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DECOMMISSIONING PLAN FOR U.S. ARMY MATERIALS TECHNOLOGY LABORATORY RESEARCH REACTOR

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DECOMMISSIONING PLAN
FOR
U.S. ARMY MATERIALS TECHNOLOGY LABORATORY
RESEARCH REACTOR

Prepared by
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EXECUTIVE SUMMARY

This Decommissioning Plan (DP) for the U.S. Army Materials Technology Laboratory Reactor at Watertown, Massachusetts, specifies that the reactor facility will be decommissioned by decontamination. The Army decided upon total dismantlement as the preferred alternative for reasons explained in this document. However, this DP describes decommissioning of the reactor facility by continuous dismantlement as required in order to decontaminate by removing contaminated materials. This will accomplish decommissioning and lead to the release of the facility for unrestricted use.

The facility to be decommissioned includes the containment building (Building 100), piping in the reactor facility laboratory (Building 97), an underground pool water retention tank and remnant components of the secondary coolant system.

The following tasks will be undertaken in order to complete the decommissioning of the reactor. Some of these may be performed in parallel. First, the Decommissioning Contractor will remove any auxiliary structures, which will include Cistern 242, the piping in Building 97, and the secondary coolant system.

Secondly, the Contractor will remove the components and systems from the reactor building itself. This will involve removing the reactor pool internals, reactor pool liner, platforms, basement piping, basement sumps, the gamma facility and storage tubes, and the reactor pool.

Following completion of decommissioning, the termination survey will be performed by the U.S. Army Environmental Hygiene Agency. The survey will include the reactor building and the soil areas where excavations were performed.

A Quality Assurance Plan will be developed and followed throughout the decommissioning process to ensure that the work is performed in compliance

with the Decommissioning Plan, the Statement of Work, procedures, and other applicable specifications and requirements.

In addition, the decommissioning program will be conducted in such a way as to be certain that the following regulations, guides, and standards are met: all applicable regulations from the Code of Massachusetts Regulations; pertinent portions of the U.S. Code of Federal Regulations; applicable Army regulations; regulatory guides from the regulating agencies (e.g., the Nuclear Regulatory Commission and the Environmental Protection Agency); and the standards set forth by certain institutions or technical societies (e.g., American Society for Testing and Materials, the International Commission on Radiological Protection, and the American National Standards Institute).

The Occupational and Radiation Protection Programs for the AMTL Reactor decommissioning will consist of a set of policies, procedures, and instructions to protect workers, the general public, and the environment. The Occupational and Radiation Protection Programs will provide occupational health, health physics, industrial hygiene, and safety elements.

The Radiation Protection Program will include requirements to monitor radiation and radioactive materials, to control distribution and releases of radioactive materials, and to keep radiation exposure within 10 CFR 20 limits and at as-low-as-reasonably-achievable limits. The related Industrial Safety and Hygiene Program will be concerned primarily with protection against nonradioactive exposures and hazards and will be administered in accordance with regulations from the Occupational Safety and Health Administration.

The decommissioning operations will be performed and managed by a decommissioning contractor. As the licensee, the AMTL Commander is responsible for the overall decommissioning project and has authority in all associated matters, including safety. The overall project management, however, will be accomplished by the U.S. Army Corps of Engineers, New England Division. The AMTL will provide a quality assurance expert to ensure that the contractor is performing work in accordance with terms of the contract.

The estimated cost of the AMTL Reactor decommissioning project ranges between $4.3 and $5.1 million. Both estimates are based on decontamination. The lower estimate is based on an assumed void volume of 10% in the packaged radioactive waste, and the higher cost estimate is based on an assumed void volume of 60%. The estimates also assume that the radioactive waste would be disposed of between 1 January 1992 and 1 January 1993, which would involve paying a state penalty surcharge of $120 per cubic foot of radioactive waste. The funding for this project will come from appropriate U.S. Army sources and will be distributed by the U.S. Army Corps of Engineers Military Programs (CEMP).

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ACRONYMS

ACGIH	American Conference of Governmental Industrial Hygienists
AHERA	Asbestos Hazardous Emergency Response Act
ALARA	as-low-as-reasonably-achievable
AMC	Army Material Command
AMCCOM	U.S. Army Armament, Munitions, and Chemical Command
AMMRC	Army Materials and Mechanics Research Center
AMTL	U.S. Army Materials Technology Laboratory Research Reactor
ANS	American National Standard
ANSI	American National Standards Institute
ARCHS	Army Reactor Committee for Health and Safety
ASTM	American Society for Testing and Materials
BeO	beryllium oxide
CCP	Contamination Control Point
CIH	certified industrial hygienist
CMR	Code of Massachusetts Regulations
COE	Corps of Engineers
COR	Contracting Officer's Representative
CPM	critical path method
CSCA	Controlled Surface Contamination Area
DOT	Department of Transportation
DP	Decommissioning Plan
dpm	disintegrations per minute
HEPA	high-efficiency particulate absorption
HP	Health Physicist
ICRP	International Commission on Radiological Protection
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear [illegible] Operations

