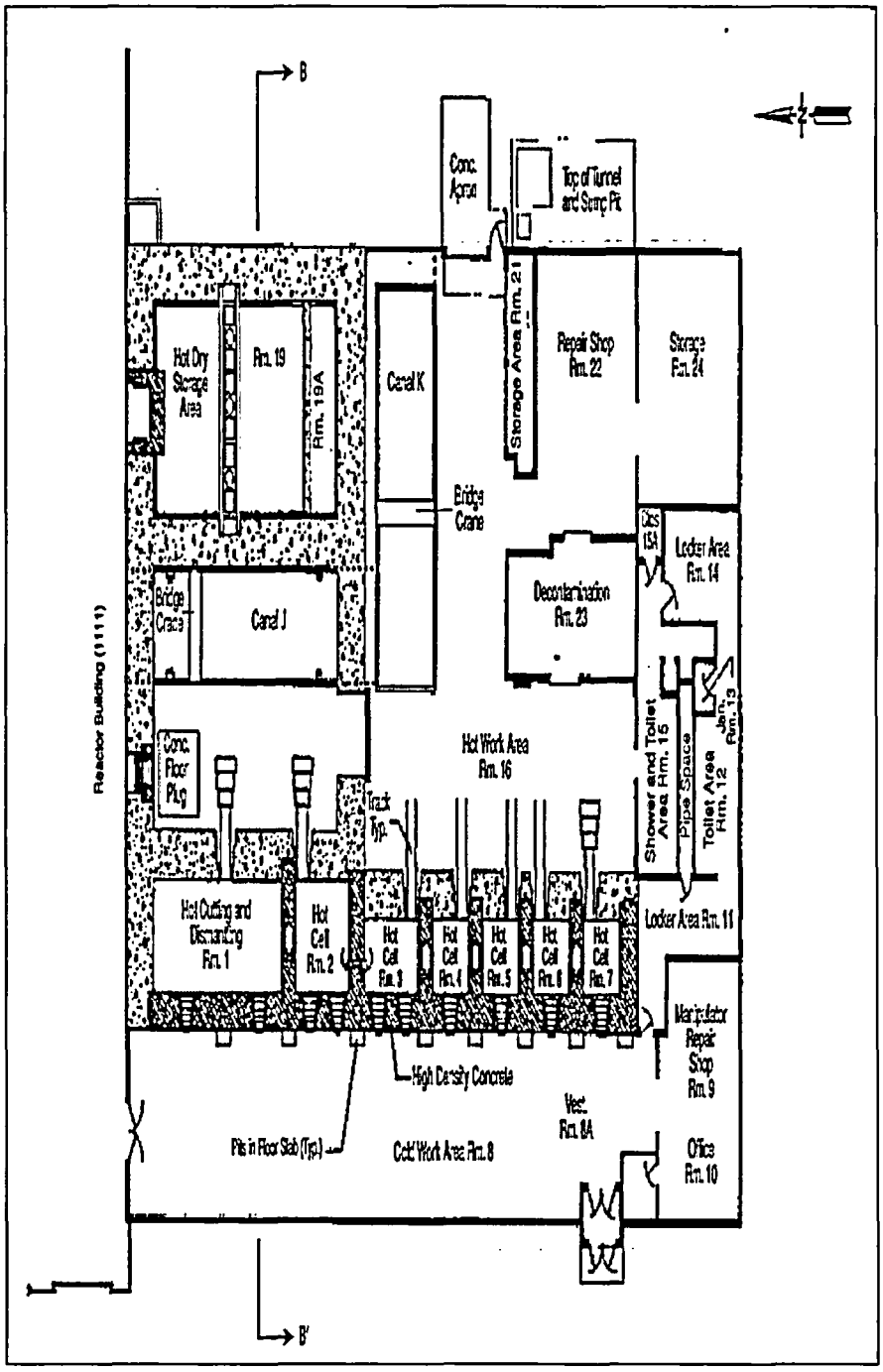


Figure 1-9. Plan View of Hot Laboratory (1112) (modified from NASA 1980b)

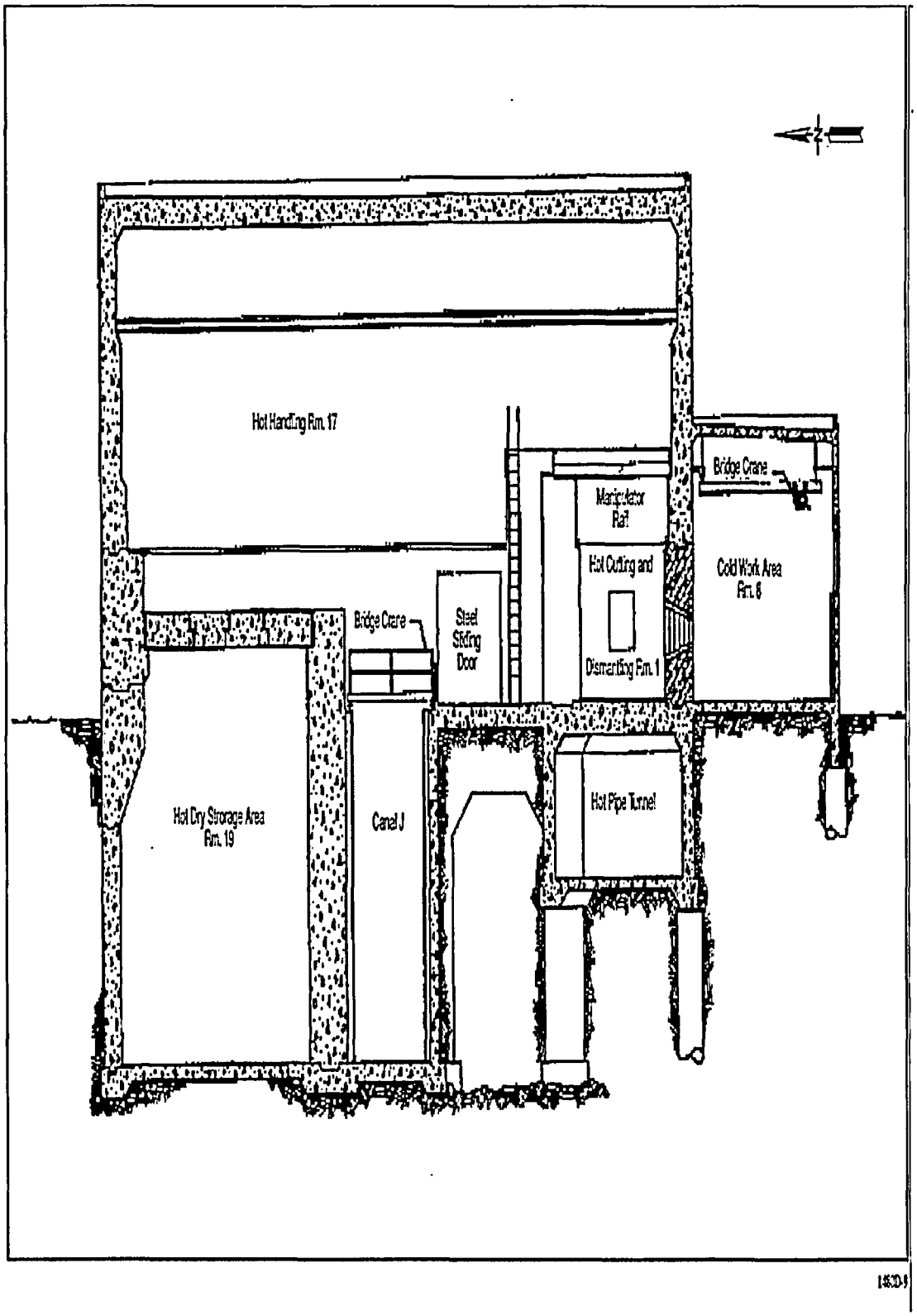


Figure 1-10. Elevation View (Section B-B') at Hot Laboratory (1112) (refer to Figure 1-8 for Section B-B') (modified from NASA 1980b)

Reactor Office and Laboratory Building (1141)

A three-story building (basement, main level, and second story level) attached to the west side of the Reactor Building, the Reactor Office and Laboratory Building housed offices (e.g., for engineering personnel), repair shops, health physics offices, a first aid facility, and radiochemistry laboratories. The layout of the Reactor Office and Laboratory Building is shown in Figure 1-11. At shutdown, the majority of the equipment in the offices and some of the equipment in the laboratories were removed. Equipment remaining in the laboratories includes fume hoods, sinks, drain lines, and a sump. Services to the building were terminated with the exception of electricity and the sumps. Sanitary systems and water were cut off, the heating system was secured, and the laboratory hoods were cabled shut to prevent entry.

Primary Pump House (Building)

A one-story building attached to the east side of the Reactor Building, the Primary Pump House contains the reactor primary pumps, heat exchangers, ion exchangers for the primary cooling system, primary coolant strainer, resin pits, and a hot sump. A floor plan of the Primary Pump House is shown in Figure 1-12.

Fan House (1132)

The Fan House is a two-story building (basement and main level) southeast of the Hot Laboratory. It houses filtration and exhaust systems for several PBRF buildings. Room air from the Reactor Building, containment vessel, Quadrants A and C, Hot Laboratory, Reactor Office and Laboratory Building basement, Primary Pump House, Waste Handling Building, Hot Retention Area, and hot pipe tunnel flowed into the Fan House, was filtered, and then exhausted through the Fan House stack. Equipment inside the Fan House includes pumps, compressors, storage tanks, scrubber, activated carbon absorbers, and monitoring system. The floor plan of the Fan House is shown in Figure 1-13.

Waste Handling Building (1133)

The Waste Handling Building is a two-story building (basement and main level) located southeast of the Fan House. It contains the liquid waste evaporator system with the associated boiler, laundry equipment, waste packaging equipment, and waste storage facilities. The floor plan of the Waste Handling Building is shown in Figure 1-14.

Hot Retention Area (1155)

The Hot Retention Area, located south of the Fan House, contains eight 230,000-L (60,000-gal) and four 28,000-L (7500-gal) steel underground storage tanks. The larger tanks, located in an underground concrete room, received all of the radioactively contaminated water from the hot drain system. The four smaller tanks were used as holding tanks. The contaminated water was treated, and the water in the holding tanks was monitored and then was discharged to the Cold Retention Basins (1154), the quadrant and canal recirculating system, or to the Water Effluent Monitoring Station (1192).

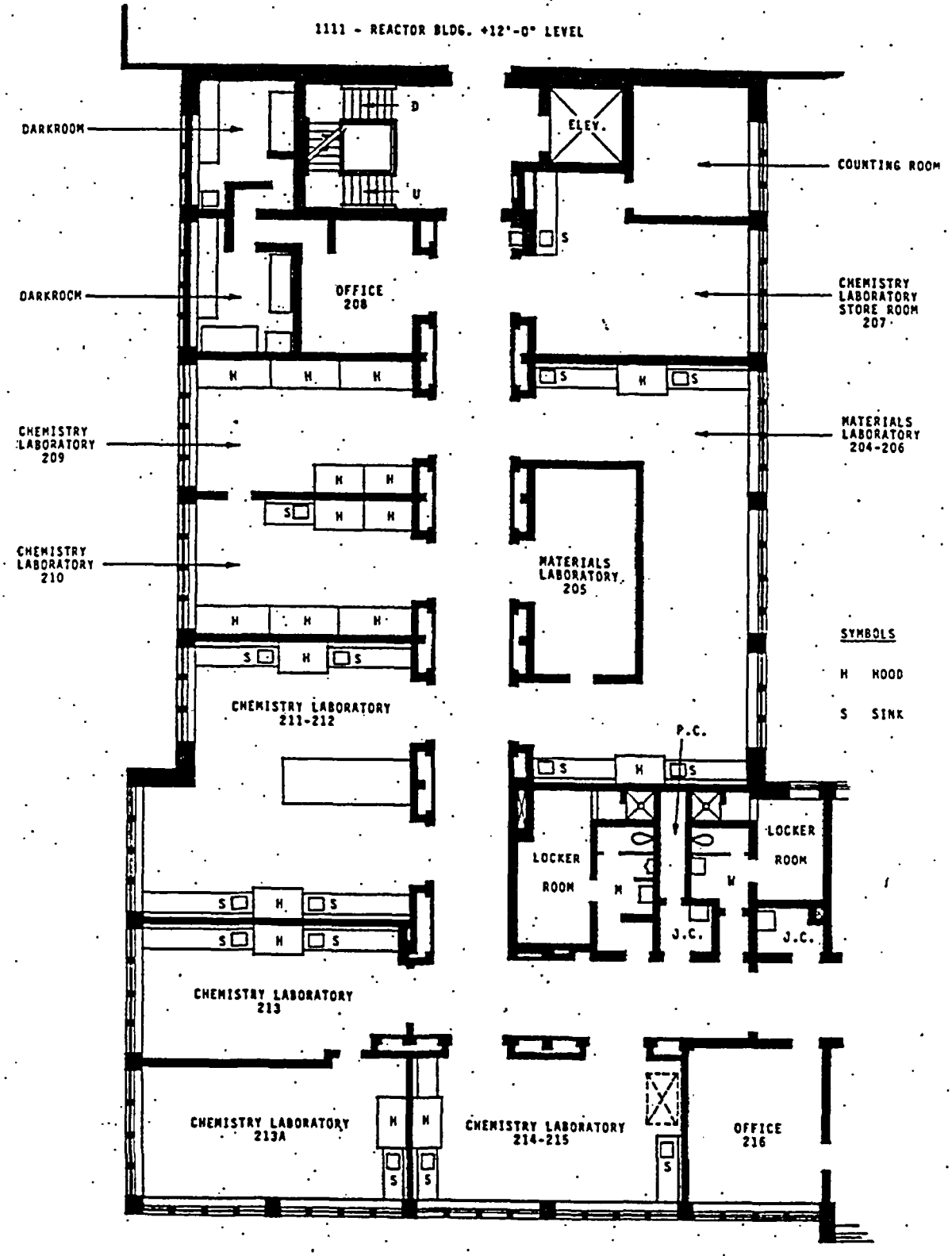


Figure 1-11. Plan View of Reactor Office and Laboratory Building (1141) (modified from NASA 1980b)

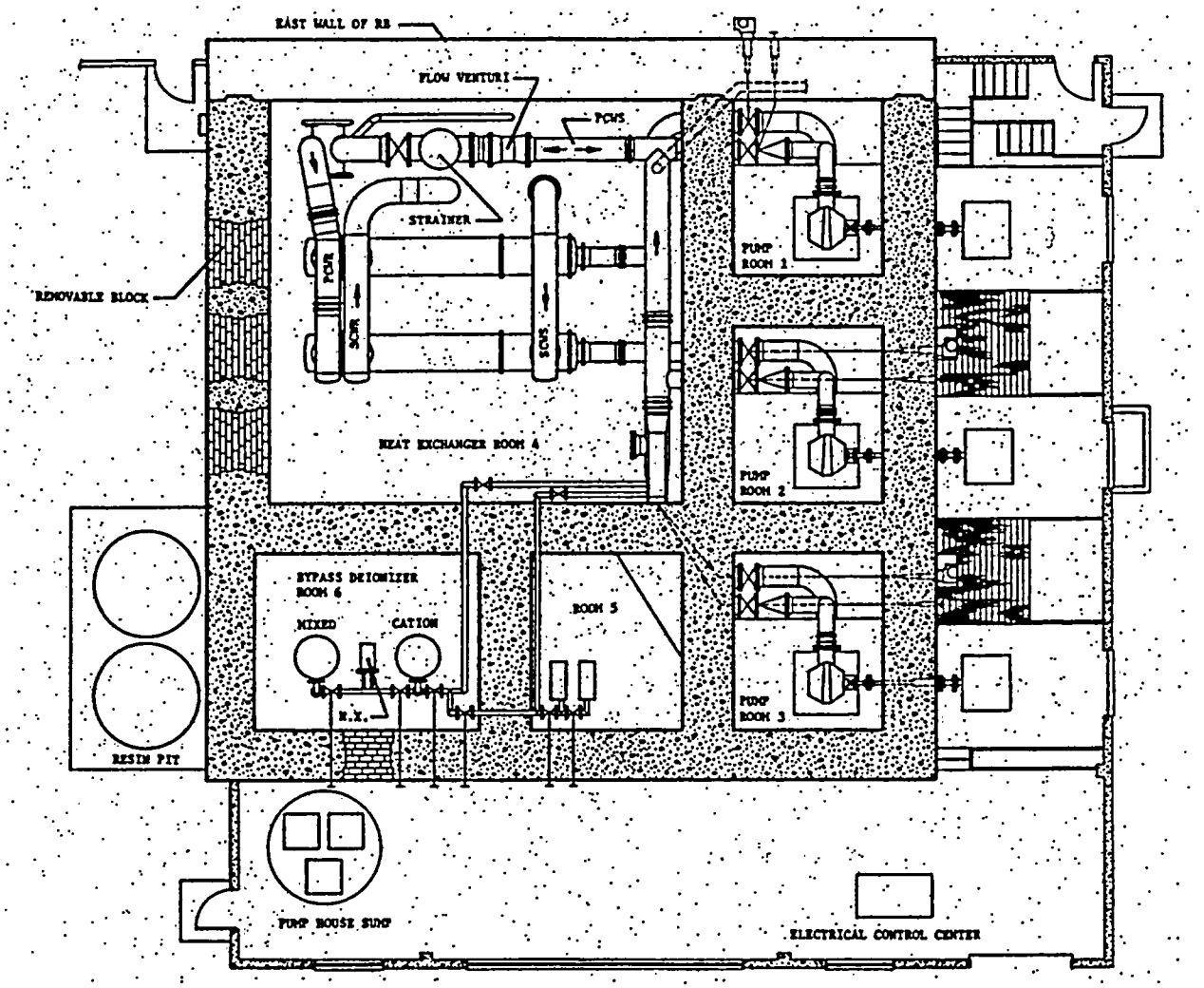


Figure 1-12. Plan View of Primary Pump House (1134) (modified from NASA 1980b)

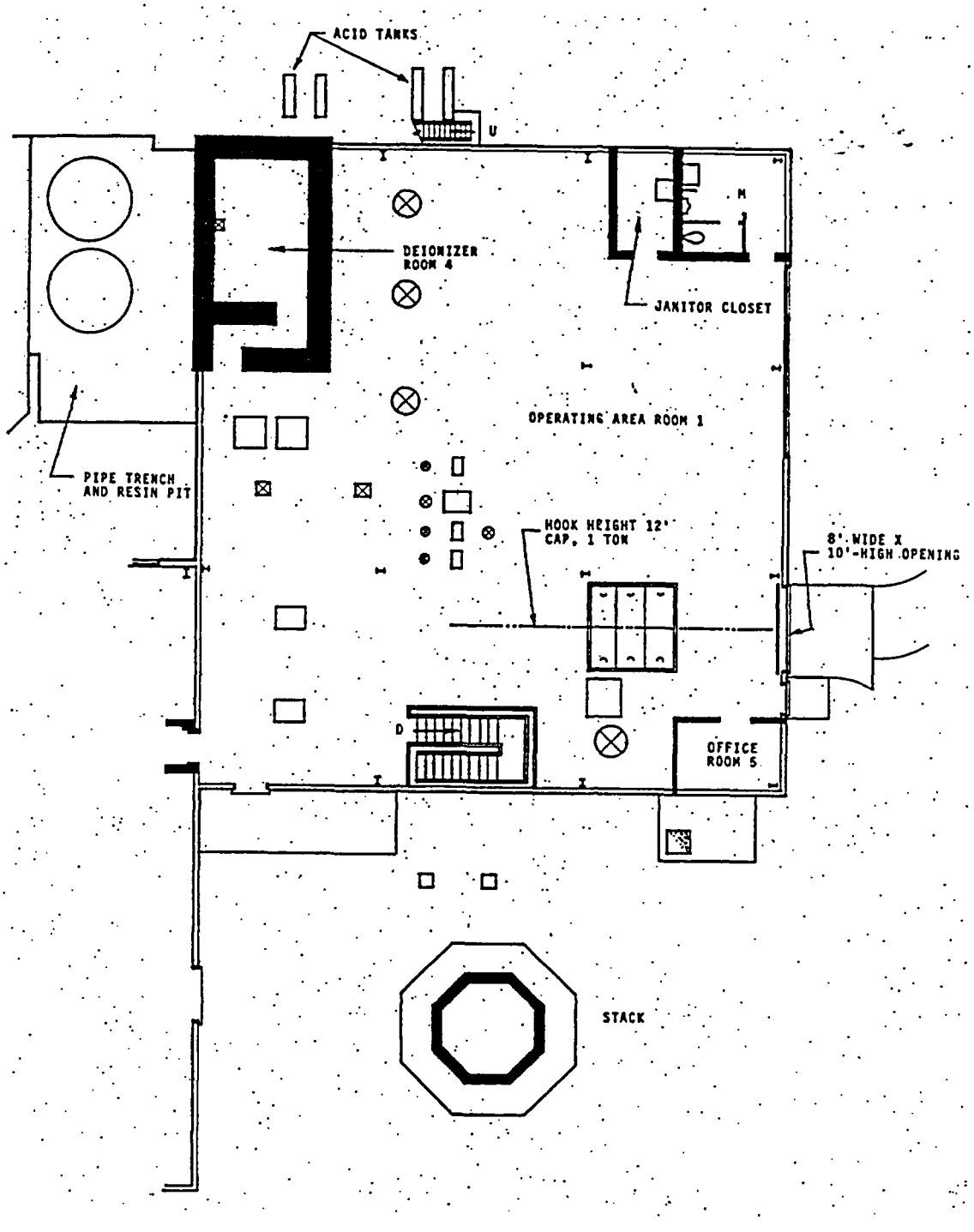


Figure 1-13. Plan View of Fan House (1132) (modified from NASA 1980b)

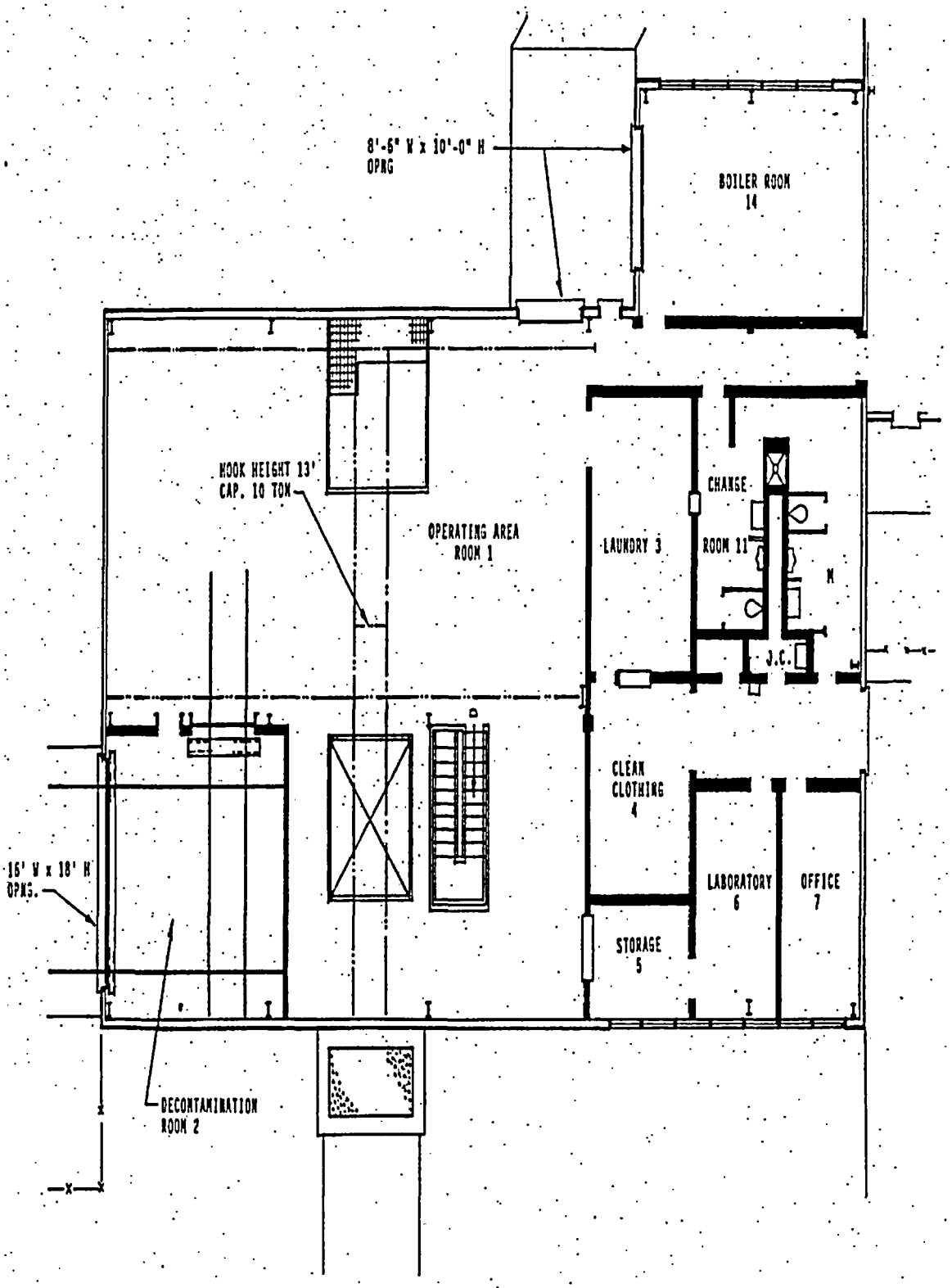


Figure 1-14. Plan View of Waste Handling Building (1133) (modified from NASA 1980b)

Cold Retention Basins (1154)

The Cold Retention Basins are two 1,900,000-L (500,000-gal) below-grade storage basins in the shape of inverted pyramids. The basins were used to store low-level radioactive water, primarily from the quadrants and canals in the Reactor Building. During facility operations the basins were suspected of leaking, so a plastic liner was installed. At the time of shutdown, the Cold Retention Basins were opened to permit groundwater to enter the structures to equalize the water levels in the groundwater and basins and prevent the basins from floating in the event of a high groundwater table. Silt and sludge accumulated on the side walls and bottoms of the basins.

Hot Pipe Tunnels

The hot pipe tunnel connects the Reactor Office and Laboratory Building (1141), the Hot Laboratory (1112), the Reactor Building (1111), and the Fan House (1132) (shown on Figure 1-2). The tunnel contains piping that was used to handle radioactive liquids (e.g., piping that was part of the hot drain systems) and gasses. The tunnel connecting the Reactor Office and Laboratory Building is a 1.8-m (6-ft) diameter corrugated steel pipe. The tunnel connecting the Reactor Building, Hot Laboratory, and the Fan House is made of concrete and is approximately 4.3-m (14-ft) wide by 3-m (10-ft) high.

Emergency Retention Basin

The Emergency Retention Basin was a 76-m × 107-m (250-ft × 350-ft), 38 million-L (10 million-gal) aboveground, earthen-diked basin located in the southeast corner of the PBRF. It was used as emergency storage for radioactively contaminated water that exceeded the allowable discharge criteria. Earth in the basin is mostly brown clay to a depth of at least 3-m (10-ft).

Drainage System

A series of open ditches, covered culverts, and more than 40 catch basins were used to collect and transport surface water runoff, building sump discharges, roof top runoff, and low-level liquid wastes (within discharge limits) to the Water Effluent Monitoring Station (1192). The ditches and culverts are shown as dotted lines on Figure 1-2.

Water Effluent Monitoring Station (1192)

The Water Effluent Monitoring Station is located in the southeast corner of the PBRF (Figure 1-2). It consists of a metal building mounted on top of a concrete trench containing metal gates and flumes. All PBRF liquid effluents flowed through the series of flumes at this station, were monitored for radioactivity, and were discharged to the Pentolite Ditch. A small amount of silt accumulated behind the weirs.

Pentolite Ditch

The Pentolite Ditch is located along Pentolite Road extending from the southeast corner of the Emergency Retention Basin approximately 840-m (2750-ft) eastward to Plum Brook (Figures 1-1 and 1-2). This ditch received all water from the Water Effluent Monitoring Station.

Uncontaminated Facilities

As shown in Table 1-1, there are several facilities within the PBRF that are not expected to be radioactively contaminated:

- Cold pipe tunnel – contains piping used for transporting uncontaminated process water from the Reactor Water Tower (1151) to the Reactor Services Equipment Building (1131) and the Primary Pump House (1134)
- Reactor Services Equipment Building (1131) – located east of the Primary Pump House, contains water processing equipment, air compressors, electrical control equipment, diesel generators for emergency electrical power, and a water chemistry laboratory that was also used for some radiological bioassay and environmental analyses.
- Reactor Gas Services Building (1135) – located just north of the Reactor Building
- Reactor Compressor Building (1136) –located north of the North Road
- Reactor Substation (1161) – located south of the Reactor Services Equipment Building
- Reactor Security Building (1191) – located on the western edge of the PBRF site
- Reactor water tower (1151) – located northeast of the Reactor Services Equipment Building
- Reactor sludge basins (1153) – located east of reactor water tower
- Reactor precipitator (1157) – located north of the Reactor Services Equipment Building
- Reactor Cyrogenic and Gas Supply Farm and Building (1195 and 9837) – located in the northernmost area of the PBRF
- Reactor Gas Storage Structure (1196) – located north of the North Road.

Historical Overview

Construction of the PBRF began in 1956. Preoperational testing of the reactor was performed during 1961 and 1962, and full power operations began in April 1963. The PBRF was used to perform nuclear irradiation testing of fueled and unfueled experiments for space program application. The reactor was operated by NASA on an essentially uninterrupted basis for almost 10 years until January 1973 when it was shut down after accumulating 98,000 Mega-Watt Days (MWD) of operation. The presence of low levels of fission products suggests that a small clad failure or recycling of treated water from the Hot Laboratory may have occurred.

The reactor was defueled from January to July 1973. During that time, the reactor fuel element assemblies (all special nuclear material, source material, and radioactive waste generated at that time) were removed from the PBRF and preliminary decontamination was performed. The fuel assemblies were transferred and reprocessed offsite, and the radioactive wastes were disposed of offsite at licensed commercial sites (NASA 1980a).

An NRC Broad Byproduct Materials License (BPL#34-06706-03) covered operation of the Hot Laboratory and the remainder of the PBRF. This license was amended on August 7, 1973, to

permit only possession and storage of the licensed material. The PBRF systems and support facilities not required for safe storage were maintained in a safe storage condition for possible future operations and the materials license was terminated.

In 1977, NASA decided that the PBRF would not be placed back into operation and that it should be decommissioned, remaining radioactive structures and materials decontaminated or disposed of, and the NRC licenses terminated (NASA 1980b).

In March 1980, NASA requested authorization to dismantle the PBRF and associated facilities, to dispose of waste generated by the decommissioning actions, and to terminate the license. The request to dismantle was submitted for the research reactor, the mock-up reactor, and the Hot Laboratory facilities and was accompanied by an environmental report (NASA 1980a). The original submittals were revised in response to NRC questions. In May 1981, the NRC issued a dismantling order to NASA and authorized proceeding with dismantlement (NRC 1981a). The NRC staff prepared a safety evaluation report before issuing the order, which concluded, "dismantling of the Plum Brook Reactor and the Plum Brook Mock-up Reactor and disposing of component parts as described in the dismantling plans will not be inimical to the common defense and security or to the health and safety of the public" (NRC 1981a). NRC staff also prepared an environmental impact appraisal for the proposed dismantling and concluded that "there will be no significant environmental impact associated with the dismantling of the Plum Brook Reactor Facility and the disposal of component parts, and that no environmental impact statement is required..." (NRC 1981a).

Although the dismantling order was received from NRC, NASA budgetary constraints prevented the Agency from proceeding with its original plans. In October 1984, NASA informed NRC that decommissioning would be delayed because of funding constraints (Dosa 1987). As a result, NASA applied for and was granted a possession-only license that allowed it to "possess but not operate" the two reactors. The "possess but not operate" licenses for the two reactors were reinstated in January 1987 (Dosa 1987).

Although the funding was inadequate to proceed with dismantlement as documented in the 1980 dismantling plan (NASA 1980b), NASA began characterizing the radiological contamination at the PBRF and evaluating alternatives to decontaminate and decommission the facility. The 1985 characterization data were used to support the 1987 engineering analysis of decontamination and decommissioning alternatives. This characterization information is a major information source for the discussion in Section 2.2.2.1 of this plan.

During 1997 and 1998, NASA management decided to decontaminate and decommission the PBRF to allow termination of the NRC licenses and release of the PBRF for unrestricted use.

Alternatives for terminating the PBRF license were analyzed using previous engineering studies. Confirmatory measurements of contamination at the PBRF were also made to confirm the earlier sample measurements. This characterization information is discussed in Section 2.2.2.1.

1.2.1 Reactor Decommissioning Overview

The PBRF is in protective safe storage under two "possess but not operate" NRC licenses. License No. TR-3, Docket No. 50-30 is for the main test reactor and License No. R-93, Docket No. 50-185 is for the mock-up reactor. The PBRF is presently shut down; building entries are controlled and buildings are contained within a locked fence. All process lines, including the primary cooling water system, have been drained, flushed, and isolated. The reactor fuel element assemblies, special nuclear material, and source material have been removed from the site. The reactor tank internals and waste in the Hot Dry Storage Area of the Hot Laboratory contain most of the radioactivity. Other than these areas, the remaining residual radioactivity at the PBRF is confined generally within equipment and piping. Much lower (several orders of magnitude lower) levels of radioactivity are contained in the support facilities. Characterization surveys show that environmental contamination is primarily in the Emergency Retention Basin, with lower levels in the Water Effluent Monitoring Station, Pentolite Ditch, and a spill area.

After evaluating decommissioning alternatives for the PBRF, NASA chose a decommissioning strategy that would lead to decontamination of the facility to a level that would permit termination of the NRC licenses and release of the facility for unrestricted use. In its final state, all contaminated materials and equipment will be removed, radioactively contaminated soils will be removed, and buildings and structures demolished to below grade level and backfilled. In some cases, buildings and structures will be demolished and the demolition debris appropriately disposed of as a means of removing residual radioactivity from the facility prior to termination of the license. In cases where safety, technical, and economic issues show it to be a more viable approach, the buildings and structures will be decontaminated to the license termination criteria and then demolished using conventional demolition techniques after license termination. Excavations and below grade portions of demolished structures will be backfilled with clean hard fill and demolition rubble that evaluation and dose modeling has shown to satisfy the license termination criteria. Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, the results submitted to the NRC with a safety or technical justification for backfilling, and concurrence obtained from the NRC that the area may be backfilled or otherwise rendered inaccessible. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and/or site visits.

The major decommissioning tasks will include:

- Characterization of the facility systems, structures surfaces, and environmental areas
- Removing friable asbestos and lead paint
- Removing reactor internals and tank
- Removing the activated material in the Hot Dry Storage Area
- Removing loose equipment, fixed equipment and components, and piping in buildings and underground areas where necessary
- Removing activated portions of the concrete biological shield and other areas of contaminated concrete inside and outside of buildings
- Removing embedded piping (i.e., piping embedded in concrete), where necessary

- Removing contaminated soil and either leveling or backfilling the areas
- Conducting final status surveys of all affected areas after decontamination to verify that radioactive material has been removed to below the license termination criteria
- Demolishing the above grade portions of decontaminated buildings and other structures (either before or after license termination)
- Backfilling the below grade portions of decontaminated buildings and in-ground structures
- Submitting reports to NRC that demonstrate compliance with the license termination.

The proposed Final Status Survey Plan, developed according to guidance jointly published by the Nuclear Regulatory Commission, Environmental Protection Agency, Department of Energy and Department of Defense, is contained in Section 4 of this decommissioning plan. As decommissioning progresses and additional characterization data is obtained, the Final Status Survey Plan will be refined. Prior to performing the Final Status Surveys, the completed Final Status Survey Plan will be submitted to the NRC for review and approval along with any additional characterization data.

The planning phase and site preparation would last about 14 months. Then decontamination and dismantling activities would begin and last for about 32 months. It is estimated that the total license termination portion of the decommissioning project would be completed by December 31, 2007.

An estimated 3170 m^3 (112,000 ft^3) of radioactive waste will be generated during PBRF decommissioning. It will be sent offsite either to a processor for decontamination or volume reduction or directly to a disposal facility. An estimated 6435 m^3 (227,200 ft^3) of non-radioactive building demolition debris (concrete and metal) that meets release criteria will be generated. Concrete will be used onsite as backfill, and metal debris will be disposed of offsite at an industrial landfill.

The total dose estimated to be received by workers from decommissioning the PBRF is approximately 70 person-rem. The estimated doses to transportation workers and the public along transportation routes from transporting radioactive waste from PBRF decommissioning are estimated to be 5 and 0.5 person-rem, respectively. The greatest radiation exposure to the public would occur if accidents occurred, releasing airborne radioactive material. Hypothetical accidents at the PBRF with unfiltered releases (very conservative assumption) are estimated to result in a maximum estimated total effective dose equivalent (TEDE) of 0.53 mrem to the average member of the public.

1.2.2 Estimated Costs

The estimated cost for decontaminating and decommissioning the PBRF, as described in Section 2.3, is provided in Appendix D. During the actual decommissioning planning phase, a detailed engineering cost estimate will be prepared.

Note: Appendix D is considered NASA pre-decisional (ref. 5 USC §552(b)(5) and 10 CFR §2.790(a)(4) and (a)(5)). This information is provided only for the purpose of the NRC's review and approval of the PBRF Decommissioning Plan. Distribution should be limited to official government purposes only. Release without prior written consent of NASA Glenn Research Center is strictly prohibited.

1.2.3 Availability of Funds

This section provides, in accordance with 10 CFR 50.75(e)(1)(iv), a statement of intent for obtaining funds for decommissioning when necessary. The Director of the Glenn Research Center is responsible for developing and submitting all budgeting and legislative requests necessary to operate, maintain, and ensure the ultimate proper disposition of all Glenn Research Center facilities, including the licensed PBRF. The Center intends to make all appropriate and timely budget submissions necessary to ensure that required funds for decommissioning the PBRF, consistent with the schedule provided in Section 2.3.2, will be requested. The Center intends to advocate appropriate priority in requesting funding for the PBRF decommissioning project.

Certain aspects of the process by which funds are appropriated are not within the control of the licensee, including most notably, but not exclusively, the role of the United States Congress in appropriating the funds by which Federal agencies operate. A project of this magnitude will require a specific line item appropriation by Congress, and the licensee cannot make commitments on behalf of Congress. If Congress does not provide the requested funding levels, NASA will evaluate impacts to the decommissioning project and notify the NRC.

1.2.4 Program Quality Assurance

This section describes the organizational structure established to ensure that Quality Assurance (QA) measures are applied to the planning, dismantlement, radiological surveys, and material shipments. The purpose of NASA's QA program for decommissioning the PBRF is to achieve the following:

1. Prevent the uncontrolled release of radioactive materials offsite
2. Ensure that the radiation exposure to workers and to the public from decommissioning activities is below the limits established in 10 CFR Part 20 and is maintained As Low As Reasonably Achievable (ALARA)
3. Meet the requirements for the packaging and shipping radioactive and hazardous materials, primarily 10 CFR Part 61 and 49 CFR Parts 172 and 173
4. Ensure that work practices employed during all phases of the project are controlled to comply with requirements, that waste is characterized and measured for proper disposition, and that the quality of radiological and chemical measurements is suitable to permit regulators to release the site
5. Prevent the unnecessary spread of radiological and hazardous contamination to uncontaminated areas.

NASA will develop a written QA Plan describing the implementation of its QA Program. The program will be based on the concept that all managers implement and support the QA program when performing daily management and supervisory functions and through the management review and approval process of project documents, such as plans, instructions, procedures, drawings, and specifications. The program will implement those criteria of 10 CFR 50 Appendix B that are applicable to the decommissioning project, and to a degree necessary to fulfill the previously stated purpose of the program. The NASA Decommissioning Project Manager and the project organization will ensure that each contractor's QA program meets the objectives and requirements of NASA's QA program, or that each contractor endorses and implements the NASA QA program in their performance of work activities and processes. NASA will perform independent reviews, as necessary, to ensure contractor compliance. The program will ensure that decommissioning activities are performed in a manner to permit the termination of the PBRF license and the release of the site for unrestricted use.

1.2.4.1 Quality Assurance

QA will include planned and systematic actions necessary to provide acceptable confidence in program results. The program will include, but not be limited to, the following:

1. Authority and Responsibility: Written definitions of authority, duties, and responsibilities of managerial, operation, and safety personnel; a defined organizational structure; assigned responsibility for review and approval of plans, specifications, designs, procedures, data, and reports; and assigned responsibility for procurement and oversight of services (e.g., analytical laboratory). Assigned authority and organizational responsibility to persons performing QA functions to allow them to identify quality problems; to initiate, recommend, and provide solutions; and to verify implementation of solutions.
2. Personnel Training: An indoctrination and training program to provide staff trained and qualified in principles and techniques of jobs assigned, aware of the nature and goals of the QA aspects of the job, and able to demonstrate proficiency maintained by retraining and/or periodic performance reviews.
3. Procedures: Written procedures for decommissioning activities that are prepared, reviewed, and approved by knowledgeable persons and that incorporate appropriate QA requirements. Procedures will be prepared for the following, as required by the Technical Specifications:
    - Routine maintenance on major components that could have an effect on radiation safety,
    - Surveillance tests and calibrations required by the Technical Specifications or those that have an effect on radiation safety,
    - Personnel radiation protection consistent with applicable regulations,
    - Administrative controls for maintenance and for the conduct of activities that could affect facility radiation safety,
    - Shipping and receipt of radioactive material,
    - Waste management,

- Quality assurance,
- Environmental protection management,
- Health and safety management.

4. Documentation and Data Management: Records to document the sequence of activities performed and to track and control a task in its progress from start to finish.
5. Data Assessment: Review and analysis of data. Examining data for reasonableness and consistency and establishing general criteria for recognizing deficiencies.
6. Root Cause Corrective Action Process: Process to investigate and correct recognized deficiencies and document corrective actions.

1.2.4.2 Quality Control

The unique requirements for decommissioning the PBRF include the need to provide a consistent basis for preparing work packages, ensure procedural compliance, and provide reliable tool and equipment calibration. Significant quality control activities will include:

- Control and calibration of radiation measurement equipment
- Receipt inspections of packaging materials and shipping containers
- Work observations and radiation work package compliance
- Control of liquid waste discharges and airborne waste discharges to the environment and consideration of exposure to the public
- Control of waste handling operations and removal of waste from the site
- Control of site surveys
- Accuracy and completeness of project records.

1.2.4.3 Audits and Assessments

To verify implementation of the QA program, qualified individuals will perform periodic reviews, assessments and audits of activities and work processes. Results will be documented and reviewed by management responsible for the area audited.

Decommissioning Contractor Team will perform periodic internal assessments and audits throughout the decommissioning program to ensure compliance with the requirements of this plan.

The NASA Decommissioning Team will perform independent audits, oversight, and assessments of decommissioning activities and processes to confirm project compliance with this Plan, and the NRC License.

In addition to the internal audits and assessments, the Chairman of the Executive Safety Board will appoint a PBRF Audit Team. These members, generally from one to three in number, will

be trained in QA procedures and will not be directly associated with the dismantling activities at the PBRF. The PBRF Audit Team will perform an annual audit of the decommissioning activities that covers all significant aspects of the dismantling, with special attention to the areas of compliance with procedures and the NRC License, and record keeping. A written report of each audit will be prepared, addressed to the Chairman of the Executive Safety Board, and copies will be sent to the NASA Decommissioning Project Manager, the Radiation Safety Officer, and the Chairman of the PBRF Decommissioning Safety Committee. The NASA Decommissioning Project Manager will take corrective action on reported audit deficiencies. The PBRF Audit Team leader or designee will be responsible for verifying that corrective actions have been completed.

2. DECOMMISSIONING ACTIVITIES

2.1 Decommissioning Process

NASA evaluated a range of alternatives for decommissioning the PBRF, from decontamination to allow free release according to Title10 Code of Federal Regulations (10 CFR) 20.1402 to decontamination to allow restricted release according to 10 CFR 20.1403.

The process selected by NASA includes three major steps. The first step will be to remove or decontaminate radioactive components of PBRF to levels that would allow unrestricted release according to the license termination criteria of 10 CFR 20 Subpart E (NRC 1998a). Items to be removed include the reactor tank and its internals, the MUR, the material in the Hot Dry Storage Area, contaminated equipment and piping in PBRF buildings and structures, and contaminated soil in areas surrounding the PBRF. Decontamination involves removal of radioactive contamination from plant structures and components. Waste generated during decontamination will be disposed of offsite. Decontamination will proceed until the residual contamination is below levels that would produce a total effective dose equivalent (TEDE) distinguishable from background that is less than 25 mrem/yr to the average member of the critical group (AMCG) and ALARA. Where feasible, structures will be decontaminated to a level that will allow termination of the NRC License, then demolition will proceed after license termination using conventional demolition and backfill techniques. Where not feasible to decontaminate structures in place due to technical, safety, and economic considerations, they will be demolished prior to license termination and the debris disposed of in accordance with applicable regulatory requirements. Decontamination goals for surface soil, building surfaces, and subsurface material are presented in Section 2.2.3.1 of this plan.

The second step in the decommissioning process will be to perform the Final Status Survey (FSS) in accordance with a FSS Plan that will be submitted to and approved by the NRC. Upon completion of the FSS, a Final Status Survey Report will be submitted to the NRC for review and independent verification, as deemed necessary by the NRC. Excavations and below grade portions of demolished structures will be backfilled with clean hard fill and demolition rubble that evaluation and dose modeling has shown to satisfy the license termination criteria. Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, the results submitted to the NRC with a safety or technical justification for backfilling, and concurrence obtained from the NRC that the area may be backfilled or otherwise rendered inaccessible. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and/or site visits. Documentation will be prepared that demonstrates compliance with the license termination criteria of 10 CFR 20.1402, and will be submitted for NRC review. Following NRC acceptance and approval of the Final Status Survey Report, the NRC may authorize termination of the license.

The final step in the decommissioning process will occur after termination of the NRC license. Remaining structures will be demolished and the below grade areas and excavations will be filled with appropriate fill material. The areas will be graded and contoured to comply with applicable environmental regulations.

2.2 Facility Radiological Status

This section presents the operational history of the PBRF and summarizes the radiological status of the PBRF. Section 2.2.1 discusses routine and non-routine events that occurred during the PBRF's operational history that contributed to facility radioactivity and contamination levels. Section 2.2.2 describes the current radiological status of the PBRF and presents quantitative information. The major historical facts relevant to decommissioning are: there is no fuel in the reactors, there were a few suspected fuel cladding leaks during reactor operations, and the PBRF previously underwent a program of waste removal and decontamination following shutdown. Although there was no confirmed fuel cladding leaks, any suspect fuel elements were removed as a precaution to minimize contamination of the reactor coolant system.

2.2.1 Facility Operating History

The PBRF operated for 98,000 MWD between 1963 and 1973. It was used to perform material and fuel testing in support of the Space Nuclear Rocket Program. All fuel was removed in 1973. The facility has been in a safe storage mode since 1973. Today there are low levels of fission products in the canal and quadrant drains, hot sumps, Hot Retention Area, and Emergency Retention Basin. This contamination could have resulted from inclusions in the fuel element cladding (tramp uranium), minor fuel leaks, or the segmentation of irradiated fuel samples.

Historical and characterization information (Teledyne Isotopes 1987) indicate the following causes of radioactive contamination from routine occurrences/operations:

- During reactor operations, irradiated fuel specimens were processed in the Hot Laboratory; so fission products were expected to be present in the hot cells, hot drains, hot sumps, etc. Characterization sampling showed that Cs-137 and Sr-90 are present in the hot cell drains and sumps.

- Portions of the biological shield nearest the reactor tank were contaminated by neutron activation.

- Operational tasks performed in the seven hot cells resulted in radioactive contamination in the Hot Laboratory air handling system and liquid drain system as well as on various equipment and building surfaces.

- There were many options for routing quadrant and canal water to and from the Hot Retention Area and Cold Retention Area, which resulted in low levels of contamination in these systems.

- The hot drain system collected water from all radioactively contaminated areas and is contaminated.

- During facility operations, it was suspected that the Cold Retention Basins leaked so a plastic liner was installed in 1969. Both basins were pumped dry during the 1985 characterization study, and some silt had accumulated both on the liner and in the

bottoms of the basins. The concrete structures and underlying soil could also be contaminated.

- During operations, the Emergency Retention Basin was used for emergency storage of radioactively contaminated water, and the stored water could evaporate, percolate into the soil, decay off and be discharged, or be diluted and discharged. Therefore, the soil contamination in the Emergency Retention Basin was expected. Characterization showed that the clay-lined base slowed or prevented contamination penetrating to the underlying soil.

- Low-level contaminated water was discharged to the drainage system (drainage pipes and catch basins) that traveled through the Water Effluent Monitoring Station trench, and then discharged to the Pentolite Ditch. Radioactively contaminated silt collected in these areas.

Throughout the operating history there were no major releases of airborne radioactive materials that resulted in detectable variances with background as verified by an extensive onsite and offsite environmental monitoring program (Teledyne Isotopes 1987). Historical and characterization information (Teledyne Isotopes 1987) indicate the following causes of radioactive contamination from non-routine occurrences/operations or accidents/spills:

- During reactor operations, tritium was produced in the various beryllium components of the reactor core. Sampling confirmed that tritium was off-gassing inside the reactor tank.

- Characterization sampling showed that Cs-134, Cs-137, and Sr-90 are present in the quadrant and canal drains, all hot sumps, resin pits, Hot Retention Area, and in the soil at the Emergency Retention Basin. This fission product contamination could have resulted from either a small clad failure or the recycling of treated water from the Hot Laboratory.

- The PBRF reactor had a poison injection safety system consisting of pressure injection of several gallons of gadolinium nitrate solution. This system was accidentally triggered on three occasions during operations; at least one of these occurred during criticality while neutron fluxes existed. The primary cooling water piping was promptly flushed and cleaned out.

- Contamination in the primary cooling water piping consists of a corrosion film deposit in the piping and loose crud and debris in components such as the strainer, pumps, valves, etc. This type of contamination found in the primary cooling water piping and components is expected to be present inside the reactor tank as well as the piping, components, pumps, tanks, etc., in the Primary Pump House.

- A low-level radioactivity spill occurred during spent resin pumping near the Primary Pump House resin pits (see Figure 1-2), resulting in a small area of contaminated soil.

- A spill occurred adjacent to the Waste Handling Building concrete pad (see Figure 1-2), resulting in contaminated pavement and underlying soil.

- A spill occurred adjacent to the Water Effluent Monitoring Station trench, resulting in contaminated soil.

- During the 1985 characterization study, the polyethylene hot cell drain line in the hot pipe tunnel was found to be broken, leaking contamination onto the floor of the tunnel. The cause of failure was believed to be the expansion/contraction due to temperature difference together with possible failure due to radiation damage. The fracture was repaired and a strippable coating formulated to remove contamination from concrete surfaces was applied to the contaminated area, and was partially successful at removing the contamination.

- Spills occurred in a small location on the floors of Room 212 and 214 in the Reactor Office and Laboratory Building that penetrated the cracks between the floor tiles.

One of the conclusions from the 1985 characterization survey (Teledyne Isotopes 1987) was that there was no other apparent leakage of any radioactive contaminated material into sub-surface soils from either surface infiltration or leakage from deep structures.

2.2.2 Current Radiological Status of the Facility

This section summarizes radiological characterization data for the facilities and areas at the PBRF. A major characterization survey for the entire PBRF was conducted in 1985, and a confirmatory survey was conducted in 1998. Section 2.2.2.1 summarizes the characterization information from the two surveys. The results from the two surveys indicate that most of the residual radioactivity at the PBRF is confined generally within equipment and piping, including the reactor tank, and in the Hot Dry Storage Area. Limited environmental contamination was found.

Facility characterization continues to be performed as decommissioning progresses. The characterization being performed has been designed to meet the guidance provided in DG-4006 and NUREG-1575. The results of these surveys will be provided when completed and will be used to guide the decontamination efforts, develop waste profiles for shipping and disposal of materials removed during the decommissioning operations, and classification of areas. The additional characterization data will also be used in the design of the Final Status Survey and in the development of the final site specific Derived Concentration Guideline Levels (DCGL's).

Sections 2.2.2.2, 2.2.2.3, and 2.2.2.4 summarize radiological characterization information collected in 1985 (Teledyne Isotopes 1987) and in 1998 (Appendix A) for the major facilities, areas of environmental contamination, and contaminated support facilities, identified in Table 1-1. Section 2.2.2.5 briefly summarizes the nonradiological characterization of the PBRF. The major facilities, areas of environmental contamination, and contaminated support facilities in Table 1-1 would be classified as impacted areas for the final status survey, and the uncontaminated support facilities in Table 1-1 would be classified as non-impacted areas. A more detailed classification of these facilities as impacted (Class 1, 2, or 3) or non-impacted is presented in Section 4.1.4 and Table 4-2 of this plan.

The following discussion emphasizes relevant information for decommissioning planning and for demonstrating compliance with the license termination criteria of 10 CFR 20.1402, which has a standard for dose "distinguishable from background." Residual contamination in the PBRF buildings and environment is from activation products (i.e., H-3 and Co-60) and fission products (i.e., Cs-137 and Sr-90). The background concentrations of these radionuclides are essentially negligible. Radionuclides such as K-40 and Ra-226 are naturally occurring and were measured during the 1985 PBRF characterization. A summary of background radionuclide concentrations is presented in Section 2.2.2.1.