ISHP	Industrial Safety and Hygiene Program
KO	Contracting Officer
LLD	lower limit of detection
LSA	low-specific activity
NCRP	National Committee on Radiation Protection and Measurement
NED	New England Division
NRC	Nuclear Regulatory Commission
NVLAP	National Voluntary Accreditation Program
OSHA	Occupational Safety and Health Administration
PPE	Personal protective equipment
QAE	Quality Assurance Evaluator
QAE/HP	Quality Assurance Evaluator/Health Physicist
RAM	radioactive material
RC&SO	Radiological Control and Safety Officer
RCA	radiation control area
RCC	Radiation Control Committee
RE/HPS	Radiological Engineer/Health Physics Supervisor
RFP	Request for Proposal
RHWPs	Radiation/Hazardous Work Permits
RPO	Radiation Protection Officer
RPP	Radiation Protection Program
RSM	Radioactive Shipment Manifest
SOW	Statement of Work
TLV	Threshold Limit Values
USAEHA	U.S. Army Environmental Hygiene Agency
USATHAMA	United States Army Toxic and Hazardous Materials Agency
WBS	Work Breakdown Structure

CHAPTER 1

BACKGROUND AND MANAGEMENT PLAN

1. INTRODUCTION

This document is the Decommissioning Plan (DP) for the Army Materials Technology Laboratory (AMTL) to use in decommissioning the AMTL Research Reactor located at Watertown, Massachusetts. Decommissioning as described in this DP is accomplished through decontamination. Release of the reactor building and site for unrestricted use will be requested from the Nuclear Regulatory Commission (NRC) following completion of decommissioning. The location of the AMTL in relation to the city of Watertown is shown in Figure 1-1. The location of the reactor building (Building 100) within the AMTL is shown in Figure 1-2.

This DP follows the format and content specified in the NRC document titled Guidance and Discussion of Requirements for an Application to Terminate a Non-Power Reactor Facility Operating License, Revision 1, dated September 15, 1984. When the DP is approved by the licensee, the DP will be submitted to the NRC for approval, and also made available to the public through local public repositories.

1.1 Summary Description

The following section describes the AMTL Reactor Facility and reactor usage during the licensed period. In addition, a brief discussion is given of the decommissioning alternatives considered and the cost, duration, and radiation exposure estimates for each alternative. Major tasks and schedules of the preferred alternative, quality assurance considerations, contractor involvement in the decommissioning, and the planned post-decommissioning characterization are also summarized in this section.

1.1.1 Reactor Facility Description

The AMTL Research Reactor consists of the reactor building, certain liquid water systems remaining inside Building 97, an underground pool water retention tank, and remaining components of the secondary coolant system.