2.2.2.1 Radiological Characterization of the PBRF

Two radiological characterization efforts have been conducted at the PBRF. A radiological survey of the PBRF was conducted in 1985. A confirmatory survey was conducted in September 1998 to verify the 1985 results and to provide additional isotopic data to use for estimating doses for license termination. During the 1998 confirmatory survey, buildings that were not expected to require decontamination were surveyed because contamination in these areas could impact decommissioning planning and costs. Two areas of environmental contamination, the Emergency Retention Basin and the Pentolite Ditch were also sampled to confirm the 1985 data.

Most of the inventory at the PBRF is contained in the reactor tank internals and the waste in the Hot Dry Storage Area. Tritium (H-3) is the primary radionuclide of concern in these areas. Outside of the reactor tank and Hot Dry Storage Area, the radionuclides of concern consist of both mixed fission products and activated materials, with the primary radionuclides expected to be Co-60, Cs-137, and Sr-90.

1985 Characterization Survey

The first survey in 1985 (Teledyne Isotopes 1987) characterized the buildings and ground surface around the PBRF. The floor and inside wall surfaces at all elevations (including basements) were surveyed in the Reactor Building, the Hot Laboratory, the Waste Handling Building, the Fan House, the Primary Pump House, the hot pipe tunnel, and the Reactor Office and Laboratory Building. The exterior surfaces of the containment vessel dome and roofs were not surveyed because both were resurfaced. The grounds within the fence line were also surveyed, including soil surfaces, paved areas, and the Pentolite Ditch from the PBRF to Plum Brook.

Background samples were collected and analyzed for eight categories of soil and six buildings that were not affected by plant operations (Teledyne Isotopes 1987). The background characterization consisted of measuring gross alpha and gross beta activity levels for all samples and direct radiation levels for a portion of the samples.

For soils, the majority of the samples had gross alpha and gross beta activity levels of 6 to 10 pCi/g and 30 to 40 pCi/g, respectively. Direct radiation levels were approximately 6 μR/hr. These levels are consistent with background levels in other areas of the U.S. One set of background soil samples was collected from a location containing an outcropping of shale. These samples had average gross alpha and gross beta activity levels three times higher than the balance of the background soil samples.

For building surfaces, background characterization included collection of smear samples and static measurement of gross alpha and gross beta activity levels. The average gross alpha activity level was 3 cpm, which is consistent with gross alpha levels reported for similar materials. The average gross beta activity level was 30 cpm, which is lower than gross beta levels reported for similar materials. For typical equipment characteristics, the reported count rates correspond to gross alpha and gross beta activity levels of approximately 25 and 250 dpm/100 cm^2, respectively.

The outdoor area and buildings were surveyed on grids for gross alpha and gross beta activity within the PBRF fence line. Direct radiation measurements were taken with a micro-roentgen meter. Surface and deep soil samples were analyzed for gross alpha and gross beta activity.

Isotopic analyses were performed on all samples containing significant quantities of radioactive material when those samples represented the systems or structures from which they came. Radioisotopes were identified by gamma pulse height analysis using germanium detectors networked in multi-channel analyzer systems. Strontium-90 was analyzed by chemical separation of strontium, holding for in-growth of the Y-90 daughter and subsequent counting and analysis. Low energy gamma or pure beta emitters, such as Fe-55 or Ni-63, were not measured during the isotopic analysis.

The 1985 characterization survey estimated the radiological inventory of the reactor tank and internals. Three core samples from the biological shield were analyzed for gross alpha and gross beta activity; some portions of the core samples were analyzed for Co-60. Piping and drain systems were also characterized. External contamination and direct dose rates were measured and corrosion films were collected. The water handling systems, including the Hot Retention Area and Cold Retention Basins, also were analyzed. External contamination and direct dose rates were measured and sludge samples were collected and analyzed.

The major conclusions from the 1985 characterization survey were:

- The majority of the radionuclide inventory at the PBRF is in two locations: (1) the reactor tank and its internals and (2) in stored waste in the Hot Dry Storage Area (in the Hot Laboratory).
- Most of the contamination inside the buildings is inside piping and equipment. Other than the internal piping and equipment contamination, residual contamination in the facilities is limited to locations where piping or equipment has leaked (e.g. the hot pipe tunnel and evaporator in the Waste Handling Building).
- In the reactor tank (exclusive of reactor internals) and the primary cooling system, Co-60 was the dominant gamma-emitting nuclide based on analysis of corrosion film samples.
- The isotope Co-60 and fission products Cs-137 and Sr-90 were detected in the canal and quadrant drains, hot sumps, resin pits, Hot Retention Area, and Cold Retention Basins.
- Areas of environmental contamination contain Co-60 and fission products.

- Residual activity levels in the MUR ranged from 1.5 mrem/hr to 13 mrem/hr with no significant alpha activity.

1998 Confirmatory Characterization Survey

In 1998, a confirmatory radiological survey (documented in Appendix A) was conducted at portions of the PBRF to support the planning for decommissioning and license termination activities. For the confirmatory survey, only the easily detected radionuclides were analyzed (by gamma spectroscopy) and quantified. As a result, beta emitters and radionuclides that are difficult to detect (i.e., Sr-90, Fe-55, Ni-63, and other low energy beta emitters) were not identified and quantified. The analysis for the primary gamma emitters (i.e., Co-60, Cs-137, and europium isotopes) was determined to be adequate to verify the 1985 characterization data. Where possible, the sampling techniques and locations used for the 1998 survey duplicated those of the 1985 survey to ensure consistency. However, because exact locations could not be duplicated, the sampling results from the 1998 investigation were compared with the 1985 investigation results primarily to identify any significant differences. Appendix A presents a description of the 1998 confirmatory survey and the survey results.

The results from the 1998 confirmatory survey generally confirmed the findings from the 1985 survey. Gamma scans of outdoor areas showed exposure rates of 5 to 10 μR/hr, which are typical for background levels. The 1998 confirmatory survey examined the Emergency Retention Basin, Water Effluent Monitoring Station, Pentolite Ditch, PBRF grounds, PBRF paved areas, catch basins, Cold Retention Area, Reactor Building outside the reactor containment vessel, Reactor Office and Laboratory Building, Service Equipment Building, Fan House, Waste Handling Building, and the cold service tunnels. The areas were surveyed to measure gross beta activity and direct radiation exposure rates. In addition, soil, sediment, and concrete samples were analyzed for gamma-emitting radionuclides.

In general, the 1998 confirmatory survey confirmed the contaminated and uncontaminated areas identified during the 1985 characterization survey. The 1998 confirmatory survey identified six additional contaminated areas: four laboratories (Rooms 207, 209, 210, and 213A) in the Reactor Office and Laboratory Building; an area of contamination on the -4.6-m (-15-ft) basement level of the Reactor Building; and on the PBRF pavement near the entrance to the Reactor Building. Within the Emergency Retention Basin, the 1998 confirmatory survey identified a high Cs-137 concentration of 200 pCi/g while the 1985 high concentration of Cs-137 was 90 pCi/g. These findings are not expected to impact the degree of remediation required at these areas.

The gamma characterization information from the 1998 survey shows that the dominant gamma sources are Cs-137 and Co-60. Other gamma-emitting nuclides are only small contributors (less than 1 percent). With the exception of a single sample from canal F, gamma activity is dominated by Cs-137 at all PBRF areas (e.g., environmental contamination, sumps, floors in the Reactor Building). In canal F, the activity is dominated by Co-60. Characterization data collected more recently indicates that Co-60 is the dominant contributor in Quadrants A, C, and E, the Lilly Pad area, and the general areas of the CV.

2.2.2.2 Major Facilities at the PBRF

This section summarizes radiological characterization information for the major facilities at the PBRF (identified in Table 1-1).

Reactor Building (Building 1111)

The majority of the radioactivity at the Reactor Building is contained inside the reactor tank. The biological shield and several piping systems are also radioactively contaminated. Radioactivity was detected on the surfaces of the quadrants, canals, and drains. The following paragraphs summarize characterization data for the parts of the Reactor Building.

Reactor Tank and Internal Components

The reactor tank has the highest radionuclide inventory of all the areas at the PBRF. Radionuclide inventory estimates of the reactor tank and its internal components were presented in the 1980 environmental report (NASA 1980a). To calculate the radionuclide inventory of the reactor tank, separate calculations were performed for each of the major components of the core box and beryllium reflector pieces (refer to Appendix A of the 1980 environmental report [NASA 1980a]). Large pieces, such as through tubes, thermal shields, and the reactor tank, were analyzed as several segments. The calculations were built on estimates of integrated neutron exposure, activation cross section for the nuclides of interest in each component, the radioisotope half-life, and the decay time. Table 2-1 identifies the isotope of interest (first column), the June 30, 1978, inventory estimates (second column), and the 1978 inventories decayed to December 31, 2003 (third column). As shown in Table 2-1, H-3 dominates the inventory.

Table 2-1. Estimated Inventory in the Reactor Tank and Internal Components

Nuclide	Inventory (curies) as of 6/30/1978[a]	Inventory (curies) as of 12/31/2003[b]
H-3	156,800	37,266
Co-60	2,640	92
Fe-55	7,340	10.5
Ni-63	45	37
Ni-59	0.5	0.5
Zn-65	115	0.0
Al-26	1.4	1.4
Cd-113m	0.8	0.2
Total	**166,943**	**37,408**

a. From NASA (1980a).
b. Calculated by decaying the 1978 inventory estimates to the year 2003.

Additional characterization information regarding the reactor tank contamination and activation associated with reactor components is presented in the report "Plum Brook Reactor Vessel Activation Analysis and Classification Report", revision 1.

Mock Up Reactor

As part of the Pre-Decommissioning investigation in the summer of 2000, readings were taken around and within the MUR. The highest readings were on the order of 2 mrem/hr, and these were found well inside the structure. No significant loose contamination was found.

Reactor Primary Cooling Water System and Primary Cooling Shutdown System.

Two corrosion film samples from valves in the primary cooling water system were analyzed in 1985. The two samples showed similar levels of activity (256 and 375 dpm/100 cm^2). A gamma pulse height analysis conducted on the sample with higher activity identified the specific nuclides Co-60, Eu-152, Eu-154, and Eu-155. No fission products, such as Cs-137, were identified. Cobalt-60 had the highest activity of the gamma-emitting radionuclides. Except for special equipment (e.g., strainers and some valves), 1985 exposure rates from piping and equipment in this area were less than 30 mR/hr.

Reactor Biological Shield

The biological shield surrounding the reactor tank was activated by neutrons that entered the concrete and interacted with elements. Three core samples were taken from the biological shield in 1985 and analyzed for gross alpha, gross beta, and gamma emitters. The samples were analyzed for europium, but only Co-60 was detected. The average Co-60 concentration in the biological shield within 25-cm (10-in.) of the reactor tank was 17.5 pCi/g. A sample of the reinforcing steel in the concrete was also analyzed for gross alpha, gross beta, and gamma emitting nuclides. Cobalt-60 was detected at a concentration of 325 pCi/g in the reinforcing steel.

Reactor Quadrants and Canals, and Their Pump-out and Recirculation Systems

The 1985 characterization data for the quadrants, canals, and their pump-out recirculation systems included alpha- and beta-gamma radiation measurements of the building wells, direct radiation readings, and collected crud samples. The characterization showed:

- Reliable direct radiation measurements from the canals and quadrants were difficult to obtain because of the radiation field from the reactor tank and biological shield.
- The average concentration of loose alpha contamination, loose beta-gamma contamination, and direct radiation readings in the canals was approximately 2 dpm/100 cm^2, 1000 dpm/100 cm^2, and 0.1 mR/hr, respectively.
- Overall, the pump-out and recirculation system were contaminated internally, but they have little or no external contamination. External dose rates from piping and valves ranged from 0.01 to 0.6 mrem/hr. Drain crud samples contained 0.1 to 1 pCi/g of gross

alpha activity and up to 20,000 pCi/g of gross beta activity. Cobalt-60 was the dominant gamma-emitting radionuclide.

- Direct radiation measurements in the canals ranged from 0.001 to 0.3 mR/hr.
- Deep underground soil samples were collected, and the analytical results verified that the canals (G and K) did not leak contaminated water into the ground.

As part of the 1998 confirmatory survey, a 10-cm (4-in.) diameter concrete core sample approximately 8-cm (3-in.) deep was taken from Canal F, located outside the containment that connects to both the mock-up reactor and the canals going into the Hot Laboratory. Cesium-137 and Co-60 were detected at concentrations of 2.7 pCi/g and 156 pCi/g, respectively.

Reactor Building Rooms

The Reactor Building rooms were surveyed in both 1985 and 1998. Loose and fixed contamination and direct radiation measurements both inside and outside the containment vessel in 1985 showed:

- Inside the containment vessel, loose alpha contamination levels ranged from 0 to 5 dpm/100 cm^2, loose beta-gamma contamination levels ranged from 0 to almost 200 dpm/100 cm^2, and direct radiation readings ranged from 0.006 to a maximum of 500 mR/hr in the sub-pile room. The average direct radiation reading in the other areas ranged from 0.01 to 0.045 mR/hr.
- Outside the containment vessel, loose alpha contamination levels ranged from 0 to 5 dpm/100 cm^2, loose beta-gamma contamination levels ranged from 0 to almost 350 dpm/100 cm^2, and direct radiation readings ranged from 0.005 to 0.230 mR/hr.

The Reactor Building rooms outside the containment vessel were also surveyed during the 1998 confirmatory survey. A total of 105 direct beta measurements and smears were taken along with a single concrete core sample at the -15-ft elevation where a hot spot was identified at the -15 ft level near the east wall (location RB056). One of the 105 beta measurements had a count rate of about 43,000 dpm/100 cm^2. Another measurement had a count rate of about 7000 dpm/100 cm^2. The remaining 103 beta measurements had count rates less than 2000 dpm/100 cm^2, and the average rate was about 100 dpm/100 cm^2.

A 10-cm (4-in.) diameter concrete core sample approximately 8-cm (3-in.) deep was taken at the hot spot (43,000 dpm/100 cm^2). Cobalt-60 and Cs-137 were detected at concentrations of 0.1 pCi/g and 0.2 pCi/g, respectively.

Hot Drains, Sumps, Pumps, and Valves

The 1985 characterization data for the hot drain system included alpha and beta-gamma radiation measurements, direct radiation readings, and collected crud samples. Direct radiation readings from the hot drain system sumps ranged from 0.007 to 2 mR/hr. Ten of the 12 sumps had average readings of 1.2 mR/hr. Crud samples from the hot sumps had elevated alpha and gamma radiation readings, with alpha activity levels ranging from 15 to 9500 pCi/g, and gamma activity levels ranging from 580 to 130,000 pCi/g. The dominant gamma-emitting radionuclides were Co-60 and Cs-137.

Hot Laboratory (1112)

Most of the radioactive contamination in the Hot Laboratory is from stored waste in the Hot Dry Storage Area. Contamination has also been identified in the hot cells and rooms surfaces.

Hot Dry Storage Area

The waste in the Hot Dry Storage Area of the Hot Laboratory has the second highest estimated radionuclide inventory of all the contaminated areas at the PBRF. This waste consists of radioactively contaminated items similar to that in the reactor tank (e.g., beryllium pieces and control rod sections). Estimates of radionuclide inventories in the Hot Dry Storage Area were presented in the 1980 environmental report (NASA 1980a) (and in Teledyne Isotopes 1987). The method for estimating the inventories is discussed in Appendix A of the 1980 environmental report and involves separate calculations for each of the major components. The calculations were built on estimates of integrated neutron exposure, activation cross section for the nuclides in the various components, the half-life of the active isotopes, and the decay time. These inventory estimates, as of June 30, 1978, are presented in the second column of Table 2-2. The 1978 inventories were decayed to December 31, 2003, and these levels are shown in the third column. As shown in Table 2-2, H-3 dominates the inventory.

During the 1985 characterization, thermoluminescent dosimeters (TLDs) were lowered into the Hot Dry Storage Area to obtain dose rate measurements. No smear samples, which indicate surface contamination levels, were taken inside the Hot Dry Storage Area.

Table 2-2. Estimated Radionuclide Inventory of the Waste in the Hot Dry Storage Area

Nuclide[a]	Inventory (curies) as of 6/30/1978[b]	Inventory (curies) as of 12/31/2003[c]
H-3	34,600	8,223
Co-60	16,100	559
Fe-55	14,600	16
Zn-65	1	0.0
Total	**65,301**	**8,798**

a. Other nuclides were calculated to be less than 1 percent of the total.
b. From NASA (1980a).
c. Calculated by decaying the 1978 inventory estimates to the year 2003.

Hot Cells

The seven hot cells in the Hot Laboratory were surveyed in 1985 using instrument scans and wipe samples. Loose alpha contamination in the cells ranged from 0 to 370 dpm/100 cm^2, and loose beta-gamma contamination ranged from 200 to 173,000 dpm/100 cm^2. Direct radiation ranged

from 1 to 450 mR/hr. Isotopic analyses of wipe samples with the highest contamination levels indicated that Co-60 and Cs-137 dominate the measured activity.

Rooms

The rooms in the Hot Laboratory include the decontamination room, repair shop, storage room, mezzanine, cold work area, hot work area, and hot handling area. The floors, walls, and ceilings of the rooms were surveyed in 1985 using instrument scans and wipe samples. The 1985 characterization data show that contamination levels in the Hot Laboratory rooms, exclusive of the decontamination room, were similar to those in the Reactor Building rooms outside of the containment vessel. For areas other than the decontamination room, the loose alpha contamination ranged from 0 to 8 dpm/100 cm^2 and loose beta-gamma contamination ranged from 0 to 18,852 dpm/100 cm^2. Direct radiation levels in these same areas ranged from 0.003 to 1 mR/hr. The decontamination room had loose alpha contamination as high as 208 dpm/100 cm^2, loose beta-gamma contamination as high as 337,000 dpm/100 cm^2, and dose rates as high as 8 R/hr.

2.2.2.3 Support Facilities at the PBRF

Radiological characterization information for the contaminated support facilities at the PBRF (identified in Table 1-1 and described in Section 1.2) are briefly discussed in the following paragraphs. The support facilities are smaller and have lower levels of contamination than the major facilities described in Section 2.2.2.2. The contamination generally is in readily removable equipment or in areas that are more simply decontaminated. The structures themselves have limited contamination. A summary of characterization information for the contaminated support facilities is presented in Table 2-3. More complete information may be found in the 1985 Teledyne Characterization Survey and the 1998 GTS Duratek Confirmation Study.

Table 2-3. Summary of Survey Results for Support Facilities at the PBRF

Building/ Structure	Summary of 1985 Characterization Survey Results	1998 Confirmatory Survey	
		No. of Survey Measurements	Results
Reactor Office and Laboratory Building (1141)	• Loose alpha contamination ranging from 0 to 4 dpm/100 cm^2 • Loose gamma-beta contamination ranging from 0 to 137 dpm/100 cm^2 • Average direct radiation less than 0.02 mR/hr	• 120 direct beta measurements • 120 smears	• Two measurements were about 50,000 dpm/100 cm^2 • Three measurements were between 5000 and 10,000 dpm/100 cm^2 • All others were less than 2000 dpm/100 cm^2
Primary Pump House (1134)	• Loose alpha contamination ranging from 0 to 2 dpm/100 cm^2 • Loose gamma-beta contamination ranging from 0 to 29 dpm/100 cm^2 • Direct radiation about 0.01 mR/hr	None	None
Fan House (1132)	• Loose alpha contamination ranging from 0 to 2 dpm/100 cm^2 • Loose gamma-beta contamination ranging from 0 to 102 dpm/100 cm^2 • Direct radiation less than 1 mR/hr	• 60 direct beta measurements • 60 smears	• One measurement was about 7000 dpm/100 cm^2 • All others were less than 2500 dpm/100 cm^2
Waste Handling Building (1133)	• Loose alpha contamination ranging from 0 to 5 dpm/100 cm^2 • Loose gamma-beta contamination ranging from 0 to 11797 dpm/100 cm^2 (the highest value is in the basement; the next highest value is 2000 dpm/100 cm^2) • Direct radiation ranges from 0.02 to than 3 mR/hr	• 60 direct beta measurements • 60 smears	• One measurement was about 7000 dpm/100 cm^2 • Most others were less than 2500 dpm/100 cm^2
Hot Retention Area (1155)	• Tanks are contaminated; concrete vault contamination was less than the levels in Regulatory Guide 1.86 (USAEC 1974) • Direct radiation ranged from 0.044 to 2.8 mR/hr	None	None
Cold Retention Basins (1154)	• Alpha contamination ranged from 0 to 3 dpm/100 cm^2 • Beta contamination ranged from 25 to 1061 dpm/100 cm^2 • Direct radiation less than 0.1 mR/hr	• 8 direct beta measurements • 8 smears	Wipe samples range from 1000 to 5000 dpm/100 cm^2
Hot pipe tunnel	• Activity primarily in the 4-in. polyethylene piping. Contact dose rates range from 6 to 2200 mR/hr • Loose alpha contamination ranged from 0 to 17 dpm/100 cm^2 • Loose beta-gamma contamination ranged from 0 to 47,363 dpm/100 cm^2 with a hot spot from line leak • Direct radiation ranged from 2 to 85 mR/hr	None	None

The highest contamination levels found in the support facilities during the 1985 survey were in the hot pipe tunnels (shown in Figure 1-2). The piping in the tunnel, which was used to handle radioactive liquid and gasses, contains radioactive contamination, and the tunnel floor is radioactively contaminated in one area.

The next highest contamination levels were in an evaporator in the basement of the Waste Handling Building (1133). Other equipment and piping in this building contain radioactive contamination, and surface contamination has been identified throughout the building. In the Fan House (1132), equipment (e.g., ducts and piping) contains measurable radioactive contamination, and contamination has been identified throughout the basement floor. In the Reactor Office and Laboratory Building (1141), radioactive contamination has been found on laboratory hoods, in piping, and on the floors of some of the radiochemistry laboratories. In the Primary Pump House (1134), equipment and piping, as well as pits and sumps, contain radioactive contamination.

At the Hot Retention Area (1155), the storage tanks and associated piping and equipment are radioactively contaminated, and low levels of contamination (i.e., less than the levels in Regulatory Guide 1.86, according to Teledyne Isotopes 1987) have been identified in the concrete vault. At the Cold Retention Basins (1154), the basin liners, concrete structures, and the silt deposits on the liners are radioactively contaminated. Underground soil samples collected in 1985 verified that the Hot Retention Area and Cold Retention Basins did not leak contaminated water into the ground.

The areas examined in the 1998 survey generally confirmed the results. For the Fan House, Waste Handling Building, and Reactor Office and Laboratory Building, the 1998 results are consistent with the 1985 results. In general, the more extensive 1985 survey and the 1998 verification survey showed that there was only localized contamination in the support structures.

2.2.2.4 Environmental Contamination at the PBRF

Areas of environmental contamination include (1) inground or earthen structures or (2) soil that was contaminated from past operations or non-routine occurrences (e.g., spills) (see Table 1-1). Radiological characterization information for these areas is summarized in the following paragraphs.

Emergency Retention Basin

Surface soil in the Emergency Retention Basin (i.e., from 0-cm to 15-cm [0-in to 6-in] below the surface) and soil from 15-cm to 30-cm (6-in to 12-in) below the surface in specific areas is radioactively contaminated.

The 1985 characterization of the Emergency Retention Basin included collecting shallow (0-m to 3.0-m [0-ft to 10-ft]) cores, near-surface (5-cm to 15-cm [2-in to 6-in]) soil samples, and surface (0-cm to 5-cm [0-in to 2-in]) soil samples. The shallow cores were analyzed for gross alpha and gross beta activity and the results indicated that the residual activity was confined to the upper 15-cm (6-in) of soil. Near-surface soil samples collected from the Emergency Retention Basin indicated that gross beta activity averaged 78 pCi/g. Surface soil samples collected at locations where the near-surface samples showed the highest activity levels were also analyzed for

gross beta activity. Radionuclide concentrations in the surface soil samples were 10 to 20 times greater than that in the near-surface samples.

The near-surface samples having the highest activity also were analyzed to determine the isotopic distribution. The average Co-60, Cs-137, and Sr-90 concentrations in the near surface samples were 22, 32, and 2.4 pCi/g, respectively.

During the 1998 confirmatory survey, a gamma scan was conducted (about 1.3-cm [0.5-in.] from the surface) and five soil samples were collected. The gamma scan showed peak exposure rates of about 50 μR/hr, with average exposure rates ranging from 20 to 30 μR/hr. These exposure rates are generally similar, but they are slightly less than those reported in the 1985 survey. The soil samples taken in 1985 were from the southern portion of the Emergency Retention Basin (the most contaminated area in the 1985 survey). The decay-adjusted 1985 concentrations and the 1998 concentrations are within a factor of 3 of each other. The differences could be due to the different sample locations and the contamination not being homogenous. The lower concentrations at the 0 to 5-cm (0 to 2-in.) depth and the higher concentrations at the 5-cm to 15-cm (2-in. to 6-in.) depth may indicate downward contaminant migration.

Drainage System

The drainage system consists of a series of open ditches, covered culverts, and catch basins (ditches and culverts are shown as dotted lines on Figure 1-2). Underground piping and silt deposits in the catch basins are radioactively contaminated.

The 1985 characterization effort reported that accumulated silt in the catch basins had gross beta activity ranging from 7 to 330 pCi/g, with an average of 44 pCi/g. Depths and areas of contamination were not reported.

The catch basins were reexamined in the 1998 confirmatory survey. The beta survey showed that one sample had a maximum concentration of 5000 dpm/100 cm^2, and the remaining samples had an average concentration of less than 1200 dpm/100 cm^2. The 1998 gross beta activity measurements are on the order of 15 to 20 pCi/g, similar to the average 1985 measurements (44 pCi/g). The 1998 sampling effort also showed that the activity in the catch basins is predominantly naturally occurring K-40, at concentrations ranging from 7 to 14 pCi/g. The concentration of Cs-137 and Co-60 ranged from 1 to 11 pCi/g and from 1 to 5 pCi/g, respectively.

Water Effluent Monitoring Station (Building 1192)

The Water Effluent Monitoring Station (WEMS) includes a metal building and a concrete trench with metal gates and flumes. The trench itself, silt entrapped behind the flumes, and an area of soil adjacent to the trench are radioactively contaminated.

The 1985 characterization survey measured contamination in the WEMS building and in silt in the WEMS trench. The 1998 confirmatory survey also measured concrete surfaces of the building and found contamination levels consistent with those measured in 1985. Isotopic analysis of gamma emitters in the 1998 survey (excluding naturally occurring gamma emitters) indicated the dominant nuclides were Cs-137 (4 to 11 pCi/g) and Co-60 (1 to 4 pCi/g)

Table 2-4 compares the 1985 and 1998 survey results for the Water Effluent Monitoring Station building.

Table 2-4. Summary of Survey Results for the Water Effluent Monitoring Station (1192)

Building/ Structure	Summary of 1985 Characterization Survey Results	1998 Confirmatory Survey	
		No. of Survey Measurements	Results
Water Effluent Monitoring Station (1192)	• Loose alpha contamination ranging from 0 to 2 dpm/100 cm^2 • Loose beta-gamma contamination ranging from 0 to 48 dpm/100 cm^2 • Direct radiation levels ranging from 0.004 to 0.04 mR/hr	• 8 direct beta measurements • 8 smears	• Three measurements were about 15,000 dpm/100 cm^2 • All others were less than 5000 dpm/100 cm^2

Pentolite Ditch

The Pentolite Ditch received all water from the Water Effluent Monitoring Station. Up to 30-cm (12-in.) of silt and soil in some areas along the Pentolite Ditch are radioactively contaminated. The contamination occurs primarily at the western end (near the Water Effluent Monitoring Station outfall), with a smaller amount near the eastern end (near the confluence with Plum Brook).

For the 1985 characterization, the Pentolite Ditch was divided into 9.1-m × 9.1-m (30-ft × 30-ft) grids. A contact beta-gamma survey was performed at the center and four surrounding points in each grid. A silt sample was then collected at the center point and a soil sample was collected at the surrounding point that had the highest contamination level. The survey results indicated that portions of the ditch nearest the Emergency Retention Basin (i.e., the west end) and nearest Plum Brook (i.e., the east end) were contaminated with higher levels of contamination than in the other portions of the ditch. Samples from four shallow (3-m [10-ft]) cores indicated that contamination was confined to depths less than 15-cm (6-in).

Sampling indicated that soil from the bottom and the banks of the Pentolite Ditch had average gross beta activities of 40 and 110 pCi/g, respectively.

During the 1998 confirmatory survey, eight sediment samples were collected along the Pentolite Ditch. The analytical results showed that the total activity in the samples ranged from 10 to 30 pCi/g. Most of the activity is from natural K-40; the residual activity from Cs-137 ranged from 2 to 15 pCi/g and from Co-60 from 0 to 1 pCi/g. The 1998 average concentration (about 20 pCi/g) is

lower than that measured in 1985 (75 pCi/g). This decrease could be due to several factors, including decay, fewer sample locations, and irregular distribution of the contamination.

The presence of low concentration, man-made radionuclides in Pentolite Ditch are believed to be the result of effluent releases during PBRF operations. Sample results indicate that the external exposure pathway for the radionuclides in soil and sediment is controlling dose (and DCGL calculation) at PBRF outdoor locations. Sensitivity analyses indicate that an additional contaminated area such as Pentolite Ditch (6700-m^2) will not impact the DCGL's in the Decommissioning Plan, given the large area (102,400-m^2) of the PBRF site used to derive the project's DCGL's. Soil and/or sediment locations associated with Pentolite Ditch that contain concentrations above the DCGL's will be remediated. An environmental monitoring Lower Limit of Detection (LLD) of 1 pCi/g for Cs-137 in sediment and soil from Pentolite Ditch will ensure detection of radionuclide contamination well below the DCGL's presented in the Decommissioning Plan.

Plum Brook

No man-made radionuclides have been identified in Plum Brook above 1 pCi/g. An environmental monitoring Lower Limit of Detection (LLD) of 1 pCi/g for Cs-137 in sediment and soil from Pentolite Ditch will ensure detection of radionuclide contamination well below the DCGL's presented in the Decommissioning Plan.

Areas of Contaminated Pavement and Soil

Two areas of known low-level waste spills have been identified: one near the Waste Handling Building (1133) concrete pad and one in the vicinity of the Primary Pump House (1134) resin pits (see Figure 1-2). The 1985 characterization effort involved collecting deep and shallow cores near the concrete pad at the Waste Handling Building. Samples from the cores showed radiological contamination to a depth of 1.8-m (6-ft). At the same location, gross beta activity measurements were 1500 pCi/g at a depth of 0.3-m (1-ft) and 100 pCi/g at a depth of 1.8-m (6-ft). Gross alpha activity measurements at the same depths were 90 and 7 pCi/g, respectively. No radiological concentration was reported for the second spill area in the vicinity of the Primary Pump House (1134) resin pits. The 1998 survey confirmed the presence of contamination near the Waste Handling Building, but no contamination was detected at the previously identified spill area near the Primary Pump House.
During the 1998 survey, an additional contaminated location was identified on the pavement near the entrance to the Reactor Building, where total beta activity up to 42,000 dpm/100 cm^2 was measured.

Facilities Expected to be Clean

Based on the 1985 and 1998 characterization information, several support facilities within the PBRF fence were determined to be uncontaminated (non-impacted areas). These facilities are:

- Reactor Service Equipment Building (1131)

- Reactor Gas Services Building (1135)
- Reactor Compressor Building (1136)
- Reactor Substation (1161)
- Reactor Security Building (1191).

Based on historical knowledge, at the time of the 1985 characterization survey, the following facilities were considered to be uncontaminated and were not surveyed. This assessment was not revisited as part of the 1998 confirmatory survey:

- Cold pipe tunnel
- Reactor water tower (1151)
- Reactor sludge basins (1153)
- Reactor precipitator (1157)
- Reactor Cryogenic and Gas Supply Farm and Building (1195 & 9837)
- Reactor Gas Storage Structure (1196).

These facilities will be surveyed as part of the ongoing characterization effort and, in some cases, in the final status survey described in Section 4.

2.2.2.5 Non-Radiological Waste Characterization of the PBRF

Asbestos and asbestos/fiberglass insulation has been identified at the PBRF. This asbestos material is on various pipes, tanks, vents, etc. Some of this material is externally contaminated and internally clean, and some is both externally and internally contaminated. The asbestos material is located in the Reactor Building (1111), Hot Laboratory (1112), Fan House (1132), Waste Handling Building (1133), and Reactor Office and Laboratory Building (1141).

Lead paint is also located throughout PBRF buildings, primarily on walls and ceilings. Lead paint is located in the same buildings as the asbestos material.

Small quantities of other non-radioactive waste, such as mercury in switches and light ballasts, and PCB containing oil are expected to be generated and will be identified and properly managed during the early phases of decommissioning.

2.2.3 Release Criteria

Consistent with 10 CFR Part 20, decommissioning means reducing residual radioactivity to a level that permits termination of the license and release of the site for unrestricted use. The PBRF license would be terminated after NASA demonstrates that the site meets the criteria for decommissioning specified in 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination." The radiological criteria for unrestricted use are identified in 10 CFR 20.1402, which specifies two criteria: (1) the TEDE from residual radioactivity that is distinguishable from background radiation must not be greater than 25 mrem/yr to the AMCG and (2) residual radioactivity levels must be ALARA. This section describes methods for dose assessment, describes methods to demonstrate that levels of residual contamination are ALARA, and presents

results that will be used in the decision framework applied to PBRF decommissioning. Results include derived concentration guidelines (DCGLs) and cost-benefit relationships for specific decommissioning activities.

Section 2.2.3.1 presents the proposed methodology for establishing the residual contamination levels that would result in a TEDE to the AMCG that is less than 25 mrem/yr. These levels, which are expressed as radionuclide concentrations, are referred to as DCGLs in Draft Regulatory Guide DG-4006, "Demonstrating Compliance With the Radiological Criteria for License Termination" (NRC 1998a). As recommended in Draft Regulatory Guide DG-4006, this section presents the methodology to obtain the NRC's approval before remediating the site and conducting the final status survey. Section 2.2.3.1 presents the estimated DCGLs for surface soils, building surfaces, and subsurface structures at the PBRF. The DCGLs will be used during the final status survey to demonstrate that the residual radioactivity at the site will result in a TEDE to the AMCG of less than 25 mrem/yr. According to NUREG-1549, "Using Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination" (NRC 1998b), a licensee can demonstrate compliance with the dose criterion either by using a generic screening model or by using site-specific analyses. Site-specific analyses have been applied to develop DCGLs for PBRF, and generic screening values are presented to provide perspective.

Section 2.2.3.2 presents the proposed methodology for demonstrating that residual contamination levels are ALARA. The methodology follows the guidance in Draft Regulatory Guide DG-4006 (NRC 1998a) and involves comparing the costs and benefits of postulated decommissioning actions. Options for using DCGLs for specific portions of the PBRF and the timing of the final status survey are described in Section 2.3.1. Section 2.3.1 also identifies the criteria that will be used for selecting DCGLs for specific portions of the PBRF.

Preliminary ALARA analysis indicates that complying with the criteria of a TEDE to the AMCG of 25 mrem/yr also achieves the criterion that residual radioactivity is ALARA.

2.2.3.1 Derived Concentration Guidelines

This section presents (1) the methods used to calculate DCGLs (the level of residual contamination that would produce a TEDE of 25 mrem/yr to the AMCG) and (2) the results of dose assessments for the PBRF to show the rate of dose decrease over time. DCGLs were estimated using existing characterization data. A characterization survey of the PBRF was completed in 1985 and confirmed in 1998 as described in Teledyne Isotopes (1987) and Appendix A of this plan, respectively. On the basis of these surveys, residual radioactivity at PBRF has been categorized as surface soil, building surface, or subsurface structure residual contamination. The subsurface structures are primarily the biological shield, canals and quadrants, and embedded piping located in the Reactor Building (Building 1111) and basement areas of the remaining structures. The definitions and locations of these types of residual contamination are summarized in Table 2-5.

Table 2-5. Types of Residual Contamination at the PBRF

Residual Contamination Type	Definition	PBRF Site Areas*
Surface soil	Residual contamination of soil within 15cm (0.5 ft) of the surface that could result in a dose to a resident /agricultural intruder.	• Emergency Retention Basin • Pentolite Ditch • Spill area adjacent to Waste Handling Building (1133) concrete pad • Clean rubble used as fill • Cold Retention Basins (1154)
Building surfaces	Fixed and removable contamination on building floors, walls, or ceilings that could result in a dose to building reuser.	• Reactor Office and Laboratory Building (1141) • Reactor Building (1111) • Hot Laboratory (1112) • Waste Handling Building (1133) • Fan House (1132) • Primary Pump House (1134)
Subsurface structures and building debris	Residual contamination associated with below grade materials that could result in a dose to resident/agricultural intruder.	• Remaining subsurface structures and building debris used as backfill

* Numbers in parentheses are the building numbers (refer to Figure 1-2 for locations).

The characterization surveys indicated that surface soils having residual contamination are present at the Emergency Retention Basin, Pentolite Ditch, Waste Handling Building (Building 1133) concrete pad, and at the entrance to the Reactor Building (1111). Isotopic analysis indicated the radionuclides of concern for surface soil were Co-60, Sr-90, and Cs-137. Low levels of contamination were identified in rooms located on the second floor of the Reactor Office and Laboratory Building (Building 1141), while higher levels were identified in the Reactor Building (Building 1111) subsurface structures. Estimates of residual contamination levels have been developed based on available PBRF characterization data and on measurements at similar facilities (Abel et al. 1986; Smith et al. 1978). Based on these data, the radionuclides of concern for the PBRF are listed in Table 2-6.

Table 2-6. Radionuclides of Concern for the PBRF

Radionuclide	Surface Soil	Building Surfaces	Subsurface Structures
H-3		✓	
C-14			✓*
Fe-55		✓	✓
Co-60	✓	✓	✓
Ni-59			✓
Ni-63		✓	✓
Sr-90	✓	✓	✓
Tc-99			✓*
Cs-137	✓	✓	✓
Eu-152			✓
Eu-154			✓

* Indicates expected nuclides from similar facilities (Abel et al. 1986; Smith et al. 1978).

DCGL estimates are based on the analysis of scenarios that could reasonably occur if a site is released for unrestricted use. A scenario is defined as a set of release modes, receptor metabolic and behavioral characteristics, environmental transport pathways, and exposure modes that result in dose to an individual or population. The dose analysis performed for this decommissioning plan assumes the PBRF site is released for unrestricted use and evaluates the case of members of the public using the site. In actuality, NASA has no plans to sell property after license termination, so realistic receptors would be members of the public located offsite and NASA employees working onsite after license termination. Thus, unrestricted use of the PBRF site is a conservative scenario and bounds the realistically expected impacts.

NRC has published guidance on methods for dose analysis supporting license termination under 10 CFR Part 20 (NUREG/CR-5512 [Kennedy and Strenge 1992], Draft Regulatory Guide 4006 [NRC 1998a] and NUREG-1549 [NRC 1998b]). The guidance allows using either generic screening or site-specific dose assessment in the decision framework. Under the generic screening approach, NRC has identified pathways, scenarios, models, and model parameter values and has provided analysis results in the form of levels of contamination consistent with the 10 CFR Part 20, Subpart E, dose criteria. The pathways and scenarios constitute the resident farmer and building reuse scenarios. The NRC generic screening analysis of the resident farmer scenario is based on the assumption that all contamination has been distributed into the upper 15 cm (6 in.) of soil. Among the options presented for site-specific analysis is the use of site-specific parameter values and existing models other than the generic NRC model. The guidance recommends that the licensee provide information supporting use of site-specific data or models other than the generic NRC model. Because the generic screening model addresses subsurface contamination through assumed redistribution to surface soil, it is very conservative for the case of residual contamination of subsurface structures. To provide a consistent level of analysis for the contamination of surface soil, building surface, and subsurface structures, a site-specific analysis approach was used for the PBRF.

The approach adopted for the PBRF dose assessment was to use a dose model other than the generic NRC model and to use site-specific data where available. The dose model selected for analyzing residual soil contamination, RESRAD Version 6.0 (Yu et al. 2000), has been formally accepted by the NRC for analysis of resident farmer scenarios. The dose model selected for analyzing residual building surface contamination, RESRAD-BUILD Version 3.0 [Yu et al. 2000), addresses pathways discussed in NRC guidance and is widely used by the U.S. Department of Energy (DOE) and the U.S. Department of Defense when analyzing building reuse scenarios. These two site-specific models include all pathways and exposure modes included in the NRC generic screening models. No conditions outside those incorporated in the site-specific models are expected to occur at the PBRF. The resident farmer scenario presumes that both the residence and garden are located on contaminated soil. Thus, the site-specific modeling is appropriate for assessing doses because of contamination of soil and building surfaces.

The RESRAD Version 6.0 model was also selected for analyzing residual subsurface structure contamination to allow a sensitivity analysis to be conducted for the concrete K_d values. No conditions outside those incorporated in the site-specific model are expected to occur at the PBRF, and no pathways have been eliminated. Thus, the site-specific model is appropriate for assessing dose because of residual contamination associated with subsurface structures.

Using existing characterization information (Teledyne Isotopes [1987] and Appendix A of this plan), site-specific pathway scenarios were used to calculate DCGLs and to develop estimates of dose over time for the AMCG at the PBRF. The generic DCGLs that have been developed in draft form in NUREG-1549 (NRC 1998b) are presented to provide perspective on the site-specific DCGLs. The site-specific DCGLs were developed by considering PBRF soils and hydrology.

Residual Contamination in Surface Soils

NRC regulatory guidance (NRC 1998a) recommends analysis of a resident farmer scenario as the basis for the DCGLs for residual contamination in site-wide surface soil. In the resident farmer scenario, an individual could contact residual contamination by establishing a home and garden on contaminated soil or by using groundwater that comes in contact with the residual contamination. The primary release modes are partitioning of contaminants from soil into infiltrating water and resuspension by wind. The environmental transport pathways include ground water transport; translocation into plants, animals, and fish; and atmospheric dispersion. Exposure modes include ingestion of water, crops, animal products, and fish; direct external exposure from the ground; inhalation of airborne material; and inadvertent ingestion of soil. Because uranium contamination is not expected based on historical knowledge or survey measurements, the radon exposure pathway is not included in the calculation of DCGLs for residual contamination of surface soils.

NUREG-1549 (NRC 1998b) identifies DCGLs (i.e., soil concentrations that would result in a TEDE of 25 mrem/yr to the AMCG) for a generic, screening-level exposure scenario. More realistic site-specific DCGLs for the PBRF were developed using RESRAD Version 6.0 (Yu et al. 2000). The site-specific DCGLs were calculated based on many variables that characterize the receptors, environmental pathways, and modes of exposure. The estimates of physical, behavioral, and metabolic parameter values were developed from either site measurements or literature review. Available site-specific characterization data include meteorological and hydrogeological data, soil type characterization, and location and extent of

contamination specifications. Thus, site-specific data for annual precipitation, saturated zone hydraulic conductivity and gradient, and thickness of the unsaturated zone were used in the RESRAD analyses. The average annual precipitation, saturated zone hydraulic conductivity, and saturated zone hydraulic gradient measurements are 0.86-m/yr (34-in/yr), 1070-m/yr (9.6-ft/day), and 0.0045-m/m (feet/foot), respectively (IT Corporation 1997). The ground level and water table elevations near the PBRF are approximately 192 m (629-ft) above mean sea level (MSL) and 189-m (620-ft) above MSL, respectively, for an unsaturated zone thickness of approximately 3-m (10-ft) (IT Corporation 1997).

Based on the description of site soils, site-specific hydrologic parameters were selected for the contaminated, unsaturated, and saturated zones. The description of the Arkport-Galen soils (IT Corporation 1997) that form the surficial soils and unsaturated soils and the field hydraulic conductivity measurements ranging from 1335 to 2670-m/yr (12 to 24-ft/day) are consistent with the characteristics of loamy sand (Beyeler et al. 1998a). For loamy sand, the NRC-recommended porosity and hydraulic conductivity values are 0.41 and 1262-m/yr (12-ft/day), respectively (Beyeler et al. 1998a), and these values were assumed for both the contaminated and unsaturated zones. For the saturated zone, the measured hydraulic conductivity of 1070 m/yr (9.6 ft/day) (IT Corporation 1997) is consistent with silt loam, and a porosity of 0.45 for the saturated zone is consistent with NRC guidance (Beyeler et al. 1998a). These values were assumed for the saturated soil. The site-specific data, including thickness and extent of the contaminated zone, hydraulic conductivity, soil types, and precipitation rate, are among the more dose-sensitive parameters. Using saturated zone hydraulic conductivity at the upper end of the range for silt loam is prudently conservative because it is consistent with the observed soil type and minimizes travel time to exposure points. Using a water table drop rate of 0-m/yr is also prudently conservative because it minimizes travel time through the unsaturated zone for the observed thickness of the zone. Using observed values for other site-specific parameters is reasonable because it is consistent with existing conditions and does not introduce a judgment bias that may be conservative or non-conservative depending on the intricacies of pathway analysis for individual radionuclides.

All the other parameters used in the dose analysis were generic screening values. These parameters were estimated based on NRC and DOE guidance for generic screening (i.e., the NUREG-1549 analysis [NRC 1998b; Beyeler et al. 1998a, 1998b] and the RESRAD computer code [Yu et al. 1993]) and are considered to be prudently conservative. Wherever possible, NRC-recommended parameter values were used unless site-specific data were available. In the absence of both site-specific and NRC-recommended values, RESRAD default values were used. The parameter values used in the analysis of the PBRF resident farmer scenario are presented in Tables 2-7 through 2-14. The generic values used for the most dose-sensitive parameter (i.e., the distribution coefficient), are relatively high. This results in retaining radionuclides in soil rather than removing them by groundwater, which produces conservative dose estimates through the external exposure pathway for the radionuclides controlling dose at the PBRF. As shown in Table 2-7, the radiation dose limit and time for calculations are 25 mrem/yr and 1000-years, respectively, as specified in 10 CFR 20.1401 and 20.1402.

Table 2-7. Resident Farmer Scenario: Contaminated Zone Parameters

Parameter	Parameter Value	Source
Area of contaminated zone	102400 m^2 for controlled area including: • 8128 m^2 for Emergency Retention Basin • 6700 m^2 for Pentolite Ditch • 58 m^2 for Waste Handling Building (1133) concrete pad spill area	Teledyne Isotopes (1987)
Thickness of contaminated zone	• 0.15 m average for site	Teledyne Isotopes (1987)
Length parallel to aquifer flow	365 m	Teledyne Isotopes (1987)
Radiation dose limit	25 mrem/yr	10 CFR 20.1402
Time since placement of material	0 years	Site specific
Time for calculations	Through 1000 years	10 CFR 20.1401

The parameters identified in Tables 2-7 through 2-14 were used in the RESRAD code to determine the corresponding radionuclides concentrations in soil (i.e., DCGLs). Table 2-15 presents the DCGLs for various radionuclides that would result in an annual TEDE of 25 mrem to a resident farmer. The first column of Table 2-15 identifies the radionuclide, the second column presents the DCGLs calculated by RESRAD using selected site-specific parameters, and, to provide perspective, the third column presents the 95th percentile concentrations from NUREG-1549 (NRC 1998b). Because none of the other area-specific parameter values (i.e., total area, contaminated zone thickness, and length parallel to flow) dominated the dose estimates, a single set of site-specific DCGLs is applicable for the surface soil, buildings, and subsurface structures.

Table 2-8. Resident Farmer Scenario: Cover and Contaminated Zone Hydrologic Data

Parameter	Parameter Value	Source
Density of contaminated zone	1.56 g/cm^3	NUREG-1549[a,b]
Contaminated zone erosion rate	0.001 m/yr	RESRAD[c]
Contaminated zone total porosity	0.41	NUREG-1549[a]
Contaminated zone effective porosity	0.2	RESRAD
Contaminated zone hydraulic conductivity	1262 m/yr	NUREG-1549[a]
Contaminated zone b parameter	1.4	NUREG-1549[a]
Evapotranspiration coefficient	0.5	RESRAD
Precipitation	0.86 m/yr	Site specific
Irrigation	1.04 m/yr	NUREG-1549
Irrigation mode	Overhead	RESRAD
Runoff coefficient	0.2	RESRAD
Watershed area for stream or pond	1×10^6 m^2	RESRAD

a. Value for loamy sand (based on site description).
b. NUREG-1549 (NRC 1998b).
c. RESRAD (Yu et al. 2000).

Table 2-9. Resident Farmer Scenario: Saturated Zone Hydrologic Data

Parameter	Parameter Value	Source
Density of saturated zone	1.46 g/cm^3	NUREG-1549[a,b]
Saturated zone total porosity	0.45	NUREG-1549[a]
Saturated zone effective porosity	0.2	RESRAD[c]
Saturated zone hydraulic conductivity	1070 m/yr	Site specific
Saturated zone hydraulic gradient	0.0045 m/m	Site specific
Saturated zone b parameter	3.8	NUREG-1549[a,b]
Water table drop rate	0.0 m/yr	Site specific
Well pump intake depth	2.01 (m below water table)	Site specific
Mixing model	Nondispersion	RESRAD[c]
Individual use of groundwater	118 m^3/yr	NUREG-1549

a. Value is for silt loam (based on comparison with well-test hydraulic conductivity).
b. NUREG-1549 (NRC 1998b).
c. RESRAD (Yu et al. 2000).

Table 2-10. Resident Farmer Scenario: Uncontaminated and Unsaturated Zone Hydrologic Data

Parameter	Parameter Value	Source
Number of unsaturated zone strata	1	Site specific
Unsaturated zone thickness	3.0 m	Site specific
Unsaturated zone soil density	1.56 g/cm^3	NUREG-1549[a,b]
Unsaturated zone total porosity	0.41	NUREG-1549[a]
Unsaturated zone effective porosity	0.2	RESRAD[c]
Unsaturated zone b parameter	1.4	NUREG-1549[a]
Unsaturated zone hydraulic conductivity	1262 m/yr	NUREG-1549[a]

a. Value for loamy sand (based on site description).
b. NUREG-1549 (NRC 1998b).
c. RESRAD (Yu et al. 2000).

Table 2-11. Resident Farmer Scenario: Distribution Coefficients

Element	Parameter Value* (mL/g)
C	21
Fe	891
Co	1,000
Ni	37
Sr	32
Tc	7
Cs	447
Eu	955

* Source: NUREG-1549 (NRC 1998b).

Table 2-12. Resident Farmer Scenario: Dust Inhalation and External Gamma Parameters

Parameter	Parameter Value	Source
Inhalation rate	8400 m^3/yr	NUREG-1549[a]
Mass loading for inhalation	6×10^{-6} g/m^3	NUREG-1549[b]
Dilution length for airborne dust	3 m	RESRAD[c]
Exposure duration	365.25 days	NUREG-1549
Shielding factor, inhalation	0.40	RESRAD
Shielding factor, external gamma	0.47	NUREG-1549[d]
Fraction of time indoors, onsite	0.66	NUREG-1549
Fraction of time outdoors, onsite	0.11	NUREG-1549
Shape factor, external gamma	1	RESRAD
Fraction of annular areas	0	RESRAD

a. NUREG-1549 (NRC 1998b).
b. Activity and time average of NUREG-1549 values.
c. RESRAD (Yu et al. 2000).
d. Sum of the product of the means for the fraction of time and shielding factor for outdoor and indoor exposure.

Table 2-13. Resident Farmer Scenario: Ingestion Pathway, Data Dietary Parameters

Parameter	Parameter Value	Source
Fruit, vegetable, and grain consumption rate	78 kg/yr	NUREG-1549[a,b]
Leafy vegetable consumption rate	15 kg/yr	NUREG-1549
Milk consumption	118 L/yr	NUREG-1549
Meat and poultry consumption	52 kg/yr	NUREG-1549[c]
Fish consumption	16 kg/yr	NUREG-1549
Soil ingestion rate	18.3 g/yr	NUREG-1549
Drinking water intake	478 L/yr	NUREG-1549
Fraction of drinking water from site	1	RESRAD[d]
Fraction of aquatic load from site	0.5	RESRAD

a. Sum of individual means for other vegetables, fruit, and grain.
b. NUREG-1549 (NRC 1998b).
c. Sum of individual means for meat and poultry.
d. RESRAD (Yu et al. 2000).

Table 2-14. Resident Farmer Scenario: Ingestion Pathway Data, Nondietary Parameters

Parameter	Parameter Value	Source
Livestock fodder intake for meat	8.5 kg/day	NUREG-1549[a]
Livestock fodder intake for milk	17 kg/day	NUREG-1549[b]
Livestock water intake for meat	50 L/day	RESRAD[c]
Livestock water intake for milk	160 L/day	RESRAD
Livestock soil intake	0.5 kg/day	RESRAD
Mass loading for foliar deposition	4×10^{-4} g/m^3	NUREG-1549[d]
Depth of soil mixing layer	0.15 m	NUREG-1549
Depth of roots	0.9 m	RESRAD
Drinking water fraction from groundwater	1	RESRAD
Livestock water fraction from groundwater	1	NUREG-1549
Irrigation fraction from groundwater	1	RESRAD

a. NUREG-1549 (NRC 1998b).
b. Sum of individual medians for forage, hay, and grain.
c. RESRAD (Yu et al. 2000).
d. Value for gardening.

Table 2-15. DCGLs for Surface Soils

Radionuclide	Site-Specific DCGL[a] (pCi/g)	Generic Screening DCGL[b] (pCi/g)
Co-60	4.6	3.7
Sr-90	32	1.2
Cs-137	18	9.8

a. Calculated by RESRAD using the parameters specified in Tables 2-7 through 2-14.
b. Source: NUREG-1549 (NRC 1998b), provided for perspective.

The PBRF radionuclides of concern, DCGLs for each, where these radionuclides are present, and the ratio of the radionuclide mix will be evaluated as more samples from the current characterization are collected, analyzed, and evaluated.

Table 2-15 shows that different nuclides have different DCGLs and that the site-specific DCGLs are greater (i.e., less restrictive) than the generic screening concentration levels. Cobalt-60 has the lowest concentration limit of the site-specific DCGLs, which means cobalt presents the greatest hazard to the resident farmer per curie.

In addition to providing a numerical value for the DCGLs, dose assessment methods were used to investigate the time dependence of dose. Perspective on the evolution of dose over time can be developed if the isotopic distribution of radionuclides is known. The time dependence of dose for a time period of 1000 years is presented in Figure 2-1. The peak dose, occurring in the first year after release of the PBRF, includes contributions from Cs-137, Co-60, and Sr-90. Most of the dose in the peak year is from external radiation; because the Co-60 dose factor for external radiation is larger than that of Cs-137 or Sr-90, Co-60 produces a dose fraction greater than its activity fraction. Figure 2-2 represents Drinking Water Pathway Dose. The values presented in Figure 2-1 and 2-2 are based upon a 1pCi/g concentration of each nuclide.

Figure 2-1. Time Dependence of Dose: All Nuclides Summed, All Pathways Summed For Surface Soils (1pCi/g)

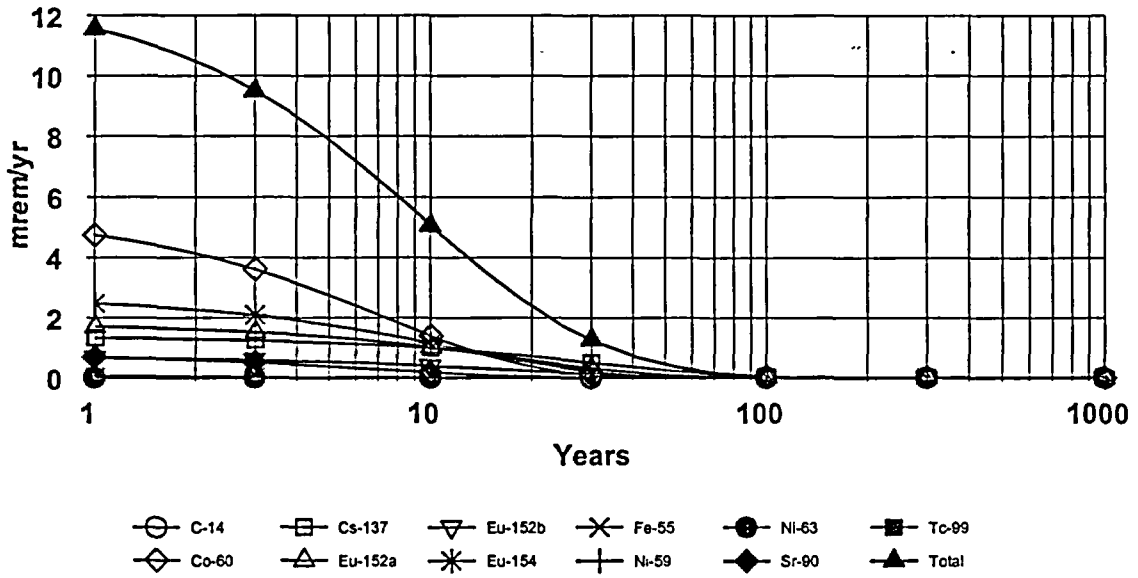
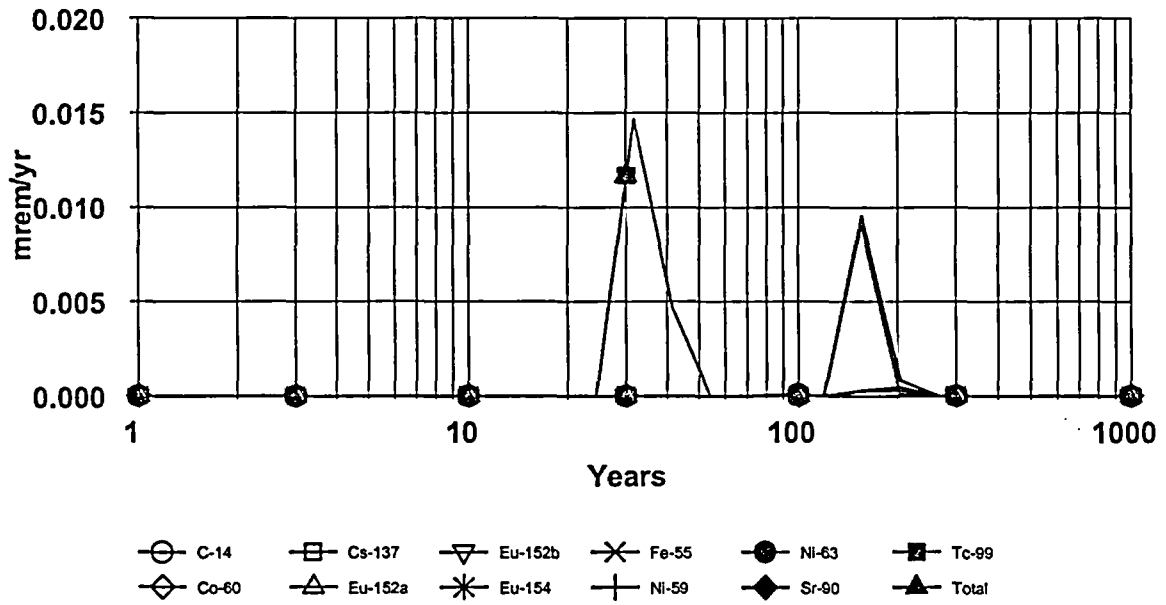


Figure 2-2. Time Dependence of Dose: All Nuclides Summed, Drinking Water For Surface Soils (1pCi/g)



Residual Contamination in Buildings

In its final state, contaminated materials and equipment will be removed as needed to meet the site release criteria, radioactive soils will be removed, and buildings and structures demolished to below grade level and backfilled. In some cases, buildings and structures will be demolished and the demolition debris appropriately disposed of as a means of removing residual radioactivity from the facility prior to termination of the license. In cases where safety, technical, and economic issues show it to be a more viable approach, the buildings and structures will be decontaminated to the license termination criteria and then demolished using conventional demolition techniques after license termination. Excavations and below grade portions of demolished structures will be backfilled with clean hard fill and demolition rubble that has been has been shown through evaluation and dose modeling to satisfy the license termination criteria. Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, the results submitted to the NRC with a safety or technical justification for backfilling, and concurrence obtained from the NRC that the area may be backfilled or otherwise rendered inaccessible. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and/or site visits. The building reuse scenario was used to develop DCGLs supporting release of PBRF buildings, such as the Reactor Office and Laboratory Building (1141), which is known to have low levels of residual contamination.

In the building reuse scenario, residual contamination is assumed to be located on building surfaces (i.e., walls, floors, and ceilings). The primary release mode is resuspension in air. Exposure modes include direct external exposure from surface and volume (i.e., surface and depth) sources; inhalation of resuspended material; and inadvertent ingestion of dust. Because uranium contamination is not expected based on historical data or survey measurements, the radon pathway is not included in the calculation of DCGLs for the building reuse scenario.

NRC has published generic screening DCGLs of common radionuclides for building surface contamination (NRC 1998c). To analyze residual contamination in PBRF buildings, a site-specific analysis was conducted using RESRAD-BUILD 3.0 (Yu et al. 2000). The analysis considered all reasonable pathways and exposure modes and is consistent with the NRC generic screening model (NRC 1998b). Table 2-16 summarizes the parameters used in the site-specific RESRAD-BUILD analysis. For the building reuse scenario, it was assumed that residual radioactivity in building walls would be sources for direct exposure, and contamination would be resuspended only from the floor. The rate of resuspension from the floor for the RESRAD-BUILD volumetric erosion source model was estimated by assuming that a concrete floor with a density of 2.4 g/cm^3 was contaminated to a depth of 1 cm, that the room was ventilated at an exchange rate of 2 room volumes per hour, and that the airborne concentration was that predicted by the NUREG-1549 surface source resuspension model with a resuspension factor of 1.8×10^{-6} m^{-1}. This approach produces an estimate of floor erosion rate of 1.3×10^{-4} cm/day. Site-specific parameters include the dimension of the room (15-m [49-ft] long, 5-m [16-ft] wide, and 3-m [10-ft] high). Using these dimensions for all rooms is prudently conservative because it represents the largest room expected to have residual contamination, it maximizes resuspension from the floors, and it maximizes direct radiation from the largest wall because of the relative narrowness of the room.

Table 2-16. Building Reuse Scenario: Parameter Values

Parameter	Parameter Value[a]
Occupancy period	365.25 days/yr[b]
Exposure time indoors	97.5 days/yr[b]
Exposure time outdoors	112 days/yr[b]
Resuspension factor	1.8×10^{-6} m^{-1}
Volumetric breathing rate	23 m^3/day
Transfer rate for ingestion	1.0×10^{-4} m^2/hr

a. Source: NUREG-1549 (NRC 1998b).
b. Time periods given as effective 24-hour days.

Table 2-17 presents the DCGLs for nuclides that would result in a TEDE of 25 mrem/yr under the building reuse scenario. The first column of Table 2-17 identifies the radionuclide; the second column presents the site-specific DCGLs calculated using RESRAD-BUILD; and, for perspective, the third column presents screening level DCGLs at the 95th percentile. Estimates of the external exposure estimated for the generic screening scenario considered a single surface of infinite extent (NRC 1998a). The site-specific estimates considered one floor and four wall surfaces of finite extent. Thus, the site-specific analysis produces DCGLs that are less restrictive than the generic screening analysis. Exposure from ceilings was not considered in derivation of the DCGLs presented in Table 2-17. If survey data collected during remediation indicate the presence of residual contamination, the DCGLs of Table 2-17 will be adjusted to reflect this condition. The individual nuclide DCGLs presented in Table 2-17 are combined using the sum-of-fractions rule to develop the single criterion used in the decision process.

Table 2-17. DCGLs for Buildings Remaining after License Termination

Radionuclide	Site-Specific DCGL (dpm/100 cm^2)[a]	Generic Screening DCGL (dpm/100 cm^2)[b]
H-3	1.3×10^8	1.3×10^8
Fe-55	1.0×10^8	4.0×10^6
Co-60	16,000	6900
Ni-63	4.2×10^7	1.6×10^6
Sr-90	199,000	7500
Cs-137	65,500	28,000

a. Calculated by RESRAD-BUILD using the parameters identified in Table 2-16.
b. Source: NUREG-1549 (NRC 1998b) and NRC (1998c)

The PBRF radionuclides of concern, DCGLs for each, where these radionuclides are present, and the ratio of the radionuclide mix will be revaluated as more samples from the current characterization are collected, analyzed, and evaluated.

In addition to providing numerical value for DCGLs, dose assessment methods were used to investigate the time dependence of dose. The time distribution for the building reuse scenario was calculated using an estimate of isotopic distribution based on characterization data. The average ratio of Cs-137 to Co-60 in the Reactor Building outside of the containment vessel was approximately 2:1, and all measurements show a Sr-90 to Cs-137 ratio of less than 0.1. Based on these data, an activity distribution of 63% Cs-137, 30% Co-60, and 7% Sr-90 was estimated. The level of residual contamination producing a maximum dose of 25 mrem/yr for this isotopic distribution has been calculated. The time dependence of dose for this inventory for a time period of 1000 years is presented in Figure 2-3. Figure 2-3 shows that the peak dose occurs in the first year after release of the PBRF and that dose decreases relatively rapidly with time. As shown by the uppermost curve in Figure 2-3, the total dose decreases to about 9 mrem/yr after 10 years.

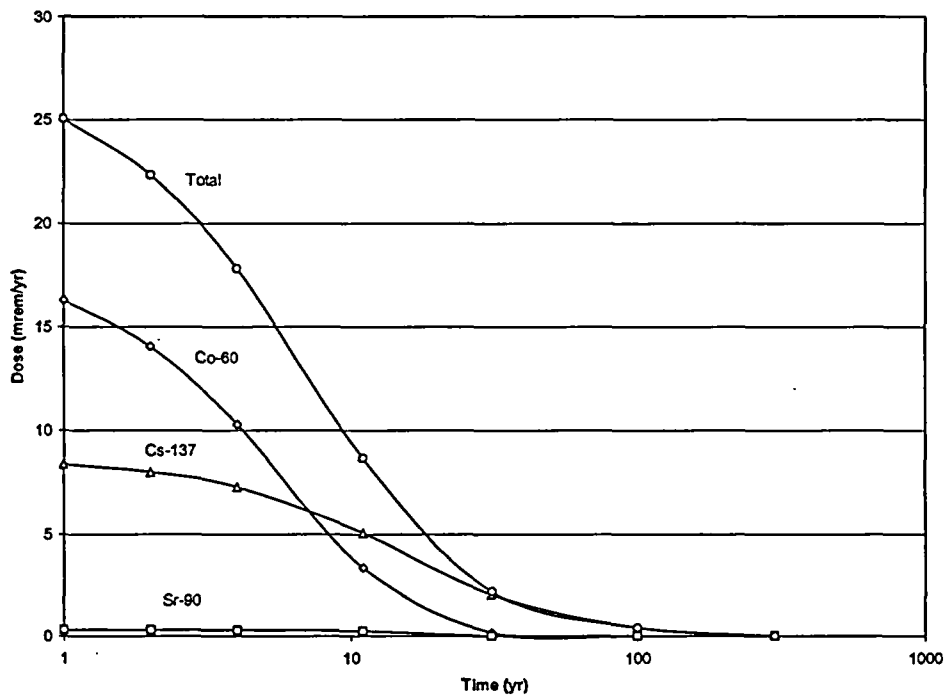


Figure 2-3. Time Dependence of Dose for Residual Contamination on Building Surfaces Producing a Maximum Annual Dose of 25 mrem/yr

Residual Contamination in Subsurface Structures

Decontamination and decommissioning of below-grade PBRF structures (e.g., the Reactor Building [1111]) will include decontamination of surfaces to building reuse DCGLs and offsite disposal of decontamination waste, removing decontaminated above-grade and below-grade structures down to 1-m (3-ft) below grade, backfilling belowground cavities with rubble generated from demolishing decontaminated above-grade and below-grade structures, and installing a cover over the backfilled area. Residual activity in the belowground portions of the various structures

could be from several sources: in crushed concrete from aboveground structures, in remaining portions of the biological shield, on the surfaces of the canals and quadrants.

The thickness of the contaminated zone is taken as 3-m (10-ft) to maximize dose through the external exposure and crop pathways. The radionuclides of concern for this scenario are those identified for subsurface structures in Table 2-6. The DCGLs derived for these radionuclides using site-specific analyses are presented in Table 2-18, and the generic screening DCGLs are provided for perspective. The DCGLs are combined using the sum-of-fractions rule to derive the single criterion needed in the decision process. The site-specific DCGLs indicate that Co-60 and Sr-90 are the dose-dominating radionuclides for the estimated radionuclide distribution.

The resident farmer scenario doses were estimated using the RESRAD code. RESRAD was designed for analysis of contamination in the unsaturated zone and the residual contamination associated with subsurface structures is in the saturated zone. To model the site in a manner consistent with the intended application of RESRAD, all of the activity was redistributed in the upper 3 meters and a value of zero was assigned to the unsaturated zone thickness. In order to be conservative, a cover layer was not included. The area used in the dose assessment was a 70-m (230-ft) cylinder (the approximate diameter of the subsurface structures), which extended vertically downward a distance of 3-m (10-ft) placing the contaminated zone directly on top of the saturated zone. The well would be located on the down gradient edge of the 70-m (230-ft.) diameter cylinder that is actually within site property.

Figure 2-4 represents total dose from all pathways summed and Figure 2-5 represents the dose associated with the drinking water pathway. The values presented in Figures 2-4 and 2-5 are based upon a 1 pCi/g concentration of each nuclide.

Table 2-18. DCGLs for Subsurface Structures

Radionuclide	Site-Specific DCGL (pCi/g)	Generic Screening DCGL (pCi/g)
C-14	39	6.5
Fe-55	71,320	9350
Co-60	4.1	3.7
Ni-59	4,240	1850
Ni-63	4,700	717
Sr-90	5.5	1.22
Tc-99	37.0	14.9
Cs-137	16.0	9.8
Eu-152	9.5	8.7
Eu-154	8.7	8.0

* Near-surface contamination at the Reactor Building is due to postulated redistribution of contamination associated with building rubble and subsurface structures.

The PBRF radionuclides of concern, DCGLs for each, where these radionuclides are present, and the ratio of the radionuclide mix will be evaluated as more samples from the current characterization are collected, analyzed, and evaluated.

Figure 2-4. Time Dependence of Dose: All Nuclides Summed, All Pathways Summed For Subsurface Structures (1pCi/g)

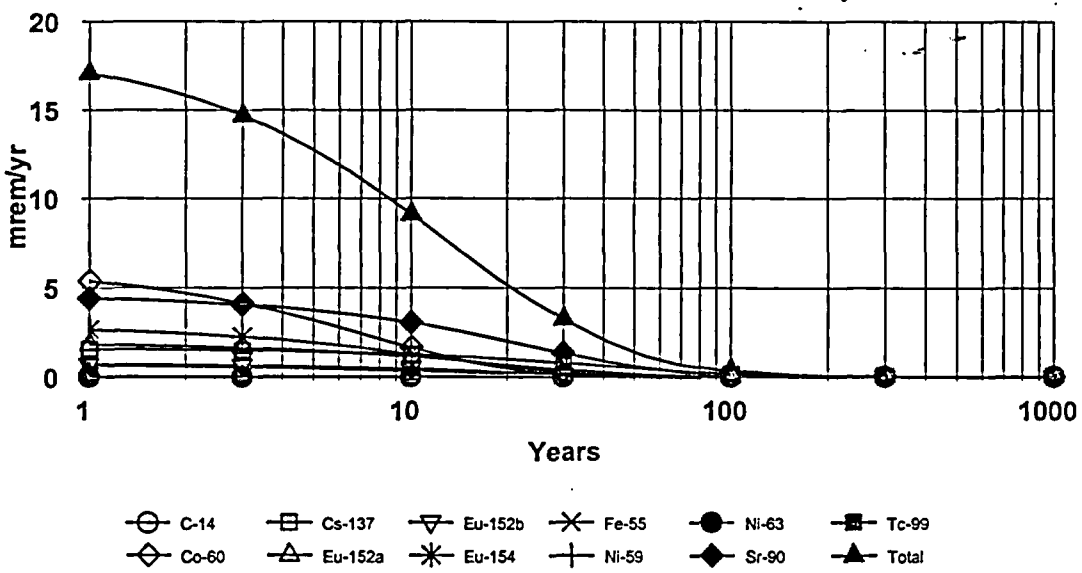
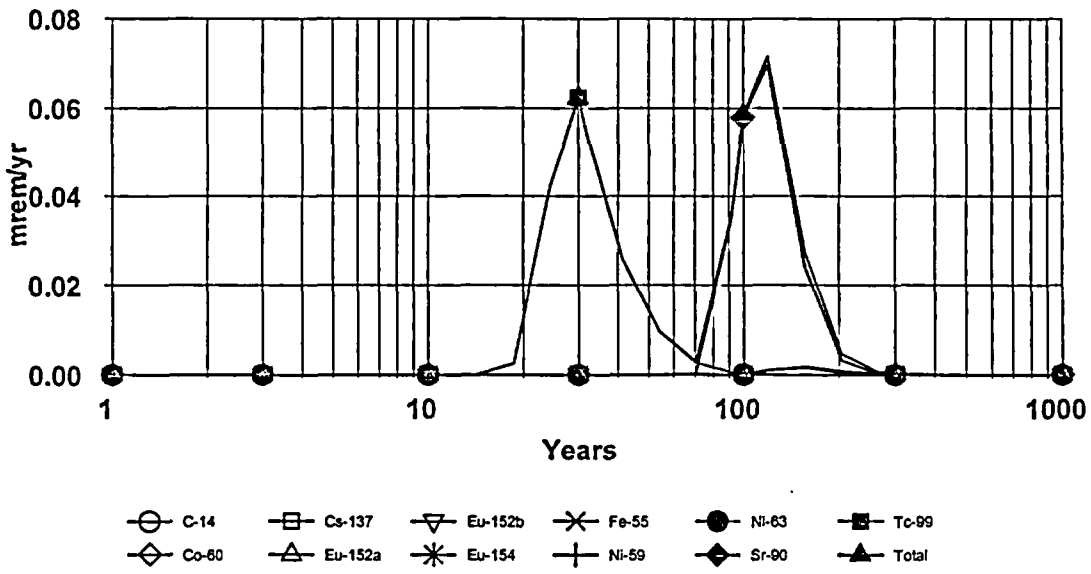


Figure 2-5. Time Dependence of Dose: All Nuclides Summed, Drinking Water For Subsurface Structures (1pCi/g)



2.2.3.2 ALARA Analysis Methodology

This section describes the proposed ALARA analysis methodology that will be conducted to demonstrate compliance with the requirements of 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination." The ALARA analysis methodology follows the concepts in NRC Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination" (NRC 1998a). The analysis compares the benefits and costs for postulated decommissioning actions and expresses them in present value economic terms (i.e., a dollar value). In accordance with Draft Regulatory Guide DG-4006, if the benefits of a particular action are greater than the costs, then an action should be taken (either the postulated action or possibly another action that has more balance between benefits and costs). If the costs for a particular action are greater than the benefits, then that action would not have to be implemented. If no action can be identified whose benefits are greater than its costs, the existing residual contamination would be considered ALARA.

Methodology for Conducting ALARA Analyses

The ALARA analysis methodology is presented as three steps:

1. Define the area of analysis and estimate the baseline population dose from existing contamination
2. Define the potential decommissioning action and estimate the population dose from residual contamination after implementing the action
3. Estimate the benefits and costs for the decommissioning action and compare them.

Each of these three steps is discussed in the following paragraphs.

Step 1: Define Area of Analysis and Estimate Baseline Population Dose—The first step is to define the area of analysis (e.g., an area of contaminated soil) and to estimate the baseline population dose (i.e., the population dose from existing contamination before implementing the action). The baseline population dose is estimated by first calculating the annual dose to the AMCG considering credible reuse scenarios as developed in Section 2.2.3.1. For example, if the contaminated area is a building, a building occupancy scenario using the existing radionuclide inventory would be evaluated to estimate the annual dose to an individual building occupant. If the area is contaminated soil, the annual dose to an individual resident farmer would be evaluated. Using the dose models described in Section 2.2.3.1, annual doses to the AMCG will be estimated for a maximum of 1000 years. Multiplying the annual individual doses by population density parameters from Draft Regulatory Guide DG-4006 yields population doses over the 1000-year period. In this way, the long-term baseline population dose as a function of time will be calculated for a specific area.

Step 2: Define Decommissioning Action and Estimate Population Dose After Implementing the Action—The second step is to define a postulated decommissioning action and then calculate what the population dose from residual contamination would be after implementing the action. This post-implementation population dose will be calculated from individual dose estimates and population density parameters in the same manner as the baseline population dose (step 1).

Step 3: Estimate Benefits and Costs and Compare Them for the Decommissioning Action—The third step is to estimate the benefits and costs of implementing the postulated decommissioning action and then compare them.

Step 3a. Estimate Benefits—The benefit of implementing a decommissioning action for a specific facility or area is the averted dose to the future population (i.e., the reduction in long-term population dose). The benefit of averted long-term population dose is calculated by subtracting the estimated long-term population dose after implementing an action (step 2) from the estimated long-term baseline population dose if no action was performed (step 1). This difference, the averted dose, will be converted to a monetary equivalent by multiplying by $2000 per averted person-rem per Draft Regulatory Guide DG-4006.

Step 3b. Estimate Costs—The costs of implementing a decommissioning action consists of the following components (including overhead costs):

- The monetary cost of performing the action
- The monetary cost of either transporting waste to a processing facility or transporting and disposing of the waste
- The monetary equivalent of worker fatalities from implementing the action
- The monetary equivalent of the dose to workers implementing the decommissioning action
- The monetary equivalent of the dose to the population from implementing the action and transporting waste
- The monetary equivalent of traffic fatalities from waste being transported to a processing or disposal facility
- Any other costs specific to the PBRF decommissioning actions.

The ALARA analysis focuses primarily on estimating the monetary cost for implementing the action and waste transport and disposal; other costs are comparatively small. If worker dose or population dose from implementing the action are included, the doses will be converted to monetary equivalents by multiplying by $2000 per person-rem. If fatalities associated with implementing the action are included, they will be converted to monetary equivalents by multiplying them by $3,000,000 per fatality according to Draft Regulatory Guide DG-4006. Input parameters used in this cost calculation (e.g., worker and traffic fatality rates) will be taken from Draft Regulatory Guide DG-4006.

Step 3c. Calculate Present Worth of Benefits and Costs—The monetary equivalent of future benefits and costs will then be discounted to determine their present worth following the guidance in Draft Regulatory Guide DG-4006. Future monetary equivalents will be discounted to determine present worth using Equation 2-1.

$$C_{pw} = \frac{C_n}{(1+d)^n} \qquad (2\text{-}1)$$

where:

C_{pw}	=	present worth of future monetary equivalent,
C_n	=	monetary equivalent at n years in the future,
d	=	discount rate,
n	=	number of years in the future that the monetary equivalent is calculated.

Discount rates will be used in accordance with Draft Regulatory Guide DG-4006. A 7 percent discount rate will be applied for the first 100 years after an action is performed, and a 3 percent discount rate will be applied beyond that time. The present worth of both benefits and costs will be discounted using Equation 2-1.

Step 3d. Compare Present Worth of Costs and Benefits—The present worth of the benefits and costs calculated in step 3c will be compared. For those actions where the present worth of benefits is greater than costs, then the existing residual contamination would not meet the ALARA requirement, and some decommissioning action (e.g., scabbling of concrete) should be taken. For those actions where the present worth of the costs are greater than benefits, then according to Draft Regulatory Guide-4006, the analyzed decommissioning action would not be necessary to comply with the ALARA requirement. If no other action can be identified that results in the benefits being greater than the costs, then the existing residual contamination level will be ALARA.

An example of the preliminary ALARA analysis conducted for the Emergency Retention Basin at the PBRF is given in Appendix C of this plan.

Examples of Preliminary ALARA Analysis for Selected PBRF Decommissioning Actions

Preliminary ALARA analyses were conducted for five postulated decommissioning actions for the PBRF facilities. These preliminary ALARA analyses considered individual resident farmer scenarios when calculating future doses because the buildings at these areas will have been demolished as part of decommissioning. Table 2-19 summarizes the selected decommissioning actions and the results of the preliminary ALARA analysis. The first two columns identify the facility or area and the postulated decommissioning action. The third column presents the calculated benefits. Because NASA intends to retain control of the site, any potential exposure would occur at a later time frame than that assumed in the analysis; therefore, the averted population dose benefit estimates presented in Table 2-19 are biased high. The fourth column presents the calculated costs, which consist of the total dollar cost of implementing the action. These cost estimates are fully burdened. The costs may be biased low because a low unit waste disposal cost was assumed for the analysis.

Table 2-19. Summary of Preliminary ALARA Analysis Results for Selected Decontamination and Offsite Disposal Actions

Facility/Area	Action	Benefit ($)	Cost ($)
Reactor Building (1111)	Remove highly activated portion of biological shield	23	69,600
	Remove primary cooling water piping	0	1,140,000
Emergency Retention Basin*	Remove contaminated surface soil	8,924	1,859,000
Pentolite Ditch	Remove contaminated surface soil	2,450	271,000
Previous spill area near the Waste Handling Building (1133)	Remove contaminated soil and asphalt	179	283,000

* Example ALARA analysis calculations for the Emergency Retention Basin are provided in Appendix C of this plan.

Preliminary ALARA analysis indicates that complying with the criteria of a TEDE to the AMCG of 25 mrem/yr also achieves the criterion that residual radioactivity is ALARA.

2.3 Decommissioning Tasks

This section (1) describes the decommissioning strategy for the PBRF, (2) provides an overview of the work scope, (3) provides a general description of the decommissioning activities associated with site preparations, and performing dismantling and decontamination activities, (4) conducting the final status survey, (5) building demolition, (6) Site Restoration, and, (7) presents the schedule for these activities. The following information related to decommissioning activities and tasks is contained in other sections of this decommissioning plan:

- The locations of facilities and areas are described in Section 1.2 and shown in Figure 1-2.
- The estimated radioactivity of PBRF facilities and areas is discussed in Section 2.2.2.
- Estimates of worker dose from decommissioning activities are provided in Section 3.1.3 (Table 3-2).
- Types of radioactive waste that will be generated, waste packaging, and waste resolution are discussed in Section 3.2.2; details are shown in Table 3-3.

2.3.1 Decommissioning Strategy

The decommissioning strategy chosen by NASA for the PBRF will lead to decontamination of the facility to a level that would permit termination of the NRC licenses and release of the facility for unrestricted use. In its final state, materials and equipment contaminated above the release criteria will be removed, radioactive soils will be removed, and buildings and structures demolished to below grade level and backfilled. In some cases, buildings and structures will be demolished and the demolition debris appropriately disposed of as a means of removing residual radioactivity from the facility prior to termination of the license. In cases where safety, technical, and economic issues show it to be a more viable approach, the buildings and structures will be

decontaminated to the license termination criteria and then demolished using conventional demolition techniques after license termination. Excavations and below grade portions of demolished structures will be backfilled with clean hard fill and demolition rubble that has been shown through evaluation and dose modeling to satisfy the license termination criteria. Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, the results submitted to the NRC with a safety or technical justification for backfilling, and concurrence obtained from the NRC that the area may be backfilled or otherwise rendered inaccessible. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and/or site visits. While the decontamination work is in process, remedial action support surveys will be made to ensure that the contamination has been removed to the limits required. Final status surveys of surface will be conducted to verify that any residual contamination results in a TEDE less than 25 mrem/yr to the AMCG, before backfilling.

NASA is considering using two options for DCGLs and decommissioning actions for above grade structures:

(1) Surfaces of above grade structures would be decontaminated to meet either building reuse or resident farmer DCGLs, the final status survey would be conducted, the above grade structure would be demolished, and the concrete that meets the requirement to be classified as clean hard fill and additional clean hard fill would be placed in the subsurface cavities within the PBRF. The final status survey measurements will be on a surface area basis.

(2) The entire above grade portion of the building would be removed. Any portions radiologically contaminated above the release criteria would be disposed of offsite as low-level radioactive waste. Non-radioactive portions of the building would be disposed of in a normal industrial landfill. Fill material acceptable to OEPA, with the exception of asphalt, would be used to backfill the PBRF.

For below-grade structures, the surfaces will be decontaminated to appropriate release criteria, the final status survey will be conducted, and the subsurface cavity will be backfilled with clean, hard fill (as defined by OEPA except no asphalt will be used) from the demolition of the above grade portion of a decontaminated PBRF building. Following placement of clean hard fill (excluding asphalt), a clean fill material will be placed as backfill, and the area will be contoured.

At the time of decommissioning the PBRF, NASA management will evaluate these options and select one that meets the regulatory requirements, safety objectives, and is cost effective.

The final status surveys will be planned and implemented in accordance with Draft Regulatory Guide 4006 (1998a), NUREG-1505 (Gogolak et. al, 1998), NUREG-1575 (USEPA et. al, 1997), and NUREG-1507 (Abelquist et. al, 1997). Verification surveys will be performed, as required, by the NRC to demonstrate the adequacy of the final status surveys. Radioactive wastes generated during the removal and decontamination activities will be shipped to either a licensed, low-level radioactive waste disposal facility or to a waste processor. Industrial waste generated by building demolition will be disposed of off site in an industrial landfill.

In each building and work area, a source term reduction strategy is planned for each task, where material having high source terms or radiation levels will typically be removed first to minimize personnel exposure during the remainder of the task. The source term reduction effort would be

modified when specific conditions are expected to result in personnel exposures that are not ALARA.

The activities comprising the PBRF decommissioning project are listed in Table 2-20. Each activity listed in Table 2-20 are the general activities to be performed during decommissioning. A Project Work Breakdown Structure (WBS) is described in Section 2.3.2.

Table 2-20. Activities and Tasks for Decommissioning the NASA Plum Brook Reactor Facility

Work Phase	Work Description[a]
Planning Activities:	
	Decommissioning Planning
	NASA Operations and Direct Support
Preparation Activities:	
	Systems Operation, Maintenance, and Deactivation
*	Site Preparation
Decontamination and Dismantling Tasks:	
	Operations Management and Support
	Security
	Health Physics
*#	Contaminated Soil Removal[b]
*	Asbestos Removal and Lead Paint Abatement
*	Loose Equipment Removal
*	Removal of Activated Material in Hot Dry Storage Area
*	Decontamination
*	Reactor Internals and Tank Removal
*	Contaminated Piping and Equipment Removal
*	Contaminated Concrete and Embedded Pipe Removal (as needed to meet the site release criteria)
	Final Status Survey
	Building Demolition
	Backfill and Site Restoration

a. Items with * include waste packaging and transportation.
b. Item with # includes backfilling excavated areas.

2.3.2 Decommissioning Scope and Work Breakdown Structure

The decommissioning activities are organized by activity and type of work through the work breakdown structure (WBS) for the project. This shows a general overview, but is subject to change as the project progresses. The first-order headings for the WBS are listed below:

- WBS 1.0 Proposals/General/Investigation/Training
- WBS 2.0 Design/Plan Development
- WBS 3.0 Execution

Although each first-level work element represents project requirements, the execution category includes the majority of activities. WBS 3.0 is organized to separate the work into geographical features or significant execution facility-wide activities at the second level. At the third level, these areas and activities are further segregated by area, subtask. Some examples of second and third level activities are provided here to clarify:

- WBS 3.1 Mobilization
  - 3.1.1 Install Temporary Services
  - 3.1.2 Site Preparation
- WBS 3.2 Reactor Building – Building 1111
  - 3.2.1 Quadrant A
  - 3.2.2 Quadrant B
  - 3.2.8 Canal G
  - 3.2.14 Pump Room Area 22
- WBS 3.9 Retention Areas
  - 3.9.1 Hot Retention Area – Building 1155
  - 3.9.2 Cold Retention Area – Building 1154
- WBS 3.15 Waste Management
  - 3.15.1 Waste Handling & Packaging
  - 3.15.2 Waste Transportation
  - 3.15.3 Waste Disposal

Planned radiological decontamination activities are presented at Level 4, as required for each building area element (Level 3). These fourth level elements are typically those shown below:

- Job Preparation
- Site/Area Preparation
- Loose Equipment Removal
- Fixed Equipment (or Major Component) Removal

- Pipe Removal
- Embedded Pipe Removal (as needed to meet site release criteria)
- Contaminated Concrete Removal
- Under Slab Soil (as needed to meet site release criteria)
- Survey/Remediation
- Area Cleanup (including placement of floor and wall coverings)
- Final Status Surveys

Specific work activities are grouped in the fourth level of the WBS. Work will be planned, executed, and controlled at this level.

A projection of the anticipated schedule for the WBS items in the Decommissioning Program is shown in Figure 2-6.

2.3.3 Decommissioning Activities

Work Breakdown Structure element 3.0 includes those tasks directly associated with decommissioning, decontaminating, and demolishing the PBRF. From mobilization to demobilization, the following tasks are required to properly dismantle the facility, dispose of generated wastes and debris, and return the site to unrestricted use per the NRC criteria. The tasks were assigned to match the work planning and activities by segregating work into building or significant functional activities. The building and areas identified for radiological decontamination include the WBS items in Table 2-21.

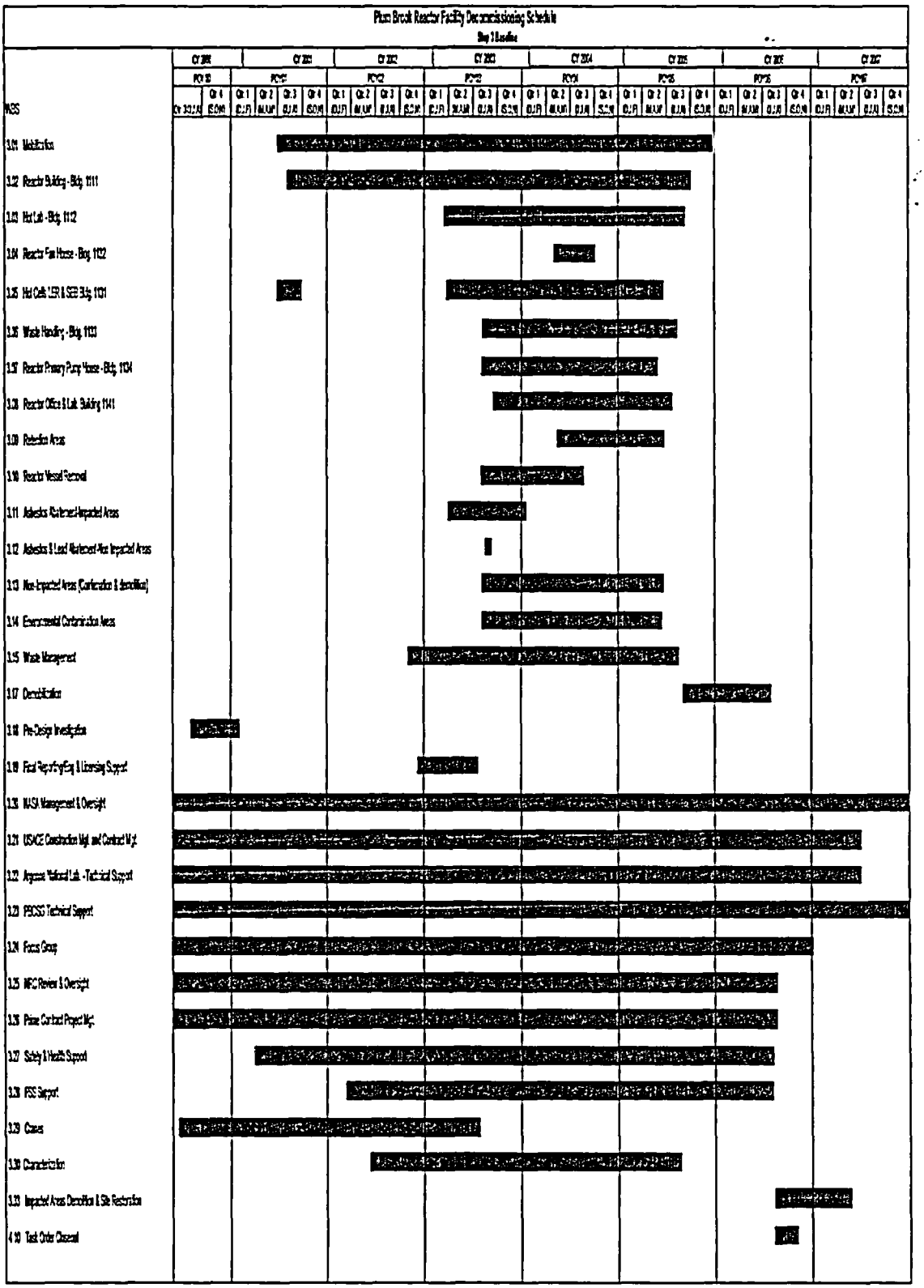


Figure 2-6. Anticipated Decommissioning Program Schedule

Table 2-21. Work Breakdown Structure for Decommissioning the NASA Plum Brook Reactor Facility

WBS Element	Task Description
WBS 3.01	Mobilization
WBS 3.02	Reactor Building – Building 1111
WBS 3.03	Hot Laboratory – Building 1112
WBS 3.04	Reactor Fan House – Building 1132
WBS 3.05	Hot Cells Loose Equipment Removal & SEB Building 1131
WBS 3.06	Waste Handling Building – Building 1133
WBS 3.07	Reactor Primary Pump House – Building 1134
WBS 3.08	Reactor Office and Laboratory – Building 1141
WBS 3.09	Retention Areas
WBS 3.10	Reactor Vessel Removal
WBS 3.11	Asbestos Abatement – Impacted Areas
WBS 3.12	Asbestos and Lead Abatement – Non-Impacted Areas
WBS 3.13	Non-Impacted Areas (Confirmation and Demolition)
WBS 3.14	Environmental Contamination Areas
WBS 3.15	Waste Management
WBS 3.17	Demobilization
WBS 3.18	Pre Design Investigation
WBS 3.19	Final Reporting/Eng & Licensing Support
WBS 3.20	NASA Management & Oversight
WBS 3.21	USACE Construction Management & Contract Management
WBS 3.22	ANL Technical Support
WBS 3.23	PBOSG Technical Support
WBS 3.24	Focus Group Community Relations Support
WBS 3.25	NRC Review & Oversight
WBS 3.26	Prime Contract Project Management
WBS 3.27	Safety and Health Support
WBS 3.28	Final Status Survey Support
WBS 3.29	Cranes
WBS 3.30	Characterization
WBS 3.33	Impacted Areas Demolition
WBS 4.1	Task Order Closeout

2.3.3.1 Site Preparations and Mobilization

Site preparations and mobilization are covered in WBS 3.01 Mobilization. Tasks associated with mobilization include the following:

Temporary Services: Power, lighting, air filtration, and HVAC will be positioned, connected, or installed prior to execution of the major decommissioning activities. Most of the decommissioning activities will be performed by using temporary services and will not depend on existing plant systems.

Site Preparations and Implementation: Plant systems not needed to support the operation of the security system and the site ventilation and monitoring equipment, will be de-energized.

Site Access Modifications and Crane Certification: Physical modifications to the PBRF will be necessary to properly access decontamination areas as well as limit access to unauthorized personnel. This task is assumed to be conducted concurrently with installation of temporary services to facilitate site modifications while ensuring that site modifications do not compromise utility availability.

2.3.3.2 Radiological Decontamination – Overview

On a fundamental level, the decontamination activities for impacted buildings are fairly consistent from area to area. However, differences exist which have an effect on project cost and schedule. As such, each building area will be characterized and will be evaluated individually to estimate the amount of debris, piping, equipment, components, concrete-embedded features, and volumes of radiological surface contamination.

In general, a work control document will be prepared for decommissioning tasks. The work control documents will provide sufficient detail to the workers to assure that the tasks can be properly and safely performed in accordance with projects plans and procedures. They will identify the hazards associated with the tasks, and provide sufficient precautions and limitations to assure that all radiological control and health and safety measures invoked by project plans and procedures are effectively implemented.

In general, asbestos will be removed from work areas before decontamination activities begin. When it is not possible to do so, appropriate protective measures will be established to minimize the worker exposure. Lead-contaminated paint will also be mitigated to the extent possible prior to decontamination activities that disturb paint surfaces

Some decommissioning tasks can produce airborne contamination. It will be controlled using appropriate engineering controls to assure that airborne activity is monitored and releases to the environment are within regulatory limits.

In general, decontamination will begin with characterization and removal of all loose debris and equipment. Fixed equipment and exposed piping is then characterized and removed. Instrument lines, electrical service connections and electrical panels will be characterized and removed utilizing appropriate tools and equipment. All equipment and piping identified for disposal will be further size reduced and be of such size as to be deposited into the appropriate waste containers. The basic sequence will be to remove equipment and components and their anchors from the floor, install scaffolding, and then remove equipment and components from the walls.

Embedded piping will be removed, when necessary, and size reduced and packaged for disposal. Large quantities of contaminated concrete may have to be excavated from walls, floors, and sumps, the biological shield, and other areas to access and then remove contaminated, embedded piping. In some cases where embedded piping can be decontaminated to a level that will meet the site release criteria, it may be left in place.

During the removal of contaminated concrete and embedded piping, the structural integrity of associated walls, floors, and ceilings may be jeopardized. Routine and specific evaluations of PBRF structures will be necessary during the decontamination phase to ensure worker safety. These structural evaluations will be conducted by qualified engineers and documented in the work execution packages.

Following necessary removal of equipment, piping, and embedded piping, contaminated surface coatings, paint, and concrete will be characterized and subsequently mechanically scraped (scabbled) from the walls and floors of each area. Where present, lead/PCB paints will be packaged as Hazardous Waste. Concrete surfaces will be decontaminated by removing the paint and surface layer of concrete by mechanical means.

Contaminated paint and concrete will be removed from the walls, ceilings and floors using a variety of powered equipment. Hand tools, floor walking and wall walking scabblers, and assorted power equipment will be used to scrape or chip concrete to a depth consistent with the amount of contamination documented in prior surveys, pre-job screening, and relative to specific activities conducted in the area being decontaminated. Where this method is not practicable, concrete will be fractured and packaged as waste or cut into sections for disposal. Removed debris will be collected using a HEPA vacuum system to thoroughly clean the walls and floor.

Steel structures requiring decontamination will be either wiped down and left for survey under the FSS Plan, or cut out and disposed of as waste. Minimal resource-hours will be expended attempting to decontaminate steel structures. Major components that may have salvage value will be decontaminated to the extent possible and surveyed for release. It is expected that only minor decontamination will be done on site. Once a useable component has been surveyed and determined to be effectively decontaminated, it will be released from the area per the Radiological Protection Plan and set aside for salvage. If components cannot be easily decontaminated on site they will be appropriately reduced in size, removed, and packaged as contaminated waste, or be sent to a commercial processor.

Following removal of contaminated surface coatings and concrete, a post decontamination survey will be completed to evaluate decontamination. Localized areas may indicate residual contamination and those areas will be re-cleaned and re-surveyed. This will continue until all surfaces are decontaminated to the site release criteria.

Following successful post-remediation survey, the area will be prepared for the FSS by isolation or control of the area to prevent re-contamination by work activities in adjacent areas.

A formal area turnover and site control protocol will be developed to ensure site integrity and minimize the potential for cross contamination. Sequencing of survey areas will be continually

evaluated to minimize the impact of parallel work activities. The Decommissioning Contractor will maintain site control protocols until confirmatory surveys are performed.

It is estimated that portions of the biological shield will be removed as bulk concrete to meet the DCGL values for subsurface structures. The exterior surface of the concrete biological shield is lined with steel plate that would be removed with the concrete.

Materials, equipment, and components will be removed from the work area and managed in a waste staging area on site. The Decommissioning Contractor will be responsible for managing waste materials and loading containers for disposal. These activities are addressed under WBS 3.15, Waste Management.

2.3.3.3 Radiological Decontamination – Reactor Vessel Removal

Feasibility studies have been conducted and results indicate that segmentation of the reactor internals and reactor tank is prudent. Therefore, these components will be segmented in place and removed. A key assumption to this conclusion is that radiation levels within the reactor tank have decayed to a level to make segmentation of the internal components practical with minimal utilization of ALARA administrative procedures. Verification of the internal radiation conditions will be performed prior to program initiation. This task will be performed after the non-embedded equipment and piping have been removed from the quadrant areas.

In preparation for removing the reactor internals and tank, temporary platforms will be constructed to support necessary activities and provide a confinement structure during segmentation operations. The platforms will also support the specialized equipment and shielding required for this task.

The reactor tank internals and core box will be disassembled by unbolting the components, where feasible. Where unbolting is not feasible, the internals and core box will be separated from the reactor tank using remotely-operated equipment (such as band saws or hydraulic shears). If components do not fit into the cavity of the licensed shipping containers needed to transport them to a radioactive waste disposal site, components will be further segmented. If thermal cutting processes, such as plasma arc cutting, are needed, the processes will be performed under suitable engineering controls to prevent the uncontrolled or unmonitored release of airborne activity.

After the reactor internals are separated from the reactor tank, the reactor internals will be moved from the reactor tank to the cask or cask liner using the polar crane and a transfer shield, if needed.. The cask or cask liner will be staged in a location where shielding will be provided if required. The cask or liner would be moved from there to be loaded onto a transport vehicle.

After the reactor internals are removed, the reactor tank will be segmented using remotely operated equipment. Mechanical cutting methods producing limited amounts of airborne contamination, such as milling machines and lathes, will be used. Thermal cutting methods, if used, will be performed under controlled conditions that assure that airborne radioactivity is controlled and monitored. The exposure of the surrounding insulation will require abatement.

The tank sections will be removed using a transfer shield, if needed, and packaged in an appropriate area within the Containment Vessel or Reactor Building (1111).

In series or parallel to segmenting and removing the reactor tank, the beam tubes, primary coolant water piping, and other penetrations in the biological shield will be vacuumed and the embedded piping (if necessary), lead shield, and supports dismantled.

The removal of the Mock-Up Reactor (MUR) will be a separate task, unassociated with the removal of the main Reactor Tank. The activity levels present in this structure are low enough that the segmentation will be fairly straightforward. Once the MUR is segmented the resulting pieces will be placed into shipping containers for transport to either a waste processor or a disposal facility.

2.3.3.4 Radiological Decontamination – Environmental Areas

Areas of environmental contamination include in-ground or earthen structures or soil that was contaminated from past operations and non-routine occurrences (i.e., spills). Radiological characterization of these areas was completed during the 1985 and 1988 surveys including several core samples and surface soil samples.

Characterization information from each of these areas was used to estimate the amount of contaminated soil to be excavated. The field approach will be to characterize soil readings as excavation progresses and discontinue excavation when acceptable measurements are obtained.

WBS 3.14.01 Emergency Retention Basin: The Emergency Retention Basin is a 250 foot by 350 foot earthen retention basin. The Emergency Retention Basin was used as emergency storage for radioactively contaminated water that exceeded the allowable discharge criteria. As a matter of course, the clayey soil of this basin became contaminated with radioactive material. That soil will be excavated as part of decontamination.

Soil will be excavated as needed to meet the site release criteria and will be appropriately packaged for disposal. As part of this activity piping and flow control equipment will be removed from the Water Effluent Monitoring Station 1192. Further excavation will be performed if survey measurements indicate that it is needed

WBS 3.14.02 Drainage System: A series of open ditches, covered culverts, and more than 40 catch basins were used to collect and convey surface water runoff, building sump discharges, roof-top runoff, and low-level liquid wastes (within discharge limits) to the Water Effluent Monitoring Station 1192. Characterization indicated that these ditches have shallow soil contamination especially in areas where surface water had a tendency to pond (catch basins) as well as the underground piping in the catch basins. Although the naturally occurring nuclide K-40 was the primary component to the radioactivity within the drainage system, measured activity from Cs-137 and Co-60 were present. As such, the ditches and catch basins will be excavated as needed to remove radioactive contamination that exceeds the site release criteria.

WBS 3.14.03 Pentolite Ditch: The Pentolite Ditch is located along Pentolite Road extending from the southeast corner of the Emergency Retention Basin eastward to Plum Brook. The ditch received all water from the Water Effluent Monitoring Station. The contamination occurs primarily at the western end (near the Water Effluent Monitoring Station outfall), with smaller amounts near the eastern end (near Plum Brook). In some areas, up to 12-inches of soil in the area radioactively contaminated. Approximately 4,500 cubic feet of soil is estimated for removal from the Pentolite Ditch in order to meet site release criteria. Further excavation of radioactively contaminated areas will be performed if survey measurements indicate it is necessary.

2.3.4 Final Status Survey (FSS)

The FSS is designed to demonstrate that decontamination activities have been effective in removing licensed radioactive materials from the PBRF structures and soil to the extent that residual levels of radioactive contamination are consistent with the approved DCGL. These DCGL values are established to ensure compliance with the unrestricted release criteria established by the NRC.

The primary objectives of the FSS are to:

- Select and verify survey unit classification
- Demonstrate that the potential dose or risk from residual contamination is less than the release criterion for each survey unit, and
- Demonstrate that the potential dose or risk from small areas of elevated activity is less than the release criterion for each survey unit.

The FSS will be performed in areas classified as one of the following contaminated or potentially contaminated areas:

Class 1 Areas: These are areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination in excess of the $DCGL_W$.

Class 2 Areas: These are areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_W$.

Class 3 Areas: These are impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$.

Final Survey Status Plan: A FSS Plan will provide the necessary detail to implement preliminary and final surveys in all areas. The FSS Plan is described in WBS 2.16, "Final Status Survey Plan". Each FSS will be completed according to the FSS Plan and will result in full documentation of conditions within each area.

Final Survey Status Approach: The design approach of the FSS is affected by the final configuration of the facility, i.e., with most systems and components removed and structures largely intact. The majority of the survey effort will occur in areas where radioactive materials were used or handled.

The FSS will conform to the project QA Plan and will be subject to review and audit by the Decommissioning Contractor and NASA.

Appropriate data tracking systems and equipment will be used. For each area, a mapping positioning systems will be used to document survey positional data. Data measurements will be concurrently tracked to combine position and measured survey concentration. Area and building data will also be taken to summarize information and for consistency. Data will be of a format that will allow for comparison with NRC or other third party verification or follow-up evaluations.

2.3.5 Facility Demolition

Facility demolition will be performed in stages as the decommissioning progresses. Where an impacted structure cannot be successfully decontaminated to meet the final license termination criteria, it will be demolished and the debris disposed of as radioactively contaminated material. Buildings that can be successfully decontaminated will be demolished after successful decontamination and remediation and successful performance and approval of the FSS. Final demolition may take place after license termination using conventional demolition processes. This will occur only after successful FSS has been completed and any independent verification required by the NRC has been completed. Demolition debris from decontaminated structures or from non-impacted structures will be processed and disposed of in accordance with applicable State and Federal environmental regulations, or will be used as backfill material.