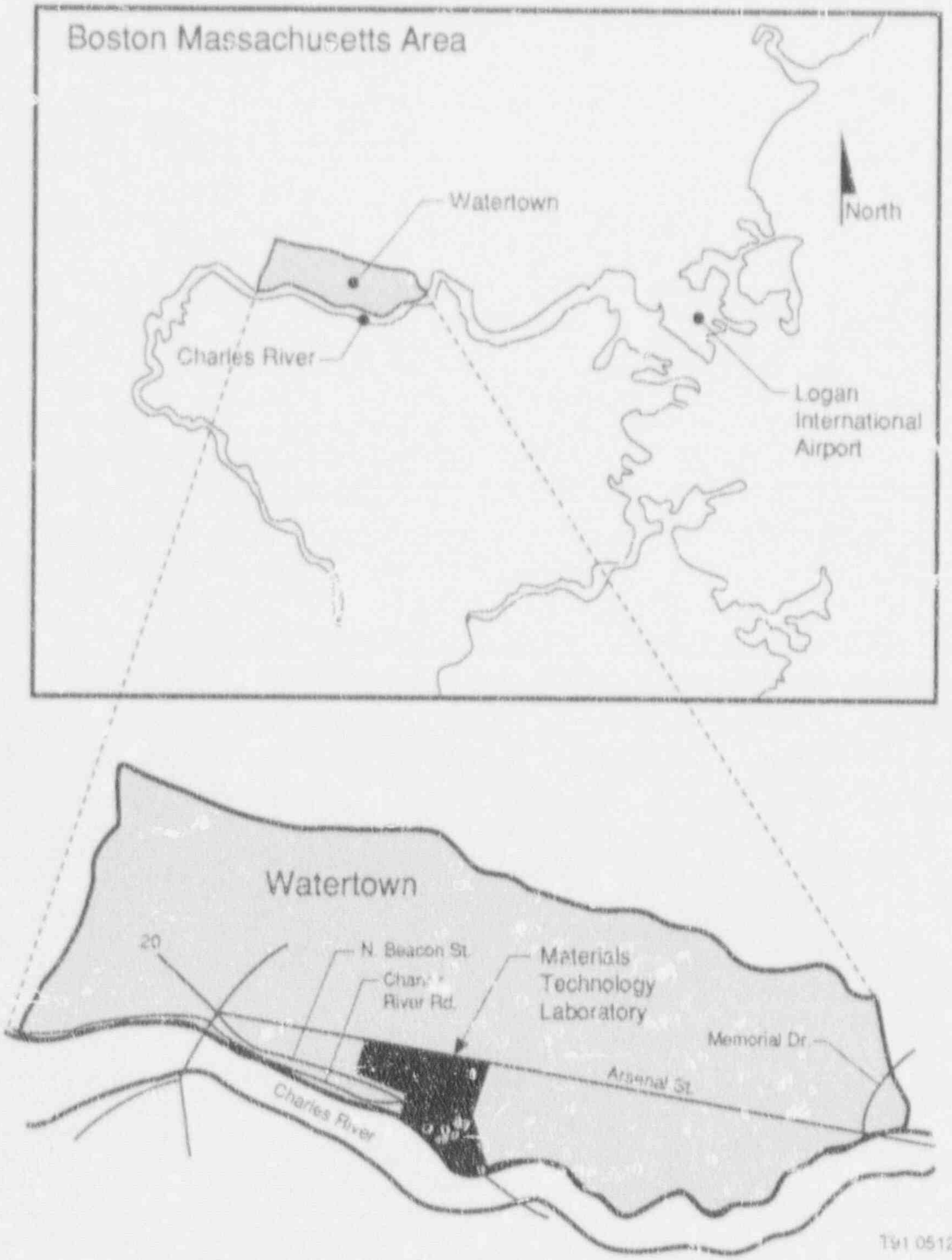

Figure 1-1. Location of Army Materials Technology Laboratory in Watertown, Massachusetts.