2.3.6 Site Restoration

After the FSS has been performed and the appropriate regulatory agencies acknowledge acceptance, the remaining structures with below grade voids will be backfilled. It is anticipated that the non-contaminated concrete and masonry bricks will make up a portion of the clean, hard fill for the site. Once the concrete and brick is used up, clean (non-contaminated) soil from the berms associated with the Emergency Retention Basin will be used. If additional material is required, an off-site backfill source will be procured to assure the below grade voids are completely brought to the appropriate grade. In order to assure proper vegetative growth, 6-inches of topsoil will be placed above all disturbed areas.

2.3.7 Safety Hazards During Decommissioning Activities

Decommissioning activities will be performed under the control of appropriate procedures that have been written, reviewed and approved in accordance with the requirements of the Facility

License and project administrative procedures. The procedure development process will include the performance of a Job Safety Analysis (JSA), and when appropriate, the preparation of Radiation Work Permits (RWPs).

Table 2-22 lists the types of hazards associated with particular decommissioning tasks and the measures to minimize potential accidents and injuries. Appropriate radiological control plans and procedures will be developed to implement an effective program of protecting the workers and the public from exposure to hazards associated with radiation and radioactive material. In addition, appropriate plans and procedures will be developed to protect the workers and public from industrial safety hazards associated with the decommissioning project. Workers will receive initial training and periodic refresher training in both radiological and industrial safety hazards and protection and will be trained on the use of any personal protection equipment used during the project.

Table 2-22. Radiological and Industrial Safety Hazards Expected During PBRF Decommissioning Activities

Hazard	Tasks Affected	Measures to Minimize Hazard
Radiological:		
High radiation exposure – direct	• Reactor Internals and Tank Removal • Removal of Activated Material in Hot Dry Storage Area	• Work will be planned considering the ALARA principle • Use of specialized shielding • Mock-up training • Special tools
Airborne radioactivity	• Decontamination of concrete and steel structures • Contaminated Piping and Equipment Removal • Reactor Internals and Tank Removal	• Worker training in Contractor Safety Procedure for Respiratory Protection and Air Monitoring • Filtered ventilation • Contamination control envelopes
Loose contamination	• Reactor Internals and Tank Removal Removal of Activated Material in Hot Dry Storage Area • Contaminated Piping and Equipment Removal • Contaminated Concrete and Embedded Pipe Removal • Decontamination	• Work will be planned considering the ALARA principle • Remedial action status surveys of work in progress • Personnel protective clothing • Portable vacuum filtration equipment • Contamination control envelopes
Industrial:		
Confined spaces	• Contaminated Piping and Equipment Removal • Decontamination of hot pipe tunnel, Cold Retention Basins, and Hot Retention Area	• Worker training in Contractor Safety Procedure for Confined Space Entry Precautions • Control access to confined spaces • Use procedures for atmospheric testing
Energized electrical systems	• Loose Equipment Removal Contaminated Piping and Equipment Removal • Contaminated Concrete and Embedded Pipe Removal	• Deenergize PBRF electrical systems • Use of ground fault circuit interrupters • Lockout/tagout of systems • Worker training in Contractor Safety Procedure
Excavation instability	• Demolition of Cold Retention Basins, Hot Retention Area Vault, and Water Effluent Monitoring Station	• Excavation permit controls • Worker training in Contractor Safety Procedure • Use of benching or sloping sides for excavations • Use of shoring or trenching shields

Table 2-22. (Continued) Radiological and Industrial Safety Hazards Expected During PBRF Decommissioning Activities

Hazard	Tasks Affected	Measures to Minimize Hazard
Industrial (Cont'd):		
Welding, cutting, burning, hot work	• Loose Equipment Removal • Contaminated Piping and Equipment Removal • Demolition	• Worker training in Contractor Safety Procedure • Hot Work Permits to control operations
Scaffolds	• Contaminated Piping and Equipment Removal • Decontamination of concrete and steel structures • Asbestos Removal and Lead Paint Abatement	• Worker training in Contractor Safety Procedures for erection, use, dismantlement, and storage of scaffolds and work platforms • Use of properly engineered scaffolds • Qualified personnel to erect and dismantle
Falls	• Contaminated Piping and Equipment Removal • Decontamination of concrete and steel structures • Asbestos Removal and Lead Paint Abatement • Building Demolition	• Worker training in Contractor Safety Procedure for Protection from Falls • Inspection of elevated working surfaces for structural integrity • Provision of guardrail and personal fall arrest systems • Provision of covers or guardrails over unprotected openings • Use of toeboards and/or canopies to prevent or protect from falling objects
Material handling	• Packaging wastes (all decontamination and dismantling tasks) • Handling waste packages (all decontamination and dismantling tasks)	• Worker training in Contractor Safety Procedure for Proper Handling of Materials
Asbestos hazards	• Asbestos Removal and Lead Paint Abatement • Decontamination of Structures • Contaminated Piping and Equipment Removal • Building Demolition	• Worker training in Contractor Hazard Awareness Program • Competent person for asbestos identification • Work practices and exposure controls • Engineering controls
Lead hazards	• Asbestos Removal and Lead Paint Abatement • Decontamination of Structures • Contaminated Piping and Equipment Removal • Building Demolition	• Worker training in Contractor Hazard Awareness Program • Competent person for lead identification • Work practices and exposure controls • Engineering controls

Table 2-22. (Continued) Radiological and Industrial Safety Hazards Expected During PBRF Decommissioning Activities

Hazard	Tasks Affected	Measures to Minimize Hazard
Industrial (Cont'd):		
Mobile equipment	• Handling waste packages (all decontamination and dismantling tasks) • Concrete excavation • Demolition	• Worker training in Contractor Safety Procedure for Safe Handling of Mobile Equipment
Power tool use	• All decontamination and dismantling tasks	• Worker training in Proper Use and Maintenance of Power Tools • Ensure tools purchased incorporate safety features
Airborne particulate	• Decontamination of concrete and steel structures • Contaminated Piping and Equipment Removal • Reactor Internals and Tank Removal	• Worker training in Contractor Safety Procedure for Respiratory Protection and Air Monitoring
Flammable and combustible liquids	• All decontamination and dismantling tasks	• Controls on storage and handling and requirements for storage areas • Restricted areas and special equipment for dispensing flammable and combustible liquids • Limitations on use • Worker training in Contractor Safety Procedure for work with flammable and combustible liquids
Toxic and hazardous substances	• All decontamination and dismantling tasks • Loose Equipment Removal (includes removal of switches containing mercury)	• Worker training in Contractor Safety Procedure for Safe Use of Toxic or Hazardous Substances • Work methods • Use of personal protective equipment
Heat stress	• Possible for all decontamination and dismantling tasks	• Worker training in Contractor Safety Procedure for Work in Hot Environments • Control of work schedule (stay time) • Use of cooling rooms • Engineering controls
Cold stress	• Contaminated Soil Removal • Building Demolition • Building Backfill	• Limit outdoor activities in adverse weather conditions • Require proper dress • Provide construction space heating for all buildings occupied during decommissioning

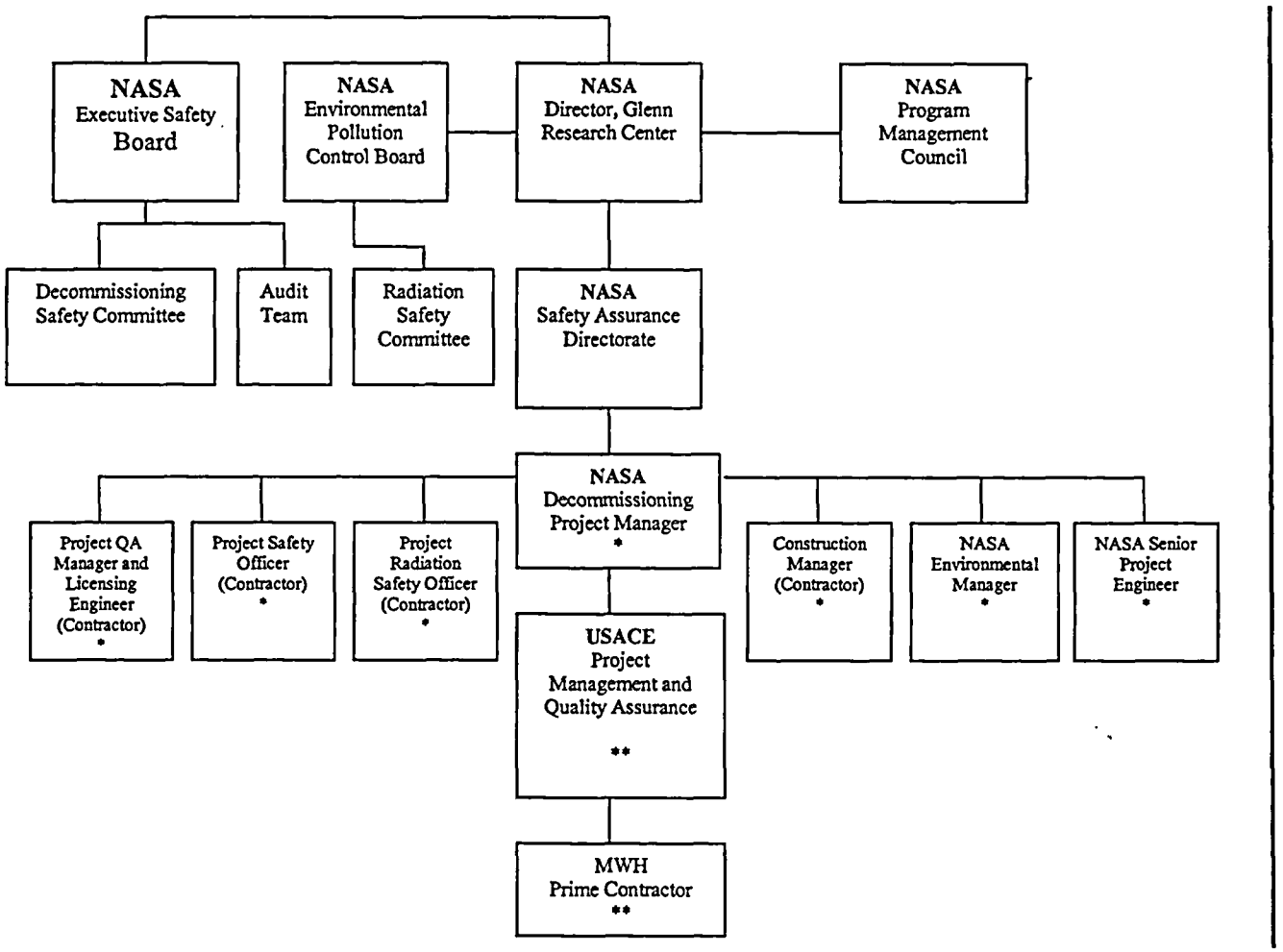
2.4 Decommissioning Organization and Responsibilities

This section describes the organizational structure that will be in place at the start of the decommissioning of the PBRF and identifies the responsibilities of key personnel in the organization. NASA is responsible for planning and managing the total decommissioning effort and has established the organizational structure to ensure that all contractors comply with the plans and programs. The NASA decommissioning organizational structure is shown in Figure 2-7. There will be two categories of contractors that will support NASA. The first category will be those individuals who work directly for NASA as part of the NASA Decommissioning Team, as described in Section 2.4.1. They will assist NASA in providing technical expertise, safety oversight, and quality assurance for the decommissioning. The second category includes the members of the Decommissioning Contractor Team. The United States Army Corps of Engineers (USACE) will manage the Decommissioning Contactor Team, and will provide the contract administration and procurement functions to acquire a Prime Contractor for NASA. USACE will also provide safety oversight and quality assurance of the decommissioning as described in Section 2.4.2.1. The Prime Contractor and other subcontractors will perform the "hands-on" decommissioning activities at PBRF. USACE has selected Montgomery Watson Harza (MWH) as the Prime Contractor. A typical Prime Contractor's organization is described in Section 2.4.2.2.

Additionally, a PBRF Decommissioning Safety Committee will be established to review and approve the administration and implementation of radiation protection and safety programs related to decommissioning as described in Section 2.4.3.

2.4.1 NASA Decommissioning Project Team

The NASA Decommissioning Project Manager will have direct responsibility for all licensed activities at PBRF including the decommissioning. The NASA Decommissioning Project Manager will be assisted by the onsite NASA Decommissioning Team. The team members will be a mix of NASA civil servant employees and NASA support contractors. The team member positions are described in Section 2.4.1.1 and include the NASA Senior Project Engineer, NASA Environmental Manager, Construction Manager, Project Safety Officer, Project Radiation Safety Officer, and QA Manager/Licensing Engineer. They will assist in directly performing the license responsibilities that cannot be delegated to the Decommissioning Contractor Team. They will also provide the QA oversight and technical support to the Decommissioning Contractor's performance. There are other NASA and NASA support service contractors that can be consulted to provide additional support as required by the NASA Decommissioning Team.



*NASA On-Site Decommissioning Team

**Decommissioning Contractor Team

Figure 2-7. Organizational Structure for the Plum Brook Reactor Facility Decommissioning Project

2.4.1.1 Key Positions in NASA Organization

Decommissioning Project Manager (NASA)

The NASA Decommissioning Project Manager will be responsible for planning and directing all decommissioning activities and will maintain ultimate responsibility for safely completing the project. The NASA Decommissioning Project Manager will review work schedules and budgets,

and will be responsible for all relevant project records. The NASA Decommissioning Project Manager will interface directly with NASA Glenn Research Center management and will serve as the primary point of contact between NASA, all members of the NASA Decommissioning Team and the USACE. The NASA Decommissioning Project Manager's responsibilities will include:

- Planning, directing, and monitoring decontamination and decommissioning activities
- Resolving work problems
- Coordinating design development for decontamination and decommissioning activities
- Reviewing decommissioning work schedules, budgets, audit reports, and other relevant documentation
- Preparing progress reports and making presentations as requested by NASA Glenn Research Center management
- Approving procurement and requests for services
- Evaluating bids and cost proposals
- Providing the licensing interface with the NRC, U.S. EPA, State of Ohio, and other regulatory agencies
- Serving as the technical spokesman for NASA on decommissioning activities
- Reporting directly to NASA Glenn Research Center management.
- Responsible for day-to-day activities at the PBRF.

The NASA Decommissioning Project Manager will have the authority to enforce safe performance of PBRF decommissioning activities and to shut down operations or activities because of either safety or environmental issues, if immediate corrective action is not taken, until a technical review has been conducted. Resumption of work will require the approval of the NASA Decommissioning Project Manager, the Senior Project Engineer, or their designees, following completion of reviews and implementation of any required corrective actions.

Minimum qualifications for the NASA Decommissioning Project Manager are twelve years in either nuclear power or decontamination and decommissioning experience, with at least four years of project management experience.

Senior Project Engineer (NASA)

The Senior Project Engineer will provide direct oversight of PBRF decommissioning for NASA Glenn Research Center management and will serve as NASA's management representative on site. The Senior Project Engineer will have direct authority over all activities that take place at the PBRF and will interface with the USACE Resident Manager. The Senior Project Engineer will serve as the normal point of contact between NASA Project Management and all USACE On-Site personnel. The Senior Project Engineer's responsibilities include:

- Providing technical oversight and guidance to the entire decommissioning process.
- Reviewing, and suggesting updates to all decommissioning Plans, Programs, and Procedures.
- Maintaining and directing the Risk Management Program for the Decommissioning Project.
- Acting as chair of the group of on-site contractor managers that direct the actions of all personnel working on site.
- Working as the interface between the Decommissioning Contractor and senior NASA Program management for all technical issues.
- Supporting the Decommissioning Safety Committee as a technical resource as Required.
- Assist the NASA Decommissioning Project Manager in ensuring that the Decommissioning Contractor's QA program is effectively implemented.
- Assisting the Decommissioning Project Manager as required, including acting as the Alternate when the Project Manager is away from the site.

The Senior Project Engineer will have the authority to enforce proper work practices during the decommissioning and will have the authority to shut down operations or activities because of safety or environmental issues, if immediate corrective action is not taken, until a technical review has been conducted. Resumption of work will require the approval of the NASA Decommissioning Project Manager, the Senior Project Engineer, or their designees, following completion of reviews and implementation of any required corrective actions.

The Senior Project Engineer will have specific knowledge of the history and condition of the Plum Brook Reactor Facility, and general knowledge of the current state of decommissioning and decontamination technology in use in industry. Minimum qualifications include a bachelor's degree in engineering, a minimum of six years experience as an engineer dealing with issues of

nuclear safety and operations, and two years experience dealing with federal and state regulatory agencies.

Environmental Manager (NASA)

The Environmental Manager will be responsible for all environmental aspects of the decommissioning project. The Environmental Manager will interface with both On-Site and Off-Site USACE Environmental personnel. Specific responsibilities include:

- Ensuring that the Decommissioning Contractor properly executes the Environmental Management Plan, and that the requirements of the Glenn Research Center Environmental Program are held as a minimum standard.
- Preparation of environmental permit applications as required by Federal, State, or Local Environmental Regulations.
- Participation in the review of contractor programs and procedures to ensure NASA programs are followed during decommissioning activities at the PBRF.
- Conducting surveillance programs and investigations to ensure that contractor's environmental programs are implemented.
- Identifying locations, operations, and conditions that have the potential for causing environmental problems.
- Assist in maintaining and directing the Risk Management Program for the Decommissioning Project.
- Working with the appropriate local and state agencies to make sure all appropriate permits are in place in a timely manner.
- Overseeing the preparation and loading of all hazardous waste (as defined in 40 CFR 261.3) shipments, and signing all EPA Hazardous Waste Manifests (required by 40 CFR 262.20) for NASA, as the waste generator.
- Assist the NASA Decommissioning Project Manager in ensuring that the Decommissioning Contractor's QA program is effectively implemented.

The Environmental Manger will have the authority to enforce proper environmental practices during the decommissioning and will have the authority to shut down operations or activities because of safety or environmental issues, if immediate corrective action is not taken, until a technical review has been conducted. Resumption of work will require the approval of the NASA Decommissioning Project Manager, the Senior Project Engineer, or their designees, following completion of reviews and implementation of any required corrective actions.

The Environmental Manager will have specific training in the environmental sciences and regulations, and will have experience applying this knowledge to managing a remediation or an environmental restoration program. Minimum qualifications for the Environmental Manger are a bachelor's degree in Biological or Environmental science or engineering or the equivalent, with a minimum of two years applied environmental management experience similar to that which will be encountered in the PBRF decommissioning project.

Construction Manager (NASA Support Contractor)

The Construction Manager will be NASA's representative in the field. The Construction Manager will work with the USACE Construction Representative and other USACE personnel to ensure that work is done in a safe, efficient manner. The Construction Manager will assist the decommissioning contractor personnel in coordination activities to prevent conflicts, and will help resolve any site issues. The Construction Manager will also have the ability to call upon the engineering resources within the NASA organization to assist in any technical issues. The Construction Manager's responsibilities will include:

- Provide direct oversight of PBRF decommissioning for NASA
- Reviewing work procedures
- Assisting the Decommissioning Contractor in technical and safety issues
- Reviewing the methodology/tooling for decontamination and decommissioning activities
- Overseeing decontamination and decommissioning activities
- Assisting the NASA Decommissioning Project Manager in all construction issues
- Coordinating Decommissioning Contractor activities on site
- Reporting to the NASA Decommissioning Project Manager on work progress and site issues
- Resolving site issues
- Drawing upon NASA engineering resources as needed
- Assisting the NASA Decommissioning Project Manager in ensuring that the QA program is effectively implemented.

The NASA Construction Manager will have the authority to enforce safe performance of PBRF decommissioning activities and to shut down operations or activities because of either safety or environmental issues, if immediate corrective action is not taken, until a technical review has been conducted. Resumption of work will require the approval of the NASA Decommissioning Project Manager, the Senior Project Engineer, or their designees, following completion of reviews and implementation of any required corrective actions.

Minimum qualifications for the Construction Manager are five years of field experience and three years of supervisory experience in either construction or decontamination and decommissioning.

Project Radiation Safety Officer (NASA Support Contractor)

The Project Radiation Safety Officer will be responsible for organizing, administering, and directing the radiation protection program at the PBRF during the decommissioning activities, including radiation safety and environmental health. The Project Radiation Safety Officer's responsibilities will include:

- Assisting the NASA Glenn Radiation Safety Officer in implementing the NASA Radiation Protection Program.
- Initiating or approving the radiation safety and health aspects of PBRF procedures, standards, and rules and ensuring the program is adequately operated
- Participating in design and decommissioning plan reviews where potential radiation exposure and safety could be affected
- Developing methods for keeping radiation exposures ALARA for workers and all facility personnel
- Conducting surveillance programs and investigations to ensure that occupational radiation exposures are below specified limits and ALARA
- Identifying locations, operations, and conditions that have the potential for causing significant exposures to radiation and initiating actions to minimize or eliminate unnecessary exposures.
- Monitoring health physics coverage of decontamination and decommissioning activities.
- Monitoring collective dose for the decontamination and decommissioning of the PBRF.

The Project Radiation Safety Officer will have the authority to enforce safe performance of PBRF decommissioning activities and to shut down operations or activities because of either safety radiological or environmental issues, if immediate corrective action is not taken, until a technical review has been conducted. Resumption of work will require the approval of the NASA Decommissioning Project Manager, the Senior Project Engineer, or their designees, following completion of reviews and implementation of any required corrective actions.

The Project Radiation Safety Officer will have specific training in the radiation health sciences and will have experience in applying this knowledge to managing a radiation protection program. Minimum qualifications for the Project Radiation Safety Officer are a bachelor's degree in physical science or biological science or the equivalent, with a minimum of five years of applied health physics experience in a program with radiation safety considerations similar to those for the PBRF decommissioning project.

Project Safety Officer (NASA Support Contractor)

The Project Safety Officer will be responsible for safety and security of the PBRF, including industrial safety, industrial hygiene, and physical security, during decommissioning activities. The Project Safety Officer will interface with the USACE Project Safety Officer on all safety issues. The Project Safety Officer's responsibilities will include:

- Assisting the Plum Brook Safety Officer and the Glenn Safety Officer in implementing the NASA Safety Program.
- Implementing the NASA industrial safety, industrial hygiene, and physical security programs through PBRF procedures, standards, and rules
- Participating in review of contractor programs and procedures to ensure NASA programs are followed during decommissioning activities at the PBRF
- Conducting surveillance programs and investigations to ensure that contractors safety programs are implemented
- Identifying locations, operations, and conditions that have the potential for causing significant exposures to industrial hazards and initiating actions to minimize or eliminate unnecessary exposures or risks
- Performing daily site walkdowns
- Reviewing contractor work procedures and safety plans.
- Assisting the NASA Decommissioning Project Manager in ensuring that the Decommissioning Contractor's QA program related to Safety and Health is effectively implemented.

The Project Safety Officer will have the authority to enforce safe performance of PBRF decommissioning activities and to shut down operations or activities because of either safety or environmental issues, if immediate corrective action is not taken, until a technical review has been conducted. Resumption of work will require the approval of the NASA Decommissioning Project Manager, the Senior Project Engineer, or their designees, following completion of reviews and implementation of any required corrective actions.

The Project Safety Officer will have specific training in the safety, security, and industrial health sciences and will have experience in applying this knowledge to managing a NASA safety program during decommissioning. Minimum qualifications for the Project Safety Officer are a bachelor's degree in physical or biological science, or the equivalent experience, with a minimum of two years of applied safety and industrial health experience similar to that which will be encountered in the PBRF decommissioning project.

Project QA Manager/Licensing Engineer (NASA Support Contractor)

The QA Manager/Licensing Engineer will assist the NASA Decommissioning Project Manager and Construction Manager in the planning and directing of decontamination and decommissioning

activities. He/She will interface directly with the USACE Resident Engineer on all technical issues and with the NRC on all licensing issues through the NASA Decommissioning Project Manager. He/She will review work procedures, and project documents, as well as project records. The responsibilities will include:

- Provide technical guidance and interpretations to project management on regulatory matters and quality issues.
- Reviewing methodology/tooling for decontamination and decommissioning activities for compliance with appropriate quality standards and regulatory requirements.
- Assuring that appropriate quality standards are invoked in project work activities
- Monitoring of daily work activities
- Preparing any licensing amendments or changes to Technical Specifications and other license basis documents
- Preparing state or local license applications or permits, as well as any amendments or changes
- Interfacing with NASA on any NRC or state licensing issues
- Maintaining the NASA Project Quality Assurance Plan and assuring that effective audit and corrective action programs are implemented
- Ensuring that the NASA and the Decommissioning Contractor's QA program are effectively implemented.

The Quality Assurance Manager/Licensing Engineer will have the authority to enforce safe performance of PBRF decommissioning activities and to shut down operations or activities because of safety, quality, regulatory, or environmental issues, if immediate corrective action is not taken, until a technical review has been conducted Resumption of work will require the approval of the NASA Decommissioning Project Manager, the Senior Project Engineer, or their designees, following completion of reviews and implementation of any required corrective actions.

Minimum qualifications for the QA Manager/Licensing Engineer are a bachelor's degree in science or engineering or equivalent, five years of nuclear or decontamination and decommissioning experience, and two years of NRC licensing experience.

2.4.2 Decommissioning Contractor Team

The Decommissioning Contractor Team consists of the USACE personnel, support service contractors, the Prime Contractor, and subcontractors. USACE personnel and their support service contractors will provide the contract administration and procurement functions to acquire a Prime Contractor for NASA. USACE will then monitor the Prime Contractor and their subcontractors and their QA program to insure it is consistent with the NASA QA program for this project. The USACE contract vehicle for the PBRF decommissioning is a Total Environmental Restoration Contractor (TERC). A TERC is a prime contractor that can manage all aspects of a large-scale environmental remediation project. The specific TERC for the PBRF decommissioning is

Montgomery Watson Harza (MWH). They may employ multiple subcontractors companies with nuclear reactor decommissioning experience and expertise.

NASA selected the USACE to build and manage the Decommissioning Contractor Teám for the PBRF decommissioning, and will ensure that all contractors subsequently selected by the USACE are selected through established procurement procedures and standards requiring a rigorous source evaluation and review process. The review and evaluation specifications will define scope and method of selection and criteria for contractor qualifications, experience, and reputation. Schedules and specific tasks to be performed by contractors will be planned in advance and detailed work procedures will be developed. Prerequisites, such as safety, health, and environmental precautions and protective clothing requirements, will be defined in writing before work is started. All contractors will adhere to NASA procedures delineating the policies and administrative guidelines applicable to the PBRF decommissioning project, and work will be performed in accordance with NASA safety and environmental requirements.

2.4.2.1 Key Positions in the USACE On-Site Organization

USACE will maintain a staff that will typically include, but not necessarily be limited to, a Project Manager, a Resident Engineer, a Health Physicist, a Project Safety Officer, a Construction Representative, a Project Scientist, a Financial Analyst, a Project Administrator and Contracts Administrative Support Staff, and Engineering Support Staff. Not all USACE staff will be stationed at the PBRF. However, USACE can use various experts and engineers from other District offices as needed based on project requirements.

Provided below are the primary roles and responsibilities of the USACE project personnel. Project personnel, particularly those onsite, may fill more than one role. Additional USACE staff/disciplines may be utilized throughout the project on an as needed basis.

Project Manager (PM)

The USACE Project Manager, is responsible for the overall management and execution of the construction phase of the project. The PM is the primary USACE interface with NASA GRC and will provide NASA GRC all required project information based on input from the USACE team. The PM will be responsible for maintaining the Operations Plan, and will review work schedules and budgets, and will be responsible for all relevant project records. The USACE Project Manager will also have the ability to call upon the engineering and construction resources within USACE to assist with any technical issues, and will interface directly with USACE senior management.

Resident Engineer (RE)

The Resident Engineer, reports to the USACE Project Manager and is responsible for the execution of all remedial activities. The RE will provide direct oversight and management of the USACE field office at Plum Brook Station and will serve as the Contracting Officer's Representative and issue technical directives to MWH for remedial activities. The RE will have

direct authority over all MWH construction activities that take place at the site and will interface directly with the MWH Project Manager and NASA's Project Manager. The RE will direct MWH in coordinating activities to prevent conflicts, and will help resolve construction issues. The RE will also have the ability to call upon the engineering and construction resources within USACE to assist with any technical issues. The responsibilities of the USACE Resident Engineer will include:

- Direct the contractor to comply with all the requirements of the Decommissioning Plan.
- Review and ensure implementation of the approved Plans and Procedures.
- Monitor safety and work schedule.
- Ensure that all required licenses or permits are approved and available on site.
- Provide technical direction to the Prime Contractor as required.

Minimum qualifications for the Resident Engineer include a bachelor's degree in physical sciences, life sciences, engineering, or equivalent, with a minimum of five years project management experience leading multi-disciplinary teams engaged in projects involving hazardous, toxic, and radioactive materials.

Health Physicist

The USACE Health Physicist will serve as the USACE Radiation Safety Officer and will be responsible for organizing, administering, and directing the Contractor's implementation of the radiation safety protection program at the site during the decommissioning activities, including radiation safety and environmental health. The Health Physicist will assist the Resident Engineer in assuring implementation of the NASA Radiation Protection Program at the PBRF during the decommissioning activities, including safety and environmental health. The Health Physicist's responsibilities will include:

- Reviewing and assuring implementation of radiological/ALARA engineering and analysis for special jobs.
- Ensuring and monitoring health physics coverage of decontamination and decommissioning activities.
- Reviewing survey reports.
- Reviewing radioactive material shipping manifests.
- Assisting in investigations of incidents and accidents.
- Ensuring implementation of the health physics procedures/guidelines for the PBRF Decommissioning Project.
- Ensuring implementation of sampling and survey plans.
- Assisting the Prime Contractor Health Physics Supervisor in the implementation of radiation protection policies and procedures.

- Monitoring and ensuring that tools and equipment will be inspected and tested by a competent mechanic or technician and certified to be in safe operating condition before use.

Minimum qualifications for the Health Physicist include a bachelor's degree in natural science or engineering, or equivalent, with a minimum of five years experience as a Health Physicist dealing with issues of nuclear safety and operations, and two years experience dealing with federal and state regulatory agencies.

USACE Project Safety Officer

The USACE Project Safety Officer will be responsible for safety and security during the decommissioning activities, including industrial safety, industrial hygiene, and physical security.

The USACE Project Safety Officer will assist the Resident Engineer in the implementation of the Prime DC's QA Program to include establishing acceptable standards of workmanship and testing. The responsibilities of the Project Safety Officer will include:

- Ensuring implementation of the approved Plans and Procedures.
- Assisting the Health Physicist in the implementation of radiation protection policies and procedures.
- Ensuring compliance with site health and safety policies and procedures.
- Observing the Prime Contractor's testing and inspection procedures, either personal observation or delegation to other Government personnel.
- Advising Prime Contractor to stop work if identified deficiencies were not corrected.
- Submitting Quality Assurance Reports.

Minimum qualifications for the Project Safety Officer include a bachelor's degree in a safety oriented program or the equivalent, with a minimum of five years experience in construction and hazardous, toxic, and/or radioactive material projects.

Construction Representative

The USACE Construction Representative will be responsible for monitoring MWH's work activities in the field, and will report directly to the Resident Engineer.

The Construction Representative will assist the Resident Engineer and Project Safety Officer in implementing the Prime Contractor's QA Program to include establishing acceptable standards of workmanship, and testing. The responsibilities of the Construction Specialist will include:

- Ensuring implementation of the approved Plans and Procedures.

- Assuring that required sampling and testing are observed, that major deficiencies are documented, and that corrective actions are taken.
- Assuring that test results are reported in the QA reports.
- Assuring that materials, supplies tools, and equipment are appropriate for the work performed.
- Assuring that work procedures are followed.
- Assuring that the Prime Contractor's required Quality Control duties are fulfilled and that necessary actions are taken to correct deficiencies.
- Performing detailed inspection or testing on work in progress to assure compliance with the Decommissioning Plan.

Minimum qualifications for the Construction Representative are five years experience in construction and hazardous, toxic, and radioactive material projects, and knowledge of OSHA safety standards, and knowledge of the project QA program.

Financial Analyst

The USACE Financial Analyst will monitor the specific financial aspects of the MWH contract such as general review and maintenance of the pay vouchers and official files, and will assist in preparing monthly financial reports as required.

Project Scientist

The USACE Project Scientist, is responsible for the quality assurance of engineering and design packages. The Project Scientist's responsibilities include scope development, preparation of the Government's cost estimates, negotiating task order modifications with MWH, monitoring schedule and budget, and other management and administrative requirements to successfully complete the work. The individual is responsible for ensuring the project complies with all federal and state environmental regulations. The Project Scientist will coordinate with regulatory agencies to determine the necessary permit requirements, and provide that information to MWH for execution. The Project Scientist also serves as an Alternate Contractor Officer's Representative.

Project Administrator and Contract Administrative Support Team

USACE will provide an Administrative Support Team to support the decommissioning of the PBRF and the administration of the MWH contract. Members of the Administrative Support Team will provide support in the areas of legal, government property management, financial management, contracts, and procurement. These individuals will support the project as necessary to keep USACE management informed of project progress and will support internal reviews and audits required for the project.

Engineering Support Team

USACE will provide engineers in various disciplines (civil, mechanical, electrical, environmental, and safety) to support the safe decommissioning of the PBRF as required. The USACE PM or RE will request the services of these individuals from various USACE District offices, as necessary, to support the project.

2.4.2.2 Key Positions in the Prime Contractor's Organization

Through use of the USACE TERC contract vehicle process, the Prime Contractor will provide all decontamination and dismantling services and related support activities during the decommissioning of the PBRF. The Prime Contractor will schedule, supervise, and perform the decommissioning operations. NASA will ensure that all contractor activities are safely performed and comply with 10 CFR Part 20 and other applicable regulations, license conditions, the decommissioning order issued by NRC, and the decommissioning plan. The Prime Contractor will be responsible for ensuring that the appropriate decommissioning contractor staff is trained in:

- Performing work in radiation areas
- Setting up work areas and the equipment and services necessary for safely accomplishing the work
- Scoping and preparing detailed procedures
- Providing sequencing and scheduling
- Processing, packaging, shipping, and disposing of radioactive materials

The Prime Contractor will have responsibility for ensuring the safety and health of their employees and for complying with Occupational Safety and Health Administration (OSHA) and NRC requirements. All these efforts will be subject to the review, approval, and authority of the NASA Decommissioning Project Team to ensure compliance with NRC requirements, license conditions, and NASA safety and health requirements.

The Prime Contractor's Organization may include, but not necessarily be limited to:

- Project administration personnel
- Project engineers
- Scheduling and field supervisors
- Support personnel including property custodians, maintenance electricians, mechanics, and janitors.

NASA envisions the key positions in the Prime Contractor's organization will be similar to the position descriptions presented below. It is understood that the Prime Contractor's organizational structure will change during the Project's lifetime and these duties may be split between a group of managers or staff and may be consolidated under fewer personnel when the workload permits.

Prime Contractor Project Manager

The Prime Contractor Project Manager will plan and direct the decommissioning of the PBRF. The Prime Contractor Project Manager will review work procedures and cost plans, and maintain all relevant decommissioning project records. The Prime Contractor Project Manager will manage and report on progress of decommissioning activities. The USACE Resident Engineer will interface directly with the NASA Decommissioning Project Manager and act as the single point of contact between NASA management and the decommissioning contractors. The Decommissioning Contractor Project Manager's responsibilities will include:

- Planning, directing, and monitoring decommissioning activities
- Resolving work problems
- Reporting directly to the USACE Resident Engineer
- Coordinating design development for decontamination and dismantlement activities
- Reviewing decontamination and dismantlement work procedures, work requests, cost plans, QA plans, and other relevant documentation
- Reviewing budgets and schedules
- Investigating potential improvements in decontamination and dismantlement methods and tooling and recommending cost-effective modifications to procedures
- Preparing progress reports and making presentations as requested by NASA management
- Approving procurement and request for services
- Evaluating bids and cost proposals.

Minimum qualifications for the Prime Contractor Project Manager are ten years of either nuclear power or decontamination and decommissioning experience, with at least five years of project management experience.

Prime Contractor Project Engineer

The Prime Contractor Project Engineer will assist the Prime Contractor Project Manager in planning and directing decontamination and decommissioning activities. The Project Engineer will prepare work procedures, cost plans, and project documents, as well as maintain the project records. The responsibilities of the Prime Contractor Project Engineer will include:

- Supervising and reporting on work progress
- Developing methodology/tooling for decontamination and dismantlement activities
- Preparing budgets and work schedules
- Planning and monitoring daily work activities
- Initiating procurement and request for services.

Minimum qualifications for the Prime Contractor Project Engineer are a bachelor's degree in science or engineering and five years of decontamination and decommissioning experience.

Prime Contractor Environmental, Health, and Safety Supervisor

The Prime Contractor Environmental, Health, and Safety Supervisor will be responsible for ensuring that the project activities are executed in compliance with site, local, and federal regulations. The Environmental, Health, and Safety Supervisor's responsibilities will include:

- Provide safety and health training including general employee training, radiation worker training, and respiratory protection training
- Coordinate baseline medical qualification and tests, whole body counts, and respiratory fit tests
- Implement, and enforce EH&S requirements
- Perform periodic site walk-downs
- Identify safety deficiencies that could result in bodily injury or damage to property
- Conduct baseline surveys on all construction type work and present the results and required personnel protection requirements in Job Safety Analyses
- Prepare the contractor EH&S plans
- Reviewing subcontractor EH&S plans
- Attend pre-job briefings to maintain cognizance of planned activities and participating in the review of safety requirements
- Perform tool and equipment inspections
- Review new tools and processes to be used during decontamination and dismantlement activities
- Ensure alarm systems are functional and tested as required
- Ensure environmental compliance
- Investigate accidents and incidents, recommend corrective actions, and report the results

- Develop required safety drills and implement such drills to meet the requirements in established plans

Minimum qualifications for the Prime Contractor Environmental, Health, and Safety Supervisor are five years of experience in industrial/construction safety or a Certified Industrial Hygienist, or a Certified Safety Professional.

Prime Contractor Health Physics Supervisor

The Prime Contractor Health Physics Supervisor will be responsible for providing basic health physics support for decommissioning the PBRF. The Health Physics Supervisor will serve as the principal interface between the Prime Contractor Project Manager and the health physics staff. The responsibilities of the Prime Contractor Health Physics Supervisor will include:

- Provide radiological/ALARA engineering and analysis functions
- Coordinate health physics oversight of decommissioning activities
- Monitor work activities to control radiation worker exposures to radiation and radioactive material
- Document radiological conditions associated with decommissioning activities
- Provide personnel protective equipment as required
- Maintain an inventory of radiological survey and analysis equipment suitable to the scope of decommissioning activities
- Perform trend analysis of dose information to identify potential problems and designing corrective actions
- Investigate radiological incidents
- Prepare health physics procedures/guidelines
- Participate in radiation protection training specific to project activities
- Maintain database of the radiological conditions of all areas of the PBRF
- Reviewing and interpreting radiation protection policies and procedures.

Minimum qualifications for the Prime Contractor Health Physics Supervisor are ten years of health physics experience in either nuclear power or decontamination and decommissioning or a Certified Health Physicist with five years of nuclear power or decontamination and decommissioning experience.

Prime Contractor Site Supervisor

The Prime Contractor Site Supervisor will be responsible for implementing the work plans associated with the PBRF decommissioning project, and will serve as the point of contact between the Prime Contractor Project Manager and the hands-on workers. The work crews performing the actual decontamination and dismantlement activities, including the various subcontractors that may be employed to perform specialized tasks (i.e., asbestos removal, lead paint abatement, equipment repair, rigging services, etc.), will report to the Prime Contractor Project Manager.

The Prime Contractor Site Supervisor will be responsible for ensuring:

- All tasks are completed in a safe and timely manner
- Timely collection of facility, environmental, safety, and health data
- Radiological protection of workers and the environment
- All workers that may be exposed to radioactive materials are properly trained in ALARA procedures
- All workers have received the training required to perform their work in a safe manner
- Site security
- All work is performed in accordance with the appropriate plans and procedures
- Coordination of work between the different work crews and subcontractors in all areas of the PBRF.
- Supervise and work progress

The minimum qualifications for the Prime Contractor Site Supervisor are ten years of supervisory experience in either construction or decontamination and decommissioning.

2.4.3 Decommissioning Safety Committee (NASA)

The Decommissioning Safety Committee will be established to conduct reviews of all matters with safety implications relative to the decommissioning of the PBRF. The Committee will have the authority to review any and all programs, plans, and procedures that may have an impact on the safety and health of workers and the public to ensure compliance with all applicable federal, state, and local regulations. The Committee will also be available to provide advice, technical expertise, and guidance to minimize health hazards associated with decommissioning activities. The authority to fulfill this responsibility and perform these functions will be granted by the Chairman of the Glenn Executive Safety Board (Figure 2-7).

The Committee will provide an executive level overview of activities at the PBRF. A prime consideration of the Committee's activities will be to ensure that all public and employee radiation

exposures are maintained as low as reasonably achievable. Members of the Decommissioning Safety Committee will include:

- Decommissioning Program Manager (NASA)
- Radiation Safety Officer (NASA)
- Chief, Construction Management Branch (NASA)
- GRC Safety Officer (NASA)
- GRC Environmental Management Office Chief (NASA)
- 2-NASA Engineers - Nuclear, Environmental, Safety, Civil, Structural, Mechanical, Electrical

One of the above committee members will serve as chair for the committee.

The following personnel will be available to support the activities of the Decommissioning Safety Committee:

- Project Radiation Safety Officer (Contractor)
- Project Safety Officer (Contractor)
- QA Manager/Licensing Engineer (Contractor)
- Any other NASA or Contractor personnel the committee deems appropriate.

The Committee will meet twice a year, with additional meetings scheduled on an as-needed basis. A quorum of the Committee shall be two-thirds of the members, but not less than three members, whichever is greater. In specific instances the Committee will designate the Chairman to act in its stead, and the Chairman will report his or her actions to the Committee at its next regular meeting. Meeting minutes will be distributed to all members and be retained on file.

The Chairman of the Decommissioning Safety Committee shall have the following qualifications:

- A bachelor's degree in engineering or a related physical science.
- Be knowledgeable in radiation hazards and radiation protection.
- Have successfully completed an orientation on the PBRF and the Decommissioning Project, as provided by the Decommissioning Project Manager.

2.5 Training Program

This section describes the training program that will be used during decommissioning of the PBRF. All field personnel (NASA and contractors) assigned to work at the PBRF will meet NASA training and certification requirements and applicable regulatory requirements. NASA employees and contractors will receive training on the decommissioning plan. More specific training for

workers will be commensurate with their duties and responsibilities and the magnitude of the potential exposure to direct radiation and contamination. The objectives of training are five-fold: (1) provide workers with information about radiologically and chemically hazardous substances, sources and types, exposure routes, and effects, (2) provide information on the radiation protection program for the decommissioning activities to enable each worker to comply with safety and health rules and to properly respond to all conditions, (3) provide instruction in the fundamentals of radiation and chemical protection to enable workers to meet ALARA objectives, (4) provide information and training on personal protection equipment, monitoring instruments, and equipment available and how to use them, and (5) instruct workers about applicable Federal, State, and PBRF radiation protection rules concerning safety and health.

The NASA Decommissioning Project Manager will assure that a policy on the personnel training program is developed and that training records are maintained. The Project Radiation Safety Officer will participate in developing and approving training programs related to work involving radiation exposure. The Project Safety Officer will participate in developing and approving training programs related to industrial safety.

All personnel assigned to work at PBRF will be given instruction in the fundamentals of radiation protection and industrial safety annually. The degree of instruction will be determined by work assignment and will ensure that workers understand how radiation protection and industrial safety relate to their jobs. The minimum training provided to any worker will include, but not necessarily be limited to, the following subjects:

- Principles of radiation protection
- Radiation monitoring techniques
- Radiation monitoring instrumentation
- Emergency procedures
- Radiation hazards and controls
- Concepts of radiation and contamination
- Provisions of 10 CFR Parts 19 and 20
- NRC license conditions and limitations
- Responsibilities of workers and supervisors
- Reporting requirements for workers
- Exposure control procedures
- Biological effects of radiation
- Radiation control zones procedures
- Radiation Work Permits
- Job Safety Analysis
- Environmental requirements and procedures including air, water, and soil

- Environmental management procedures
- Solid and hazardous waste management
- Confined space entry awareness
- Lead awareness
- Asbestos awareness
- Hazardous materials awareness
- Quality Assurance principles and requirements

Personnel will also be instructed in NASA's management commitment to implement ALARA, what ALARA means, why it is important, and how they implement it on their jobs.

Copies of applicable Federal regulations and PBRF radiation protection rules and procedures will be available where workers will have access to them for review and for use as needed. Workers will be tested upon the conclusion of training and retested on their understanding of the training at designated frequencies.

Records of individual training and qualifications will be maintained and will include the trainee's name, training date, subjects covered during training, equipment for which training was received, written test results, and the instructor's name.

The Executive Safety Board (Figure 2-7) will be responsible for ensuring that the Decommissioning Safety Committee or Audit Team conducts reviews or audits so that certification requirements and minimum standards are met and for personnel training and certification documentation are proper and consistent with applicable requirements.

2.6 Decontamination and Decommissioning Documents and Guides

The Decommissioning Plan for the PBRF has been written using the guidance and format specified in Chapter 17 of "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" (NUREG-1537) (NRC 1996). The radiological criteria for license termination to allow unrestricted use will be as set forth in 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination," and will follow the NRC guidance in Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination" (NRC 1998a). NASA will use these main documents for its decommissioning effort. NASA will also use the other regulations, regulatory guides, and standards listed below.

Code of Federal Regulations:

10 CFR Part 19	"Notices, Instructions and Reports to Workers; Inspections"
10 CFR Part 20	"Standards for Protection Against Radiation"

10 CFR Part 30	"Rules of General Applicability to Domestic Licensing of Byproduct Material"
10 CFR Part 50	"Domestic Licensing of Production and Utilization Facilities"
10 CFR Part 51	"Licensing and Regulatory Policy and Procedures for Environmental Protection"
10 CFR Part 61	"Licensing Requirements for Land Disposal of Radioactive Waste"
10 CFR Part 71	"Packaging of Radioactive Material for Transport and Transportation of Radioactive Material under Certain Conditions"
10 CFR Part 140	"Financial Protection Requirements and Indemnity Agreements"
29 CFR Part 1910	"Occupational Safety and Health Standards"
29 CFR Part 1926	"Occupational Safety and Health Standards for Construction"
49 CFR Parts 170-199	"Department of Transportation Hazardous Materials Regulations"
40 CFR Parts 260-265	"Protection of the Environment"

NRC Regulatory Guides:

DG-4006	"Demonstrating Compliance with the Radiological Criteria for License Termination"
1.86	"Termination of Operating Licenses for Nuclear Reactors"
1.187	"Guidance for Implementation of 10 CFR 50.59, Changes, Test, and Experiments"
8.2	"Guide for Administrative Practices in Radiation Monitoring"
8.4	"Direct-Reading and Indirect-Reading Pocket Dosimeters"
8.7	"Occupational Radiation Exposure Records Systems"
8.9	"Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program"
8.10	"Operating Philosophy for Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable"
8.13	"Instruction Concerning Prenatal Radiation Exposure"
8.15	"Acceptable Programs for Respiratory Protection"

ANSI Standards:

ANSI N323-1978	"Radiation Protection Instrumentation Test and Calibration"
ANSI/ANS 15.1-1990	"The Development of Technical Specifications for Research Reactors"
ANSI/ANS 15.16-1982	"Emergency Planning for Research Reactors."
ANSI N42.17A-1989	Performance Specifications for Health Physics Instrumentation – Portable Instrumentation for use in Normal Environmental Conditions
ANSI N42.17B-1989	Performance Specifications for Health Physics Instrumentation – Occupational Airborne Radioactivity Monitoring Instrumentation
ANSI N42.17C-1989	Performance Specifications for Health Physics Instrumentation – Portable Instrumentation for use in Extreme Environmental Conditions
ANSI N42.12-1980	Calibration and Usage of Sodium Iodide Detector Systems
ANSI N42.14-78	Calibration and Usage of Germanium Detectors for Measurement of Gamma-Ray Emission of Radionuclides
ANSI/IEEE STD 325-1986	IEEE Standard Test Procedures for Germanium Gamma-Ray Detectors

Regulatory Guidance and Documents:

NUREG-1505	"A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys"
NUREG-1507	"Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions"
NUREG-1537	"Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors"
NUREG-1549	"Using Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination, Draft"
NUREG-1575	"Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)"
NUREG/CR-1756	"Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors and Addenda"

NUREG/CR-6410 "Nuclear Fuel Cycle Facility Accident Analysis Handbook"

NUREG/CR-6676 "Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Codes"

Other Standards and Documents:

EM 385-1-1	U. S. Army Corps of Engineers Safety and Health Requirements Manual
NASA-STD-8719.9	NASA Technical Standard for Lifting Devices and Equipment
LeR-M0530.001	Glenn Research Center Safety Manual

3. PROTECTION OF THE HEALTH AND SAFETY OF RADIATION WORKERS AND THE PUBLIC

3.1 Radiation Protection

This section describes NASA's ALARA and health physics programs that will be in effect during decontamination and decommissioning of the PBRF. Related sections of the decommissioning plan include Section 2.4, which describes organization and responsibilities, including those related to radiation protection; Section 2.5, which describes training, including that related to radiation protection; and Section 3.1.3, which provides estimates of doses resulting from decontamination and decommissioning activities.

A radiation protection program will be provided under the cognizance of the Project Radiation Safety Officer and the Decommissioning Safety Committee (shown in Figure 2-7). This program will be implemented by trained and experienced supervisory, technical, and service contractor personnel. Radiation safety personnel will normally be present at the site when decommissioning activities are in progress to provide support and health physics supervision. These services include, but are not limited to, implementing ALARA principles, radiation worker training, monitoring personnel for occupational exposures, controlling exposure, waste disposal, providing radiation monitoring equipment, performing station area and environmental surveys, and maintaining records and generating of reports as necessary to comply with NRC and license requirements.

3.1.1 Ensuring ALARA Radiation Exposures

NASA management is committed to the policy of ALARA. Every reasonable effort will be made to maintain exposure to radiation as far below the limits specified in 10 CFR Part 20 as is reasonably achievable. This goal includes not only minimizing the dose to the worker but also the collective dose to the entire decommissioning staff. This goal will be accomplished by establishing a radiation protection program that applies sound health physics principles and uses supporting equipment, facilities, and instrumentation where applicable. NASA management will ensure that departures from this policy are not made and that good radiation control practices are implemented. Periodically audits will be performed to determine how exposures might be reduced, and based on the audit results, recommend steps to reduce exposures.

In developing the decommissioning plans for the PBRF, the potential effects that specific actions would have on the environment and the general public were examined. This examination will continue throughout the decommissioning process to ensure that discharges to the environment and exposure to the public and the workers are ALARA, which is the primary radiation protection goal. All worker activities in radiologically controlled areas will be planned ahead of time to minimize exposures. Training will reinforce the principles of radiation protection of the worker. The primary elements of the ALARA program are (1) control work activities through the use of a Job Safety Analysis and Radiation Work Permit, (2) conduct pre- and post-job reviews, (3) establish the decommissioning Radiation Safety Committee for oversight, and (4) track occupational exposures for future reference.

All personnel assigned to work at PBRF will be given instruction in the fundamentals of radiation protection annually. The minimum training provided to any worker is discussed in Section 2.5. As noted, personnel will be instructed in NASA's management commitment to implement ALARA.

3.1.2 Health Physics Program

The health physics program will be implemented under the authority of the NASA Radiation Safety Officer. The Project Radiation Safety Officer or a designee will inspect and evaluate the effectiveness of procedures, rules and regulations, license conditions, standards, and radiological health safety practices. The health physics program will satisfy the following radiation protection program commitments: (1) ensure radiological safety of the public, occupationally exposed personnel, and the environment, (2) monitor radiation level and radioactive materials, (3) control distribution and releases of radioactive materials, and (4) maintain potential exposures to the public and occupational radiation exposure to individuals within the limits of 10 CFR Part 20 and at levels that are ALARA.

3.1.2.1 Dose Limits

Annual dose limits for occupational exposure and members of the public are contained in 10 CFR Part 20. Site administrative limits for exposure to radiologically trained workers will be set well below the regulatory limits to ensure compliance with the annual dose limits and for maintaining exposures ALARA. Administrative limits will be established in the project procedures for implementation of the radiological health safety program. These procedures will also specify the means by which a worker's administrative limit may be increased, and actions to be taken when exposure limits are reached or exceeded.

Occupational Exposures

The PBRF annual occupational dose equivalent limits will be consistent with 10 CFR 20.1201(a):

(1) An annual limit, which is the more limiting of:

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are:

(i) A lens dose equivalent of 15 rems (0.15 Sv), and

(ii) A shallow-dose equivalent of 50 rems (0.50 Sv) to the skin or to any extremity.

The annual occupational dose limits for minors will be 10 percent of the dose limits specified above in 10 CFR 20.1201(a). Additionally, the dose limit for the embryo/fetus during the entire gestation period because of occupational exposure of a declared pregnant woman will be 0.5 rem.

Public Exposures

The annual limit for members of the public is 0.1 rem TEDE, exclusive of the dose contributions from background radiation, medical administrations, and disposal of radioactive material in sewerage (10 CFR 20.2003) in accordance with 10 CFR 20.1301. Air emissions of radioactive material to the environment, excluding Rn-222 and its daughters, will be managed so the member of the public likely to receive the highest dose will not exceed 0.01 rem/yr in accordance with 10 CFR 20.1101.

3.1.2.2 Personnel Monitoring

All onsite personnel will be required to participate in the monitoring program for the decommissioning project. Personnel monitoring of occupational radiation exposure will be performed for all individuals who might receive a dose in excess of 10 percent of the annual limits contained in 10 CFR 20.1201(a) (see Section 3.1.2.1). Personnel may be monitored at a greater frequency depending on the requirements contained in the applicable Radiation Work Permit or as required by the NASA Project Radiation Safety Officer.

External Monitoring

External dose will be monitored using thermoluminescent dosimeters (TLD), electronic dosimeters, self-reading pocket dosimeters, or portable survey instruments. The TLD processing program will be accredited by the National Voluntary Laboratory Accreditation Program for the energies and types of radiation expected to be encountered at the site. Personnel exiting the PBRF areas having a potential for removable surface contamination will be subject to personnel surveys designed to detect contamination by use of a count rate instrument with a thin window Geiger-Mueller (G-M) detector (pancake G-M, or equivalent). If personnel contamination is identified, decontamination will be conducted and the potential skin dose equivalent will be assessed.

Internal Monitoring

Internal dose may be monitored with air samples, *in vitro* or *in vivo* bioassay techniques, or a combination of air monitoring and bioassay in accordance with 10 CFR 20.1204. If the primary method of compliance is by air monitoring, personnel with the greatest potential for intakes of radioactive material will be sampled at a frequency determined by the Project Radiation Safety Officer and based on the pulmonary retention class (days, weeks, years) of the radionuclides of concern to evaluate the effectiveness of the air monitoring program. If respiratory protection equipment is used for protection against airborne radioactive material, air monitoring and bioassays will be performed to evaluate actual intakes in accordance with the requirements of 10 CFR 20.1703 (a)(3)(ii).

3.1.2.3 Exposure Control

The primary methods to control occupational exposures at the PBRF will be by controlling facility access, communicating area hazards through proper training and postings; maintaining knowledge

of the current radiological conditions by facility monitoring; using personnel protection equipment (e.g., protective clothing and respirators); and using a Job Safety Analysis and Radiation Work Permits.

Facility Access Control

Entry to the fenced area surrounding the PBRF will be controlled by security personnel during operating hours. During non-operating hours, the gates in the fence will be locked and routine security surveillance of the PBRF will be performed. Facility access control is described in more detail in Section 6 of this plan.

Area Posting

Areas within the PBRF designated as restricted areas, radiation areas, high radiation areas, very high radiation areas, airborne radioactivity areas, and radioactive material areas will be posted in accordance with the provisions contained in 10 CFR Part 20. Control of access and locking, where required, of these areas will be as specified in the Facility Technical Specifications.

Facility Monitoring

Facility monitoring is the routine, periodic determination of the direct radiation level and radioactivity within the PBRF. Facility monitoring will establish the radiological conditions, provide for a permanent record of these conditions, and permit evaluation of radiological trends during the decontamination and decommissioning efforts.

Representative samples of airborne radioactive material, water, and transferable surface radioactive contaminants will be routinely collected and analyzed to ensure that the radioactive materials at PBRF are being adequately contained. Direct radiation monitoring will also be performed.

Portable direct-reading radiation survey instruments and air sampling equipment will be available for facility monitoring. Types and frequency of surveys will be scheduled to comply with 10 CFR 20.1501.

Respiratory Protection Program

A respiratory protection program will be established to support decommissioning activities and will be designed to comply with the guidelines in NRC Regulatory Guide 8.15, *Acceptable Programs for Respiratory Protection* (NRC 1999). Wherever practicable, engineering controls will maintain airborne concentrations ALARA. Unwarranted use of respiratory protective equipment will not be permitted and is considered contrary to the ALARA principle because of the increased time required to perform individual tasks and the increase in physiological stress. Where there is a potential for significant intakes of radioactive material and the TEDE may be maintained ALARA, respiratory protection equipment will be worn and allowance will be made for its use in estimating exposures.

The Decommissioning Contractor will select respiratory protection equipment that provides a protection factor greater than the multiple by which peak concentrations of airborne radioactive materials in the working area are expected to exceed the values specified in 10 CFR Part 20 Appendix B, Table 1, column 3. If selecting such a respiratory protection device is inconsistent with the goal of keeping the TEDE ALARA, the Decommissioning Contractor may select respiratory protection equipment with a lower protection factor. Before selecting respiratory protection equipment, the Radiation Safety Officer, or designee, will conduct a hazard assessment of operations that use radioactive materials to determine the need for radiological respiratory protection. When assessing area(s) and condition(s), the following (as a minimum) will be taken into consideration before selecting the appropriate equipment:

- Radioactive materials sampling results
- Removal efficiency of ventilation controls
- Removable contamination levels
- Radionuclides
- Resuspension factors
- Area dose rates
- General conditions, including equipment and materials used and worker activity
- 10 CFR Part 20-derived air concentrations
- Feasibility of engineering controls to reduce employee exposure below the exposure limit
- Degree of protection provided by the respirator.

Hazard Analysis

The potential hazards presented to workers performing decommissioning tasks are assessed and controlled using two processes. Radiological hazards are evaluated by trained and qualified radiation protection personnel. Where radiological hazards are determined to be present, a Radiation Work Permit (RWP) is prepared. The RWP identifies the radiological conditions that are present for the task being performed and specifies the radiological protection and monitoring measures to be imposed during performance of the task. Similarly, the industrial safety hazards associated with proposed tasks are assessed by qualified industrial health and safety staff. The hazards are identified in a Job Safety Analysis that includes the compensatory measures that are to be imposed to assure the safety of the workers during performance of the work. The processes for preparing RWPs and JSAs will be controlled by project procedures that implement the Radiation Protection Plan and the Health and Safety Plan. Workers will be trained on the requirements for using and complying with these documents and the other types of work control permits that may be specified.

Personnel Decontamination

The project will establish procedures for decontamination of personnel who may become radiologically contaminated during decommissioning work. The procedures will provide requirements for documentation of personnel contaminations, guidance on methods for decontamination, and will specify requirements on how dose assessments are to be performed as a result of contaminations. In addition, the project will establish procedures and guidance for the handling of personnel who may be injured in areas or under circumstances that could involve radioactive contamination. This guidance will also involve coordination with local offsite medical facilities that may need to become involved in dealing with these situations. A personnel decontamination station will be maintained on site for use in performing personnel decontaminations. It will be readily available for use and will provide collection and control of any radioactively contaminated liquids that may be produced during personnel decontaminations.

3.1.2.4 Radiation Monitoring Equipment

Appropriate radiation monitoring equipment will be available to measure the types and energies of radiation present on the site. Instruments will be controlled and tested in accordance with industry standards such as ANSI-N323-1978, "Radiation Protection Instrumentation Test and Calibration". Records of maintenance and calibrations will be maintained in accordance with these standards.

3.1.2.5 Station and Environmental Monitoring

A pre-decommissioning, environmental monitoring program will be established to provide baseline radiological data on the Plum Brook Reactor Facility and nearby off-site environment. These data will be used to ensure that decommissioning operations do not negatively impact the environment. Pre-decontamination and decommissioning environmental monitoring will include measurements of gross alpha, gross beta along with site specific radionuclides in air, soil, sediment, surface water and ground water. These measurements will be continued during the decommissioning phase to permit evaluation of radiological trends over time. Action levels shall be established on all sampling media to provide trigger levels for further investigation. Because some of the monitoring results may be subject to seasonal changes, the program will be initiated prior to commencement of the decommissioning phase.

Air monitoring is one of the major components of an environmental monitoring program, as radioactive material may become airborne. Continuous airborne monitoring will be performed at the PBRF fence line (north, south, east and west) and offsite air sampling locations. Off-site stations will be located upwind or southwest of the PBRF and downwind or northeast of the PBRF. The weekly measurement of gross alpha and gross beta airborne radioactivity will be used as a screening technique to determine the need for specific radionuclide analysis. In addition, monthly composite samples from each air monitoring station will be analyzed for gross alpha/beta and gamma spectroscopy. Direct radiation exposure shall be measured quarterly by placing a TLD at each on-site and off-site air sampling station.

Monthly groundwater sampling will be performed at up and down gradient monitoring well locations. The measurement of gross alpha and gross beta radioactivity will be used as a screening technique to determine the need for specific radionuclide analysis.

Monthly surface water sampling will be conducted in areas of surface water runoff and upstream and downstream of the Plum Brook Reactor Facility, as well as upstream and downstream of Plum Brook. The measurement of gross alpha and gross beta radioactivity will be used as a screening technique to determine the need for additional specific radionuclide analysis.

Monthly sediment samples will be taken at locations where surface water samples are collected. These samples will be analyzed by gamma spectroscopy and gross alpha/beta. The measurement of gross alpha and gross beta radioactivity will be used as a screening technique to determine the need for additional specific radionuclide analysis.

Background sediment and soil samples will be collected from a background reference area or areas having similar physical and geological characteristics as impacted areas of the PBRF site, and that are assumed not to be impacted by PBRF operations.

The environmental monitoring results obtained during decommissioning operations will be compared with baseline survey data obtained during pre-decommissioning monitoring. Analytical results from the environmental monitoring program will be reviewed against established Project Specific Action Limits. If data begins to approach or exceed these action limits, additional sampling may be performed. In addition more detailed analyses (radionuclide specific) may be performed. If confirmed, a review of operations will be performed to determine the cause of the increased values, as well as to determine appropriate measures to mitigate any further impacts. After analyses results are obtained and reviewed, the results will be available on-site for NRC review.

3.1.2.6 Records and Reports

NASA will maintain records in the following categories:

- Personnel exposure records, including results of bioassays and incidents of skin contamination
- Incidents of overexposure or injuries involving radioactive materials
- Work area, facility, station, and environmental monitoring survey records indicating sampling information and analysis results
- Survey instrument calibration records and inventory
- Personnel training in radiation safety and control.

Routine reports of conditions relating to health and safety will be prepared for NASA management. In addition, reports required in 10 CFR Part 19 and 10 CFR Part 20, relative to exposures of personnel or the release of radioactive material will be submitted to the NRC. NASA will submit an annual status (progress) report to the NRC.

3.1.3 Dose Estimates

This section presents estimated doses to radiation workers and discusses potential exposure pathways and doses to the public. Doses to workers performing decommissioning activities were estimated using the estimated labor hours for each work element identified in Section 2.3.1 and the average 1985 exposure rates documented by Teledyne Isotopes (1987). The 1985 exposure rates were corrected for decay to the year 2003, the year during which decontamination and waste removal are planned to occur.

A source term reduction strategy will be used during decontamination and removal activities. The sources of the highest radiation levels in each area will be removed first, and remaining activities will be performed in areas with much lower radiation levels. The material with the highest radiation levels (i.e., the reactor tank and internals and the activated material in the Hot Dry Storage Area) will be removed first. When removing contaminated piping and equipment, the components (e.g., valves) having the highest radiation levels will be removed first, so the remaining work can be performed at substantially lower radiation levels. The strategy of removing the highest radiation source first is generally consistent with the ALARA principle. If this strategy does not achieve ALARA principles because of space limitations, system/equipment configuration, or contamination control, the procedure that results in the least exposure will be followed.

It was assumed that all radiation doses to workers will occur through direct external exposure to ionizing radiation. Doses received from inhalation of radioactively contaminated airborne material will be mitigated by:

- Developing work procedures for decontamination and decommissioning activities, including activities that could potentially result in airborne contamination. The work procedures will incorporate ALARA concepts (see Section 3.1.1) and health physics input (see Section 3.1.2). The work procedures may require that personnel receive specific training before performing the work and may require that access to the work area be controlled during cutting and burning operations.
- Performing continuous air monitoring or air sampling in all active work areas before, during, and after all decommissioning activities.
- Using appropriate personnel protective equipment, such as respirators and supplied air respirators.
- Implementing engineering controls, such as contamination control envelopes that ensure positive containment of contamination by physical barriers or a flow of air from non-contaminated areas to contaminated areas, and discharged through filters.

During all decommissioning activities, worker doses will be controlled to stay within established project administrative limits which are well below the 10 CFR Part 20 maximum allowable annual worker dose of 5 rem/yr.

The estimated cumulative worker doses for each work element of the PBRF decommissioning project are presented in Table 3-2. These worker doses were estimated using the assumed labor hours for each task and the exposure rates measured during characterization surveys. Estimated worker doses considered only external exposure and did not include inhalation or dermal absorption

pathways. Reactor Tank Removal, and Contaminated Piping and Equipment Removal, will result in the highest worker doses. Work elements for which the estimate worker dose is zero would not involve radiation. The total dose estimated to be received by workers from decommissioning the PBRF is approximately 70 person-rem.

To estimate doses from transporting radioactive wastes, transportation doses presented for the reference light power water reactor in the "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities" (NUREG-1496) (NRC 1997) were scaled using PBRF radiation levels, exposures, and waste volumes. The scaled estimates of doses to transportation workers and the public along transportation routes from transporting radioactive waste from PBRF decommissioning are estimated to be 5 and 0.5 person-rem, respectively.

As discussed above, controls will be used to ensure that doses to the public do not exceed the TEDE constraint of 0.01 rem, or 10 mrem/yr from emissions of airborne radioactive material (10 CFR 20.1101). The release of any airborne radioactive material would be minimized as described above, and any released particles would undergo dispersion as they travel 0.8 km (0.5 mi.) to the site boundary. As shown by the accident analysis in Section 3.3.2, accidents with unfiltered releases would result in a maximum estimated TEDE of 0.53 mrem to the average member of the public. The dose is significantly lower than the 15.0 mrem whole body dose identified as the lowest action level in Table 1 of ANSI/ANS 15.16-1982.

Table 3-2. Estimated Worker Doses From Decommissioning the NASA Plum Brook Reactor Facility

Work Description	Estimated Worker Dose (person-rem)[a]
Decommissioning Planning	NE[b]
NASA Operations and Direct Support	NE
Operations Management and Support	NE
Security	NE
Health Physics	1.8
Systems Operation, Maintenance, Deactivation	0.16
Contaminated Soil Removal	0.019
Site Preparation	0.75
Asbestos Removal and Lead Paint Abatement	0.11
Loose Equipment Removal	0.19
Removal of Activated Material in Hot Dry Storage Area	3.44
Decontamination	0.75
Reactor Internals and Tank Removal	23.9
Contaminated Piping and Equipment Removal	38.1
Contaminated Concrete and Embedded Pipe Decontamination or Removal	0.47
Final Status Survey	NE
Building Demolition	NE
Building Backfill	NE
Reactor Building Backfill	NE
Total	**69.5**

a. These values are doses above the doses due to background radiation. These values are the collective dose among all crew members during the multiyear decommissioning period.

b. NE = The worker dose for these activities were not estimated. Expected doses are less than the levels requiring personnel monitoring.

3.2 Radioactive Waste Management

This section summarizes the types and volumes of radioactive waste and the processes that will be used for characterizing, packaging, transporting, processing, and disposing of radioactive waste.

Based on existing characterization information (described in Section 2.2) and planned decommissioning activities (described in Section 2.3.1), the estimated waste types, volumes, disposition, packaging, transportation method, and disposal strategy are presented in Table 3-3. The information presented in Table 3-3 is a preliminary estimate. This information will be modified based on new information and regulatory status as the project progresses.

Table 3-3. Estimate of Radioactive Waste Volumes, Packaging, Transportation, and Disposition

Waste Type (Source)	Waste Classification	Waste Volume ft^3	Typical Package Types	Transport Method	Disposition
Asbestos (Bldg. 1111, 1112, 1132, and 1133)	Class A	410	B-25 boxes	Truck or rail	Direct disposal at Class A facility
Loose paint scrapings/chips (Bldg. 1111, 1112, 1132, 1133, and 1134)	Mixed Waste (Class A radioactive waste, RCRA hazardous waste, TSCA toxic waste)	75	Drum (55-gal)	Truck or rail	Stabilization, then disposal at Class A facility
Loose Equipment (Mock-Up Reactor and low specific activity from Bldg. 1111, 1112 [including Hot Dry Storage Area waste], 1133, 1134, and 1141)	Class A	9,600	Sea-Lands (20 ft)	Truck or rail	Offsite processor for survey; release, decontamination, or Class A facility
Reactor core box and internals and activated metals (including waste in Hot Dry Storage Area)	Class A, B, & C	1,500	Metal liners (120 ft^3 & 50 ft^3)	Truck	Direct disposal at Class B/C facility in shielded Type B cask
Reactor tank and tank components (sectioned)	Class A	2,200	Sea-Lands (20 ft)	Truck or rail	Offsite processor for survey; decontamination or Class A disposal, or recycle
Fixed Components (Bldgs. 1111 and 1112)	Class A	1,900	Sea-Lands (20 ft)	Truck or rail	Offsite processor for survey; decontamination or recycle
Concrete scabbling debris & decontamination wastes (several PBRF buildings and inground structures)	Class A	3,200	B-25 boxes	Truck or rail	Direct disposal at Class A facility
Contaminated piping & equipment (incl. heat exchangers, pumps, vessels, valves) (several PBRF buildings and inground structures)	Class A	29,100	Sea-Lands (20 ft)	Truck or rail	Offsite processor for survey; decontamination, Class A disposal, or recycle
Embedded piping & contaminated concrete (several PBRF buildings and in-ground structures)	Class A	11,600	Inter-modals (25 yd^3)	Truck or rail	Direct disposal at Class A facility
Contaminated soils (Emergency Retention Basin, Pentolite Ditch, spill area)	Class A	48,200	Inter-modals (25 yd^3) or covered gondola	Truck or rail	Direct disposal at Class A facility
Dry active waste (including personnel protective clothing)	Class A	4,600	B-25 boxes or Sea-Lands (20 ft)	Truck or rail	Direct disposal at Class A facility

3.2.1 Fuel Removal

The PBRF was shut down in 1973. All nuclear fuel assemblies were removed from the PBRF between January and July 1973 and transported to a DOE facility. There has been no fuel on site since 1973.

3.2.2 Radioactive Waste Management

Waste management is an integral part of the decommissioning process and the plan includes provisions for minimizing the amount of waste generated as well as for waste collection, treatment, packaging, and shipment offsite for processing and disposal. The most cost-effective radioactive waste disposal strategy, consistent with the ALARA principle, will be selected based on evaluating available methods for processing, packaging, and transporting radioactive waste in conjunction with available disposal facilities and their waste acceptance criteria.

The decommissioning process will generate large quantities of solid radioactive waste. The major sources of solid radioactive waste are removal of contaminated and activated equipment and components, structure demolition, and excavation of contaminated soils.

There is also the possibility of generating some small quantities of liquid radioactive waste. Even though the facility's systems were previously drained, small amounts of contaminated liquids may be encountered in system piping and building sumps. In addition, liquids may be generated during decommissioning activities, such as concrete cutting operations, personnel decontamination, or possible use of decontamination solutions.

The general approach to an effective waste management program includes the following considerations:

- Minimize onsite processing and focus on efficient removal and packaging
- Optimize task planning to avoid multiple handling of waste materials, to minimize the generation of new wastes, and to efficiently control the removal, characterization, segmentation, and packaging of waste materials generated during removal of equipment and components
- Use low cost metal processors for decontamination at an offsite location
- Use concrete from above-grade portions of buildings as onsite fill to the extent possible under the NRC license termination criteria
- Dispose of waste at the licensed low-level radioactive waste disposal facility that has the lowest total cost (including processing, packaging, transportation, and burial costs)
- Recycle waste whenever feasible

- Use an efficient shipping strategy that minimizes the need to stage radioactive waste on site awaiting shipment (where practical and appropriate for ALARA considerations)

Solid radioactive waste is expected to be primarily Class A waste. Contaminated systems, equipment, and components will be segmented to facilitate packaging for shipment to a licensed vendor providing decontamination, survey for release, and volume reduction services, or for disposal. Contaminated bulk commodities (i.e., concrete, construction debris, soil, and dry active waste) will be placed in proper disposal containers for shipment directly to the appropriate Class A disposal facility.

Liquid wastes will be collected and adequately characterized to determine proper treatment and disposal necessary to comply with Federal, State, and Local regulatory requirements.

Asbestos removed during decommissioning will be properly labeled and packaged in accordance with regulatory requirements. The packaged asbestos waste that is radioactively contaminated can be directly disposed of at a Class A disposal facility without further processing.

Radioactively-contaminated lead will either be decontaminated (either onsite or at an offsite vendor facility) for recycling or packaged and shipped to a Class A disposal facility for appropriate processing and disposal.

Mixed wastes that are encountered during decommissioning activities will be handled on a case by case basis. These wastes will be properly labeled, marked, transported, and disposed of at an appropriate facility in accordance with the appropriate regulatory requirements and the disposal site waste acceptance criteria.

3.2.2.1 Waste Characterization

Proper characterization is important for waste minimization and it is required by Federal and State regulations that relate to transportation and disposal facilities. Waste materials will be surveyed and characterized as they are generated and then packaged for shipment and disposal. Procedures will be developed that adequately implement the waste acceptance criteria imposed by the licenses held by disposal sites and waste processors used by the project.

Processes will be implemented to assure that non-radioactive building demolition debris disposed of in commercial landfills meets the disposal criteria imposed by Regulation or permit requirements at the disposal facility.

Waste characterization will also include those analyses necessary to demonstrate that hazardous wastes comply with the Land Disposal Restrictions specified in Ohio EPA hazardous waste management regulations. Any waste streams suspected of containing PCBs, asbestos, or special wastes, will be characterized in accordance with Ohio EPA and other applicable regulations to ensure proper storage and disposal.

3.2.2.2 Waste Packaging and Transport

The packaging and transport of radioactive and other hazardous materials will be in compliance with the applicable NRC, DOT, and state regulations. Each type of waste is controlled by different regulations, and within the regulations there are opportunities for developing lowest cost solutions. A general strategy for packaging and transporting waste generated from decommissioning is discussed below.

Radioactive waste processors will provide reusable containers necessary to transport material to the processing facility. For one-way shipments of waste for direct disposal, various containers such as inter-modals, high integrity containers, B-25 boxes, metal liners, and special design strong tight containers will be used. Typical containers used for packaging of waste are shown in the fourth column of Table 3-3. Some types of radioactive wastes may require licensed or certified packaging and/or require transportation in shielded transport casks. The reactor tank internals could potentially be sectioned, packaged in standard 3.4-m^3 (120-ft^3) metal containers, and shipped as a Type B package in a licensed cask.

3.2.3 Radioactive Waste Disposal

Procedures will be used for the handling, staging, and shipping packaged radioactive waste in accordance with 10 CFR 20.2006, "Transfer for Disposal and Manifests"; 49 CFR 100-177, "Transportation of Hazardous Materials"; 10 CFR 61, "Licensing Requirements for Land Disposal of Radioactive Waste;" and the disposal or processing facility license conditions. The disposition of wastes generated from PBRF decommissioning activities are shown in the last column of Table 3-3. Wastes may be shipped to a licensed processing facility for disposition or may be disposed of directly at a licensed disposal facility.

3.2.3.1 Radioactive Material or Waste Shipment Manifest

Each shipment of radioactive material or waste must be accompanied by shipping documents specified in 10 CFR 20, and 49 CFR 100-177. Radioactive waste generated from PBRF decommissioning activities will be manifested consistent with its waste classification and appropriate regulations. Records of shipments will be prepared by and approved by staff that are properly trained and qualified and will be maintained as project records.

3.2.3.2 Waste Minimization

Waste disposal costs are directly related to the activity, volume, and weight of the materials requiring disposal. Strategies for minimizing waste will be implemented during the project. Examples of waste minimization methods that could be used are discussed below.

Source Reduction

Ongoing sampling and analysis activities during decommissioning will better define the range of contamination and further reduce the quantity of specific waste streams. Chemical and radiological characterization will be used throughout decommissioning to verify levels of

contamination for waste classification and disposal purposes. Characterization will ensure that waste containers leaving the PBRF are properly classified for transportation and disposal.

Use of chemicals and cleaning solutions will be minimized as much as practical. Radioactive solutions that become contaminated with chemicals or cleaners may be separated from other aqueous wastes before disposition.

Reuse

Reuse of materials in radioactively contaminated areas will minimize waste generation. Items such as 55-gal drums, spray bottles, tools, equipment, radiation sign postings, water, and air hoses, will be reused wherever possible. Water used for cutting operations and cooling processes will be filtered, monitored, and reused to the extent possible. Decontaminated areas will be maintained clean to prevent contamination of items planned to be reused.

Decontamination

Onsite decontamination will be performed only when shown to be cost effective. Decontamination efforts may include vacuuming, solvent and/or wet wiping, and scabbling to remove surface contamination, where practical. Where concrete is contaminated to limited depths, scabbling techniques may be used to separate the contaminated surface layer from the rest of the concrete so that when demolished, the concrete debris can be used onsite as backfill. Personnel protective clothing will be packaged and transported offsite to either a licensed vendor where contamination will be removed by washing or to a licensed disposal facility.

Volume Reduction

During dismantling activities, equipment, piping, and ductwork will be volume reduced, where practical, by crushing and cutting to size to eliminate void spaces in the waste packages. During loading activities, other wastes will be used to fill void spaces in burial containers, when appropriate. Techniques such as efficient packaging will be used to minimize the number of containers.

Waste Stream Segregation

Waste streams will be kept separate to the extent practical to reduce the potential for cross contamination. Solid material and trash from radiologically controlled areas will be classified onsite or shipped to an offsite processor for classification and disposition. Decontaminated areas and equipment will be clearly marked to prevent reintroduction of radioactive material.

3.2.3.3 Generation and Disposal of Liquid Radioactive Waste

The D&D process is not expected to generate appreciable volumes of radioactively contaminated water. Small volumes may be produced as a result of liquids encountered in system piping and sumps. Liquid wastes will be collected and adequately characterized to determine proper treatment and disposal necessary to comply with Federal, State, and Local regulatory requirements.

3.2.4 General Industrial Safety Program

This section describes the industrial safety program that NASA will apply during the decommissioning of PBRF. Non-radiological hazards associated with decommissioning will be managed according to the requirements of the latest revisions of the *NASA Glenn Research Center Safety Manual* and the *NASA Glenn Environmental Programs Manual.* These manuals define NASA safety and environmental requirements.

Specific authority and responsibility for industrial safety are discussed in Section 2.4 of this plan. The industrial safety program during decommissioning activities will comply with the *NASA Glenn Research Center Safety Manual* and be implemented in accordance with 29 CFR Part 1910, "Occupational Safety and Health Standards," and 29 CFR Part 1926, "Safety and Health Regulations for Construction." The NASA Decommissioning Team will oversee all industrial safety, industrial hygiene, and other environmental health services and related support activities during the decommissioning of the PBRF. The day-to-day safety oversight will be provided by the Project Safety Officer, the Construction Manager, the Project Radiation Safety Officer, and the NASA Environmental Engineer (as appropriate) as described in Section 2.4. The Decommissioning Safety Committee will conduct safety reviews of all matters with safety implications relative to the decommissioning of the PBRF, including environmental safety, industrial hygiene, and industrial safety. Any worker has the authority to shut down any operation or activity within the PBRF on a question of occupational health and safety if immediate corrective action is not taken until an appropriate technical review has been conducted.

Decommissioning activities will be performed under the control of approved procedures that specify applicable requirements and limitations to assure that industrial and radiological safety are not compromised.

The planning process for any decommissioning activity will include the preparation of a Job Safety Analysis (JSA). When appropriate, supplemental JSAs may be prepared as a work task progresses. The JSA will identify all safety risks associated with the job. Typical risks might include entry into confined spaces, electrical lock-out/tag-out, fall hazard, or work in a radiological area. If there are no radiological aspects to the job the JSA will be sufficient to cover all of the safety issues associated with the job, including the required countermeasures and permits (such as a Confined Space Permit). If a job contains a radiological risk it will also require preparation of a Radiation Work Permit (RWP). The RWP will fully address all of the radiological safety aspects of the job.

Industrial hazards may include the handling of decontaminating chemical agents, cutting with oxy-acetylene and arc-type torches, rigging for component removal, and the routine industrial hazards normally associated with construction or decommissioning. Industrial hazards will be minimized to the extent practical as described in Section 2.3.1.4. Industrial hazards that cannot be eliminated will be managed by worker training; use of written and approved procedures; and NASA overview of all work. Compliance with procedures will be audited.

Personnel trained to provide first aid and prompt response to an accident situation will be available during decommissioning activities.

The following sections outline typical provisions for industrial safety and hygiene, which will be implemented during decommissioning activities.

3.2.4.1 Occupational Health and Environmental Control

The occupational health of workers during decommissioning activities will be protected by providing adequate facilities and systems as follows:

- Training and procedures as discussed in Section 2.5
- First aid supplies within work areas
- Emergency shower and eye wash facilities
- Trained first aid personnel and ambulance service readily available from off site
- Environmental controls in the work space to include adequate ventilation and dust control, temperature control, illumination, noise control, potable water, and sanitary facilities
- Fire protection in accordance with Section 3.2.4.5.

3.2.4.2 Personal Protective Devices

Protective devices provided for workers involved in decommissioning activities will include

- Hard hats and safety shoes for protection against impact and penetration of falling or flying objects
- Gloves
- Hearing protection devices
- Eye and face protection for workers exposed to potential injury from physical or chemical agents
- Respiratory protection devices.

Operations involving protective equipment will be reviewed to ensure that the workers will not be subjected to hazards as a result of using protective equipment. Heat stress controls will be in effect whenever conditions for heat stress exist.

3.2.4.3 Hearing Conservation Program

A hearing conservation program will be established for all workers who are exposed to noise levels of 80 decibel A-weighted (dBA) or greater (as an 8-hour, time-weighted average exposure). The hearing conservation program will be in accordance with 29 CFR 1910.95, and Chapter 11 of Glenn Safety Manual (GSM). Noise control measures, including the requirement to wear hearing protection equipment, will be determined by the Decommissioning Contractor. Personnel who are assigned tasks in known noise hazardous areas (≥ 85 dBA) will be required to use hearing

protection devices. Records will be maintained that document the implementation of noise monitoring, employee training, control measures, and protective equipment.

3.2.4.4 Respiratory Protection Program

Regulatory requirements and safety program procedures will be followed to prevent worker exposure to occupational dusts, fumes, mists, radionuclides, gases, and vapors above OSHA limits as stated in 29 CFR 1910.1000. Respiratory protection measures, including the requirement to wear respirators, will be determined by industrial safety personnel in accordance with 29 CFR 1910.134 "Respiratory Protection." Only respirators approved by the National Institute for Occupational Safety and Health (NIOSH) will be used. Individual workers will be tested and certified prior to being allowed to work while wearing a respirator. Records will be maintained to document air monitoring conducted, employee training conducted, medical monitoring done, control measures implemented, and protective equipment used.

3.2.4.5 Fire Protection and Prevention

Fire protection devices will be made available during decommissioning tasks. Portable fire extinguishers will be strategically located throughout the PBRF to serve areas for the various decommissioning activities. Decommissioning employees will be trained in the use of fire extinguishers to ensure that each shift is staffed with people trained in the use of fire extinguishers. Fire protection measures will be implemented to avoid ignition hazards from electrical wiring and equipment and from combustible materials. Smoking will not be permitted within the PBRF fence. Job Safety Analysis will be performed for burning, welding, cutting, and other fire potential operations. A Hot Work Permit program will be established and implemented when appropriate.

3.2.4.6 Hand and Power Tools and Cutting Equipment

The condition of the hand and power tools used during decommissioning activities will be routinely checked for proper operation and for use in compliance with the applicable provisions of 29 CFR Part 1926, Subpart I, "Tools-Hand and Power" and 29 CFR Part 1926, Subpart J, "Welding and Cutting."

3.2.4.7 Fall Protection

Decommissioning operations will be conducted in accordance with the applicable provisions of 29 CFR 1926 Subpart M, "Fall Protection" and Subpart L, "Scaffolds."

3.2.4.8 Lifting Equipment

Lifting equipment used in the decommissioning activities will comply with the NASA Lifting Standard 8719.9, which meets or exceeds the applicable provisions of 29 CFR Part 1926, Subpart N, "Cranes, Derricks, Hoists, Elevators and Conveyors," and 29 CFR Part 1926, Subpart H,

"Materials Handling, Storage Use, and Disposal." Maintenance program requirements will be performed in accordance with this Standard and these regulatory requirements.

3.2.4.9 Excavations

Excavations required during decommissioning activities will comply with applicable provisions of 29 CFR Part 1926, Subpart P, "Excavations, Trenching and Shoring." These provisions include tapered sides of excavations, daily inspections of excavations, and bracing of sides when heavy equipment is used in the vicinity.

3.2.4.10 Working in Confined Space Areas

Operations required in the confined spaces will comply with applicable provisions of 29 CFR 1910.146. The *NASA Glenn Research Center Safety Manual* establishes specific requirements for confined space entry, and these requirements will apply to all PBRF confined space entries.

3.4.2.11 Lockout/Tagout

Lockout/Tagout procedures will be implemented in accordance with OSHA requirements in 29 CFR 1910.147. The *NASA Glenn Research Center Safety Manual* establishes specific requirements for lockout/tagout that will be applied to PBRF decommissioning activities.

3.4.2.12 Asbestos Removal

Asbestos removal operations required during decommissioning activities will comply with applicable provisions of OSHA standards 29 CFR 1926.1101 and 1910.1001 and applicable State laws. The *Glenn Environmental Programs Manual* establishes specific requirements for working with asbestos. These requirements will apply to all PBRF asbestos handling operations.

3.4.2.13 Lead Paint Removal

Lead paint removal operation during decommissioning will comply with the applicable provisions of OSHA standards 29 CFR 1926.62. The *Glenn Environmental Programs Manual* establishes specific requirements for handling, removal, and disposal of lead.

3.2.4.14 Demolition

Demolition operations will comply with the applicable portion of 10 CFR 1926 Subpart T, "Demolition."

3.3 Radiological Accident Analyses

This section identifies potential radiological accidents that could occur during decommissioning of the PBRF and affect the public or occupational health and safety. Conclusions are presented as to the acceptability of the results of the accident analysis. A systematic approach to hazard evaluation and accident analysis that is consistent with the method described in the NRC's recently updated *Nuclear Fuel Cycle Facility Accident Analysis Handbook* (SAIC 1998) was used. The approach adopted is a screening analysis at a level of detail consistent with existing information about the radiological hazards at the PBRF.

A screening analysis approach is appropriate for accident analysis because the radioactive inventories at the PBRF are very small compared to those in operating reactors (both power and non-power) and in various kinds of fuel cycle facilities subject to NRC regulation. The screening analysis for the PBRF consists of identifying and analyzing plausible accident scenarios that could occur during decommissioning activities. A key conservative assumption used for all the scenarios was to neglect the impact of potential design and/or procedural controls, such as air filtering systems. The analyses show that the doses to the public from potential accidents are below the lowest action level in Table 1 of ANSI/ANS 15.16-1982 developed to protect members of the public from the consequences of accidents. Also, doses to workers from potential accidents are below the permitted annual exposure limits. Therefore, no new protective measures are required to protect public or occupational health and safety.

Section 3.3.1 identifies potential radiological accidents at the PBRF based on decommissioning activities and radiological hazards. Section 3.3.2 contains an analysis of potential accident scenarios to estimate the TEDE to a member of the public at the PBRF site boundary. Section 3.3.3 presents bounding estimates of worker exposure from worst-case accident scenarios.

3.3.1 Potential Radiological Accidents

Identifying potential accident scenarios included evaluating PBRF areas that contain the highest inventories of radioactive material, describing energy sources and external events, reviewing proposed activities, and considering combinations of these elements that could lead to a release of radioactive material. This identification process was supplemented by reviewing experience at other decommissioning projects and reviewing lists of potential accident scenarios developed for decommissioning activities at reactor facilities (Murphy 1978) and fuel cycle facilities (Schneider and Jenkins 1977). This process is consistent with the hazard evaluation steps identified in the *Nuclear Fuel Cycle Facility Accident Analysis Handbook* (SAIC 1998). Because of the limited inventory, the evaluation of accident scenarios conservatively assumed that no design or procedural controls would be available to prevent or mitigate accidental releases, even though such controls will be implemented during decommissioning activities. This assumption allows for a worst-case accident analysis to be performed.

3.3.1.1 Highest Radionuclide Inventories at the PBRF

The radiological inventories contained at the PBRF are summarized in Section 2.2.2, "Current Radiological Status of the Facility," of this plan. The reactor tank in the Reactor Building (Building 1111) has the highest inventory of any interior building area at the PBRF, an estimated

37,408 curies (Ci) (in the year 2003). Most of the inventory is tritium (H-3), together with 92 Ci of Co-60, and smaller inventories of other radionuclides. Therefore, potential accidents during decommissioning of the reactor tank are assigned the highest priority for this accident analysis. The Hot Dry Storage Area in the Hot Laboratory (Building 1112) has the second largest inventory, an estimated 8798 Ci, most of which is H-3. Most (559-Ci) of the remainder inventory is Co-60.

All of the other buildings and structures at the PBRF have small radioactive inventories compared to these two areas. In addition, no types of accidents were identified in these other buildings and structures that differ from those discussed in Section 3.3.1.2. Therefore, the results of accident analyses conducted for decommissioning the reactor tank and Hot Dry Storage Area bound the potential impacts of inside accidents during decommissioning of the PBRF.

Accidents could also occur outside buildings in soil at areas of past environmental contamination. These areas include the Emergency Retention Basin, the drainage system, the Water Effluent Monitoring Station, the Pentolite Ditch, and two known low-level waste spill areas. The area with the greatest radionuclide inventory is the Emergency Retention Basin, estimated to contain 0.15 Ci of Cs-137, 0.015 Ci of Co-60, and 0.011 Ci of Sr-90. Therefore, the results of accident analyses conducted for decontaminating the Emergency Retention Basin bound the potential impacts of exterior accidents at the PBRF.

3.3.1.2 Potential Accident Scenarios

Considering the planned decommissioning activities, accident scenarios that could result in releasing radioactive material as airborne particles small enough to be respirable were evaluated. Such releases could occur during cutting operations, dropping of a radioactively contaminated component, or dropping of a container of radioactively contaminated dust or soil. Because all PBRF buildings are outside of the 500-year floodplain and releases from the Emergency Retention Basin will be monitored and controlled, extreme precipitation events are not expected to cause offsite radiological impacts. The potential onsite and offsite impacts of accidents will be mitigated by emergency procedures required by PBRF technical specifications (Mendonca 1998). The emergency procedures include providing personnel trained to respond to fires, floods, and tornadoes.

Based on the decommissioning activities outlined in Section 2.3.1 and the radiological inventories identified in Section 3.3.1.1, the following accident scenarios were evaluated:

- The reactor tank will be dismantled by cutting it into pieces, and the pieces will be lifted and put into containers for transport off site. An accident during a cutting operation could result in small, radioactive particles becoming airborne.
- The Hot Dry Storage Area contains various reactor components that remained after the reactor was shut down and the core removed. During decommissioning, these components will be lifted and placed into transport containers. This is a very simple operation, and the worst-case accident scenario would be dropping one of these components as it is being lifted.
- The walls of some of the buildings will likely be decontaminated. The most intrusive decontamination method would be to scrape the concrete walls, producing a fine dust.

This dust would be placed in a container, such as a 55-gal drum, which could potentially be dropped and then burst.

- Areas of environmental contamination outside of buildings, such as the Emergency Retention Basin, will be decontaminated by digging up contaminated soil and placing it into containers. Either the digging operations or dropping of a container that then bursts could produce airborne particles.
- The potential for fires was also considered. The materials in the Reactor Building and Hot Dry Storage Area are metals, concrete, or similar materials. It is considered highly unlikely that a fire will start or that a fire could become intense enough to release radioactive material. Impacts of releases from a fire involving dry solid waste (i.e., rags, wipes, and anti-contamination clothing) were considered.

3.3.2 Evaluation of Public Impact from Accident Scenarios

This section further develops and analyzes accidents having the greatest potential for offsite impacts to estimate the TEDE to a member of the public. The following accident scenarios were analyzed:

- Release during cutting of the reactor tank
- Dropping a component from the Hot Dry Storage Area
- Dropping a drum of contaminated concrete dust
- Release while removing contaminated soil from the Emergency Retention Basin.

3.3.2.1 Assumptions

The following assumptions were used in all of the accident analyses:

- Except where otherwise stated, the radionuclide inventories were decayed to the year 2003, when it was assumed the decommissioning activities would occur. (A variation of a few years before or after this date will only result in a small percentage change in the inventory estimates.)
- No credit was taken for a filter located between the source of the release and the external atmosphere. Because filters will be used in indoor work areas, this is a conservative assumption for releases occurring inside PBRF buildings, such as the Reactor Building. Generally, HEPA filters having in-place tested removal efficiencies of 99.95% will be used.
- When released material travels through the Reactor Building, no plate-out (i.e., adsorption onto solid surfaces) or other deposition mechanisms were assumed to occur before the release reaches the external atmosphere.
- To be conservative, unfavorable weather conditions for atmospheric dispersion were assumed. In accordance with the Nuclear Fuel Cycle Facility Accident Analysis Handbook (SAIC 1998) and for purposes of analysis, atmospheric stability class F with

a wind speed of 2 m/s (6.6 ft/s) was assumed, which represents a "severe meteorological condition." In addition, the radioactive material was assumed to be released at ground level and to remain airborne as it travels downwind.

- Radioactive material was assumed to be transported to the closest site boundary (i.e., a distance of approximately 0.8 km (0.5 mi). This is a conservative assumption because the nearest dwelling is located outside the PBRF fence at a distance of 0.9 km (0.55 mi), and more dispersion would occur before the material reaches the dwelling.

3.3.2.2 Methodology for Calculating Total Effective Dose Equivalent

The consequences of accidents were quantified by calculating the TEDE to a member of the public at the site boundary. Then the calculated TEDE was compared to the lowest action level identified in Table 1 of ANSI/ANS 15.16-1982, "Emergency Planning for Research Reactor" (15.0 mrem whole body dose), to determine whether or not the calculated exposure is acceptable. Equation 3-1 was used to calculate the TEDE:[a]

$$TEDE_i = CEDE_i + Ext_i \tag{3-1}$$

where

TEDE = total effective dose equivalent
CEDE = committed effective dose equivalent
Ext = contribution from external irradiation
i = radionuclide.

The committed effective dose equivalent (CEDE) is the dose contribution from inhalation as the cloud passes by the receptor. Consistent with the lung model developed by the International Commission on Radiological Protection (ICRP 1979), the CEDE is found by

$$CEDE_i = Q_i\,(\chi/Q) \times B \times D_i \tag{3-2}$$

where

Q_i = the total released activity of nuclide i, in Ci

χ/Q = the airborne dosage (concentration integrated over the duration of cloud passage) per unit activity released, in s/m^3. The derivation of χ/Q presented in Appendix B shows that for a distance of 0.8 km (0.5 mi) in atmospheric stability class F with a windspeed of 2 m/s, $\chi/Q = 5 \times 10^{-4}$ s/m^3.

B = the breathing rate, typically 3.3×10^{-4} m^3/s. (This is the breathing rate for adults during light activity [ICRP 1979]).

D_i = the factor that converts the amount of activity inhaled into the CEDE. Values of D_i are given in Federal Guidance Report No. 11 (USEPA 1988).

[a] This estimate of the TEDE neglects any contribution from gamma rays emitted by radionuclides deposited on the ground. Such doses build up relatively slowly and, if necessary, can be controlled by various countermeasures.

The dose contribution from external irradiation is found by

$$Ext_i = Q_i\,(\chi/Q)F_i \qquad (3\text{-}3)$$

where

F_i = the dose coefficient for air submersion. Values of F_i are given in Federal Guidance Report No. 12 (USEPA 1993).

3.3.2.3 Scenario 1: Cutting Reactor Tank Internal Components with a Plasma Torch Releases Activation Products

The estimated inventory in the reactor tank at the time decommissioning is expected to occur (refer to Section 2.2.2.2) is 37,408 Ci, most of which will be H-3. The remaining inventory will be Co-60 (92 Ci); Fe-55 (10.5 Ci); Ni-63 (37 Ci); and relatively small amounts of Ni-59, Al-26, and Cd-113m. During decommissioning, the reactor tank will be cut up into pieces and the pieces removed and disposed of as described in Section 2.3.3.3. The engineering details of this activity have not been finalized, but for purposes of analysis, the following assumptions were made:

1. Cutting, disassembly, and packaging operations inside the reactor tank would be performed using remotely operated equipment. (It was assumed that the reactor tank was dry.)
2. The cutting operation would be performed using mechanical tools or flame cutting. However, to be conservative, it was assumed that a plasma torch would be used because it would vaporize more of the radioactive materials in the reactor tank internals than the technologies identified in Section 2.3.3.3.
3. It is assumed that the plasma torch cutting operation completely vaporizes a portion of the reactor tank or its internals equivalent to a 6.5-cm^2 (1-in^2) area with a 0.64-cm, (0.25-in.) thickness (i.e., a volume of approximately 4 cm^3 [0.24 in^3]). This volume corresponds to a 10-cm (4-in.) long cut, 0.64-cm (0.25-in.) wide and 0.64-cm (0.25-in.) deep. This is a cross section that is typical of plasma torches. The volume of 4 cm^3 (0.24 in^3) far exceeds the volume of material that might become airborne from mechanical cutting operations. A fine airborne particulate is produced that is assumed to be entirely within the respirable range (i.e., with particle sizes <10 μm).
4. It is assumed that the item in the reactor tank with the highest radionuclide contamination is involved in the accident. Table A-4 (Radioactivity Analysis) in NASA's 1980 *Environmental Report, Plum Brook Reactor Dismantling* (NASA 1980a) gives the activity levels in and masses of 38 different items in the reactor tank. The most contaminated items are 75 miscellaneous 304-stainless steel bolts with a total mass (all 75 bolts) of 1940 g (68 oz). The 1980 inventory estimates were adjusted to the year 2003, at which time the bolts would contain approximately 0.276 Ci of Fe-55, 0.923 Ci of Co-60, 1.28 Ci of Ni-63, and much lower activities of other radionuclides. The inventory of each nuclide was divided by the total mass, resulting in radionuclide concentrations of 1.4×10^{-4}, 4.76×10^{-4}, and 6.61×10^{-4} Ci/g for Fe-55, Co-60, and Ni-63, respectively. The concentrations of these beta- and gamma-emitting

radionuclides were used as an upper bound for activities that could be released by a cutting accident.

The above assumptions are considered to be conservative.

The calculation of the TEDE for Scenario 1 is summarized in Table 3-4. Q_i is the total released activity in curies of nuclide *i*. Using a typical steel density of 7.86 g/cm^3 (0.28 lb/in.^{3}), the mass of the nuclide that is released from the 4 cm^3 (0.24 in.^{3}) of steel vaporized is $4 \times 7.86 = 31.44$ g (or 1.1 oz). Multiplying the activity densities given in assumption #4 above by 31.44 g (1.1 oz) yields the total activities released in curies, Q_i, as shown in the second column of Table 3-4. The third column of Table 3-4 presents the values of D_i from U.S. EPA (1988): Using values of $\chi/Q = 5 \times 10^{-4}$ s/m^3 and $B = 3.3 \times 10^{-4}$ m^3/s (see Section 3.3.2.2) and the values of Q_i and D_i from Table 3-4 in Equation 3-2 yields the values of CEDE in the fifth column of Table 3-4.

Table 3-4. Values used to Calculate TEDE for Scenario 1: Cutting Reactor Tank Internal Components with a Plasma Torch

Nuclide *i*	Q_i (Ci)	D_i (mrem/Ci)	F_i ([mrem/s]/[Ci/m^3])	$CEDE_i$ (mrem)	Ext_i (mrem)	$TEDE_i$ (mrem)
Fe-55	0.0044	2.69×10^6	0	0.002	0	0.002
Co-60	0.015	2.19×10^8	466.2	0.5	0.0035	0.5035
Ni-63	0.0208	6.29×10^6*	0	0.022	0	0.022
Total	--	--	--	0.524	0.0035	0.5275

* This is the value for Ni-63 as a vapor. This is a conservative assumption because the Ni-63 will condense into fine particles as it mixes in the air in the reactor building.

The values of F_i from U.S. EPA (1993) are presented in the fourth column of Table 3-4. F_i is zero for both Fe-55 and Ni-63 because neither of these nuclides emit gamma rays. Using Equation 3-3, the dose contribution from external irradiation, Ext_i, was calculated and presented in the sixth column of Table 3-4. As shown in Table 3-4, the total dose contribution from external irradiation (0.0035 mrem) is only a small fraction (< 0.01) of that from the inhalation pathway (0.524 mrem).

The fifth and sixth columns of Table 3-4 show that the dominant radionuclide from a postulated release during reactor dismantling operations would be Co-60. The TEDE (whole body dose) is the sum of the internal (CEDE) and external (Ext) doses. The last column of Table 3-4 shows that the total TEDE is very small, approximately 0.5 mrem. Thus, the TEDE from Scenario 1 is significantly lower than the 15.0 mrem whole body dose identified as the lowest action level in Table 1 of ANSI/ANS 15.16-1982, "Emergency Planning for Research Reactor".

3.3.2.4 Scenario 2: Cutting a Beryllium Component in the Reactor Tank with a Plasma Torch Releases Tritium

As discussed in Section 2.2.2.2, the reactor tank and the Hot Dry Storage Area are estimated to contain 37,266 Ci and 8,222 Ci of H-3, respectively. In the reactor tank, most of the H-3 is in

irradiated components that contain beryllium. For example, the set of "RA pieces with plugs" [Item 21 in Appendix A of NASA's environmental report (NASA 1980a)] was estimated to contain 16,900 Ci of H-3 in 1993 and weighs 113,000 g (249 lb). By 2003, this inventory will have decayed to approximately 9100 Ci. The corresponding activity density is 9100 Ci/113,000 g = 0.08 Ci/g.

A cutting accident where it was assumed that 4 cm^3 (0.24 in^3) of the irradiated components is vaporized was analyzed. The density of beryllium is 1.8 g/cm^3; therefore, 4 $cm^3 \times 1.8$ g/cm^3 = 7.2 g (0.25 oz) would become airborne. The total activity released, Q_i, is 7.2 g × 0.08 Ci/g = 0.576 Ci.

The TEDE from this release can be estimated using Equation 3-1. Using a D_i value of 6.40×10^4 mrem/Ci for H-3, and the χ/Q and B values from Section 3.3.2.2, the CEDE was calculated using Equation 3-2:

$$CEDE_{H\text{-}3} = (0.576)(5.0 \times 10^{-4})(3.3 \times 10^{-4})(6.40 \times 10^{4})$$

$$= 0.006 \text{ mrem.}$$

Using an F_i value of 0.0012 mrem-m^3/s-Ci for H-3, the contribution from external irradiation is found by Equation 3-3:

$$Ext_{H\text{-}3} = (0.576)(5.0 \times 10^{-4})(0.0012)$$

$$= 3.46 \times 10^{-7} \text{ mrem.}$$

Thus, the TEDE = CEDE + Ext = 0.006 + 3.46×10^{-7} = 0.006 mrem, which is significantly lower than the 15.0 mrem whole body dose identified as the lowest action level in Table 1 of ANSI/ANS 15.16-1982. Also, the TEDE calculated for Scenario 2 is nearly a factor of 100 lower than the TEDE calculated for Scenario 1 in Section 3.3.2.3.

3.3.2.5 Scenario 3: Dropping a Component Stored in the Hot Dry Storage Area

Most of the activity in Hot Dry Storage Area is contained in stored contaminated components [refer to Table A-5 of NASA (1980a)] that include control rods, beryllium L-shaped pieces, instrument thimbles, etc. Some of these individual pieces were estimated to contain hundreds of curies in 1993 (assuming 20 years of decay after the 1973 reference point). Each beryllium control rod contains approximately 800 Ci (mostly H-3, but approximately 10% Co-60).

Because no cutting operations are planned for the components stored in the Hot Dry Storage Area, no cutting accident was postulated. However, a stored component could be dropped while it is lifted for placement into a shipping container. It would be highly unlikely for a component to break. If it did break, the diameters of any particles produced would be large enough that it is unlikely that the particles would remain airborne and be respirable. Therefore, no plausible accident was postulated that could result in measurable exposures at the site boundary. The TEDE would be much less than the 0.5 mrem calculated for Scenario 1 in Section 3.3.2.3; therefore, it

would be significantly lower than the 15.0 mrem whole body dose identified as the lowest action level on Table 1 of ANSI/ANS 15.16-1982.

3.3.2.6 Scenario 4: Dropping a 55-Gallon Drum of Contaminated Concrete Dust Generated from the Biological Shield or Hot Cells

Radiological survey data described in Section 2.2.2 indicate the greatest activity contained in concrete structures at the PBRF is in the biological shield surrounding the reactor tank and the wall and floors of the hot cells. In 1985, measured activity levels of Co-60 in the biological shield varied from less than 1 to 33 pCi/g (Teledyne Isotopes 1987). Using the 33 pCi/g as a conservative upper bound and calculating radioactive decay for another 20 years (i.e., to about the year when decommissioning activities would occur) leads to a predicted activity concentration of 2.4 pCi/g.

It was assumed that a 55-gal drum (a volume of 0.21 m^3 [7.5 ft^3]) of concrete dust or fine particulate was generated from decontaminating either the biological shield or the hot cells. Assuming a conservative concrete density of approximately 3 g/cm^3, the 55-gal drum would contain approximately 6.4×10^5 g of dust. Assuming a Co-60 concentration of 2.4 pCi/g, there would be $(6.4 \times 10^{-5}) \times (2.4 \times 10^{-12}) = 1.5 \times 10^{-6}$ Ci in the drum. This activity is many orders of magnitude less than the 0.015 Ci calculated for Scenario 1 in Section 3.3.2.3, which resulted in a TEDE of 0.5 mrem. Therefore, even if the contents of a whole drum were spilled and became airborne in respirable form (which is physically unrealistic), the predicted TEDE at the site boundary would be significantly lower than the 15.0 mrem whole body dose identified as the lowest action level in Table 1 of ANSI/ANS 15.16-1982.