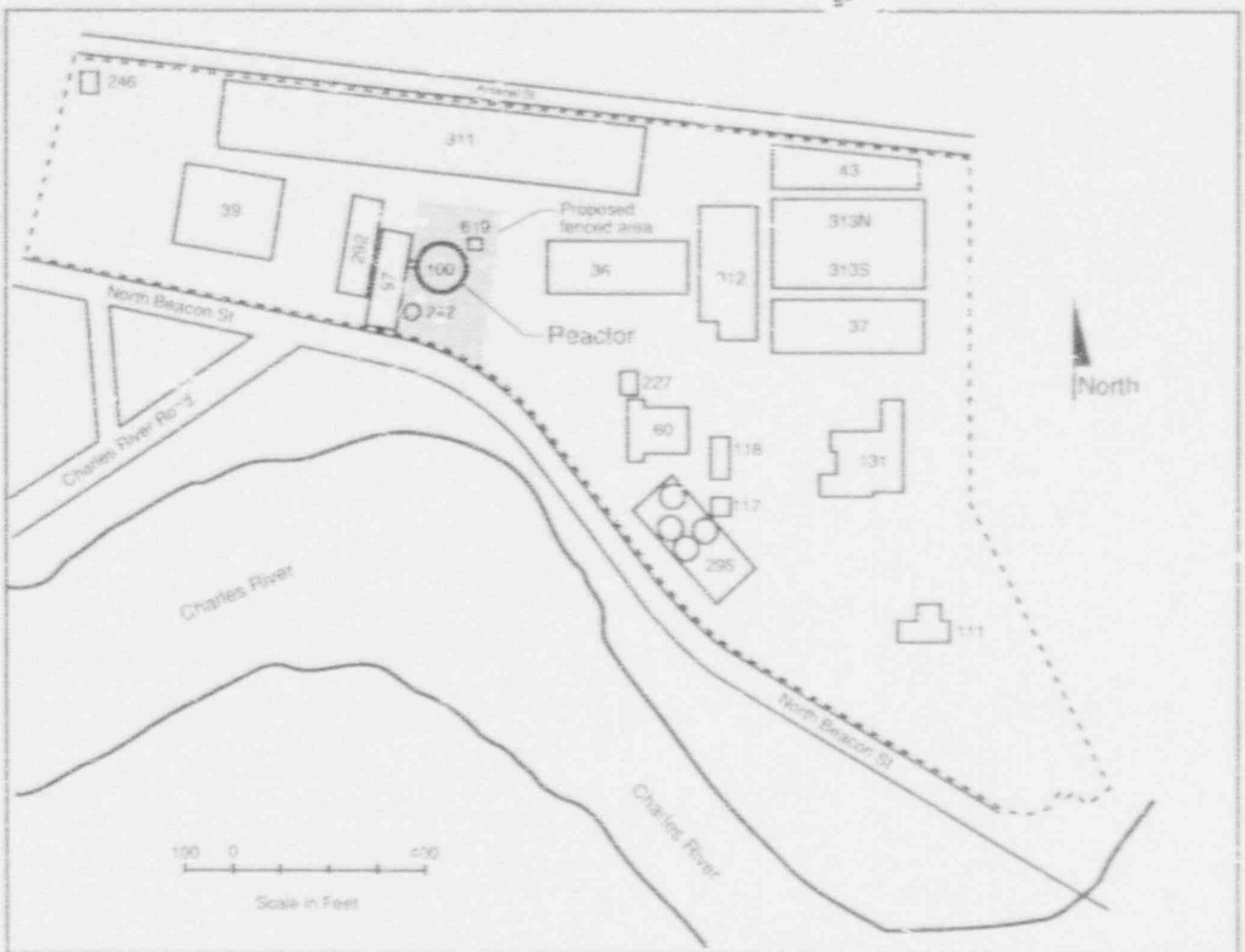

Figure 1-2. Army Materials Technology Laboratory general site map.

1.1.1.1 Building 100, Reactor Containment Building. Building 100, the reactor containment structure (see Figure 1-3), is a cylindrical pressure vessel 80 ft in diameter and approximately 67 ft high from ground level, with an elliptical top. A cross-section of the containment shell is shown in Figure 1-4. The basement foundation is approximately 19 ft below ground level; a 6-ft-diameter gamma-ray facility extends an additional 9 ft below the foundation level. The perimeter walls of the gas-tight containment shell are 2-ft-thick concrete and extend up to the crane rails 44 ft above ground level. The aboveground walls and the roof are covered with a 1/2-in. welded steel plate. There are two large penetrations in the perimeter wall for personnel airlocks permitting access to the interior of the shell. In addition, there is a large double door in the southwest portion of the containment shell. The electrical utilities are brought in through seals, which consist of a conduit box filled with a sealant and a metal tube welded to the shell. This allows the electrical cables to pass through the containment shell. The water inlet lines (city water and secondary cooling water) and outlet lines (liquid waste and secondary cooling water) pass through pipes welded to the shell. The air, steam, return condensate, and demineralized water lines also enter the containment shell from Building 97 through pipes welded to the steel shell. Air intake and exhaust are accomplished through steel ducts with flanges welded to the shell and provided with automatic closing dampers. Overpressurization protection for the shell is provided by a 2-in. line with a water trap equivalent to a 5-ft head of water installed between the shell and the atmosphere. The containment shell completely encloses the reactor and all of its associated equipment, with the exception of Cistern 242 and the secondary coolant system.

The containment-shell floor plans are shown in Figure 1-5. The main operating floor, which is at ground level, is about 76 ft in diameter. During operations, the first platform provided access to the six slant beam tubes, which exit at this level from the reactor vessel. The second platform provided an area for the reactor control room and space for personnel and equipment for loading and unloading the reactor. A standard 10-ton crane, mounted on a circular track, was used to service the main floor, the platforms, and part of the basement through floor openings.