3.3.2.7 Scenario 5: Contaminated Soil Released from the Emergency Retention Basin

The area of contaminated soil at the PBRF having the highest radionuclide inventory is the Emergency Retention Basin. At the Emergency Retention Basin, the estimated concentrations of Cs-137, Co-60, and Sr-90 are approximately 200, 20, and 20 pCi/g, respectively. If an entire 55-gal drum of this contaminated soil became airborne in respirable form (i.e., approximately 3.3×10^5 g, assuming a soil density of 1.56 g/cm^3 from Section 2.2.3.1), the airborne quantities of Cs-137, Co-60, and Sr-90 would be 6.6×10^{-5}, 6.6×10^{-6}, and 6.6×10^{-6} Ci, respectively. Using the values of X/Q and B given in Section 3.3.2.2 in Equations 3-1 through 3-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 3-5.

Table 3-5. Values Used to Calculate TEDE for Scenario 5: Contaminated Soil Released from the Emergency Retention Basin

Nuclide i	Q_i (Ci)	D_i (mrem/Ci)	F_i ([mrem/s]/[Ci/m^3])	$CEDE_i$ (mrem)	Ext_i (mrem)	$TEDE_i$ (mrem)
Cs-137	6.6×10^{-5}	3.2×10^{7}	101	0.0003	2×10^{-5}	0.0003
Co-60	6.6×10^{-6}	2.19×10^{8}	466.2	0.0002	9×10^{-6}	0.0002
Sr-90	6.6×10^{-6}	1.30×10^{9}	0.0278	0.0014	5×10^{-10}	0.0014
Total	--	--	--	0.0019	2.9×10^{-6}	0.0019

As shown in Table 3-5, the TEDE is 0.0019 mrem, to which the external dose is a negligible (0.1%) contributor. The TEDE of 0.0019 mrem is significantly lower than the 15.0 mrem whole body dose identified as the lowest action level in Table 1 of ANSI/ANS 15.16-1982.

3.3.2.8 Scenario 6: Fire Involving Dry Solid Waste

Dry solid waste generated from decontamination activities will include contaminated rags, wipes, and anticontamination clothing that are combustible. As discussed in Section 3.2, such dry solid waste material is categorized as Class A waste, with a total volume of 130 m^3 (4600 ft^3) (see Table 3-3). Reactor decommissioning studies estimate that radionuclide concentrations of 1.4×10^{-3} Ci/m^3 of Co-60, 2.8×10^{-4} Ci/m^3 of Sr-90, and 0.35 Ci/m^3 of Cs-134 are representative of this material (Murphy 1978). Combustion of this type of material would release approximately 0.05% of the contamination (SAIC 1998). Thus, combustion of the entire inventory of this dry, solid waste would release 9×10^{-5} Ci of Co-60, 1.8×10^{-5} Ci of Sr-90, and 0.02 Ci of Cs-137. Because impacts of accidental releases are dominated by the external exposure pathway and the dose factor for external exposure for Cs-137 is less than that of Co-60, potential impacts of this fire-initiated scenario are bounded by those of Scenario 1 in Section 3.3.2.3. The TEDE for this fire accident is much less than the 15.0 mrem whole body dose identified as the lowest action level in Table 1 of ANSI/ANS 15.16-1982.

3.3.3 Evaluation of Worker Exposure from Accident Scenarios

The accident scenario that would have the greatest potential release during decommissioning is Scenario 1 (Section 3.3.2.3), reactor tank dismantling operations. This accident could result in 0.015 Ci of Co-60 becoming airborne, along with smaller quantities of Fe-55 and Ni-63. As described in Section 3.3.2.3, cutting the reactor tank would likely be performed by remotely operated equipment. The engineering details of the remote cutting operation have not been finalized, but it is assumed that if radioactive material was released during cutting operations, it would bypass workers controlling the cutting operations. Therefore, it is unlikely that accidents in the reactor tank would affect workers. Conservatively, in case the remote cutting arrangement does not protect the worker from exposure, it was assumed that a worker inhales a fraction (i.e., 1×10^{-6} [one millionth]), of the radioactive material released following the cutting accident

(Brodsky 1980). If 0.015 Ci of Co-60 becomes airborne, the worker would inhale 1.5×10^{-8} Ci. Using the dose conversion factor, D_i, for Co-60 of 2.19×10^{8} mrem/Ci, the dose to the worker would be $(1.5 \times 10^{-8}$ Ci) $(2.19 \times 10^{8}$ mrem/Ci) = 3.3 mrem. This dose is well below the 15.0 mrem whole body dose identified as the lowest action limit in Table 1 of ANSI/ANS 15.16-1982.

The accidents discussed in Sections 3.3.2.3 through 3.3.2.8 would result in even less severe consequences than the 3.3 mrem calculated above. Therefore, it is unlikely that an accident could occur where a worker would accumulate a significant fraction of the 5-rem annual exposure limit. As described in Section 3.1.2 the radiation protection program will include worker protection and approved work control permits and procedures.

3.3.4 Conclusions

The accident analysis shows that the postulated accident scenarios would result in TEDEs to a member of the public at the site boundary that are significantly lower than the 15.0 mrem whole body dose identified as the lowest action level in Table 1 of ANSI/ANS 15.16-1982. Also, doses that workers could receive from an accident are much less than the allowable annual exposure for workers, 5 rem (5000 mrem) (NRC 1991) and the lower NASA administrative limits for worker exposure.

Also, as stated in Section 3.3.2.1, the accident analysis did not take credit for protective features (e.g., presence of building structures and filters). However, because the accident analysis shows that predicted offsite consequences to a member of the public are small, there is no need to develop technical specifications for filter performance.

4. PROPOSED FINAL STATUS SURVEY PLAN

This section describes the approach that will be used in the Final Status Survey Plan for the PBRF. This approach has been developed according to the guidance in Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination" (NRC 1998a); NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" (USEPA et al. 1997); and NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys" (Gogolak et al. 1998). Consistent with this guidance, the final status survey plan has been designed incorporating the Data Quality Objectives (DQO) process. This process is iterative because it is applied from a current base of information and developed as information is revised or collected. Applying the DQO process ensures that the type and quality of radiological data needed to support license termination are considered early in the decommissioning process. Section 4.1 describes the final status survey design and the DQO process and identifies criteria and methods that will be used to support the decision on terminating the license and releasing the PBRF site for unrestricted use. Section 4.2 briefly discusses how the final status survey plan will be documented.

As the decommissioning process proceeds, a revised FSS Plan will be prepared and submitted to the NRC for review and approval.

Because of the iterative DQO process, implementing the final status survey plan will incorporate additional information available during decommissioning. The final status survey plan will use remediation plans, decision errors, and statistical parameters that have not been subject to regulatory review, while implementing the plan will use approved cost and post-remediation statistical parameters. The information available for the final status survey plan presented in this section, as well as the additional information expected to be available to implement the final plan, is illustrated in Figure 4-1.

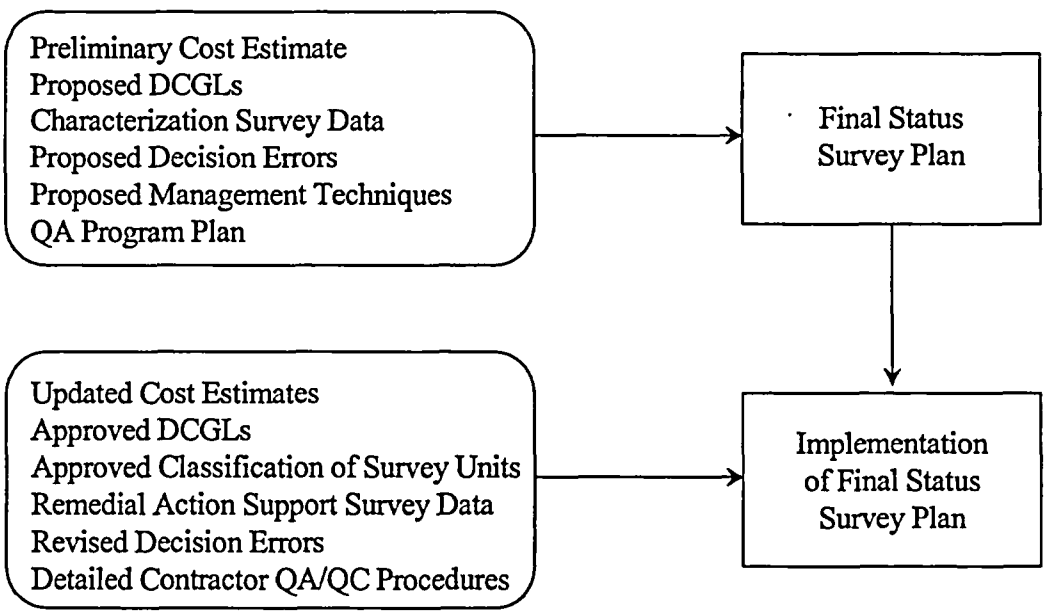


Figure 4-1. Evolution of the Final Status Survey Plan

4.1 The Data Quality Objectives Process

The DQO process is a series of planning steps that have been defined by EPA (USEPA 1994) to ensure that the type, quantity, and quality of environmental data used in decision making are appropriate for the intended application. DQOs are qualitative and quantitative statements that clarify the study objective, define the most appropriate data to collect, determine the most appropriate conditions for collecting the data, and specify acceptable levels of decision errors that will be used to establish the quantity and quality of data needed to support the decision. The DQO process is iterative, so specifications may change as new information is obtained during the course of site remediation, until the final status survey is actually performed. The DQO process comprises the seven steps identified in Figure 4-2. These seven steps are discussed in Sections 4.1.1 through 4.1.7.

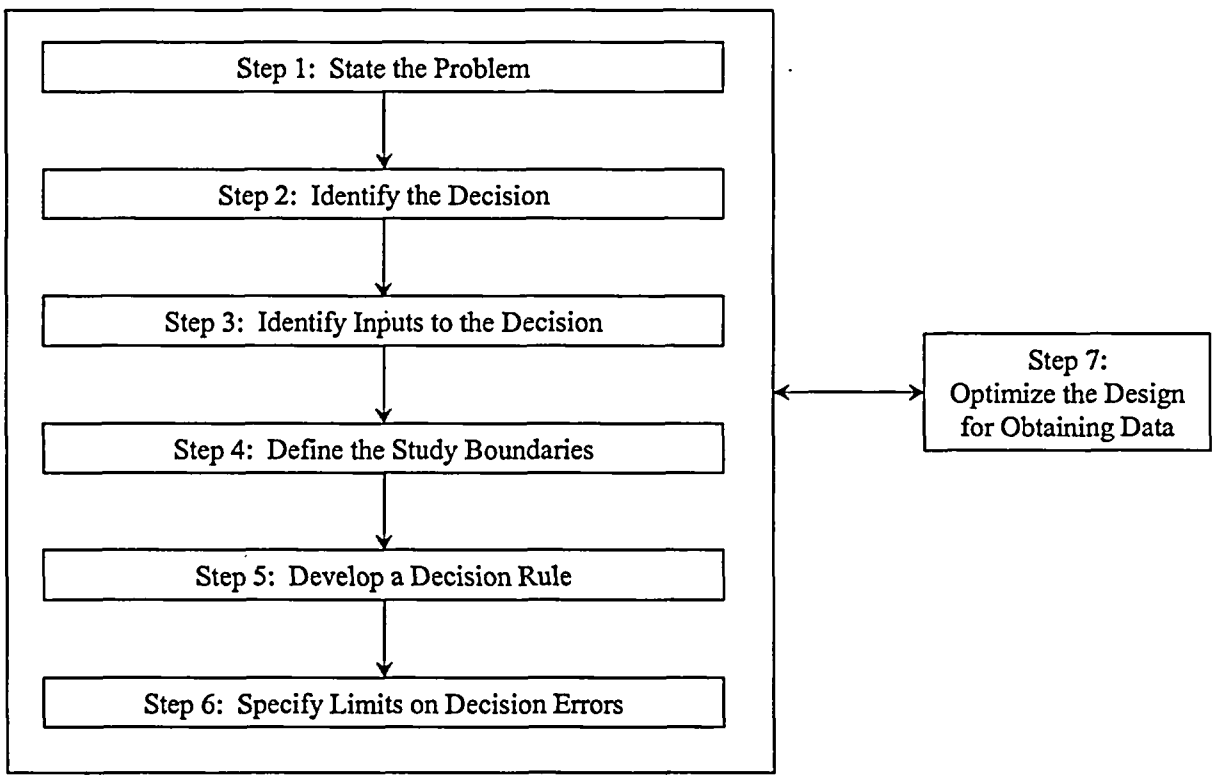


Figure 4-2. The Data Quality Objectives Process (adapted from Figure D.1 of NUREG-1575 [USEPA et al. 1997])

4.1.1 Step 1: Stating the Problem

The objective of decommissioning the PBRF is to reduce the residual radioactivity to a level that permits unrestricted release of the property and termination of the license. Data will be needed to support this objective to demonstrate that residual radioactivity remaining at the PBRF results in a dose less than the release criterion. This objective will be met by performing a final status survey in individual survey units. A separate decision will be made for each survey unit about whether the

release criterion has been met. The information currently available to describe the nature and extent of the contamination is described in Section 2.2 of this plan. However, additional information on the general location and extent of residual radioactivity and estimated concentration levels will be gained during the characterization steps of remediation. The NRC will make the final decision to terminate the license and release the PBRF. The NASA Decommissioning Project Manager will make the final decision about decommissioning activities and developing the final status survey plan. Stakeholders in the project include NRC, NASA, and local residents. NASA's organizational structure and responsibilities are discussed in Section 2.4 of this plan.

4.1.2 Step 2: Identifying the Decision

The primary decommissioning criterion is that the TEDE to future occupants at the PBRF site from residual radioactivity that is distinguishable from background radiation must be less than 25 mrem/yr. In addition, analysis must demonstrate that levels of residual radioactivity are ALARA. The decision statement is:

Has the decommissioning dose criterion been met in individual survey units?

Dose assessment modeling will be used to translate the dose criterion into levels of residual contamination that are acceptable at each survey unit. These levels, termed DCGLs, were estimated for the PBRF using the methods, site-specific source terms, and site information described in Section 2.2.3.1 of this plan. DCGLs were derived for individual nuclides for residual contamination of surface soils, building surfaces, and subsurface structures. The numerical release criterion proposed for demonstrating that the dose criterion has been met will be that the sum-of-fractions of quotients of concentrations and DCGLs of contributing radionuclides shall be less than unity. If a survey unit fails to meet this numerical release criterion, the need for additional sampling or remediation will be evaluated.

The DCGLs assume that the level of residual radioactivity is uniformly distributed across the survey unit; they are designated $DCGL_W$[b] in this plan. A nonparametric statistical test will be applied to the sampling data taken at distinct locations in the survey unit to determine whether this level meets the release criterion. The test will be based on the probabilities of rejecting a true null hypothesis (Type I error) and accepting a false null hypothesis (Type II error) established in the sixth step of the DQO process (Section 4.1.6).

In addition, a separate $DCGL_{EMC}$ (the DCGL used for the elevated measurement comparison) will be calculated if it is assumed that residual radioactivity is concentrated in a much smaller area (i.e., in only a small percentage of the entire survey unit). The $DCGL_{EMC}$ will be calculated for survey planning purposes and will trigger further investigation of a portion of the survey unit. Any measurement from the survey unit will be considered elevated if it exceeds the $DCGL_{EMC}$. However, the elevated measurement alone does not indicate that the survey unit fails to meet the release criterion, only that further investigation will be necessary to determine the actual extent and

[b] The "W" in $DCGL_W$ stands for Wilcoxon Rank Sum test, which is the statistical test recommended in MARSSIM for demonstrating compliance when the contaminant is present in background. The Sign test recommended for demonstrating compliance when the contaminant is not present in background also uses the $DCGL_W$.

concentration level of the elevated area. This information may be used with further modeling to demonstrate that the release criterion has been met.

4.1.3 Step 3: Identifying Inputs to the Decision

The purpose of Step 3 is to identify the information needed to resolve the decision statement identified in Step 2 and sources of this information. The primary inputs to the decision statement are the DCGLs and average radionuclide concentrations at each survey unit. This information will be developed using site and survey unit characteristics data, decision error magnitudes, and radionuclide concentration data. Sources of data are discussed in Sections 4.1.3.1 and 4.1.3.2.

4.1.3.1 Derived Concentration Guidelines

DCGLs for individual radionuclides were developed using data on radiological and physical characteristics of the PBRF site for receptor scenarios that quantify modes and rates of exposure. Section 2.2 of this plan describes radiological and physical characteristics of the site, including radionuclides of concern, and Section 2.2.3.1 describes exposure scenarios and how DCGLs were calculated. DCGLs were developed for residual contamination of surface soils, building surfaces, and subsurface structures (Section 2.2.3.1). Direct measurements of residual contamination of surface soils and of abovegrade and belowgrade building surfaces will be compared with the DCGLs. For subsurface structures, measurements of surface and volumetric contamination levels will be volume averaged to calculate a concentration to demonstrate that building surface DCGLs are protective for all scenarios.

4.1.3.2 Measurement of Radionuclide Concentrations

Radionuclide concentrations are a primary input to the decision rule. Measuring radionuclide concentrations involves delineating discrete survey units, identifying the nature and number of measurements, and selecting measurement techniques. Delineating survey areas is discussed in Step 4 of the DQO process (Section 4.1.4).

Nature and Number of Measurements

The decision rule and the site physical and radiological characteristics will direct the nature of measurements taken. The decision rule described in Step 5 of the DQO process (Section 4.1.5) requires knowledge of individual radionuclide concentrations in volumes of soil and rubble and on surfaces of buildings and subsurface structures. In addition, the decision rule requires assessing the potential for elevated concentrations. Thus, types of samples will include volumes of soil and rubble, scrapings and smears of surfaces, and scans of surfaces.

The number of samples for each survey unit will be determined by balancing costs and decision errors using the error magnitudes specified in Step 6 of the DQO process (Section 4.1.6). The approach for calculating the number of samples will be the same as that recommended in Chapter 5 of the MARSSIM (USEPA et al. 1997). Before acceptable limits on decision errors and the number of measurements necessary to meet them can be established, an estimate of the expected variability of the measurement data will be necessary. Information from scoping, characterization, and remedial action support surveys will be used in estimating the mean and standard deviation

expected for residual radioactivity in a survey unit. A summary of data from the 1985 PBRF characterization survey and the 1998 PBRF confirmatory survey is presented in Section 2.2.2.2 of this plan. These data provide a basis for radiological classification of all structures, systems, and grounds at the PBRF, but they do not provide a basis for estimating the standard deviation for all survey units. An example of the method for estimating the number of final status survey samples for a survey unit has been developed using the limited radionuclide-specific measurements collected for the Emergency Retention Basin during the 1985 characterization survey. Concentrations of measured Co-60, Sr-90, and Cs-137 in soil are reported (Teledyne Isotopes 1987) for 9, 58, and 8 samples, respectively. The measurement standard deviation of the weighted sum, calculated using Equation (11-7) of NUREG-1505 (Gogolak et al. 1998), is large, producing a small relative shift and a large number of final status survey samples following Equation (5-2) of MARSSIM (USEPA et al 1997). Thus, data from remedial action support surveys will be combined with experience and scientific judgment to estimate the measurement variability. As more information is available during decommissioning, the measurement and statistical methods needed to meet release criteria will be refined.

A more representative example of applying sample estimation following remediation has been developed using the 1985 measurements of Sr-90 in the Emergency Retention Basin in isolation. Measured concentrations are reported (Teledyne Isotopes 1987) for 58 locations, one of which is an outlying point with a concentration 30 times the mean of the other 57 samples. If the outlying point is neglected (a situation that could reflect the post-remediation condition of the Emergency Retention Basin), the mean and standard deviation of the measurements are 2.3 and 3.4 pCi/g, respectively. The DCGL calculated for Sr-90 in surface soils is 30 pCi/g (see Section 2.2.3.1). Following the guidance in the MARSSIM (USEPA et al. 1997) and using a Lower Boundary of the Gray Region (LBGR) (see definition in Section 4.1.6.3) of two-thirds of the DCGL, yields a relative shift of 3.0. For Type I and II errors of magnitude 0.05 and 0.10, respectively (see Section 4.1.6), applying Equation (5-2) of the MARSSIM yields an estimate of approximately 15 samples for the Emergency Retention Basin. It is anticipated that this procedure will be applied for all survey units of the PBRF using remedial action support survey data.

Identification of Measurement Techniques

Radionuclide-specific measurement techniques will be needed for both gamma- and beta-emitting radionuclides in surface soil and on building surfaces. The gamma-emitting radionuclides are projected to dominate the dose for the residential farmer scenario for surface soils and the building reuse scenario. The beta-emitting radionuclides are projected to dominate the dose for the residential farmer scenario applied to subsurface structures. A list of candidate measurements is presented in Table 4-1, and techniques used in the radionuclide-specific and scanning measurements are discussed below. Gas proportional detectors with alpha and beta probes also are considered appropriate for direct measurement of gross levels of activity on building surfaces.

Table 4-1. Survey Instrumentation

Measurement	Instrument Type
Scanning:	
• Alpha	• Gas proportional, Zn S(Ag) scintillation
• Beta	• Gas proportional, Geiger-Mueller
• Gamma	• NaI (Tl) scintillation
Radionuclide-specific:	
• Beta	• Liquid scintillation
• Gamma	• ISOCS Ge solid state or equivalent

*ISOCS- In Situ Object Characterization System – ISOCS is a specific example of a portable, solid-state detector based spectroscopy system that provides in-situ, quantitative and qualitative information on the types and amounts of radiation present.

Scanning Measurements

Scanning will be performed to locate radiation anomalies that might indicate elevated areas of residual activity and that will require further investigation or action. Scanning will be performed using a gamma detector for surface soils and a beta detector for building surfaces. If the scanning results exceed an investigation level determined for the detector and survey parameters, further investigation will be performed using direct measurement or sampling. Scanning will be performed to provide 100% coverage for Class 1 areas and 10% to 100% coverage for Class 2 areas. Scanning will be performed as judged necessary for Class 3 areas (Class 1, 2, and 3 areas are defined in Section 4.1.4).

Direct Field Measurements

Direct field measurements on building surfaces will be made at fixed locations using a gas proportional detector and an exposure rate instrument. This will provide a quantitative measure of radioactivity present in surface soils and on building surfaces. A portable insitu gamma spectrometer may be used in direct measurements of surface soils to verify sample results. Gamma spectrometry will allow direct measurement of all gamma-emitting radionuclides, including Cs-134, Cs-137, Co-60, Eu-152, and Eu-154. Other radionuclides that have been detected at the PBRF include H-3, Ni-59, Ni-63, and Sr-90. Although these other radionuclides do not have significant gamma radiations, their concentrations were inferred from the concentrations of the measured radionuclides based on established ratios in each survey unit. The established ratios will be confirmed through further sampling and laboratory analysis.

The probability sampling performed by field measurements will be systematic sampling on a systematic grid, with a random start for Class 1 and Class 2 areas and simple random sampling for Class 3 areas. It is anticipated that only these measurements will be used in conducting the nonparametric statistical test. However, results from scanning, direct field measurements, and laboratory analysis of samples may be used for elevated measurement comparison against an upper limit value.

Sampling and Laboratory Analysis

Sampling and laboratory analysis will be required during the final status survey to confirm the established ratios for the non-gamma-emitting radionuclides, to further define the areal extent of potential contamination, and to determine maximum radiation levels within an area. For surface soils and building surfaces, it is expected that ratios of non-gamma-emitting radionuclides to gamma-emitting radionuclides can be developed using field measurements. Probability sampling using locations chosen on a random or random start systematic grid basis will be limited to direct field measurements for these surfaces. For subsurface structures, probability sampling and laboratory analysis of beta-emitting radionuclides will be conducted. If it is determined through further characterization or confirmation sampling that any of the ratios are not constant, probability sampling will be employed for laboratory analysis of the non-gamma-emitting radionuclides.

Background Determination

Radionuclides contaminating the PBRF site do not occur in significant natural background concentrations. Therefore, reference area measurements will not be compared to radionuclide-specific direct field measurements or laboratory analyses. Some comparison to background levels will be required for the scanning measurements and topographical considerations will be weighed for this background comparison.

4.1.4 Step 4: Defining Study Boundaries

Defining spatial and temporal boundaries helps ensure the samples taken during the final status survey are representative of the survey unit. The spatial area under consideration for release is the entire PBRF. Because statistical methods will be used to define the number of samples taken and extent of surveys performed, it will be important to classify survey areas and to define their constituent survey units to minimize variability of concentrations. Furthermore, concentration levels of residual radioactivity before remediation will be used to define the type of statistical sampling and the extent of scanning coverage for each survey unit.

The survey areas are classified as either non-impacted areas or impacted areas. Non-impacted areas have no potential for residual contamination. Impacted areas are further divided into one of three classifications:

> ***Class 1 Areas***: These are areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination in excess of the $DCGL_W$.
>
> ***Class 2 Areas:*** These are areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_W$.
>
> ***Class 3 Areas***: These are impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$.

4.1.4.1 Spatial Boundaries

Section 2.2.2 of this plan describes the PBRF and its current radiological status. Because contaminated equipment and piping will be removed and disposed of, contamination levels on them will not be used to classify each facility. Although the above-grade portions of the buildings and below-grade portions of buildings within 1 m (3 ft) of the surface will be demolished after decontamination, the concrete rubble will be used as fill for the remaining below-grade portions of buildings and in-ground structures, such as the Hot Retention Area and Cold Retention Basins. Therefore, the contamination levels found on the floors and walls of the facilities before remediation will be used to classify subsurface structures. The facilities and grounds have been defined as separate survey areas and placed in the above classifications based on data from the 1985 and 1998 characterization surveys as shown in Table 4-2. The above-grade portions of the Reactor Building (1111), the Hot Laboratory (1112), and the Fan House (Building 1132) have been classified as shown in Table 4-2. However, it is expected that the above-grade portions of these buildings (and below-grade portions within 1 m (3 ft) of the surface) will be demolished, size-reduced, and collapsed into the remaining below-grade cavity of the buildings after completion and confirmation of the final status survey. Pending approval by regulatory authorities before demolition, the final status survey of the remaining below-grade structures will be validated before backfilling these areas. As a conservative measure, outside areas not designated as impacted areas will be surveyed as Class 3 survey units.

4.1.4.2 Temporal Boundaries

Some remedial action support and survey measurements of building surfaces will be made before demolition and backfilling to aid in design of the final status survey and further decontamination if necessary. This is especially true of below-grade surfaces that will not be demolished. Also, further sampling of the concrete rubble fill may be performed after demolition but before a cap is placed on the fill areas. Likewise, environmental media (e.g., soil, and water) will be sampled in the remedial action survey to aid in design of the final status survey and further decontamination if necessary.

4.1.4.3 Reference Coordinates

Reference coordinate systems will be established at the PBRF site to select and relocate measurement and sampling locations. A diagram showing each survey unit will be prepared.

4.1.4.4 Sampling Grids

Sampling locations in Class 1 and Class 2 survey units will be placed on random start systematic grids. These grids will be used as the sampling locations for the direct field measurements. However, if it is determined through further characterization or confirmation sampling that any of the ratios are not constant, probability sampling will be employed for laboratory analysis of the non-gamma-emitting radionuclides. An equilateral triangular will be used, with the distance between the sample points, L, determined by the number of samples or measurements that will be taken for the survey unit as dictated by statistical test requirements.

Table 4-2. Area Classification

Facility or Area within PBRF Fence	MARSSIM[a] Classification
Reactor Building (1111):	
• Inside containment vessel	Class 1
• Outside containment vessel	Class 2
• Canals	Class 1
Hot Laboratory (1112):	
• Cold work area floors, walls, and ceiling	Class 2
• Hot Work Area floor	Class 1
• Hot Work Area walls and ceiling	Class 2
• Decon Room floors and walls	Class 1
• Decon Room ceiling	Class 2
• Repair Shop floors and walls	Class 1
• Repair Shop ceiling	Class 2
• Storage room floors, walls, and ceiling	Class 2
• Mezzanine floors	Class 1
• Mezzanine walls and ceiling	Class 2
• Hot handling floors and walls	Class 1
• Hot handling ceiling	Class 2
• Hot pipe tunnel	Class 1
• Canals	Class 1
Fan House (1132):	
• First floor floors	Class 1
• First floor walls and ceiling	Class 2
• Basement floors, walls, and ceiling	Class 1
Waste Handling Building (1133):	
• First floor floors	Class 1
• First floor walls (white)	Class 2
• First floor walls (controlled)	Class 1
• First floor ceilings	Class 2
• Basement floors, walls, and ceiling	Class 1
Primary Pump House (1134) floors walls and ceilings	Class 2
Reactor Office and Laboratory Building (1141) all areas (exclusive of lab hoods and hood filter housings)	Class 2
Water Effluent Monitoring Station (1192) all areas	Class 2
Cold Retention Basins (1154)	Class 1
Hot Retention Area (1155)	Class 1
Emergency Retention Basin	Class 1
Drainage System	Class 1
Pentolite Ditch	Class 1
Areas of past spills	Class 1
Cold pipe tunnel	Class 3
Reactor sludge basins (1153)	Class 3
Reactor precipitator (1157)	Class 3
Reactor Service Equipment Building (1131)	Non-Impacted
Reactor Gas Services Building (1135)	Non-Impacted
Reactor Water Tower (1151)	Non-Impacted
Reactor Substation (1161)	Non-Impacted
Reactor Security Building (1191)	Non-Impacted
Reactor Compressor Building (1136)	Non-Impacted
Reactor Cryogenic and Gas Supply Farm and Building (1195 & 9837)	Non-Impacted
Reactor Gas Storage Structure (1196)	Non-Impacted

a. USEPA et al. (1997).

4.1.5 Step 5: Developing a Decision Rule

A decision rule relates the concentration of residual radioactivity in the survey unit to the release criterion so that decisions can be made based on the results of the final status survey. The decision rule proposed in this final status survey plan consists of a statistical test and an elevated measurement comparison. Because radionuclide-specific measurements will be made, if all of the measurements are below the $DCGL_W$, the survey unit will meet the release criterion. However, if the average of the measurements is above the $DCGL_W$, the survey unit will not meet the release criterion. When the average is below the $DCGL_W$ and some of the measurements are above the $DCGL_W$, a Sign test and the elevated measurement comparison will be used to determine if the release criterion has been met. Sections 4.1.5.1 and 4.1.5.2 define the parameters that will be used with the methods presented in NUREG-1575 (MARSSIM) (USEPA et al. 1997) and NUREG-1505 (Gogolak et al. 1998) for determining the number of samples (direct field measurements) that will be necessary for the statistical test to be valid.

4.1.5.1 The Statistical Test

The sign test for statistical analysis does not use background radiation level data. Therefore, statistical tests will only be performed on direct field measurements for radionuclides that are not present at significant background levels. Also, because it is expected that the variability in the data will be small relative to the $DCGL_W$, the following hypotheses have been chosen for the statistical test:

The null hypothesis, H_0 = the survey unit does not meet the release criterion.
The alternative hypothesis, H_a = the survey unit meets the release criterion.

Although the null hypothesis may require additional remediation when it is not strictly necessary, this is acceptable for the following reasons: (1) the contamination below the $DCGL_W$ is expected to be measurable, (2) additional remediation may still have some benefit in the form of reduced radiation exposure, and (3) additional remediation is preferable to releasing a survey unit that really should be remediated further.

4.1.5.2 Elevated Measurement Comparison

The decision rule for the elevated measurement comparison will be a two-stage process. In the first stage, areas will be flagged as potentially elevated at the specified investigation levels. Investigation levels will be established in consultation with NRC staff (USEPA et al. 1997). In the second stage, the actual average concentration over the actual extent of elevated area will be compared to the release criterion. The level at which measurements should be flagged will depend on the unit classification. For Class 1 survey units, areas will be flagged if the direct measurement or scanning measurement indicates concentrations above the $DCGL_{EMC}$. For Class 2 survey units, areas will be flagged if the direct field measurement or scanning measurement indicates concentrations above the $DCGL_W$. For Class 3 survey units, areas will be flagged if the direct measurement indicates concentrations above one-half of the $DCGL_W$ or the scanning measurement indicates concentrations above the minimum detectable concentrations.

4.1.6 Step 6: Specifying Limits on Decision Errors

4.1.6.1 Measurement Technique Detection Capabilities

Based on draft NUREG-1507 (Abelquist et al. 1997), it is expected that concentrations of 6.4 pCi/g of Cs-137 and 3.4 pCi/g of Co-60 will be detected with 95 percent confidence by scanning measurements made with a 5-cm x 5-cm (2-in. × 2-in.) NaI Detector on land areas with typical background levels. Based on the same information source, it is also expected that concentrations of approximately 0.05 pCi/g will be detected with 95 percent confidence for Co-60, Cs-137, and Eu-152 by direct field measurements made with a typical 25 percent relative efficiency p-type germanium detector and a 10-minute count time at typical background levels. These minimum detectable concentrations are only a fraction of the DCGL. Therefore, the measurement variability will be expected to be small at the DCGL. Minimum detectable concentrations of gross alpha and gross beta contamination on building surfaces are expected to be approximately 50 and 1000 dpm/100 cm^2, respectively. These levels also are fractions of the DCGLs.

4.1.6.2 Type I and Type II Errors

A Type I error is made when the null hypothesis, H_0, is rejected when it is true. A Type II error is made when the null hypothesis is not rejected when it is false. The error rates are expressed as the probability that a survey unit passes when it should fail (α for this scenario) or fails when it should pass (β for this scenario). Because the measurement variability is expected to be small at the DCGL, the α for this project has initially been chosen to be 0.05, or 5 percent, probability. The β for this project initially has been chosen to be 0.10, or 10 percent, probability.

4.1.6.3 The Gray Region

A LBGR also will need to be selected to apply the statistical test. The LBGR is the concentration level below which further remediation is not reasonably achievable. The statistical test uses the LBGR to define the level that above which false positive rates greater than that specified by the limits on decision errors are accepted. The LBGR is limited by the variability exhibited by the measurements and the decision errors chosen. Because the detection limits expected to be achieved by the direct field measurements are low relative to the $DCGL_W$, it is estimated that an LBGR equal to one-half of the $DCGL_W$ can be achieved for the PBRF decommissioning project. The concentration range between the LBGR and the DCGL defines the gray region of residual radioactivity concentrations in which the consequences of decision errors are relatively minor.

4.1.7 Step 7: Optimizing the Design

The DQO process is neither static nor sequential. New information will be gathered during remediation that will be incorporated into the planning process. The final status survey will be optimized by examining all of the factors that affect the decision errors and sample sizes so that costs and potential risks are balanced. This may include further evaluating the $DCGL_W$, the $DCGL_{EMC}$, and the measurement standard deviation. The estimate of the measurement standard deviation will include both the uncertainty in the measurement process and any anticipated spatial and temporal concentration variations.

4.2 Documentation of the Final Status Survey Plan

The final status survey plan will be documented in a report that summarizes PBRF operations, site characterization data, remediation activities, and all elements of the DQO process. The description of PBRF operations, site characteristics, and remediation will provide perspective and allow the report to function as a stand-alone document. The report will include a description of QA and QC procedures for all elements of the process. The primary focus of the report will be describing the decision process followed to evaluate each survey unit. Detail will be sufficient to recreate the decision in the future.

5. TECHNICAL SPECIFICATIONS

The current Technical Specifications will be adhered to during the decontamination and decommissioning activities. NASA may chose to revise the Technical Specifications as part of a separate licensing action prior to or during the Decommissioning Plan approval process or during decommissioning. However, NASA will comply with only NRC-approved Technical Specifications.

6. PHYSICAL SECURITY PLAN

This section describes the physical security provisions that will be in place during decommissioning of the PBRF. All nuclear fuel has been removed from the PBRF and shipped off site. Therefore, there is no requirement for safeguarding special nuclear material. During decommissioning, industrial security will be provided. Security provisions will provide access control for protection from radiation and industrial hazards and protect capital assets.

During decommissioning activities, access to the PBS will be controlled by physical barriers and security personnel. Access to the PBRF located within the PBS site boundaries will be controlled by means of two fences. The outer fence surrounds the PBS with entry controlled through a single entry control point, which is staffed by guards 24 hours per day, 7 days per week. The inner fence surrounds the PBRF. Access to keys for the PBRF fence gates is limited to personnel authorized by the NASA Decommissioning Project Manager on a need-to-possess basis and is regularly audited.

During decommissioning activities, access to the PBRF will be limited to those personnel required to perform work. Access control requirements for radiologically controlled areas are based on 10 CFR 20 requirements.

Visitors and non-radiological workers must be escorted by a trained radiation worker whenever they are inside the PBRF fenced area.

During non-working hours, the PBRF fenced area gates and all PBRF buildings will be secured. Security personnel will conduct routine inspections of PBRF areas during non-working hours.

7. EMERGENCY PLAN

This section reviews the PBRF licensing history with respect to emergency plans and provides information that supports the conclusion that developing an emergency plan to support decommissioning activities is not required.

The PBRF has been in a standby mode for over 25 years. The facility was shut down in 1973 and the reactor fuel assemblies, all special nuclear material, and source material were removed; the fuel assemblies were transferred off site and much of the facility was decontaminated (NASA 1980b). An emergency plan was not prepared to support NRC-authorized demolition in 1981 (NRC 1981b) nor to support the change in the license status to a possession-not-operate status in 1987 (Dosa 1987). The NRC license was renewed in 1998 with no formalized emergency plan. Technical specifications, issued as part of the 1998 license renewal, require emergency procedures for emergencies arising from fire, floods, and tornadoes and procedure approval by the PBRF Safety Committee (Mendonca 1998).

Section 3.3 of this plan presents a conservative accident analysis that shows offsite impacts are much less than the 15 mrem whole body dose identified as the lowest action level in Table 1 of ANSI/ANS 15.16-1982, "Emergency Planning for Research Reactors." The offsite doses for the accident analysis are low for two reasons. First, the radionuclide inventory at PBRF is limited because the fuel has been removed and much of the facility has been decontaminated. Second, the operations associated with decontamination and decommissioning (localized cutting and decontamination of surfaces) are not the type that would result in large releases of material into the atmosphere.

While no emergency plan is required to deal with accidents involving fuel, the established procedures for conventional emergencies have been enhanced. These procedures address the responsibilities of all parties and the proper actions for a variety of situations, including the following:

a) Medical Emergencies (including a contaminated injured worker)
b) Fire (in both radiological and non-radiological areas)
c) Severe Weather
d) High Airborne Radioactivity
e) Spills
f) Evacuation
g) Natural Disasters

NASA has coordinated the response to various emergencies with the local community emergency responders, including the hospitals, police, and fire departments. Training sessions have been held. Formal written agreements have been signed as well to document the nature of support to be provided.

8. ENVIRONMENTAL REPORT

8.1 Purpose and Need for Action

The NASA PBRF was shut down in 1973. NASA currently has a "posses but do not operate" license for the facility, and has decided to decontaminate the facility to levels that would allow unrestricted release of the PBRF and termination of the license.

8.2 Facility Description

The Plum Brook Station site, which includes the PBRF, is 2614-hectares (6454-acres) in size (NASA 1997). The site is located in a rural area in west central Erie County, Ohio, approximately 6 km (4 mi) south of Sandusky. The major roads near the site are Route 2 to the north, Route 250 to the east, and Interstate 80 to the south (Figure 8-1).

Most of the Plum Brook Station site is in Perkins and Oxford Townships, with some land in Huron and Milan Townships to the east. The site boundaries are Bogart Road to the north, Mason Road to the south, U.S. Highway 250 to the east, and County Road 43 to the west.

The Plum Brook Station includes five major testing facilities: the inactive PBRF and four space testing facilities. Table 8-1 describes the testing facilities.

Figure 8-2 shows the specific buildings and facilities associated with the PBRF. The specific buildings and facilities of PBRF are described in more detail in Section 1.2 of the PBRF decommissioning plan. Radiological contamination of these facilities is primarily inside equipment and waste storage locations. There is limited contamination outside the buildings in the areas of former spills and water handling systems. The dominant radionuclides are H-3, Co-60, Cs-137, and Sr-90. A summary of the radiological contamination in these facilities and the immediate environment is presented in Section 2.2.2 of the PBRF decommissioning plan. In addition to the radiological contaminants, friable lead paint and asbestos or asbestos/fiberglass insulation will have to be managed during decontamination and decommissioning. This is also addressed in the PBRF decommissioning plan.

The Plum Brook Station includes other facilities. The Engineering Building (7141) provides office space and ninety-nine storage bunkers in the southeast (originally used for storing munitions) are now used for warehousing and storing records and equipment that is in the NASA "Hold Storage" system. The Ohio Air National Guard stores munitions in one bunker. Two raw water pumping stations are located offsite and supply water for fire protection and cooling equipment. There is also a small grass airstrip, as well as buildings for mechanical and process equipment, shipping and receiving areas, substations, and cooling towers.

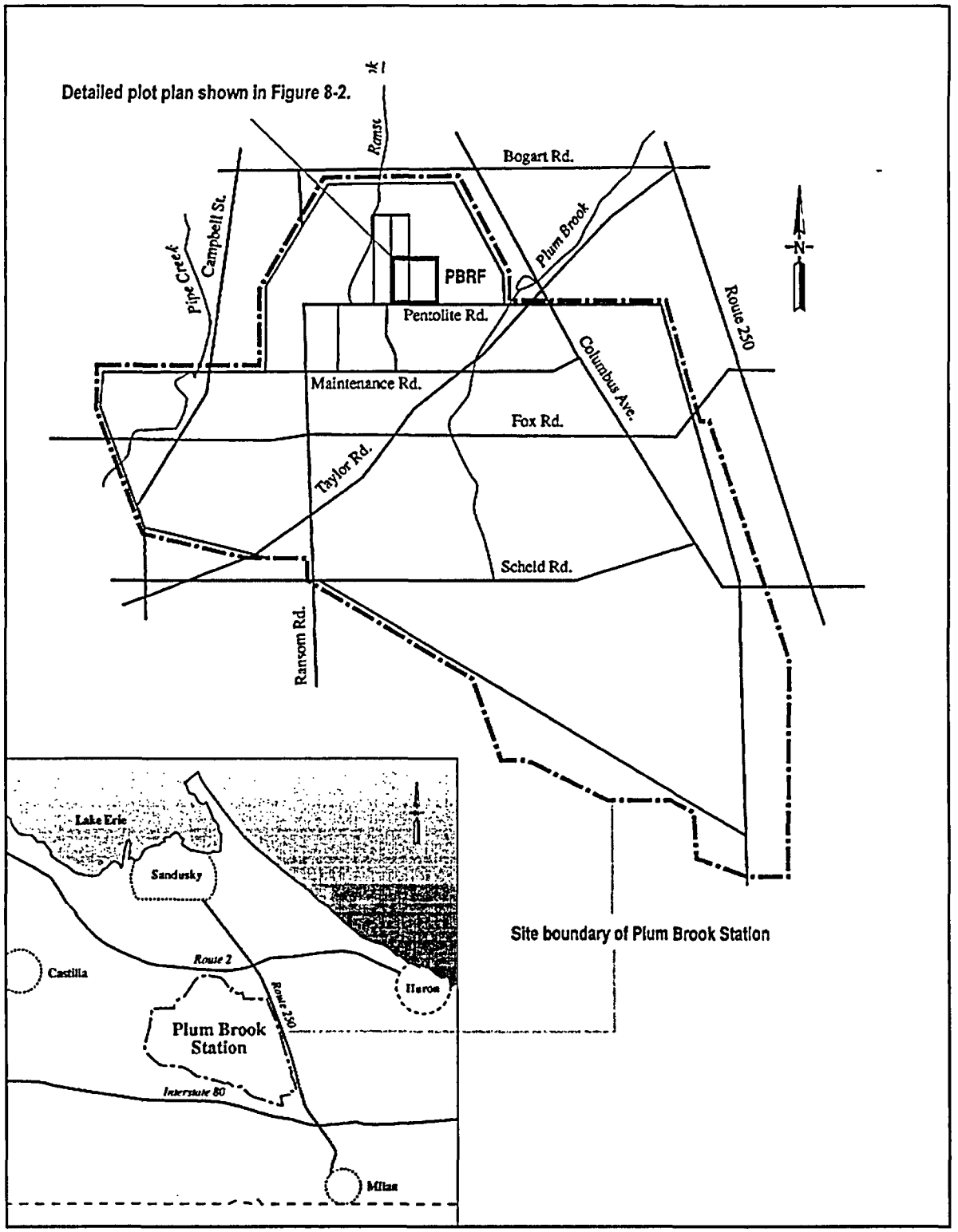


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Figure 8-1. Location of the Plum Brook Reactor Facility at Plum Brook Station (modified from NASA 1980a)

Table 8-1. Plum Brook Station Testing Facilities

Facility	Description
Plum Brook Reactor Facility	Facility includes a 60 MW test reactor, a 100 kW mockup reactor, and a hot laboratory that was used for examination of irradiated material. The reactor and lab facilities were supported by liquid waste management systems, fan house, waste handling building, and office space and analytical laboratories.
Space Power Facility	Facility for testing space power generation and propulsion systems and space hardware under simulated conditions. The aluminum test chamber measures 30 m (100 ft) in diameter and 37 m (122 ft) high with a volume of 23,000 m^3 (800,000 ft^3) and is enclosed by concrete.
Spacecraft Propulsion Research Facility	Chamber for research, development, and validation testing of spacecraft and space propulsion systems that can perform full-mission profile simulation testing for large upper stage rocket engines and complete launch vehicles. Vacuum, cryogenic background temperatures, and solar heating conditions found in near-earth orbit can be simulated.
Cryogenic Propellant Tank Research Facility	Tests rocket propellant tank systems, including cryogenic fluid slush tests, tank fill and expulsion tests, and performance testing of tank insulation systems. The 7.6-m (25-ft) diameter test chamber has a volume of 269 m^3 (9500 ft^3). Propellants types tested there include liquid hydrogen, liquid nitrogen, gaseous hydrogen, gaseous nitrogen, and gaseous helium. Used intermittently for densified hydrogen research programs for advanced engine development.
Hypersonic Tunnel Facility	Used to support the development of air-breathing engines for use in hypersonic aircraft. The aero-thermodynamics of the flight environment is simulated. Propulsion systems and components with synthetic air speeds up to Mach 7 (approximately 8700 kph [5400 mph]) can be tested. Approximately 20,000 m^3 (700,000 ft^3) of gaseous nitrogen and a smaller quantity of gaseous oxygen are stored for use in creating the desired atmospheric test conditions.

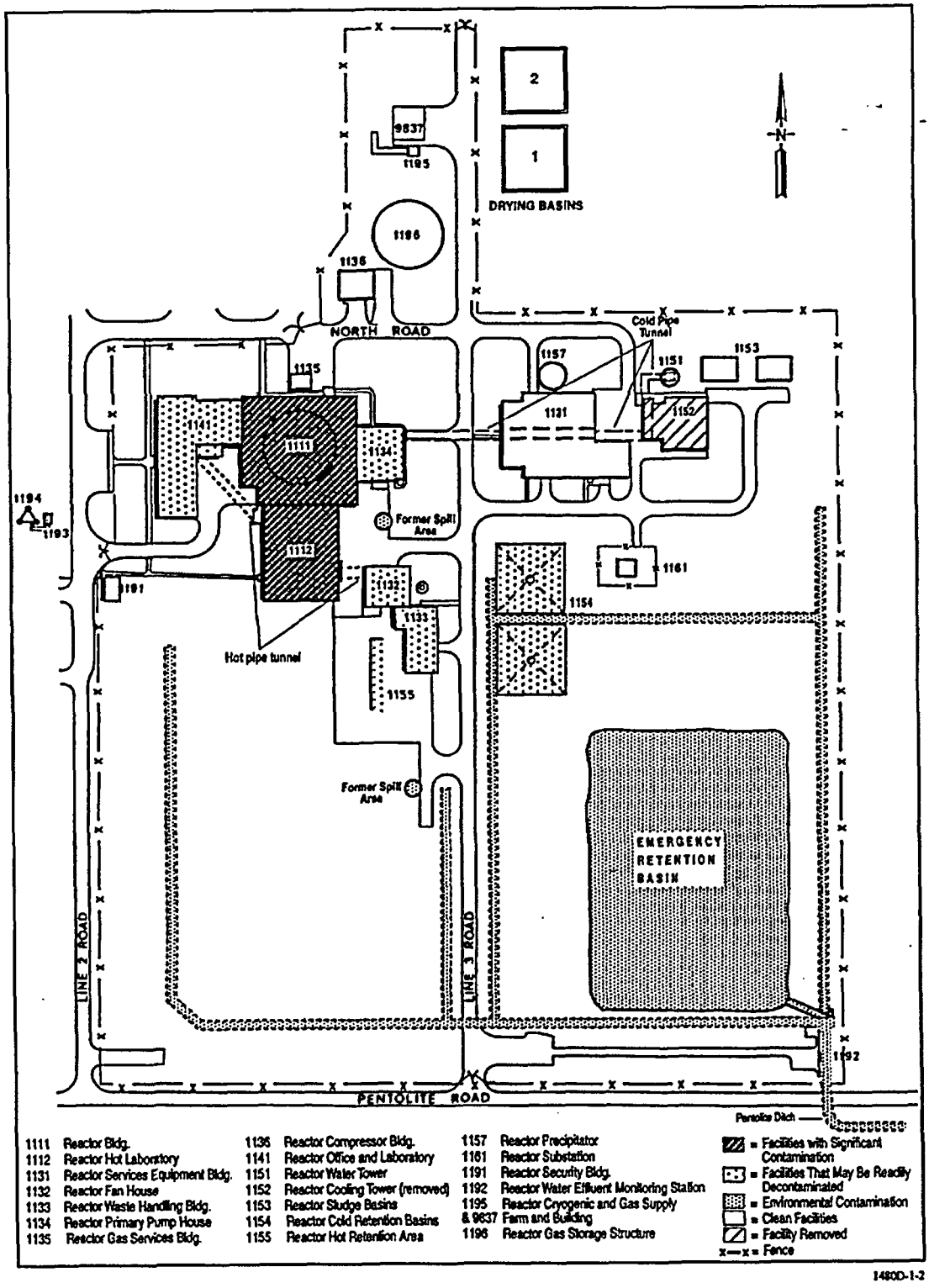

Figure 8-2. Plot Plan of Plum Brook Reactor Facility (modified from NASA 1980b)

8.3 Proposed Action and Alternatives

The proposed action and the alternatives for PBRF decommissioning are as follows:

- Proposed Action (DECON)—Decontamination and decommissioning of the PBRF followed by the release of the site for unrestricted use by the U.S. Nuclear Regulatory Commission (NRC).
- Alternative 1 (SAFSTOR)—In safe storage, the PBRF would continue to be maintained in a condition that allows it to be safely stored and subsequently decontaminated to a level permitting release of the property by the NRC.
- Alternative 2 (ENTOMB)—In entombment, radioactive materials would be encased in a structurally long-lived material such as concrete. The entombed structure would be appropriately maintained and surveillance would continue until the radioactivity decayed to a level permitting release of the property by the NRC.

Implementation of the proposed action would involve performing the following major tasks:

- Removing contaminated equipment, components, and systems
- Removing contaminated material and soil
- Decontaminating buildings and structures
- Demolishing structures to an elevation 1 m (3 ft) belowgrade
- Backfilling belowgrade portions of buildings with clean soil and/or concrete and masonry rubble.

While the decontamination work is in process, remedial action status surveys would be conducted to ensure that the contamination has been removed to the limits required. Final status surveys would also be conducted. Further details on the actions that would be taken to implement the Proposed Action (DECON) are presented in Section 2.3 of the PBRF decommissioning plan.

8.4 Description of the Affected Environment

8.4.1 Topography, Geology, Soils, and Seismicity

8.4.1.1 Topography

Plum Brook Station is situated in the Ohio Lake Plain physiographic region. The topography at Plum Brook Station is relatively flat and slopes gently northward toward Lake Erie. The average slope of the land is less than 6%. Elevations range from about 191-m to 207-m (625-ft to 680-ft) above sea level. The elevation at the PBRF is about 191-m (625-ft) above sea level (SAIC 1991).

8.4.1.2 Geology

Bedrock formations underlying the site consist of carbonates and clastics (sandstones and shales) of Devonian age: Columbus Limestone, Delaware Limestone, Plum Brook Shale/Prout Limestone, and Ohio Shale. The depth to bedrock varies from 0.7 to 7.6 m (2 to 25 ft) across the site and outcrops at certain locations on the site. The depth to bedrock is about 7.6 m (25 ft) in the vicinity of the Reactor Building, where soils have filled in a bedrock low in that area (IT 1999).

8.4.1.3 Soils

Two soil associations occur at the site. The Arkport-Galen association occurs in the northern and western areas of the site, including the area of the PBRF, and the Prout association occurs in the southern and eastern areas. Soils are highly variable in thickness and permeability.

The Arkport-Galen association is characterized by deep, nearly level to moderately sloping, well-drained to moderately well-drained soils that have a subsoil of loamy fine sand and fine sand and occur on sand hills and ridges (SAIC 1991). The Arkport soils are gently to moderately sloping and well drained. The Galen soils are nearly level and moderately well drained. The minor soils occur in level to depressional areas and in the flat areas between the sand hills and ridges. The minor soil associations are either very poorly or somewhat poorly drained.

The Prout association has moderately deep to deep, nearly level to gently sloping, somewhat poorly drained soils that have a subsoil of heavy silt loam to silty clay loam. This association occurs on uplands, such as the sides of stream valleys, shale outcrop ridges, along drainage ways, and in some steeper areas. The Prout soils are nearly level to gently sloping, dark colored, and somewhat poorly drained. These soils are underlain by shale bedrock at a depth ranging from 51 to 102 cm (20 to 40 in.) for the Prout soils and 102 to 152 cm (40 to 60 in.) for deep variant Prout soils. The minor soils in this association include a broad spectrum from nearly level to depressional and very poorly drained to nearly level to gently sloping and well drained (SAIC 1991).

8.4.1.4 Seismicity

Occasional earthquakes in Ohio appear to be associated with ancient zones of weakness in the Earth's crust. The historic record suggests a risk of moderately damaging earthquakes in the western, northeastern, and southeastern parts of the state. The Plum Brook Station site is located in Seismic Zone 1 according to the 1990 Ohio Building Code.

8.4.2 Climate and Air Quality

The climate at the Plum Brook Station is continental in character and influenced by its proximity to Lake Erie. Summers are moderately warm and humid, with temperatures occasionally exceeding 32°C (90°F). Winters are cold and cloudy, with temperatures falling below –18°C (0°F) an average of 5 days per year. Annual temperature extremes typically occur after late June and in December. The first frost typically occurs in October (NASA 1997). The predominant wind direction is southwest throughout the year. In spring and summer, northerly and northeasterly breezes also blow from the lake (NASA 1997). Average annual precipitation at the Plum Brook Station is 86-

cm (34-in.) (1951–1980 data). The 2-year, 24-hour rain event is 6.2-cm (2.45-in.). Average annual water loss is estimated at 57-cm (22.5-in.) (NASA 1997).

Plum Brook Station is located in Erie County, which is in attainment for all National Ambient Air Quality Standards. The Ohio Environmental Protection Agency (EPA) maintains a monitoring station in Erie County for total suspended particulate levels. The site is not classified as a major emission source under the Clean Air Act Title V permitting program.

Emission sources in the surrounding area include the Ford Motor Company in Sandusky and some large coal-fired institutional boilers (NASA 1997).

8.4.3 Hydrology

8.4.3.1 Groundwater

Two principal bedrock aquifers underlie the site (Morrison Knudsen 1994). A fractured limestone aquifer occurs in the western portion of Erie County, and groundwater flow is to the north. A fine-grained shale aquifer to the east has low yields, and the Ohio Department of Natural Resources (ODNR) has delineated three groundwater zones based on well yield. The PBRF is located in an area where wells with a capacity of 19-L to 95-L (5-gal to 25-gal) per minute can be developed (NASA 1997).

One hundred seventy-nine private drinking water wells are located within a 6-km (4-mi) radius of Plum Brook Station based on a record search of the Erie County Health Department (Morrison Knudsen 1994). No downgradient groundwater wells are known to be used for industrial or agricultural purposes. The closest recorded downgradient well for the entire Plum Brook Station is at 6115 Schenk Road, but this is crossgradient from the PBRF. The 1991 survey of permitted wells did not identify any well downgradient of the PBRF (SAIC 1991).

8.4.3.2 Surface Water

Plum Brook Station is located in the Lake Erie watershed. The Huron River and its branches constitute the major surface water system. Eleven streams cross the site, the largest of which are Pipe Creek, Kuebler Ditch, Ransom Brook, and Plum Brook. Streams generally flow northward and converge into Ransom Brook, Storrs Ditch, Plum Brook, and Sawmill Creek and eventually flow north into Lake Erie. Seventeen isolated ponds and reservoirs are located on the site (NASA 1997).

All of the 27-acre PBRF Site is graded to cause surface water to drain out through the Waste Efficient Monitoring System (WEMS), to Pentolite Ditch, and then into Plum Brook.

The largest surface water body near Plum Brook Station is Sandusky Bay on Lake Erie, approximately 6-km (4-mi) to the north. The lake is an important fresh water fishery with a combined commercial and sport fishery catch estimated to exceed 20 million fish. Most commercial fishing takes place near Sandusky Bay. Lake Erie is also used for recreational purposes.

8.4.4 Biologic Resources

Plum Brook Station is part of a regional ecosystem encompassing Sandusky, parts of Lake Erie, and several Lake Erie islands. Several natural areas are found in the general vicinity. The Milan State Wildlife Area is located approximately 5-km (3-mi) to the south. The Erie Sand Barrens State Nature Preserve is approximately 305-m (1000-ft) to the south. The Sheldon Marsh State Nature Preserve is approximately 6-km (4-mi) to the northwest, and the Resthaven Wildlife Area is approximately 10-km (6-mi) to the northwest. Another local natural area is Old Woman Creek, a National Estuarine Research Reserve and State Nature Preserve, which is east of the city of Huron (NASA 1997).

8.4.4.1 Vegetation

Plum Brook Station contains significant areas of grassland, bushland, and woodland. A biological survey conducted in 1994 determined that no significant plant communities were located at Plum Brook Station. About 330 vascular plant species were collected or observed during the 1994 survey, and of these, 251 species are considered indigenous to the area. Areas of greatest plant diversity are in the central and southern portion of Plum Brook Station and not near PBRF (NASA 1997). Open burning is conducted annually for weed control and to assist in establishing field grasses.

8.4.4.2 Wildlife

Wildlife at the site includes white tailed deer, raccoons, woodchucks, moles, starlings, pigeons, coyotes, hawks, Canada geese, and turkey vultures. Periodic controlled deer hunting occurs to manage wildlife populations and to control overgrazing. A total of 116 bird species has been identified at the site (NASA 1997). Of these, 92 species were either confirmed or likely nesters. Five species were considered to be late migrants and nine species visitors only. Common birds at Plum Brook Station include the American robin, song sparrow, field sparrow, indigo bunting, common yellowthroat, blue jay, and house wren. Nineteen reptile and 13 fish species have been identified. All of the fish species are common State-wide and tolerant of water quality and habitat degradation except for the brook stickleback.

The biological survey identified one Federally listed species, the bald eagle (*Haliaeetus leucocephalus*), and three State-listed endangered, four threatened, six potentially threatened, and three species of special concern. The bald eagle is classified as transient; none have ever been seen to nest at Plum Brook Station. These are summarized in Table 8-2. Other rare bird species identified at Plum Brook Station include the Cooper's hawk, Alder flycatcher, Least flycatcher, Marsh wren, Brewster's warbler, Black-throated green warbler, and Henslow's sparrow.

The Indiana bat (*Myotis sodalis*) is a Federally listed endangered species that has been reported in Erie County. Other State protected species reported in the County include the western banded killifish (*Fundulus diaphanous menona*), least bittern (*Ixobrychus exilis*), eastern pondmussel (*Ligumia nasuta*), and the common tern (*Sterna hirundo*) (NASA 1997).

Table 8-2. Special Status Animals and Plants Residing at the Plum Brook Station

Status	Species	Common Name
Endangered	*Hypericum gymnanthum*	Least St. John's-wort
	Cistothorus platensis	Sedge wren
	Carex cephaloidea	Thin-leaf sedge
Threatened	*Arenaria laterifolia*	Grove sandwort
	Carex conoidea	Field sedge
	Helianthus mollis	Ashy sunflower
	Bartramia longicauda	Upland sandpiper
Potentially threatened (plants)	*Baptisia lactea*	Prairie false indigo
	Carex alata	Broad-winged sedge
	Gratiola virginiana	Round-fruited hedge-hyssop
	Hypericum majus	Tall St. John's-wort
	Rhexia virginiana	Virginia meadow-beauty
	Viola lanceolata	Lance-leaved violet
Special concern (animals)	*Emydoidea blandingii*	Blanding's turtle
	Elaphe vulpina gloydi	Eastern fox snake
	Opheodrys vernalis	Smooth green snake

Source: NASA (1997).

8.4.5 Population and Land Use

The 1990 population of Erie County was 76,779, with a total of 32,827 housing units. During the summer, the population at Sandusky increases by approximately 50% because of tourism (NASA 1997).

The area surrounding Plum Brook Station is largely rural and agricultural. Some food processing facilities, including dairy and meat processing operations, are located in the area. During the summer, tourism and recreation are important economic influences in the Sandusky area.

Most of the land at the Plum Brook Station consists of forestland and old fields. About 25% of the acreage is used for offices, test facilities, roads, and infrastructure. The remaining portions of Plum Brook Station are unused. Other organizations maintaining offices or using space at Plum Brook Station include the U.S. Department of the Interior, U.S. Department of Agriculture, the Federal Bureau of Investigation, U.S. Department of Labor, U.S. Department of Defense, Immigration and Naturalization Service, U.S. Coast Guard, U.S. Army Reserve, and Ohio Air and Army National Guards (NASA 1997).

8.4.6 Cultural and Historical Resources

The Spacecraft Propulsion Research Facility (B-2 Facility) at Plum Brook Station has been designated a National Historic Landmark. Approximately 133 Native American archaeological

sites outside the Plum Brook Station fence line have been placed on the Ohio Historic Society Register (NASA 1997). The Ohio State Historic Preservation Officer has informed NASA that the PBRF is not considered an historic site.

8.4.7 Socioeconomics and Environmental Justice

Plum Brook Station employs about 120 people (NASA 1997). Large employers in the area include the Ford Motor Company, Delco-Chassis NDH, Imperial Clevite, Sandusky Plastics, and Sandusky Foundry and Machine. NASA's presence in the area provides local economic impacts and benefits nonetheless.

The *Environment Justice Implementation Plan for NASA Lewis Research Center* (Jones Technologies, Inc. 1996) determined that that there were no substantial offsite impacts from the Plum Brook Station. The plan also identified that the minority populations (4200 blacks and 450 Hispanics) were located in the town of Sandusky, which has a total population of approximately 30,500. Sandusky and these populations are located 8 km (5 mi) or more from the Plum Brook Station.

8.4.8 Transportation

Plum Brook Station includes a 101-km (62.5-mi) internal paved road system. There is also a 25-km (15.7-mi) rail line that is currently unused (Morrison Knudsen 1994). Several State roads service the area. Route 250 is just to the east of the site and serves as a major route to the Plum Brook Station. The Ohio Turnpike (Interstate 80 and 90) is located 8-km (5-mi) south of the main entrance to the Plum Brook Station. Two major railroads, Conrail and Norfolk & Southern, serve the area.

8.4.9 Noise

Sources of noise at Plum Brook Station include an airstrip, transient noise blasts from test facilities, construction activities, and traffic noise. The Army Reserves and the Ohio Air National Guard also discharge pyrotechnic devices at Plum Brook Station (NASA 1997). None of these activities is a significant noise source, in part because impacts are mitigated by the large distances to offsite receptors. None of the activities occurring during decommissioning would result in sustained offsite noise impacts.

8.4.10 Background Radiation Levels

The public is continuously exposed to radiation from natural sources, primarily from cosmic radiation; external radiation from natural material in the earth and global fallout; and internal radiation from natural radioactive materials taken into the body via air, water, and food. The public receives and accepts the risks associated with radiation exposures from medical x-rays, nuclear medicine procedures, and consumer products. On average, a member of the public in the United States receives approximately 300 mrem/yr from natural sources of radiation; 50 mrem/yr from medical procedures; and 10 mrem/yr from consumer products, for a total of 360 mrem/yr (NCRP 1987).

8.5 Environmental Impacts of Proposed Action and Alternatives

This section discusses the potential direct and cumulative effects of the proposed action on human health and the environment.

8.5.1 Environmental Impacts of the Proposed Action

8.5.1.1 Human Health Effects

This section identifies and discusses expected impacts to workers and people offsite from normal PBRF decommissioning activities and potential accidents. The general nature of industrial and radiological hazards associated with PBRF decommissioning are identified in Section 2.3.7 and Table 2.21 of the PBRF decommissioning plan.

Industrial Hazards

The decontamination and decommissioning operations will involve several hundreds of thousands of hours of labor. Activities will include soil excavation, concrete removal, piping and equipment removal, and building demolition. Workers will be exposed to industrial hazards and there is the potential of occupational accidents. The hazards associated with these activities will be identified and managed as discussed in the PBRF decommissioning plan.

Radiological Hazards

Estimated Worker Exposure

The collective dose equivalent estimate to workers for the entire decommissioning project is about 70 person-rem over the approximate 4-year decommissioning project (see Section 3.1.3 of the PBRF decommissioning plan). Total person hours involving radiological exposure is estimated to be 100,000 hours. The estimated occupational exposure for the DECON alternative of the reference test reactor in the *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities* (NUREG-0586) is 344 person-rem (NRC 1988). These exposure estimates are consistent given the fact the estimate in the PBRF decommissioning plan is for DECON after about 30 years of decay, while the NRC estimate is for a DECON alternative shortly after shutdown of the reactor.

Occupational exposure associated with shipment of the low level waste was estimated in NASA's 1980 environmental report to be 18 person-rem (NASA 1980). This is similar to the estimate of 22 person-rem for the reference test reactor presented in NRC (1988). Because of the decay that has occurred since reactor shutdown, the actual doses to offsite populations are expected to be even less. A scaled estimate of the occupational exposure associated with waste transportation based on the "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NEC-licensed Nuclear Facility" (NUREG-1496) is 5 person-rem.

Estimated Public Exposure

The dose to the offsite public from routine releases is expected to be small. NUREG-0586 estimates this dose to be negligible (less than 0.1 person-rem) (NRC 1988). This is consistent with general conclusion drawn from the conservative accident analysis presented in Section 3.3 of the PBRF decommissioning plan. The largest accident analyzed resulted in an offsite dose of about 0.5 mrem.

The cumulative offsite dose because of shipping of radioactive waste will be small. In its 1980 Environmental Report (NASA 1980a), NASA estimated the population dose due to shipment of waste to be 8.2 person-rem. NUREG-0586 estimates population dose for waste shipment to be 22 person-rem (NRC 1988). These estimates are generally consistent because the 1980 estimate was based on waste shipments after about a decade of decay while the NRC estimate assumed the shipments were made a few years after reactor shutdown. The population dose that will occur during planned decommissioning should be less than these estimates because of the decay that has occurred since reactor shutdown. A scaled estimate of public exposure associated with waste transportation based on the "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NEC-licensed Nuclear Facility" (NUREG-1496) is 0.5 person-rem.

The dose due to potential accidents will also be small. The conservative accident analysis presented in Section 3.3 of the PBRF decommissioning plan shows that offsite doses to the maximally exposed individual should be less than 0.5 mrem. This is consistent with the assessment made in NUREG-0586, which showed the maximum dose due to onsite accidents to be 0.25 mrem to the lung (NRC 1988).

The anticipated potential exposures to the public after license termination is also negligible. The site will have been released to unrestricted use. This means the maximum dose to the "average member of the critical group" will be less than 25 mrem. In fact, any realistic estimate of dose will be much less than 25 mrem/yr because decontamination will be more extensive than required to meet minimum license termination requirements and exposure will not occur for some time because NASA has no plans to make the site available for public reuse. The projected dose after license termination will be dominated by nuclides such as Cs-137 and Co-60, which have half-lives 30 and 5 years, respectively.

Nonradiological Transportation Impacts

Transportation would be conducted in accordance with applicable U.S. Department of Transportation, U.S. EPA, and NRC regulations. The radiological impacts of incident-free transportation will be minimal as discussed above. During such transport, hazardous and radioactive materials will be effectively packaged to prevent significant radiation external to the truck or rail car. The primary nonradiological impacts would be due to emissions and noise from the trucks or trains themselves and potential accidents resulting in injuries or fatalities.

8.5.1.2 Environmental Impacts

Air Quality

Several decommissioning-related activities could minimally impact air quality because of both mobile and stationary source emissions. A small amount of mobile source emissions, such as carbon monoxide and nitrogen oxides, could be released from contractors' equipment, such as backhoes, cranes, trucks, and cars. The impact of these sources should be minimal.

Hydrology

The site elevation is approximately 191-m (625- ft) above mean sea level. It is not within either the 100-year or 500-year flood plain. Groundwater is currently pumped at the PBRF to prevent it from entering the basements of buildings, such as the Reactor Building. The pumping has created a localized cone of depression in the groundwater surface, but it has no impact on the larger groundwater flow. At some time, groundwater pumping will be terminated, the local groundwater depression will cease, and the general groundwater flow pattern (flow to the north and north east) will establish itself over the entire PBRF area.

Biologic Resources

The PBRF is an industrial area with no known sensitive or endangered species.

Population and Land Use

The proposed action will involve less than 100 additional employees at Plum Brook Station for the duration of the project. There will be no change in land use as a result of the decommissioning project. The area of the PBRF will remain as part of the buffer zone for the Plum Brook Station.

Cultural and Historical Resources

There are no cultural resources on the site of the PBRF and the decommissioning project will not impact other portions of the Plum Brook Station (NASA 1997)

No historical survey is reported for the Plum Brook Station. There are Native American archaeological sites outside the Plum Brook Station fenceline. There may be similar sites on the Plum Brook Station grounds, but no undisturbed ones would be expected at PBRF because of the extensive site construction that occurred during the late 1950s and early 1960s (NASA 1997)

Aesthetics

The PBRF has minimal visibility from offsite locations. The decommissioning project will remove most of the structures in the area of PBRF and restore the land closer to its condition before the PBRF was constructed. The remainder of the Plum Brook Station will not be impacted by the proposed action.

Socioeconomics and Environmental Justice

During decommissioning of PBRF, less than 100 people will be employed for the duration of the project. This labor is a small fraction of the total Erie County labor force, which is about 40,000 (PeopleVision, 1996). The offsite impacts of PBRF decommissioning will be minimal and there would be no disproportionate impacts on minority populations.

Noise

During PBRF decommissioning activities, local noise will be generated by equipment such as jackhammers, scabblers, and concrete saws. Backhoes and other heavy equipment could also be used for partial dismantling activities. Onsite workers will be outfitted with ear protection devices. The closest offsite receptors are over 914 m (3000 ft) away. Noise from PBRF decommissioning activities should have no impact offsite.

8.5.1.3 Cumulative Effects

Cumulative impacts for PBRF decommissioning will be minimal. The only impact will be the small impacts associated with the disposal of radioactive and nonradioactive waste at licensed facilities. Only small amounts of hazardous waste are expected to be generated as a result of PBRF decontamination and decommissioning. The majority of the hazardous waste is expected to be generated from removing friable lead paint. The waste will be removed by a licensed contractor and disposed of at a licensed facility.

The total estimated volume of low-level radioactive and mixed waste from Plum Brook decommissioning is about 3100 m^3 (110,000 ft^3). Most of this volume is Class A. A small fraction would be classified as Class B or C. It is also estimated that there will be a very small amount (about 2.1 m^3 [75 ft^3]) of mixed waste comprised of contaminated lead paint scraping or chips.

Some nonhazardous solid waste will be generated during decontamination and decommissioning. The material that has scrap value (e.g., copper wire and steel plate) will be recycled. Clean demolition debris will be used as fill material for decontaminated belowgrade structures. Material that has no scrap value and is not acceptable for fill will be disposed offsite in an industrial landfill.

The impacts of waste disposal actions should be within the limits of impacts analyzed when the waste disposal facilities such as Barnwell or Envirocare were granted their licenses.

8.5.2 Environmental Impacts of Alternatives

8.5.2.1 Safe Storage

Alternative 1 to the proposed action is Safe Storage (SAFSTOR). Implementing this alternative would necessitate continued surveillance and maintenance of the PBRF over a period of time. Impacts during the storage period would be minimal although there would be substantial monitoring and maintenance costs. Eventually, decontamination and decommissioning would be

required. The impacts of delayed decontamination and decommissioning would be comparable or slightly less than those of the proposed action.

8.5.2.2 Entombment

Alternative 2 to the proposed action is Entombment (ENTOMB). Implementing this alternative would necessitate continued surveillance and maintenance of the PBRF over a substantial time period until the activity has decayed to minimal levels. The time period for this level of decay has not been determined for PBRF. Information presented in NUREG-0586 (NRC 1988) and preliminary dose analyses conducted by NASA suggest entombment would have to last for timeframes on the order of a hundred years. There would be costs associated with such long-term monitoring and maintenance.

9. CHANGES TO THE DECOMMISSIONING PLAN

NASA may want to make changes to portions of the decommissioning plan, including the description of actions that will be taken during decommissioning, the organizations that will be involved in decommissioning and their specific role, procedures in effect during decommissioning, or specific programs that will be maintained during decommissioning. NASA may want to make such changes to improve safety or the cost effectiveness of the overall operation.

NASA will prepare a change control procedure to determine if such a change can be made without prior NRC approval. This procedure will require applying the test identified in 10 CFR 50.59 as they apply to non-power reactors in decommissioning (i.e., Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Test, and Experiments"). This procedure will (1) identify the criteria and methods to be used to determine whether a proposed change can be implemented without prior NRC approval, (2) specify the review and approval process, and (3) identify the documentation and reporting requirements. The impacts of the proposed change will be determined by conducting an analysis comparable to that presented in Section 3.3 of this plan. If the analysis concludes that the proposed change will not (1) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the decommissioning plan as approved; (2) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the decommissioning plan as approved; (3) Result in more than a minimal increase in the consequences of an accident previously evaluated in the decommissioning plan as approved; (4) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the decommissioning plan as approved; (5) Create a possibility for an accident of a different type than any previously evaluated in the decommissioning plan as approved; (6) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final decommissioning plan as approved; (7) Result in a design basis limit for a fission product barrier as described in the decommissioning plan as approved; or (8) Result in a departure from a method of evaluation described in the decommissioning plan as approved used in establishing the design bases or in the safety analyses; then the proposed change can be made by NASA without NRC approval.

In addition, the proposed changes will be evaluated for conformance to License Condition 3.A.1. This section stipulates the licensee may make changes to the Decommissioning Plan without prior NRC approval provided that the changes do not (1) Require Commission Approval pursuant to 10 CFR 50.59; (2) Reduce the coverage requirements for scan measurements; (3) Increase the DCGL and related minimum detectable concentrations (for both scan and fuel measurement methods); (4) use a statistical test other than the Sign test or Wilcoxon Rank Sum test for evaluation of the final status survey; (5) Result in significant environmental impacts not previously reviewed; (6) Increase the radioactivity level, relative to the applicable DCGL, at which an investigation occurs; (7) Increase the Type I decision error; or (8) Decrease an area classification (i.e., impacted to unimpacted; Class 1 to Class2; Class 2 to Class 3; or Class 1 to Class 3).

The NASA change will be contingent upon review and approval of the analysis by the Decommissioning Safety Committee, the Project Radiation Safety Officer, and the Decommissioning Project Manager.

NASA will maintain records of decommissioning plan changes made until decommissioning activities have been completed. NASA will submit an annual report to NRC that identifies the changes that were made.

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APPENDIX A

1998 CONFIRMATORY CHARACTERIZATION SURVEY FOR THE PLUM BROOK REACTOR FACILITY

APPENDIX A

1998 CONFIRMATORY CHARACTERIZATION SURVEY

A.1 Survey Objective

A radiological survey of the Plum Brook Reactor Facility (PBRF) was conducted in 1985 as documented in "An Evaluation of the Plum Brook Reactor Facility and Documentation of Existing Conditions, Volume 3: Physical Characterization of Radioactive/Contaminated Areas of the PBRF," by Teledyne Isotopes, December 1987. A confirmatory survey was conducted in September 1998 to verify the 1985 results and to provide additional isotopic data to use for estimating doses license termination.

During the 1998 confirmatory survey, the areas known to require remediation (e.g., the Emergency Retention Basin [ERB] and the Pentolite Ditch) were sampled to confirm the 1985 data. The 1998 survey also examined areas and buildings that were expected to not require decontamination. These areas were examined because contamination in these areas could impact decommissioning planning and costs.

A.2 Survey Sampling Design

Surveys were performed according to project-specific procedures and the *Sampling and Survey Plan for the Plum Brook Test Reactor Facility*. The procedures identified survey instrument requirements, measurement and sample collection methods, and data reduction and evaluation methods. The sampling and survey plan identified the survey protocols. Implementation of the sampling and survey plan included the following:

- Developing survey packages (portfolios) for the survey areas
- Mapping the survey locations as applicable
- Collecting survey measurements and analyzing samples using appropriately calibrated instruments
- Downloading the survey data into a database for storage and processing
- Reviewing completed survey packages to ensure that all required surveys were performed and that the completed survey packages contained all necessary information
- Comparing the survey results with the 1985 characterization data
- Identifying areas that were not previously identified during the 1985 survey.

The sampling and survey design is described in Sections A.2.1 through A.2.5.

A.2.1 Survey Package Development

For each area surveyed, a survey package, or portfolio, was developed by performing a walk-down and preparing a worksheet/tracking sheet outlining the general survey instructions, location codes,

and specific survey instructions for any abnormal conditions within the area. Survey progress was tracked using completion and review signature blocks. The PBRF areas sampled during the 1998 confirmatory survey (shown on Figure A-1), along with a brief characterization of the 1985 results, are listed below.

- *Emergency Retention Basin*—Slightly contaminated water from the plant effluent was diverted to the Emergency Retention Basin and allowed to evaporate or percolate into the ground. The 1985 characterization identified low levels of contamination in this area.
- *Water Effluent Monitoring Station (Building 1192)*—The site effluent was continuously monitored and discharged at the Water Effluent Monitoring Station. If radioactivity was detected in the effluent, the gates were closed and the water was diverted to the Emergency Retention Basin. The 1985 characterization identified no activity on the building, but small amounts of activity were identified in the sediments.
- *Pentolite Ditch*—The Pentolite Ditch receives all site effluent after it passes through the Water Effluent Monitoring Station and discharges the effluent at Plum Brook. The 1985 characterization identified areas of low-level radioactivity requiring remediation.
- *PBRF Grounds*—The outdoor PBRF areas enclosed within the fence were extensively surveyed and sampled during the 1985 characterization to assess the potential of ground contamination from stack fallout, spills, and spread by personnel traffic. The 1985 characterization identified a few localized areas of contamination.
- *PBRF Pavement*—No characterization data from the 1985 survey were available for the access ways paved with asphalt and concrete.
- *Catch Basins*—Approximately 40 catch basins and stormwater basins at the PBRF drain to the Water Effluent Monitoring Station. No significant activity was found in these basins during the 1985 characterization, except for low levels of radioactivity in the sediments that had collected in the basins.
- *Cold Retention Basins (1154)*—The Cold Retention Basins consist of two retention basins designed for storing low-level contaminated water from the reactor quadrants and canals. The silt that had accumulated in the Cold Retention Basins was found to be contaminated.
- *Reactor Building (1111)*—The Reactor Building has two basement levels, a main level, and some second floor offices. The Reactor Building houses the containment vessel that encloses the reactor tank and associated testing equipment and reactor systems. The 1985 characterization showed that the quadrants and canals were contaminated, as well as some surface areas of the rooms.
- *Reactor Office and Laboratory Building (1141)*—The Reactor Office and Laboratory Building houses personnel offices, laboratories, drafting, photographic developing, health physics, and electronic services. The 1985 characterization identified some contamination in Rooms 212 and 214, as well as inside several of the laboratory hoods.

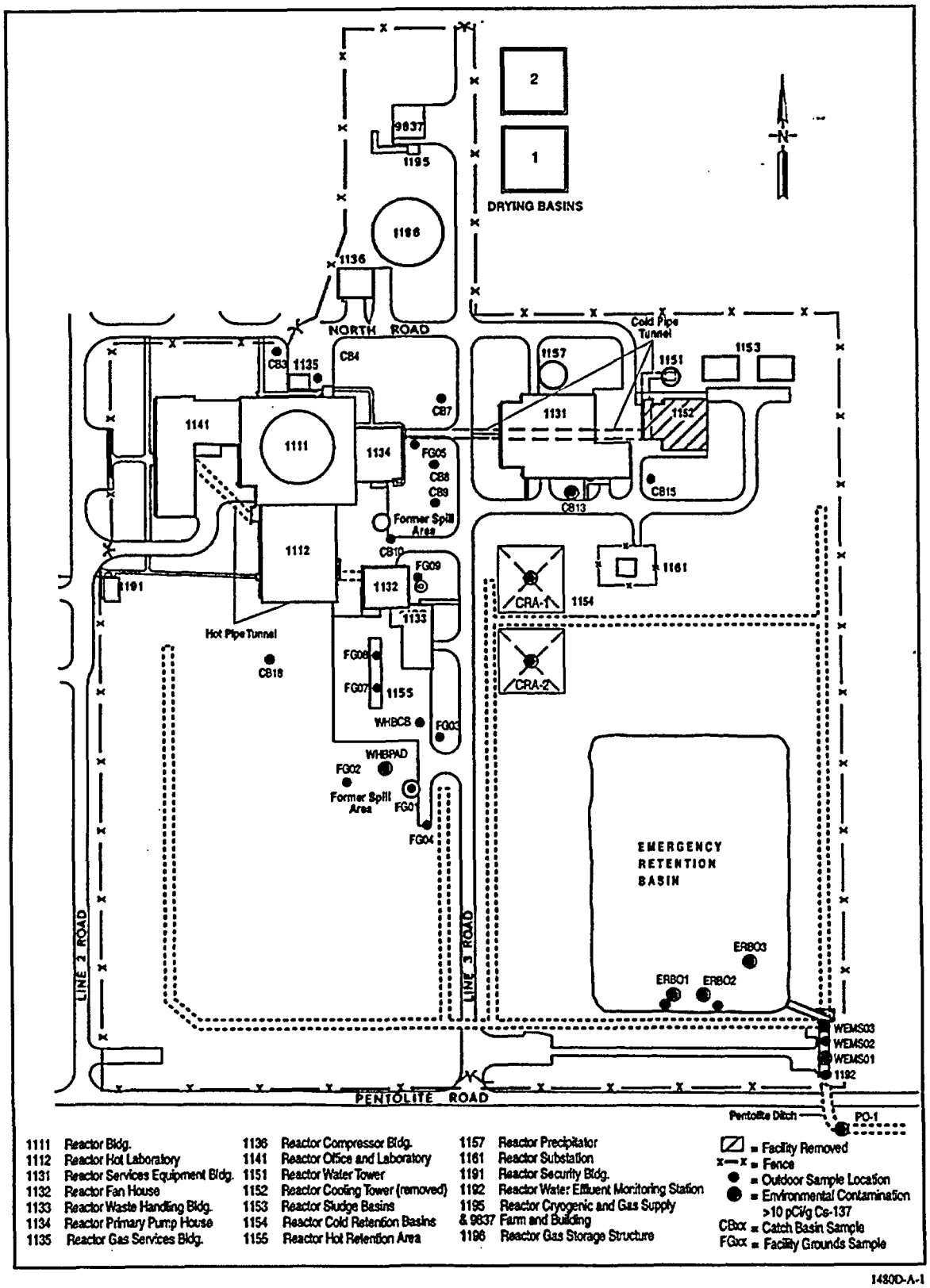

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Figure A-1. PBRF Outdoor Sample Locations

- *Reactor Service Equipment Building (1131)*—The Reactor Service Equipment Building was not surveyed during the 1985 characterization because it was considered to be a clean building.
- *Fan House (1132)*—The Fan House contains the ventilation systems for the Reactor Building and hot cells. The 1985 characterization (Teledyne Isotopes, 1987) showed no contamination on the building structure, but contamination was found in the ventilation equipment.
- *Waste Handling Building (1133)*—The Waste Handling Building contained the waste evaporator, contaminated systems, and small quantities of low-level radioactive waste. The 1985 characterization showed no contamination on the building structure, but contamination was found in some of the waste handling systems.
- *Service Tunnels*—Three underground service tunnels, excluding the hot pipe tunnel, connect the Reactor Building with the other PBRF buildings. Minimal contamination was identified in these tunnels during the 1985 characterization.
- *Canal F*—Canal F is located outside of the containment vessel and was found to contain contamination.
- *Petri Dish Samples*—In addition to samples collected in 1998, NASA provided samples taken from sumps during the 1985 survey for isotopic analysis.

The levels of contamination at these areas are identified in Section A.4.

A.2.2 Survey Requirements

The survey protocols, based in part on the 1985 characterization surveys, are specified in the *Sampling and Survey Plan for the Plum Brook Test Reactor Facility*. These protocols consist of both direct beta surface activity measurements and sampling. Where applicable, the survey protocols (e.g., sampling depths and intervals) used during the 1985 characterization were duplicated to confirm the results.

The areas that were not surveyed during 1998 are the hot cells, inside the reactor containment vessel, hot pipe tunnels, inside the Hot Retention Area, resin pits, and other primary systems. These areas are known to be contaminated, and additional surveys would not result in changes to the decommissioning costs.

A.2.3 Radionuclides of Concern

The radionuclides of concern at the PBRF consist of both mixed fission products and activated materials, with the primary radionuclides being Co-60 and Cs-137. Other mixed fission products and activated materials are also present; however, the quantities of these radionuclides constitute a small percentage of the total activities.

For the confirmatory survey, only the easily detected radionuclides were analyzed (by gamma spectroscopy) and quantified. As a result, beta emitters; radionuclides that are difficult to detect (i.e., Sr-90, Fe-55, and Ni-63); and other low-energy beta emitters were not identified and

quantified. The analysis for the primary gamma emitters (Co-60 and Cs-137) and europium were considered adequate to verify the contaminated and uncontaminated areas identified from the 1985 characterization data.

A.2.4 Gridding

Because the 1998 survey was designed to confirm and supplement the 1985 survey, new gridding was not performed. Most measurements and samples were taken at biased locations based on the 1985 characterization data and the requirements of the sampling and survey plan.

A.2.5 Survey Records

Survey records were maintained in the area-specific survey packages according to project procedures. Each survey package included the following records, if appropriate:

- Survey Package Worksheet, giving the survey package identification, survey location information, general survey instructions, and any specific survey instructions
- Survey Comment Addendum, containing comments from the survey technician regarding any unusual situation encountered while surveying
- Survey Unit Diagram, a diagram of the area to be surveyed, if available
- Printout of laboratory analysis results
- Ludlum Model 2350 data files and Paradox® converted values for all radiation survey measurements.

Total beta surface activity was directly measured using the Ludlum Model 2350 Data Logger system. After completing a survey, the contents of the Data Logger's memory were downloaded to a database.

A proprietary computer program was used to generate a survey report that presented the raw data and converted data by survey location. The survey technician and supervisor reviewed these reports for completeness, accuracy, and suspect entries and compared the data to the 1985 characterization data.

Data and document control included maintaining the raw data files and translated data files (Paradox® database files) and documenting all corrections made to the data. The databases were backed up daily.

A.3 Survey Instrumentation

The survey instruments used had sensitivities that were sufficient to detect the identified primary radionuclides at the minimum detection requirements. Table A-1 provides a list of the survey instruments, types of radiation detected, and calibration sources used during the 1998 confirmatory survey.

Table A-1. Survey Instrumentation

Instrument/ Detector	Detector Type	Radiation Detected	Calibration Source	Use
Ludlum Model 2350/43-68, 43-98 or 43-94	Gas-flow proportional (126 cm^2)	Alpha or beta	Tc-99 (β) Th-230 (α)	Direct measurements and smear counting
Ludlum Model 2350/44-2	NaI scintillator	Gamma	Cs-137	Gamma exposure rate
Ludlum Model 2350/44-40	Shielded GM (15.5 cm^2)	Beta	Tc-99 (β)	Direct measurements
Ludlum Model 2350/43-5	ZnS scintillator (45.5 cm^2)	Alpha	Th-230 (α)	Direct measurements
Tennelec LB 5100 Planchet Counter	Gas-flow proportional	Alpha and beta	Tc-99 (β) Th-230 (α)	Smear counting
Gamma Spectroscopy (Lab)	HPGe	Gamma	Mixed	Nuclide identification and quantification

The Ludlum Model 2350 Data Logger, along with a variety of detectors, was used for directly measuring total beta surface activity as well as measuring exposure rates. This data logger is a portable, micro-processor, computer-based counting instrument capable of operating with NaI(Tl) gamma scintillation, gas-flow proportional, GM, and ZnS scintillation detectors.

The detector selected depended on the survey to be performed, surface contours, and the size of the survey area. The 126-cm^2 (19.5 $in.^2$) gas-flow proportional detector was used for direct beta measurements, and a 2.5-cm × 2.5-cm (1-in. × 1-in.) NaI(Tl) gamma scintillation detector was used for exposure rate measurements.

A.3.1 Instrument Calibration

Ludlum Measurements, Inc., calibrates the data loggers and associated detectors semiannually using National Institute of Standards and Technology (NIST)-traceable sources and calibration equipment. Calibration of the data loggers includes:

- High voltage calibration
- Discriminator/threshold calibration
- Window calibration
- Alarm operation verification
- Scaler calibration verification.

Calibration of the detectors includes:

- Operating voltage determination
- Calibration constant determination
- Dead time correction determination.

Calibration labels with the instrument identification number, last calibration date, and next calibration due date were attached to all portable field instruments. The user checked the instrument calibration label before each use.

A.3.2 Sources

Radioactive sources used for calibration or for determining efficiencies were representative of an instrument's response to the identified nuclides and are traceable according to NIST. Radiation protection technicians controlled the radioactive sources used for instrument response checks and efficiency determinations. The sources were stored securely and were signed out by survey technicians when needed in the field. A sign-out log was used to track the location of all sources when they were removed from the field office.

A.3.3 Minimum Detectable Activity

The minimum detectable activity (MDA) is defined as the smallest amount or concentration of radioactive material in a sample that will yield a net positive count with a 5 percent probability of falsely interpreting background responses as true activity. The MDA is dependent upon the counting time, geometry, sample size, detector efficiency, and background count rate. As a data quality objective, the MDAs for the 1998 confirmatory survey were set to be approximately equal to or less than 50 percent of the site-specific guideline values developed in accordance with NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, December 1997. The equation used for calculating the MDA for field instrumentation is

$$MDA = \frac{\frac{2.71}{t_s} + 3.29\sqrt{\frac{R_b}{t_s} + \frac{R_b}{t_b}}}{E\left(\frac{A}{100}\right)} \qquad \text{(A-1)}$$

where

MDA = minimum detectable activity (dpm/100 cm^2)
t_s = sample count time (minutes)
R_b = background count rate (cpm)
t_b = background count time (minutes)
E = detector efficiency (counts/disinegrations)
A = detector area (cm^2).

A priori MDAs were established for the gamma spectroscopy equipment by counting an empty cave or a field blank. Count times were established to detect 0.1 pCi/g Cs-137 and Co-60 for shielded laboratory instruments on an *a priori* basis. Table A-2 provides a list of the typical MDAs for the primary radionuclides in the gamma spectroscopy analytical library for low activity samples. Samples with higher activity have higher MDAs. The other gamma-emitting radionuclides in the analytical library that are not shown in Table A-2 have MDAs similar to those in Table A-2 (i.e., less than 1 pCi/g).

Table A-2. Typical Analytical Minimum Detectable Activities*

Radionuclide	Minimum Detectable Activity (pCi/g)
Cs-137	0.1
Co-57	0.1
Co-60	0.1
Am-241	0.1
Eu-154/155	0.2
Mn-54	0.2
Zn-65	0.4
Sb-125	0.3

* Typical for low activity samples.

A.4 Survey Data Summary

Where possible, the sampling techniques and locations used for the 1998 survey were duplicated from the 1985 survey to ensure consistency. However, because of the lack of benchmark and reference locations, exact locations could not be duplicated. Therefore, the results were compared primarily to identify any significant differences. As long as the results were the same order of magnitude and agreed well, it was assumed that the 1985 analytical results were valid. A summary of the 1998 survey results, as well as comparisons with the 1985 results, where applicable, is provided in Sections A.4.1 through A.4.15.

It should be noted that K-40, a naturally occurring radionuclide, was reported in the 1998 survey results to provide a data quality check; K-40 levels should not change over time and should be generally between 10 and 20 pCi/g. Potassium-40 concentrations are presented in the analytical results for each area and can indicate the validity of the data.

A.4.1 Emergency Retention Basin

Samples were taken at five locations at the Emergency Retention Basin at the same depth intervals used during the 1985 survey (i.e., 0 to 5 cm, 5 to 15 cm, and 15 to 30 cm [0 to 2 in., 2 to 6 in., and 6 to 12 in.]). Three samples (ERB01, ERB01, and ERB03) were collected from the basin itself, while the other two samples (ERB04 and ERB05) were collected from the berm surrounding the basin. The five sample locations are shown in Figure A-2.

The highest contamination levels were identified at ERB03 within the southern half of the basin. Contamination was also contained within the basin and on the downslope along the inside of the berms. Contamination levels within the basin ranged from 10 to 200 pCi/g Cs-137 and 1 to 30 pCi/g Co-60, with some detectable levels of Co-57. Table A-3 provides the 1998 analytical results from samples ERB01, ERB02, and ERB03. Samples ERB04 and ERB05 taken along the berm of the basin showed only detectable cesium activity in the range of 0 to 1 pCi/g. The 1985 decay-corrected results for the 0 to 5-cm (0 to 2-in.) depth interval are provided for comparison. The contamination levels identified at the Emergency Retention Basin during the 1998 survey are consistent with the 1985 analytical results.

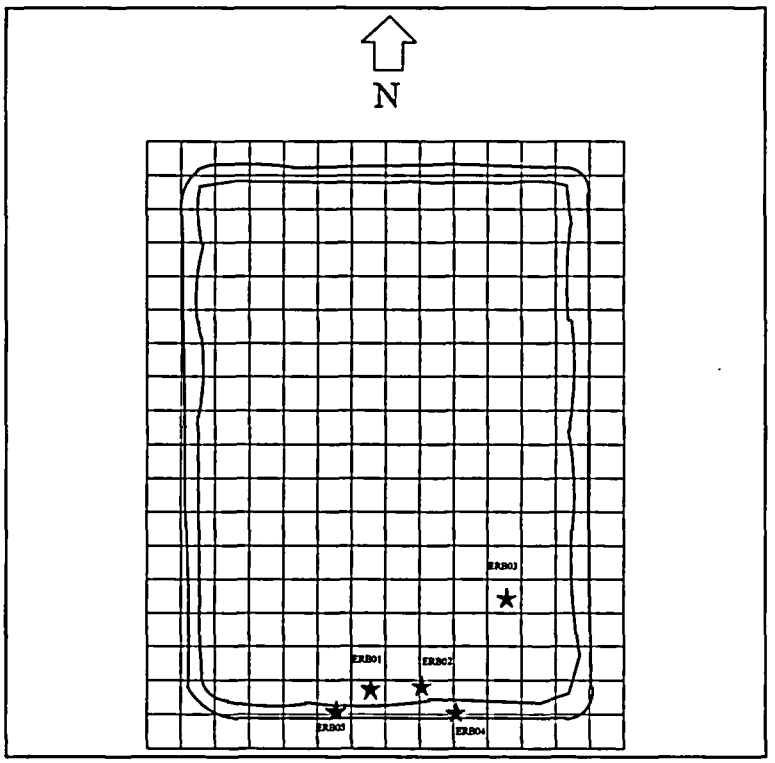

Figure A-2. Emergency Retention Basin Sampling Locations

Table A-3. Emergency Retention Basin Isotopic Results

Depth Sampled	Range of Activity (pCi/g), September 1998						
	Cs-137	Co-60	Co-57	Eu-154	Eu-155	Am-241	K-40
0–5 cm (0–2 in.) 1985 Results	35–200 20–90	2–30 2–22	0–1 ND*	< 1 ND	< 1 ND	< 0.5 ND	15–20 13–18
5–15 cm (2–6 in.)	75–120	2–30	0–1	< 1	< 1	< 0.5	15–20
15–30 cm (6–12 in.)	3–11	1–3	< 1	< 0.5	< 1	< 0.2	15–20

* ND = not detected or less than the lower limit of detection.

The 1998 results in Table A-3 show that contamination was identified up to 30 cm (12 in.) below the surface of the Emergency Retention Basin (3 to 11 pCi/g Cs-137 and 1 to 3 pCi/g Co-60). While the 1985 results indicated that contamination was within the top 5 cm (2 in.) with some contamination in selected areas up to 15 cm (6 in.) deep, the 1998 results indicate a greater depth of contamination. This could be a result of several factors, including the use of more sensitive counting equipment and, more importantly, a result of contamination migration because of weathering over the last 10 to 15 years.

A.4.2 Water Effluent Monitoring Station

Direct and removable beta surveys were performed at the Water Effluent Monitoring Station, and three sediment samples were collected (refer to Figure A-3 for sample locations). (The point where effluent discharges to the Pentolite Ditch, PD1, was sampled as part of the Pentolite Ditch verification survey.)

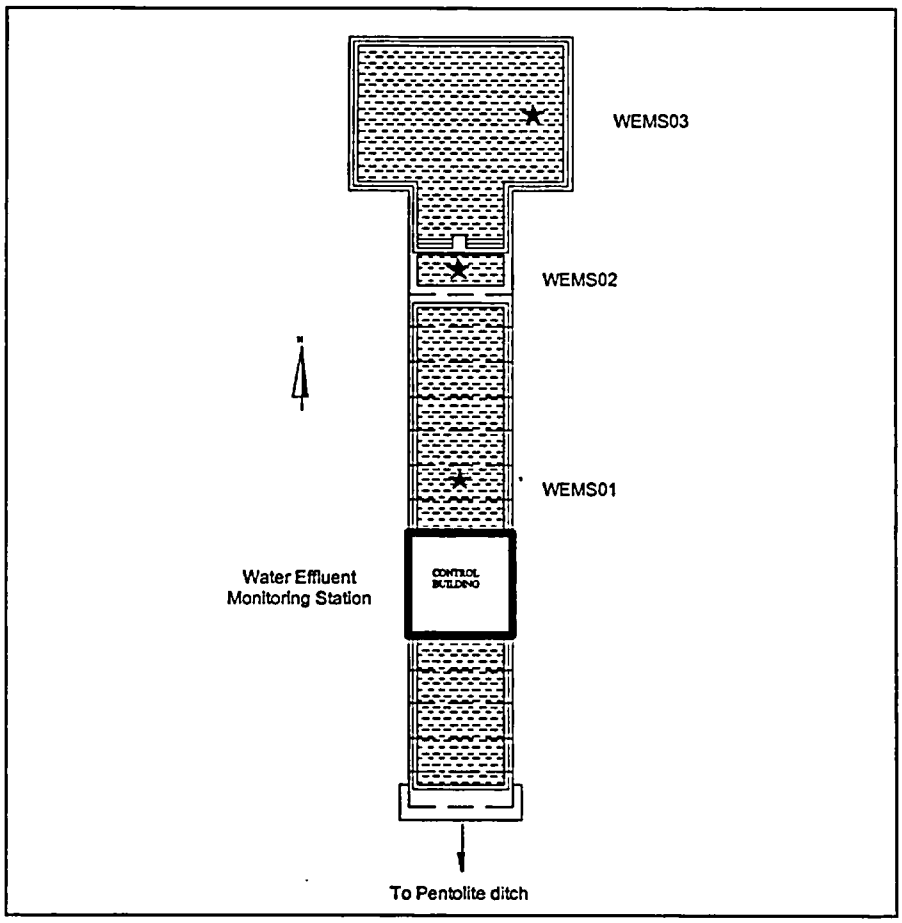

Figure A-3. Sample Locations at the Water Effluent Monitoring Station

Low levels of detectable beta contamination (1000 to 4000 dpm/100 cm^2) were identified on the concrete surfaces. All sediment samples had activities ranging from 4 to 11 pCi/g Cs-137 and 1 to 4 pCi/g Co-60. The analytical results are summarized in Table A-4. No equivalent data were reported in the 1985 characterization study with which to compare. However, the report indicated that there were low levels of detectable activity within the concrete trench, and the 1998 survey verifies this finding.

Table A-4. Water Effluent Monitoring Station Isotopic Results

Area Sampled	Range of Activity (pCi/g), September 1998						
	Cs-137	Co-60	Co-57	Eu-154	Eu-155	Am-241	K-40
Sediment	4–11	1–4	< 0.1	< 0.5	< 0.4	< 0.2	10–13

A.4.3 Pentolite Ditch

Verification sampling of the sediment in the Pentolite Ditch was performed along the entire length of the Pentolite Ditch as shown in Figure A-4 (PD-1 through PD-8). The 1985 sediment samples were taken from the bottom of the ditch, so the 1998 samples were also collected on the bottom of the ditch using long-handled tools.

The verification survey identified low levels of contamination in sediment (2 to 15 pCi/g Cs-137 and 0 to 1 pCi/g Co-60) along the first 305 m (1000 ft) of the ditch, with the highest levels being at the head of the ditch where the plant effluent was discharged. Table A-5 presents the 1998 analytical results from the eight 1998 Pentolite Ditch sediment samples. The 1985 decay-corrected isotopic results for two sediment samples along the first 305 m (1000 ft) of the ditch are provided for comparison. Review of the 1985 sediment and soil samples showed that of 97 samples analyzed in the laboratory, all but one sample had low levels of gross beta activity. The sample that had a high gross beta value was analyzed for isotopic composition and results showed about 71 pCi/g of Cs-137. The multiple 1985 samples that had low levels of gross beta are considered to be representative of contamination at the Pentolite Ditch. The 1998 survey confirmed that the Pentolite Ditch has levels of contamination slightly above background.

A.4.4 PBRF Grounds

Gamma scans of outdoor areas showed exposure rates of 5 to 10 micro-R/hr, which are typical for background levels. Ten surface and subsurface samples were taken across the outdoor areas (refer to Figure A-1 for sample locations FG01 through FG10). These locations correspond to the previous grid coordinate system locations H/I-9, G-10, J-10, I-8, I-19, D-18, H-12, H-13, I-15, and F-21. The 1985 characterization survey identified three contaminated areas located at grids H-18 (near the Primary Pump House resin pits), I-9/10 (known spill area near the Waste Handling Building concrete pad), and U-3 (near the Water Effluent Monitoring Station outlet). The 1998 survey confirmed the presence of contamination at grid H/I-9/10 (sample location FG01); however, the other two areas of elevated activity were not located. These other two areas may have been small, which would explain why they could not be located during the confirmatory survey. Soil was sampled at two locations (FG07 and FG08) inside the Hot Retention Area fence, but no contamination was found.

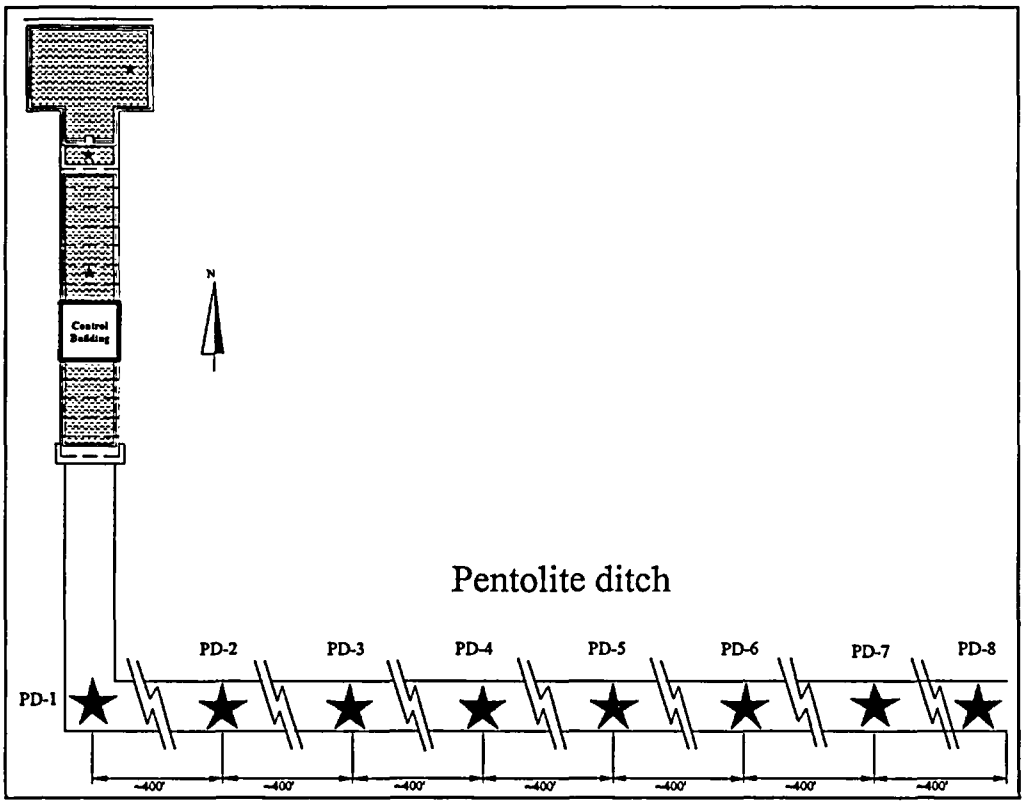

Figure A-4. Sample Locations at the Pentolite Ditch

Table A-5. Pentolite Ditch Sediment Isotopic Results

Area Sampled	Range of Activity (pCi/g), September 1998						
	Cs-137	Co-60	Co-57	Eu-154	Eu-155	Am-241	K-40
0 to 305 m (0–1000 ft) 1985 Results	2–15 3–71	0–1 0–0.1	< 0.1 ND*	< 0.5 ND	< 0.5 ND	< 0.2 ND	12–17 17–28
305 m (1000 ft) to End	0–2	< 0.2	< 0.1	< 0.5	< 0.5	< 0.2	14–22

* ND = Not detected or less than the lower limit of detection (LLD).

A summary of the 1998 results for the facility grounds is provided in Table A-6. In the area of the Waste Handling Building concrete pad (FG01, which corresponds to grids H/I-9), the Cs-137 activity level was up to 200 pCi/g. There were no 1985 isotopic results with which to compare levels of contamination. Also, the depth of contamination could not be verified because asbestos was encountered. In general, the results of the 1998 survey showed very little activity on the facility grounds, which confirmed the 1985 characterization results.