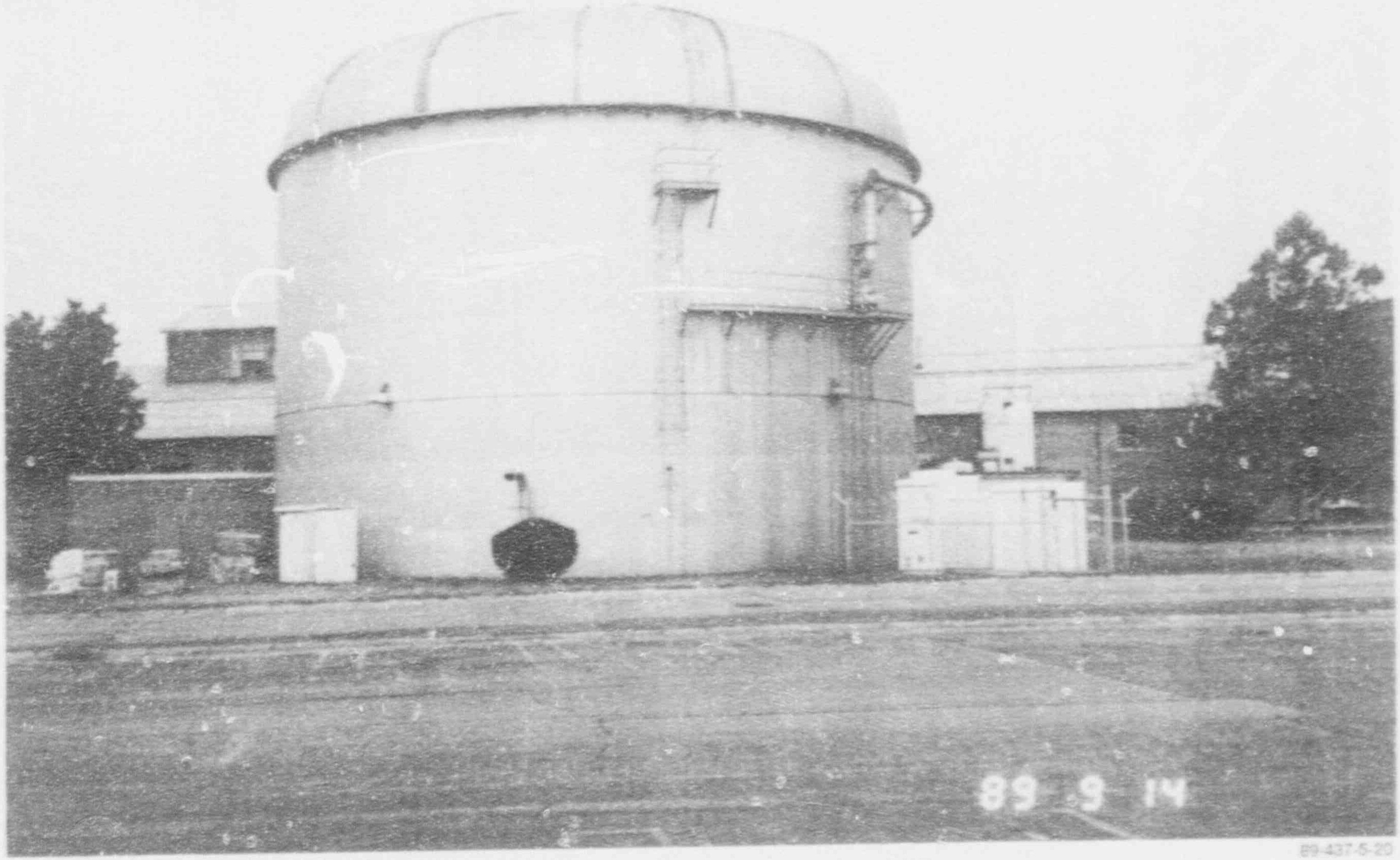

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Figure 1-3. View of Building 100, looking west. Building 97 is in the background.

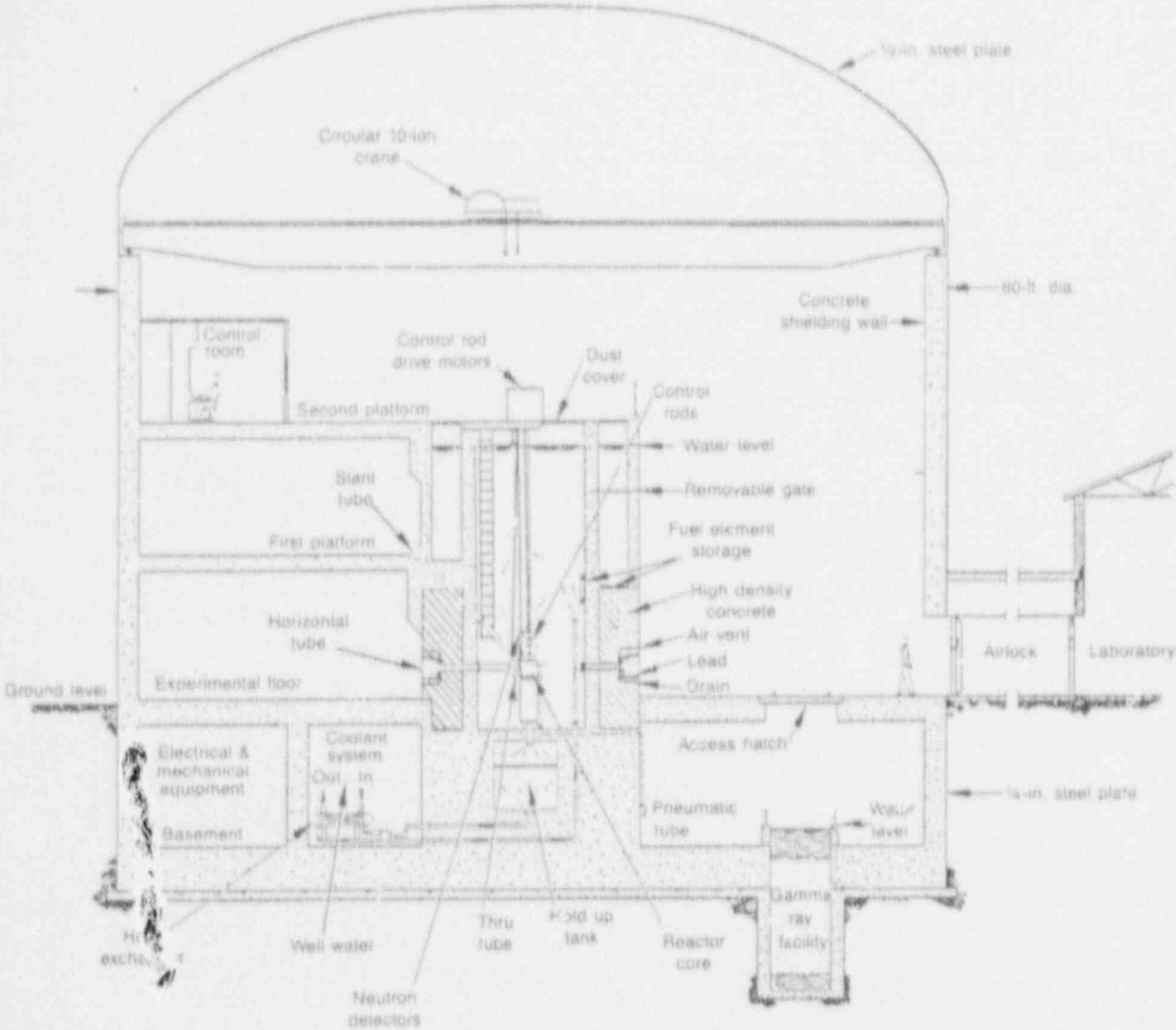

Figure 1-4. Reactor containment-shell cross-sectional view.