Table A-6. Facility Grounds Isotopic Results

Area Sampled	Range of Activity (pCi/g), September 1998						
	Cs-137	Co-60	Co-57	Eu-154	Eu-155	Am-241	K-40
Known spill area near Waste Handling Building concrete pad (FG01)	201	0.3	< 0.2	< 0.3	< 0.2	< 0.6	16.5
Other areas (FG02 through FG10)	0-2	< 0.2	< 0.1	< 0.3	< 0.2	< 0.2	6-18

A.4.5Facility Pavement

The asphalt and concrete pavement was sampled at two locations during the 1998 verification survey: (1) near the main entrance to the Reactor Building (1111), near the Reactor Gas Services Building (1135) and (2) near the Waste Handling Building concrete pad (WHBPAD) (refer to Figure A-1 for sample locations). Also, direct surveys were performed on paved and concrete areas throughout the site. Spotty contamination was identified over the asphalt and pavement in the area between the Primary Pump House (1134), Hot Laboratory (1112), and the Reactor Building (1111), as well as around the Waste Handling Building (1133) and Fan House (1132). This is consistent with the historical spill areas (grids H/I-9/10 and H-18) and the areas of contaminated soil identified during the 1985 characterization survey. Direct beta contamination levels ranged from 10,000 to 13,000,000 dpm/100 cm^2, with the highest levels on the pad to the south of the Waste Handling Building (1133). No data from the 1985 characterization survey were found for the asphalt and concrete pavement at the PBRF.

In addition to these areas that were previously suspected of being contaminated, a spot of contamination was also identified near the main entrance to the Reactor Building (1111) near the access gate above the service tunnel. The surface activity at this spot measured 42,000 dpm/100 cm^2.

A pavement sweeping (debris) sample was taken from the area with the highest direct surface activity near the Waste Handling Building (1133) (location on WHBPAD Figure A-1). This sample had 1300 pCi/g of Cs-137. Another sweeping sample was taken from the paved area near the Reactor Gas Services Building (1135) (see Figure A-1) where contamination was not detected using direct survey techniques. The results of these two samples are presented in Table A-7.

Table A-7. Facility Pavement Isotopic Results

	Range of Activity (pCi/g), September 1998						
Area Sampled	Cs-137	Co-60	Co-57	Eu-154	Eu-155	Am-241	K-40
Waste Handling Building (1133) concrete pad	1300	< 0.6	< 0.7	< 0.5	< 2.1	< 0.8	11.1
Reactor Building (1111) and Reactor Gas Services Building (1135)	0.2	< 0.1	< 0.04	< 0.3	< 0.2	< 0.1	15.4

A.4.6 Catch Basins

Ten catch basins at the PBRF were surveyed (CB3, CB4, CB7, CB8, CB9, CB10, CB13, CB15, CB18, and WHBCB), which included direct and removable activity surveys and collecting sediment samples (Figure A-1). The basins surveyed were in the areas where pavement contamination was identified, which included areas between the Primary Pump House (1134) and Hot Laboratory (1112), as well as around the Waste Handling Building (1133). Stormwater generally flows from north to south, and water from catch basins and drainage ditches (indicated by dotted lines on Figure A-1) is collected at the Water Effluent Monitoring Station and then discharged to the Pentolite Ditch.

Low levels of contamination ranging from 1 to 11 pCi/g Cs-137 and 1 to 5 pCi/g Co-60 were identified in the catch basin sediments. Direct activity surveys were performed, but very little direct activity was identified. The highest levels (5000 dpm/100 cm^2) were identified in catch basin 13A, located immediately south of the Waste Handling Building (1133). This finding is consistent with the 1985 survey results. A summary of the sediment sample results is provided in Table A-8.

Table A-8. Catch Basin Sediment Isotopic Results

Area Sampled	Range of Activity (pCi/g), September 1998						
	Cs-137	Co-60	Co-57	Eu-154	Eu-155	Am-241	K-40
Sediment	1–11	1–5	< 0.05	< 0.4	< 0.3	< 0.2	7–14

A.4.7 Cold Retention Basins

Direct and removable activity surveys were taken and sediment samples were collected from the two Cold Retention Basins (1154) (sample locations CRA-1 and CRA-2 are shown on Figure A-1). Low levels of uniform direct beta contamination were identified within basin 2 (CRA-2), ranging from 1000 to 5000 dpm/100 cm^2. These contamination levels are consistent with the contamination levels identified in 1985. (The bottom of the Cold Retention Basins could not be directly surveyed because of standing water.)

In addition to the direct surveys, sediment samples were collected from the bottom of both basins using long-handled tools. CRA-1 from basin 1 had activity levels near 20 pCi/g Cs-137, 80 pCi/g Co-60, and 6 pCi/g Eu-154, while CRA-2 from basin 2 had 5 pCi/g Cs-137 and 6 pCi/g Co-60.

These levels are much less than the levels identified during the 1985 characterization survey. The 1998 confirmatory survey confirms that the two basins are contaminated.

A.4.8 Reactor Building

Verification surveys were performed within the Reactor Building (1111) but outside the reactor containment vessel. These surveys consisted of direct beta and removable alpha and beta measurements at areas on floors and walls that were previously surveyed in 1985 as indicated by existing markers. Measurements were taken at all elevations (i.e., the -7.6 m [-25 ft], -4.7-m [-15-ft], main floor, and 3.7-m [+12-ft] elevations).

Activity was identified on the floors of both the -7.6-m (-25-ft) and -4.7-m (-15-ft) elevations at locations RB013 and RB056, respectively. The level of beta activity at RB013 was about 2000 to 10,000 dpm/100 cm^2, while the level at RB056 was 45,000 dpm/100 cm^2. No removable activity was identified. A concrete core was taken from the floor at the RB056 location, and a summary of the analytical results are presented in Table A-9.

Table A-9. Reactor Building Floor Core Isotopic Results

Location Sampled	Activity (pCi/g), September 1998						
	Cs-137	Co-60	Co-57	Eu-154	Eu-155	Am-241	K-40
RB056, -7.6-m (-25-ft) Level	0.2	0.1	< MDA*	< MDA	< MDA	< MDA	5.0

* MDA = minimum detectable activity.

The 1985 characterization only identified low levels of activity on the -7.6-m (-25-ft) elevation. The levels of direct beta contamination in 1998 were higher than the levels measured in 1985. This may be a result of using more sensitive equipment during the 1998 survey. No activity was identified on the other elevations.

A.4.9 Reactor Office and Laboratory Building

The surveys performed within the Reactor Office and Laboratory Building (1141) consisted of direct beta and removable alpha and beta measurements on the floors at existing survey markers, where applicable. The 1985 characterization identified activity only on the second floor in labs 212 and 214. The 1985 confirmatory survey also identified activity only on the second floor; however, activity was identified in labs 207, 209, 210, 213A, and 214/215. Activity levels ranged from 5000 to 70,000 dpm/100 cm^2. In addition, removable alpha activity was identified in labs 207, 213A, and 214/215 up to 160 dpm/100 cm^2, and removable beta activity was identified in lab 210 up to 150 dpm/100 cm^2.

A.4.10 Reactor Service Equipment Building

The Reactor Service Equipment Building (1131) was not surveyed in 1985 because the building was determined to be clean based on previous measurements. As part of the 1998 confirmatory

survey, this building was surveyed for direct and removable activity in the basement, first floor, and mezzanine. No activity was identified in this building.

A.4.11 Fan House

The confirmatory survey of the Fan House (1132) included performing surveys on the basement floor and on the first floor. The 1985 characterization survey identified that the activity in the Fan House was confined to the basement. The 1998 confirmatory survey confirmed this. Low levels of direct beta activity were identified in the basement ranging from 1000 to 10,000 dpm/100 cm^2. In addition, removable beta activity was found throughout the basement floor ranging from 20 to 150 dpm/100 cm^2. No direct activity was identified on the first floor.

A.4.12 Waste Handling Building

The confirmatory survey of the Waste Handling Building (1133) was performed on the basement floors and on the first floor. Low levels of detectable activity were identified throughout the building ranging between 1000 and 10,000 dpm/100 cm^2. Although these levels are slightly higher than those identified in the 1985 characterization report and the activity was distributed across a larger area, the 1998 results generally confirm the 1985 characterization results.

In addition to the direct activity, removable activity was identified throughout the building both in the basement and the first floor. Removable activity levels ranged from 50 to 600 dpm/100 cm^2.

A.4.13 Service Tunnels

Surveys were performed throughout the north and east service tunnels. In general, no activity was identified in these two tunnels. However, there was an elevated measurement of 2000 dpm/100 cm^2 at the entrance to the east tunnel from the Reactor Building.

A.4.14 Canal F

A concrete core was taken from Canal F for isotopic analysis. A summary of the analytical results are presented in Table A-10.

Table A-10. Canal F Concrete Core Isotopic Results

Area Sampled	Activity (pCi/g), September 1998						
	Cs-137	Co-60	Co-57	Eu-154	Eu-155	Am-241	K-40
Canal F core	2.7	156	<MDA*	<MDA	<MDA	<MDA	5.3

* MDA = minimum detectable activity.

A.4.15 1985 Petri Dish Samples

In addition to the areas surveyed as summarized in Sections A.4.1 through A.4.14, NASA provided several petri dish samples that were collected from sumps during the 1985 characterization effort.

A summary of the gamma isotopic analyses is provided in Table A-11. Isotopic analyses from the 1985 characterization were not available for comparison.

Table A-11. 1985 Petri Dish Samples Isotopic Results

Area Sampled	Activity (pCi/g), September 1998						
	Cs-137	Co-60	Co-57	Eu-154	Eu-155	Am-241	K-40
Fan House sump	13,923	7,707	< 8.36	< 60.6	< 27.9	< 16.9	< 202
Reactor decontamination sump	35.3	9.7	< 0.38	< 3.43	< 2.11	< 1.60	< 18.6
Reactor hot sump	379	183	< 1.35	< 5.99	< 5.05	< 4.52	< 34.8
Waste Handling Building Laundry Sump	348,340	35,732	< 40.9	< 3870	< 1600	< 1200	< 7600
Containment vessel sump	119	240	< 3.16	< 19.8	< 3.85	< 2.97	< 25.3

A.5 Conclusions

In general, the results of the 1998 confirmatory survey confirmed the contaminated and uncontaminated areas identified during the 1985 characterization survey. The confirmatory survey did identify a couple of additional contaminated areas: labs 207, 209, 210, and 213A in the Reactor Office and Laboratory Building (1141); an area of contamination on the -4.7-m (-15-ft) elevation in the Reactor Building (1111); and the PBRF pavement near the entrance to the Reactor Building. A greater depth of contamination was identified within the Emergency Retention Basin during the 1998 confirmatory survey than in 1985, but could be a result of contamination migration over the last 10 to 15 years and using more sensitive counting equipment.

The results from the 1985 and 1998 characterization surveys were analyzed to determine the distribution of radionuclides. The gamma characterization information from the 1998 survey shows that the dominant gamma sources are Cs-137 and Co-60. Other gamma-emitting nuclides are only small contributors (less than 1 percent). At all PBRF areas (e.g., environmental contamination, sumps, floor in the Reactor Building) except the single sample taken from Canal F, the gamma activity is dominated by Cs-137. In the 15 samples taken throughout the PBRF (not including Canal F), the percentage of gamma activity because of Cs-137 ranged from 64 to 100 percent ,with an average of 79 percent. (In Canal F, the activity is dominated by Co-60 rather than Cs-137)

APPENDIX B

DERIVATION OF χ/Q

APPENDIX B

DERIVATION OF χ/Q

In the accident analysis presented in Section 3.3, the quantity χ/Q is used to express the dilution of the released effluent as it travels 0.8 km (0.5 mi) to the site boundary. χ/Q is calculated using the well-established formula for Gaussian Dispersion, which is applicable when the effluent is released at such a rate that it does not perturb the existing pattern of turbulent eddies in the atmosphere. This is the expected case for small releases such as are evaluated in Section 3.3 of the decommissioning plan. χ/Q was calculated using the formula:

$$\frac{\chi}{Q} = \frac{1}{\pi \sigma_y \sigma_z u} \qquad \text{(B-1)}$$

where:

σ_y = the crosswind standard deviation (meters)
σ_z = the vertical standard deviation (meters)
u = the windspeed (meters/second) measured at a height of 10 meters.

Equation B-1 is valid for a ground-level release, which is the most conservative case when dry deposition is neglected, as was done for the accident analysis in Section 3.3.

In NRC's Accident Analysis Handbook (SAIC 1998), the NRC has published the following standard deviations as a function of distance, d (in meters) downwind in severe, category F meteorological conditions:

$$\sigma_z = \frac{0.016\text{d}}{(1 + 0.0003\text{d})} \qquad \text{(B-2)}$$

$$\sigma_y = \frac{0.04\text{d}}{\sqrt{1 + 0.0001\text{d}}} \qquad \text{(B-3)}$$

These parameters are generally regarded as being conservative.

Using d = 800 meters in Equations B-2 and B-3 gives σ_z = 10. 32 meters and σ_y = 30.79 meters. Using these values of σ_z and σ_y and a windspeed, u, of 2 m/s, Equation B-1 yields

$$\frac{\chi}{Q} = \frac{1}{\pi(30.79\,\text{m})(10.32\,\text{m})(2)} = 5 \times 10^{-4}\,\text{sec/m}^3$$

APPENDIX C

SAMPLE ALARA CALCULATION FOR THE EMERGENCY RETENTION BASIN

APPENDIX C

SAMPLE ALARA CALCULATION FOR THE EMERGENCY RETENTION BASIN

This appendix gives an example of how an as low as reasonably achievable (ALARA) analysis was conducted for the Emergency Retention Basin at the Plum Brook Reactor Facility (PBRF). This analysis uses the three steps of the ALARA methodology described in Section 2.2.3.2 of the decommissioning plan. As noted in Section 2.2.3.1 of the plan, a resident farmer scenario was postulated for the Emergency Retention Basin because the primary exposure pathway is contaminated soil. The dose to an individual resident farmer was calculated using the dose assessment methods described in Section 2.2.3.1 of the plan. The postulated actions at the Emergency Retention Basin include removal of contaminated surface soil from 0 to 10 cm (0 to 4 in.) below the surface, followed by selected removal of contaminated surface soil from a depth of 10 to 15 cm (4 to 6 in.) below the surface.

The spreadsheet in Table C-1 summarizes the ALARA analysis calculations for the Emergency Retention Basin. Column A gives the year after the contaminated soil has been removed. The benefit of the averted population dose is calculated in columns B through K. Columns B and C give the discount rate and discount factor, respectively, used in the calculation.

The annual individual dose resulting from the existing condition (i.e., not removing contaminated soil in the Emergency Retention Basin) is given in column D. Column E gives the estimated annual individual dose after the contaminated soil has been removed. Using the RESRAD code (Yu et al. 1993), a set of points (year, individual dose) were calculated assuming the resident farmer scenario for both the leave-as-is and after soil removal conditions. Each set of points was fitted with an exponential curve of the form $y = a \times e^{bt}$, where y is the dose in millirem per year, *a* and *b* are constants, and *t* is the specific year after the action would be completed (refer to Figures C-1 and C-2). The values for *a* and *b* for each equation are shown in rows 9 and 10 of Table C-1. The equation for the exponential curve was used to calculate the individual doses by year in columns D and E.

The contaminated area (column G, row 10) multiplied by the population density from Draft Regulatory Guide DG-4006 (NRC 1998) for land (column G, row 11) gives the number of people that would be exposed to residual contamination (column G, row 13). The annual individual doses in columns D and E were then multiplied by the population density to convert the individual dose to an annual population dose in columns F and G. The annual population dose resulting from existing conditions is given in column F; the annual population dose after the action has been implemented is given in column G.

The benefit of averted population dose is calculated in column H (value in column F – value in column G). Following the methodology described in Section 2.2.3.2, this benefit was multiplied by $2000/person-rem to convert it to a monetary equivalent (shown in column I).

Table C-1. Refined ALARA Analysis: Removal of Contaminated Soils from the Emergency Retention Basin

	A	B	C	D	E	F	G	H	I	J	K
1	ALARA Analysis for: Removal of contaminated soil from the Emergency Retention Basin										
2	Individual scenario: Resident farmer										
3											
4	Dose (person-rem) Cost Equivalent ($)			$2,000							
5											
6											
7			a=154		1098		CALCULATION OF BENEFITS				
8			b=-0.0896		-0.04	area (sqm)=	8129				Total Benefit
9			y=a*exp(b*t)			pop density/sqm=	0.0004	(land)			$8,924
10		first 100 yrs	7%								
11		after 100 yrs	3%			#people=	3256				
12	Year	Discount rate (%)	Discount factor	Annual individual dose for existing conditions (mrem/yr)	Annual individual dose from final conditions after alternative is implemented (mrem/yr)	Annual population dose for existing conditions (person-rem)	Annual population dose from final conditions after alternative is implemented (person-rem)	Annual benefit of avoided population dose (person-rem)	Annual monetary equivalent of the benefit of avoided population dose	Annual present worth of benefit of avoided population dose	Cumulative present worth of benefit of avoided population dose
13	0	7%	1.00E+00	1.68E+02	1.18E+01	5.47E+01	3.84E+02	5.08E+01	$1,016	$1,016	$1,016
14	1	7%	9.33E+01	1.57E+02	1.10E+01	5.11E+01	3.58E+02	4.75E+01	$949	$883	$1,899
15	2	7%	8.66E+01	1.46E+02	1.01E+01	4.76E+01	3.28E+02	4.30E+01	$860	$744	$2,643
16	3	7%	8.04E+01	1.39E+02	9.83E+00	4.52E+01	3.19E+02	4.21E+01	$840	$676	$3,319
17	4	7%	7.48E+01	1.31E+02	9.28E+00	4.27E+01	3.02E+02	3.97E+01	$794	$594	$3,913
18	5	7%	6.96E+01	1.26E+02	8.91E+00	4.11E+01	2.91E+02	3.82E+01	$764	$531	$4,444
19	6	7%	6.47E+01	1.21E+02	8.55E+00	3.95E+01	2.78E+02	3.67E+01	$734	$475	$4,919
20	7	7%	6.02E+01	1.17E+02	8.21E+00	3.80E+01	2.67E+02	3.53E+01	$706	$425	$5,344
21	8	7%	5.60E+01	1.12E+02	7.88E+00	3.65E+01	2.56E+02	3.39E+01	$678	$380	$5,723
22	9	7%	5.21E+01	1.08E+02	7.56E+00	3.51E+01	2.46E+02	3.26E+01	$652	$339	$6,063
23	10	7%	4.84E+01	1.04E+02	7.26E+00	3.37E+01	2.36E+02	3.13E+01	$627	$303	$6,366
24	11	7%	4.50E+01	9.98E+01	6.97E+00	3.24E+01	2.27E+02	3.01E+01	$603	$271	$6,637
25	12	7%	4.19E+01	9.58E+01	6.69E+00	3.11E+01	2.17E+02	2.90E+01	$579	$242	$6,880
26	13	7%	3.89E+01	9.20E+01	6.42E+00	2.99E+01	2.09E+02	2.78E+01	$557	$217	$7,097
27	14	7%	3.62E+01	8.83E+01	6.16E+00	2.88E+01	2.00E+02	2.68E+01	$535	$194	$7,290
28	15	7%	3.37E+01	8.50E+01	5.91E+00	2.76E+01	1.92E+02	2.57E+01	$514	$173	$7,464

Table C-1. Refined ALARA Analysis: Removal of Contaminated Soils from the Emergency Retention Basin (Continued)

	L	M	N	O	P
1					
2					
3					
4					
5					
6					
7					
8					
9	*CALCULATION OF COSTS*			*COMPARISON*	
10			Total Cost		Benefit - Costs
11			$1,859,000		($1,850,076)
12					
13					
14	Implement. Cost	Annual present worth of cost of impl. the action ($)	Cumulative present worth of cost of impl. the action ($)	Annual present worth of benefits - costs	Cumulative present worth of benefits - costs
15	$1,859,000	$1,859,000	$1,859,000	($1,857,984)	($1,857,984)
16	$0	$0	$1,859,000	$883	($1,857,101)
17	$0	$0	$1,859,000	$744	($1,856,357)
18	$0	$0	$1,859,000	$676	($1,855,681)
19	$0	$0	$1,859,000	$594	($1,855,087)
20	$0	$0	$1,859,000	$531	($1,854,556)
21	$0	$0	$1,859,000	$475	($1,854,081)
22	$0	$0	$1,859,000	$425	($1,853,656)
23	$0	$0	$1,859,000	$380	($1,853,277)
24	$0	$0	$1,859,000	$339	($1,852,937)
25	$0	$0	$1,859,000	$303	($1,852,634)
26	$0	$0	$1,859,000	$271	($1,852,363)
27	$0	$0	$1,859,000	$242	($1,852,120)
28	$0	$0	$1,859,000	$217	($1,851,903)

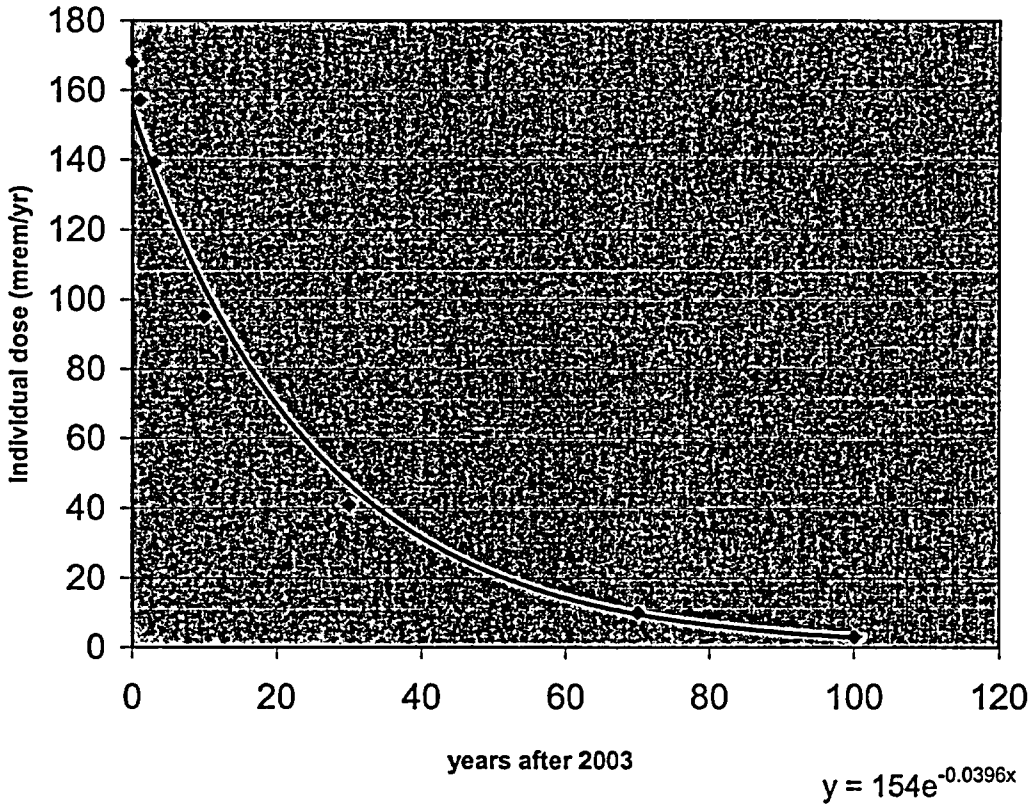

Figure C-1. Individual Dose from Emergency Retention Basin If Left As Is

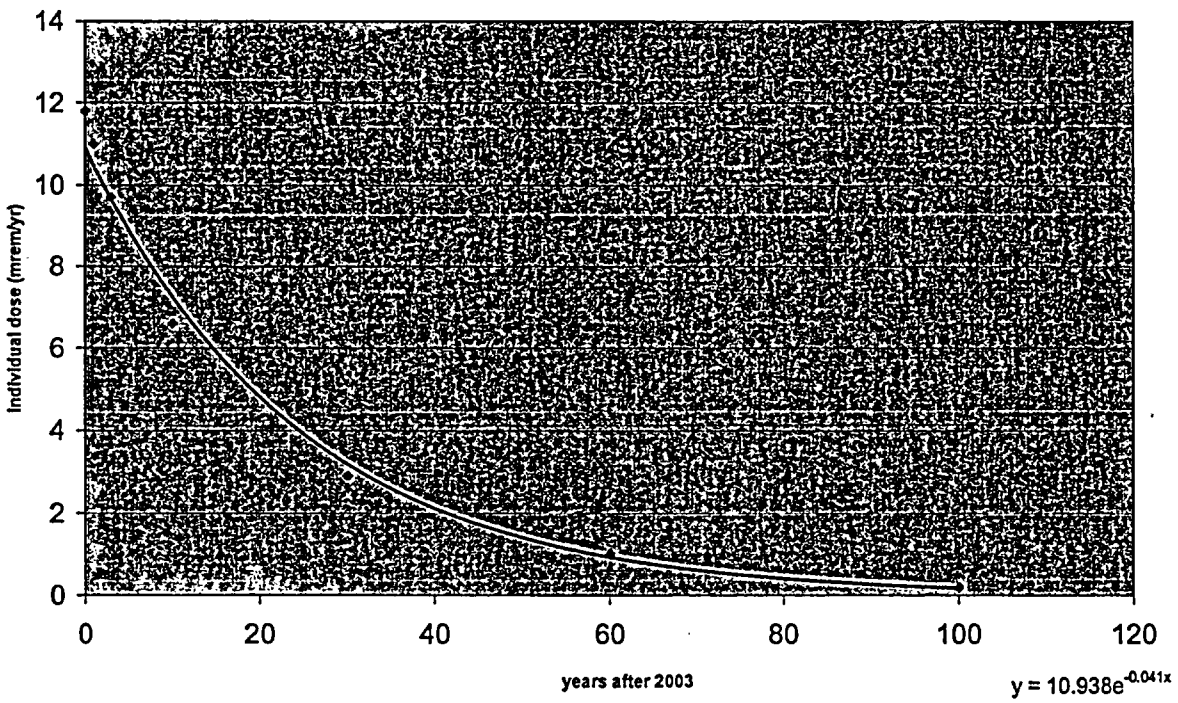

Figure C-2. Individual Dose from Emergency Retention Basin After Remediation

The present worth value of the benefit was then calculated by multiplying the monetary equivalent (column I) by a discount rate. Discount rates of 7% were applied for the first 100 years and a discount rate of 3% applied thereafter, consistent with the guidance in Draft Regulatory Guide DG-4006 (NRC 1998) (column B). The discount factor was calculated by year (column C) using Equation C-1:

$$\text{Discount factor}_{\text{year } n+1} = \text{Discount factor}_{\text{year } n} \times (1 - \text{Discount rate}_{\text{year } n+1}). \qquad \text{(C-1)}$$

The annual present worth of the benefit (column J) was calculated by multiplying the annual monetary equivalent (column I) by the annual discount factor (column C). The cumulative present worth (column K) is the sum of the annual present worth (column J). Column K, row 11 shows that the present worth of the cumulative benefit over 1000 years is $8,924.

The cost of implementing the action was then evaluated. Scoping calculations indicated that the cost equivalent of occupational dose and fatalities and population dose are very small contributors (much less than 1 percent) to the overall cost. Therefore, these costs were ignored for this calculation. The implementation cost for removing contaminated soil at the Emergency Retention Basin is shown in column L, row 15. The implementation cost (column L, row 15) was multiplied by the annual discount factors (column C) to calculate an annual present worth (column M). The annual present worth was then summed to calculate a cumulative present worth in dollars (column N). Because the action would occur in 1 year, there are no additional costs after the first year.

The total benefit and total cost were then compared (columns K and N, respectively). The total benefit of $8,924 is 3 orders of magnitude less than the cost of $1,859,000. The annual present worth of benefits minus the cost (column O) is calculated by subtracting column N from column K. The cumulative value of the benefit minus the cost ($1,850,076) in column P indicates the costs greatly exceed the benefit. Therefore, the action of removing the contaminated soil from the Emergency Retention Basin to meet the residual dose criteria would, in the process, comply with the ALARA requirement.

References:

NRC (U.S. Nuclear Regulatory Commission), 1998. Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," August.

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APPENDIX D

ESTIMATED COST FOR DECOMMISSIONING THE PBRF

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APPENDIX D

ESTIMATED COST FOR DECOMMISSIONING THE PBRF

This appendix presents the cost estimate by principal task for decommissioning the PBRF. The cost estimate was based on the current radiological status discussed in Section 2.2.2, the proposed criteria for unrestricted release discussed in Section 2.2.3, and the planned decommissioning tasks identified in Section 2.3. During the actual decommissioning planning phase, a detailed engineering cost estimate will be prepared. The total project cost estimate for decommissioning PBRF is $[ ] million in current year dollars escalated to the mid-project.

The estimated cost for decontaminating and decommissioning the PBRF, as described in Section 2.3, is approximately $[ ] million in 1999 dollars. The cost estimate is summarized in Table D-1. The cost estimate assumes that contaminated buildings and structures will be decontaminated, contaminated material and soil will be removed, decontaminated buildings and structures will be demolished, and the remaining belowgrade portions of buildings will be backfilled. Also, it was assumed that all radioactive wastes generated during the removal and decontamination activities will be shipped to a licensed low-level radioactive waste disposal facility, and building demolition wastes that have met the criteria for unrestricted release will be disposed of offsite in an industrial landfill.

Starting with the cost estimate of $[ ]million in 1999 dollars a historical contingency factor was added to represent the historical data from initial cost estimate to actual project completion costs for reactor decommissioning projects to give a project cost estimate of $[ ] million in 1999 dollars. Finally, the project costs were escalated to current year dollars using a project start date in fiscal year (FY) 2001 and project duration of 5 years. The current funding profile is estimated at $[ ] million in FY 2001, $[ ] million in FY 2002, $[ ] million in FY 2003, $[ ] million in FY 2004, and $[ ] million in FY 2005 to give an estimated total project cost in current year dollars of $[ ] million summarized in Table D-2.

Table D-1. Estimated Cost of Decommissioning the NASA Plum Brook Reactor Facility

Work Phase	Work Description[a]	Cost ($)[b,c]
Planning Activities:	Decommissioning Planning	[ ]
	NASA Operations and Direct Support	[ ]
Decontamination and Dismantling Tasks:	Operation Management and Support#	[ ]
	Security#	[ ]
	Health Physics#	[ ]
	Systems Operation, Maintenance, and Deactivation#	[ ]
	Contaminated Soil Removal*#	[ ]
	Site Preparation*#	[ ]
	Asbestos Removal and Lead Paint Abatement[d] *#	[ ]
	Loose Equipment Removal*#	[ ]
	Removal of Activated Material in Hot Dry Storage Area*#	[ ]
	Decontamination*#	[ ]
	Reactor Internals and Tank Removal*#	[ ]
	Contaminated Piping and Equipment Removal*#	[ ]
	Contaminated Concrete and Embedded Pipe Removal*#	[ ]
	Final Status Survey#	[ ]
	Building Demolition#	[ ]
	Building Backfill#	[ ]
	Reactor Building Backfill#	[ ]
	Radioactive Waste Disposal	[ ]
	Industrial Waste Disposal	[ ]
TOTAL	Without contingencies	[ ]
TOTAL	With historical contingencies for decommissioning projects	[ ]

a. The estimated costs for items marked with * include size reduction, packaging, and transportation costs to a disposal facility.

b. In 1999 dollars.

c. The costs of items mark with # include a prorated portion of the indirect costs, for example, a fee of 20 percent, an Ohio franchise tax of 5 percent applied to the fee, and a performance bond of 1.15 percent.

d. Assumes that a single contractor will perform both asbestos and lead paint abatement.

NASA PRE-DECISIONAL INFORMATION
Note: Information on this page is considered pre-decisional (ref. 5 USC §552(b)(5) and 10 CFR §2.790(a)(4) and (a)(5)). This information is provided only for the purpose of the NRC's review and approval of the PBRF Decommissioning Plan. Distribution should be limited to official government purposes only. Release without prior written consent of NASA Glenn Research Center is strictly prohibited.

Table D-2. Funding Profile for NASA Plum Brook Reactor Facility

TASK	Fiscal Year	SM*
Pre-Decommissioning	2001	[ ]
Design & Planning	2002	[ ]
Decommissioning	2003	[ ]
Decommissioning	2004	[ ]
Final Survey & Demolition	2005	[ ]
License Termination	2006	
	Total	[ ]

* Costs in current year dollars escalated from the 1999 dollar cost estimate to mid-project

NASA PRE-DECISIONAL INFORMATION
Note: Information on this page is considered pre-decisional (ref. 5 USC §552(b)(5) and 10 CFR §2.790(a)(4) and (a)(5)). This information is provided only for the purpose of the NRC's review and approval of the PBRF Decommissioning Plan. Distribution should be limited to official government purposes only. Release without prior written consent of NASA Glenn Research Center is strictly prohibited.

ATTACHMENT 1

REGULATORY REVIEW PACKAGE
DECOMMISSIONING PLAN
REVISION 4

This revision to the Decommissioning Plan incorporates changes related to the requirements for NRC review of Final Status Surveys for areas that will be backfilled or rendered inaccessible. The changes are the result of a License Amendment Request dated January 14, 2005. The revision will not become effective until the NRC has formally issued the License Amendment requested in that submittal.

The Facility Licenses, TR-3 and R-93, Docket Numbers 050-00030 and 050-00185, allow changes to the Decommissioning Plan without prior approval by the USNRC as long as the changes meet certain criteria. First, the revision must meet the criteria for 'changes to the facility' made without NRC approval specified in 10CFR50.59. Second, condition 3.A.1 of the facility licenses stipulate that changes to the Decommissioning Plan may be made without prior NRC approval as long as eight conditions relating to Final Status Surveys and site release criteria are satisfied.

This document presents the evaluation of the changes against the criteria of 10CFR50.59 using the evaluation process specified in Procedure PSC-02, "Facility Modification Review and Approval", revision 0, effective September 5, 2002. This is followed by a written evaluation of the changes against the eight license criteria. The final section of this document is a detailed description of the changes that were made to the Decommissioning Plan.

This evaluation concludes that the changes incorporated in Revision 4 to the "Decommissioning Plan for the Plum Brook Reactor Facility" may be implemented without prior approval by the USNRC pursuant to 10CFR50.59. It also concludes that the changes incorporated in Revision 4 to the "Decommissioning Plan for the Plum Brook Reactor Facility" may be implemented without prior approval by the USNRC pursuant to condition 3.A.1 of License Numbers R-93 and TR-3.

REGULATORY REVIEW CHECKLIST

Document/Activity Title Decommissioning Plan for the PBRF
Document/Activity Number: Decommissioning Plan **Revision No.** 4
Type of Activity: Revision to the Decommissioning Plan

This Regulatory Review Package includes the following (check all that apply):

1.	Section I, Environmental Determination (questions 1 through 7)----------	☒Yes	☐No
2.	Written Environment Impact Assessment/Evaluation----------	☐Yes	☒No
3.	Section II, 10 CFR 50.82 Evaluation (questions 8 through 11)----------	☒Yes	☐No
4.	Section III, 10 CFR 50.59 Applicability Determination (questions 12 through 16)---	☒Yes	☐No
5.	Attached evaluations required by other regulations (question 15) ----------	☒Yes	☐No
6.	Section IV, 10 CFR 50.59 Screening (questions 17 through 21)----------	☒Yes	☐No
7.	Written justification for 'NO' answers to 10 CFR 50.59 Screening questions---------	☒Yes	☐No
8.	Section V, 10 CFR 50.59 Evaluation (questions 22 through 28)----------	☐Yes	☒No
9.	Written justification to 'NO' answers to 10 CFR 50.59 Evaluation----------	☐Yes	☒No
10.	Section VI, Independent Review Determination (questions 29 through 39)----------	☒Yes	☐No

Based upon the results of this regulatory review: (select one of the following)

This activity may be implemented without obtaining a license amendment; ☒ **(As Indicated)**

or

This activity requires a license amendment and may not be implemented without prior NRC approval. ☐ **(As Indicated)**

Based upon the results of this regulatory review:

This activity requires review by a Responsible Technical Reviewer.	☒Yes	☐No
This activity requires review by an Independent Safety Reviewer	☐Yes	☒No

Review Performed by:	John A. Thomas 11/25/05	(print name and sign/date)
Responsible Technical Review by:	William Stoner (RSO) 11/25/05	(print name and sign/date)
Independent Safety Review by PSC:	Keith Peecook 12/2/05	(print name and sign/date)

In addition to the requirements of Procedure PSC-02, "Facility Modification Review and Approval", revision 0, a 10CFR50.59 evaluation for a revision of the Decommission Plan issued without prior approval of the USNRC is required by Decommissioning Plan Chapter 9 to be reviewed and approved by the NASA Decommissioning Project Manager and the Decommissioning Safety Committee. Accordingly, this addendum page is inserted into the regulatory review package for the aforementioned approval signatures.

Regulatory Review Approved: _____
Decommissioning Safety Committee

Regulatory Review Approved: _____
Decommissioning Project Manager

I **ENVIRONMENTAL DETERMINATION**: The following questions are used in determining whether the proposed activity involves any environmental concerns that will necessitate the generation of a written Environmental Impact Assessment or Evaluation.

1. Will implementation of this document/activity result in an increased potential to release any chemicals (gas, liquid, solid or semi-solid) to the environment?

   ☐Yes ☒No

2. Will implementation of this document/activity compromise existing capability to control, treat, or monitor releases to the environment?

   ☐Yes ☒No

3. Will implementation of this document/activity cause a physical or chemical change in the characteristics of facility discharges, effluents, or withdrawals?

   ☐Yes ☒No

4. Will implementation of this document/activity result in the permanent or temporary storage (for use, disposal, or transfer) of any hazardous or other regulated waste or any chemical(s). And, will such storage be outside of the established and previously evaluated handling facilities or outside of the limits of existing procedures where the margin of control or containment will increase the potential of a release to the environment?

   ☐Yes ☒No

5. Will implementation of this document/activity result in an increase in the amount or a change in the type of hazardous waste(s) typically generated, and/or previously evaluated for the type of activity?

   ☐Yes ☒No

6. Will implementation of this document/activity result in land disturbance (e.g. excavation work, grading), or modification or alteration of storm water drainage systems that would change site storm water runoff or increase sediment loading of storm water runoff?

   ☐Yes ☒No

7. Will implementation of this document/activity result in a physical alteration to a wastewater treatment facility or other facility system(s) or component regulated by environmental permit (e.g. discharge to groundwater permit, discharge to surface water permit etc.)?

   ☐Yes ☒No

If any of the questions above are checked 'Yes', obtain a written Environmental Impact Assessment/Evaluation from the NASA PBRF Environmental Manager with the support of the USACE Environmental Engineer and attach the document to this Regulatory Review Package.

Continue with Section II

II **<u>10 CFR 50.82 EVALUATION</u>:** The following questions are used to determine if the proposed activity conforms to the limitations for decommissioning activities specified by 10 CFR 50.82.

8. Does this activity/document accomplish or control the conduct of a decommissioning activity?

☒Yes ☐No

Note: If the answer to question 8 is NO, questions 9, 10, and 11 do not apply and should be answered as 'NA'. Proceed to question 12. If in doubt, answer question 8 'YES' and continue with questions 9, 10, and 11.

9. Will the proposed decommissioning activity foreclose release of the site for possible unrestricted use?

☐Yes ☒No ☐ NA

10. Will the proposed decommissioning activity result in significant environmental impacts not previously reviewed?

☐Yes ☒No ☐ NA

11. Will the proposed decommissioning activity result in there no longer being reasonable assurance that adequate funds will be available for decommissioning?

☐Yes ☒No ☐ NA

If the answers to questions 9, 10, and 11 are 'NO' or 'NA', proceed with question 12. If the answer to any of questions 9, 10, or 11 is 'YES', the proposed activity may not be consistent with the requirements of 10CFR50.82(a)(6). The PSC must be consulted prior to proceeding with document approval.

III **<u>10 CFR 50.59 APPLICABILITY DETERMINATION</u>**: The following questions are used to determine if the requirements of 10 CFR 50.59 are applicable to the proposed activity.

12. Is this activity or document a change request to the NRC License or Technical Specifications?

☐Yes ☒No

If the answer to this question is 'Yes', the document must be reviewed and processed in accordance with the requirements of 10 CFR 50.90. **Continue with question 13.**

13. Will this activity or document control a maintenance activity, which will restore a structure, system or component to its design basis configuration?

☐Yes ☒No

14. Will this activity or document control a maintenance activity which implements a design change where all aspects of the design change have been evaluated per 10CFR 50.59?

☐Yes ☒No

15. **(N/A NASA PBRF)** Is the proposed activity a change to any of the following programs or plans that have other specific regulatory requirements applicable to their changes:
NOTE: YES, see attached explanation
Continue with question 16.

16. Is the proposed activity a change to a procedure, even one described in the Facility Safety Analysis Report (as updated by the Decommissioning Plan), that does not contain information on how systems, structures, or components are controlled or operated?

☐Yes ☒No

If the answer to any of the above questions is 'YES', the review requirements of 10CFR50.59 do not apply unless there are multiple aspects of the proposed activity and one or more of those aspects falls outside of the 'YES' answer. Exercise care in verifying that the 'YES' answer is applicable to every aspect of the activity before concluding that 10CFR50.59 is not applicable.

IV **10 CFR 50.59 SCREENING**: The following questions are used to determine if the proposed activity constitutes a 'change' as defined in 10 CFR 50.59.

17. Does the proposed activity or document adversely affect an SSC design function described in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

☐Yes ☒No

18. Does the proposed activity or document involve a change to a procedure that adversely affects how a design function described in the Facility Safety Analysis Report (as updated by the Decommissioning Plan) is performed or controlled?

☐Yes ☒No

19. Does the proposed activity or document involve a change to the methodology used in establishing the design basis or in a safety analysis used in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

☐Yes ☒No

20. Does the proposed activity or document involve a test or experiment not described in the Facility Safety Analysis Report (as updated by the Decommissioning Plan) where a SSC is used or controlled in a manner that is outside the reference bounds of its design or is inconsistent with the analyses or descriptions in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

☐Yes ☒No

21. Does the proposed activity or document require a change to the Technical Specifications or the License?

☐Yes ☒No

If questions 17 through 21 are all answered 'NO', then the activity or document does not constitute a 'change' as defined in 10 CFR 50.59 and the activity may be implemented without NRC approval. On attached pages, write a justification for all of the 'NO' answers.

If any of the questions 17 through 21 are answered 'YES', then a 10 CFR 50.59 evaluation is required by completing the next section of the questionnaire.

If question 21 is answered 'YES', a Technical Specification Change or a License Amendment must be obtained before implementing the proposed activity.

V **10 CFR 50.59 EVALUATION**: The following questions are used to determine if the change can be implemented without obtaining an NRC approved change to the license or the Technical Specifications.

22. Does the proposed activity result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

☐Yes ☐No

23. Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

☐Yes ☐No

24. Does the proposed activity result in more than a minimal increase in the consequences of an accident previously evaluated in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

☐Yes ☐No

25. Does the proposed activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

☐Yes ☐No

26. Does the proposed activity create a possibility for an accident of a different type than any previously evaluated in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

☐Yes ☐No

27. Does the proposed activity create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

☐Yes ☐No

28. Does the proposed activity result in a departure from a method of evaluation described in the Facility Safety Analysis Report (as updated by the Decommissioning Plan) used in establishing the design basis or in the safety analysis?

☐Yes ☐No

If the answers to all of the questions of this section (questions 22 through 28) are 'NO', then attach a written explanation to this form explaining the justification for the answers.

If the answer to any of these questions is 'YES', then the proposed activity can not be implemented without obtaining NRC approval through the License/Technical Specification amendment process.

VI **INDEPENDENT REVIEW DETERMINATION:** The following questions are used to determine the level of independent review required by Technical Specifications for the proposed activity:

29. Is the document a procedure or substantive change to a procedure that meets one of the following:

a. Is a characterization, decommissioning, operating, administrative, or maintenance activity determined to be within the scope of the NASA PBRF Decommissioning Project QA Plan

☒Yes ☐No

b. Is an access control, emergency action (including fire protection program implementation), or facility inspection or audit procedure

☐Yes ☒No

c. Involves radiological exposure control, survey activities, or radwaste shipping and handling

☐Yes ☒No

d. Involves an activity which could result in a measurable release to the environment

☐Yes ☒No

30. Is the document a proposed change to the License or Technical Specifications

☐Yes ☒No

31. Does the document involve a proposed test or experiment

☐Yes ☒No

32. Is the document a proposed modification to facility structures systems or components determined to be within the scope of the NASA PBRF Decommissioning Project QA Plan

☐Yes ☒No

33. Does the document involve an investigation of a violation of Technical Specifications (this could include the documentation of such an investigation under a report of an unusual occurrence or other NRC report of event)

☐Yes ☒No

34. Does the document involve reporting of a reportable event per Technical Specification 3.8

☐Yes ☒No

If any the items in questions 29 through 34 are marked 'YES', the document requires an Independent Technical Review and must be signed by a designated Responsible Technical Reviewer. Continue with Question 35.

35. Does the document involve a written safety evaluation (i.e. does it include section V of this form and written justification for the 'NO' answers)

☐Yes ☒No

36. Does the activity require a change in the Technical Specifications or license, or does it require NRC approval per 10CFR50.59 prior to implementation

☐Yes ☒No

37. Is the document a proposed change to the License or Technical Specifications

☐Yes ☒No

38. Does the document involve a review of a violation, deviation, or reportable event which requires reporting to the NRC in writing

☐Yes ☒No

39. Is the document a written summary of audit reports identified in Technical Specification 3.5.4

☐Yes ☒No

If any of the items in questions 35 through 39 are 'YES', the documents require review by a qualified Independent Safety Reviewer.

EXPLANATION OF 'YES' ANSWER TO QUESTION 15 ON THE 10CFR50.59 APPLICABILITY DETERMINATION:

Q 15 Is the proposed activity a change to any of the following programs or plans that have other specific regulatory requirements applicable to their changes?

Although there are no additional regulatory requirements specified in 10CFR50 related to revisions to the Decommissioning Plan that are applicable to this revision, there are additional evaluations required as stipulated in condition 3.A.1 of the NRC Facility License. The License condition states, "The licensee may make changes to the above plan and revisions without prior U. S. NRC approval provided the changes do not:" It then lists 8 criteria. The written evaluation that demonstrates compliance with these eight criteria is attached to this regulatory review package.

JUSTIFICATION OF "NO" ANSWERS ON THE 10CFR50.59 SCREENING:

Question 17: *Does the proposed activity or document adversely affect an SSC design function described in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?*

The activity being evaluated is an administrative change to the Decommissioning Plan. The revision incorporates wording from the revised license condition 3.A.4 approved by the NRC in Amendments 12 and 8 to licenses TR-3 and R-93 respectively. This license condition relates to the degree of NRC confirmation of Final Status Survey required prior to backfilling an excavated area. There are no changes to the considerations in the accident analyses, and no changes to the descriptions of the operations of structures, systems, or components.

Therefore, since all SSC's involved in this activity will be operated, maintained, and ultimately dismantled in the same manner as described and within the parameters described in the previous revision to Decommissioning Plan, this revision will have no adverse affect on these SSC's.

Question 18: *Does the proposed activity or document involve a change to a procedure that adversely affects how a design function described in the Facility Safety Analysis Report (as updated by the Decommissioning Plan) is performed or controlled?*

This revision to the Decommissioning Plan has no impact on any SSC's described in the previously approved Decommissioning Plan. Since no SSC design functions are impacted, there will be no procedure changes that affect SSC design functions. Therefore, this activity does not involve any change to procedures that would adversely affect the control or performance of SSC's as previously described.

Question 19: *Does the proposed activity or document involve a change to the methodology used in establishing the design basis or in a safety analysis used in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?*

The revision is an administrative change to the Decommissioning Plan. The revision incorporates wording from the revised license condition 3.A.4 approved by the NRC in Amendments 12 and 8 to licenses TR-3 and R-93 respectively. This license condition addresses the degree of NRC confirmation of Final Status Survey required prior to backfilling an excavated area. The revision has no affect on the design basis for SSC's, and makes no changes to the accident analysis or safety analysis. This revision has not changed the methodology to be used in the Final Status Survey. In addition, this revision to the Decommissioning Plan implements an amendment to the Facility Licenses that has been evaluated and approved by the U.S. NRC and has been determined to involve none of the Significant Hazards Considerations specified in 10 CFR 50.92. All previous bases used in the previous safety and accident analyses and the methodologies used in evaluating these design bases remain intact. Therefore, the proposed activity does not involve a change to the methodology used in establishing a design basis or in a safety analysis used in the Facility Safety Analysis Report, and requires further evaluation.

Question 20: *Does the proposed activity or document involve a test or experiment not described in the Facility Safety Analysis Report (as updated by the Decommissioning Plan) where a SSC is used or controlled in a manner that is outside the reference bounds of its design or is*

inconsistent with the analyses or descriptions in the Facility Safety Analysis Report (as updated by the Decommissioning Plan)?

A "test or experiment" is an activity where an SSC is utilized or controlled in a manner which is either outside the reference bounds or inconsistent with analyses or descriptions in the Decommissioning Plan. This revision to the Decommissioning Plan does not have any affect on the methods or manner of performance of decommissioning activities. This revision does not result in the performance of any additional activities other than those discussed in the previously approved Plan, and results in no tests or experiments. Therefore, this activity does not involve a test or experiment where an SSC is used or controlled in a manner that is outside the reference bounds of its design or inconsistent with the analyses in the Decommissioning Plan.

Question 21: *Does the proposed activity or document require a change to the Technical Specifications or the License?*

The proposed revision to the Decommissioning Plan implements a revised License condition approved by the NRC by the issuance of amendments 12 and 8 to Licenses TR-3 and R-93 respectively. Therefore, the proposed activity does not require a change to the Technical Specifications to the License.

EVALUATION OF REVISION AGAINST CRITERIA OF LICENSE CONDITION 3.A.1:

Condition 3.A.1 of the Facility License stipulates that "The licensee may make changes to the above plan and revisions without prior U. S. NRC approval provided the proposed changes do not:

a. Require Commission Approval pursuant to 10 CFR 50.59;

b. Reduce the coverage requirements for scan measurements;

c. Increase the DCGL and related minimum detectable activity (for both scan and fixed measurement methods);

d. Use a statistical test other than the Sign test or Wilcoxon Rank Sum test for evaluation of the final status survey;

e. Result in significant environmental impacts not previously reviewed;

f. Increase the radioactivity level, relative to the applicable DCGL, at which an investigation occurs;

g. Increase the Type I decision error; or

h. Decrease an area classification (i.e., impacted to unimpacted; Class 1 to Class2; Class 2 to Class 3; or Class 1 to Class 3).

The previously presented evaluation has shown that the proposed revision to the Decommissioning Plan does not require a Technical Specification change and can be implemented without prior NRC approval pursuant to 10CFR50.59.

Criteria b through h above relate to the collection, evaluation, and analyses of data that will be collected during Final Status Survey to demonstrate that NRC approved radiological criteria are met in order to terminate the Facility License and release the site for unrestricted use.

The proposed revisions to the Decommissioning Plan do not affect Final Status Survey techniques, evaluations or analyses of data collected as discussed in the approved Decommissioning Plan. The proposed revisions further clarify that the Final Status Surveys will be performed for areas that have been excavated, the results submitted to the NRC for review, and NRC concurrence obtained for backfilling of the affected areas prior to backfilling or otherwise rendering the area inaccessible. Prior to the NRC approval of the amended license condition 3.A.4, the license condition required NRC confirmation of the FSS results rather than review and concurrence.

Since this revision incorporates a revised Condition 3.A.4 in the NRC License that has already been reviewed and issued by the NRC, these proposed changes in revision 4 to the Decommissioning Plan may be implemented without prior approval by the U.S. Nuclear Regulatory Commission.

The following is a detailed list of the changes that were incorporated by Revision 4 to the Decommissioning Plan:

1. Page 1-23 - The second paragraph was revised ***from*** "Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, confirmed, and approved by the NRC. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and site visits." ***to*** "Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, the results submitted to the NRC with a safety or technical justification for backfilling, and concurrence obtained from the NRC that the area may be backfilled or otherwise rendered inaccessible. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and/or site visits."

2. Page 2-1- The third paragraph was revised ***from***, "Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, confirmed, and approved by the NRC. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and site visits." ***to*** "Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, the results submitted to the NRC with a safety or technical justification for backfilling, and concurrence obtained from the NRC that the area may be backfilled or otherwise rendered inaccessible. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and/or site visits."

3. Page 2-30, the first paragraph was revised ***from***, "Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, confirmed, and approved by the NRC." ***to*** "Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, the results submitted to the NRC with a safety or technical justification for backfilling, and concurrence obtained from the NRC that the area may be backfilled or otherwise rendered inaccessible. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and/or site visits."

4. Page 2-30, the first sentence of the second paragraph was revised by deletion of the word "either" to correct a typographical error.

5. Page 2-39, the first paragraph was revised ***from***, ""Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, confirmed, and approved by the NRC." ***to*** "Prior to any backfill operations, the Final Status Survey of the area to be backfilled will be completed, the results submitted to the NRC with a safety or technical justification for backfilling, and concurrence obtained from the NRC that the area may be backfilled or otherwise rendered inaccessible. It is anticipated that the project will require backfilling, incrementally over the course of the project, which may require multiple NRC survey reviews and/or site visits."