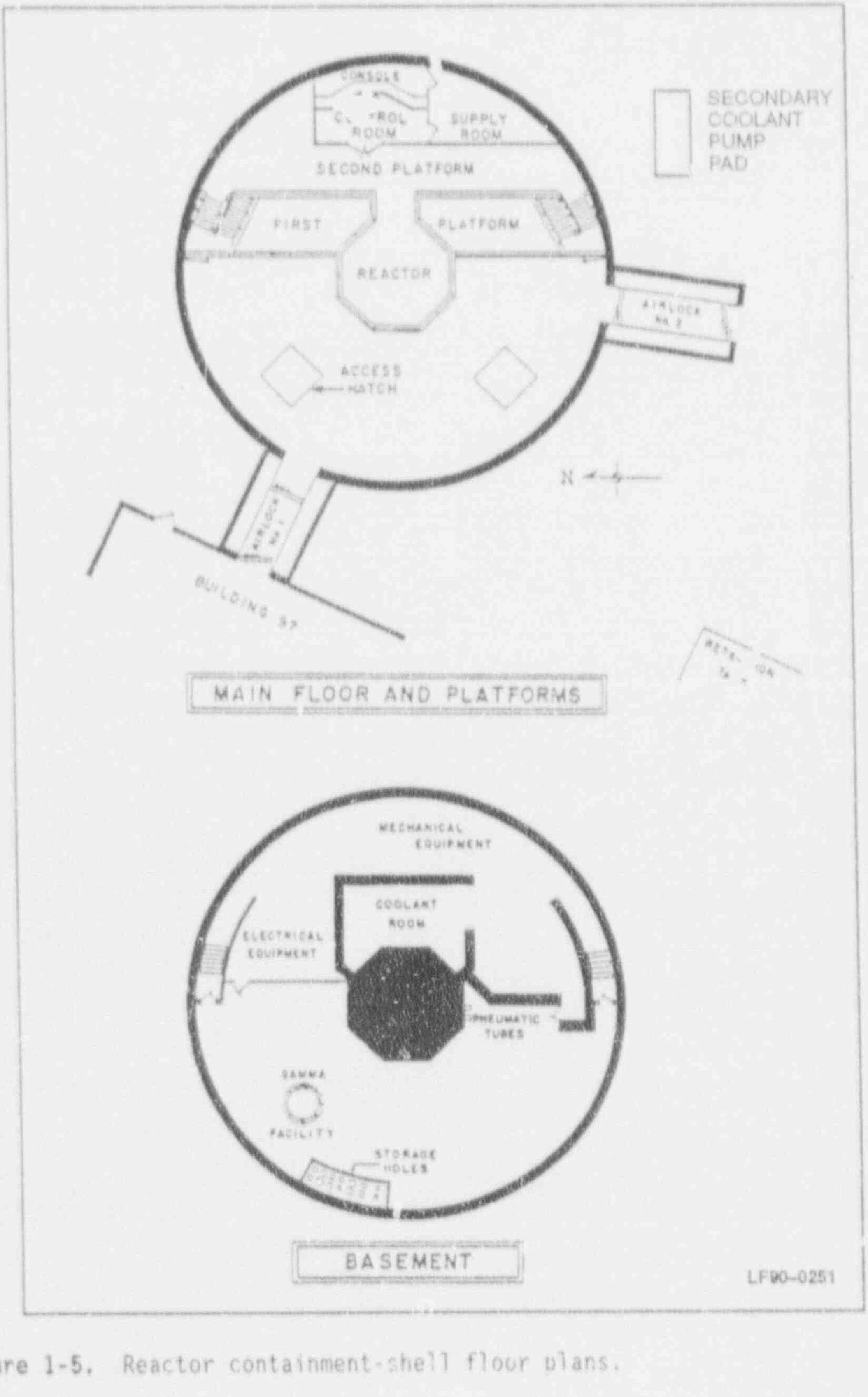

Figure 1-5. Reactor containment-shell floor plans.

The basement area was used for experimental and operational purposes. A gamma-ray exposure experimental facility was built below the basement floor level. The gamma-ray facility could also be used for fuel storage. There are 16 vertical storage tubes in one part of the basement floor that could be used for storing radioactive materials, such as beam tube plugs, collimators, and irradiated samples. These 16 vertical tubes are 4 ft 3 in. deep. Two tubes are 12 in. in diameter, two are 10 in. in diameter, eight are 8 in. in diameter, and four are 4 in. in diameter.

All the primary-coolant equipment for the operation of the reactor is located in the basement and is separated from occupied areas by 2-ft-thick, ordinary concrete walls. A main sump is also located in the basement, to which all liquid waste within the containment shell drained. The liquid waste was then automatically pumped to a liquid-waste storage tank, which was part of the liquid-waste handling system that was located in Building 97 before its removal.

The reactor is a version of the "swimming pool" type, the pool having been replaced by an octagonal open tank, completely above ground. The internal dimensions of the tank are 10-1/2 ft in diameter by a depth of 30 ft, which provided 4 ft of water shielding from the fuel horizontally and 22 ft of water shielding above the centerline of the fuel vertically. The concrete biological shield consists of an approximate inner 16 in. of ordinary concrete and 4 ft of high-density concrete. Maximum neutron beam facilities were utilized by the addition of 16 horizontal 6-in. beam tubes, one horizontal 6-in. through tube, and six slant 6-in. beam tubes. See Figure 1-6 for a three-dimensional model of the reactor.

A cover made up of hinged metal plates closed off most of the top of the pool. This allowed access to the pool for transfer of fuel elements between the reactor and the lead-lined recessed storage positions available in the annulus. The annulus was built around the upper portion of the reactor pool and is accessible from the pool through a removable, watertight gate which permitted the movement of fuel elements under water.

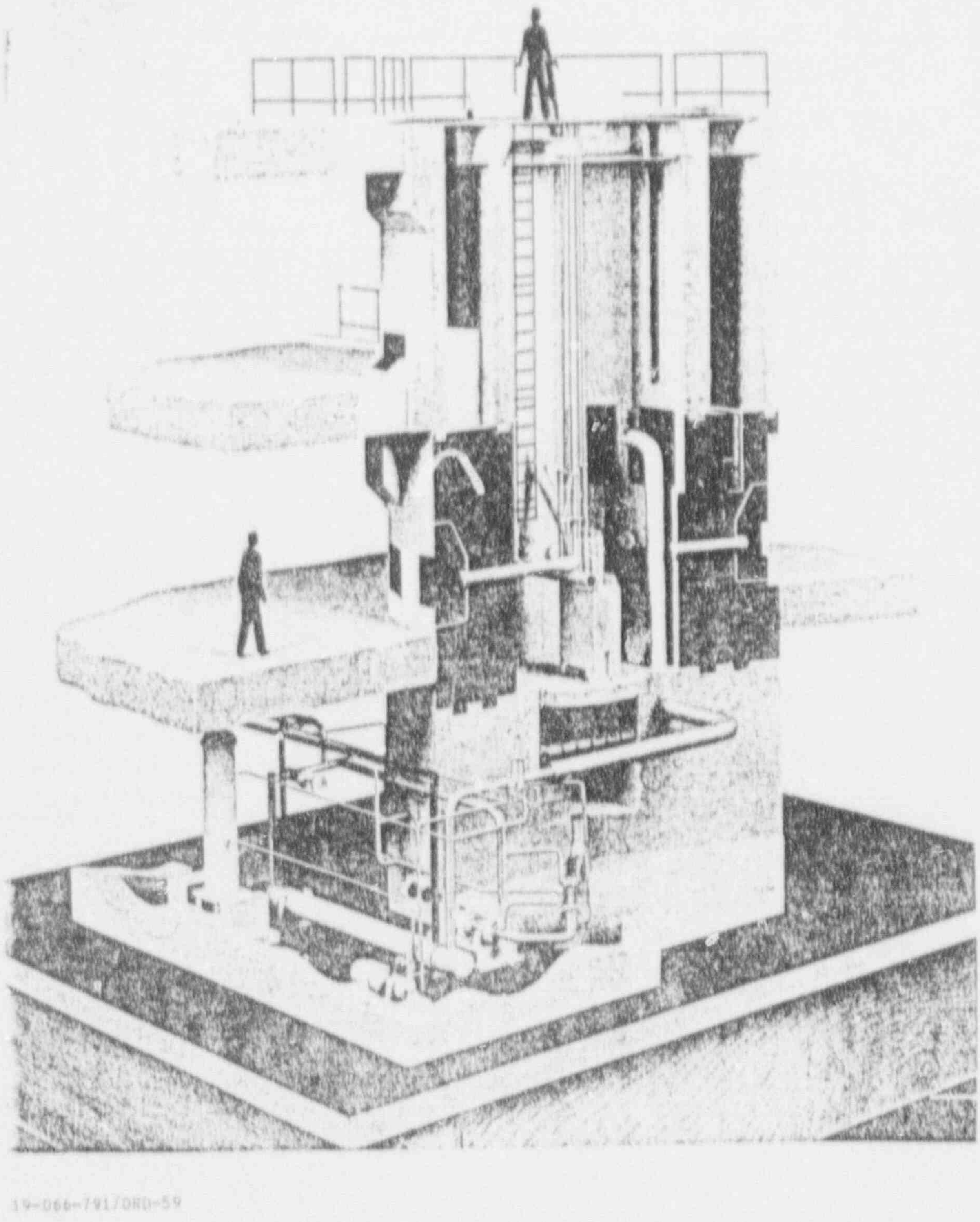

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Figure 1-6. Army Materials Technology Laboratory Research Reactor model.

The fuel elements used in the open-top, tank-type, thermal, heterogeneous, H_2O-cooled and moderated reactor were Materials Testing Reactor-type assemblies fitted into a gridplate and expanded to a 7 x 9 array of fuel elements, reflector elements, and plugs as shown in Figure 1-7. The increase in grid plate size permitted changes of core configuration and provided the greater fuel loading needed to overcome reactivity losses to the beam tubes and for operation at higher power levels. The reactor grid plate is supported on a pedestal on the bottom of the pool as shown in Figure 1-7.

1.1.1.2 Building 97, Reactor Facility Laboratory. The reactor facility included portions of Building 97, which provided access to Building 100 through an airlock and which also contained offices and laboratories for support of the reactor operations (see Figure 1-5). The liquid-waste handling system for the reactor was also contained in the south end of this building. This system consisted of three aboveground 3000-gal wastewater storage tanks and a disposable cation, anion, and mixed ion-exchange system for storing and treating contaminated water from the reactor. Contaminated water stored in the retention tank, Cistern 242, could also be pumped to this system for processing. (According to operating history records, contaminated water was stored in the retention tank, Cistern 242, at least once.) The three storage tanks, mixed ion-exchange system, and pool fill, make-up, and laboratory demineralizer system were removed after deactivation of the reactor to make room for a particle accelerator. The portion of the piping that remains between Buildings 97 and 100 and going to Cistern 242 will be addressed in this report. Building 97 presently contains chemistry laboratories, an ion-implantation facility, and a particle accelerator for neutron production.

1.1.1.3 Cistern 242. A buried pool water retention tank, Cistern 242, located approximately 25 ft southwest of Building 100, served as the low-level waste storage tank for the reactor (see Figure 1-8). The 23-1/2-ft-square tank is constructed of 1-ft-thick concrete and is 15 ft deep. A manhole cover provides access to the tank's interior. The tank was used to hold the reactor pool water during reactor maintenance to minimize the time and expense of supplying demineralized water to refill the pool. Had the reactor pool water

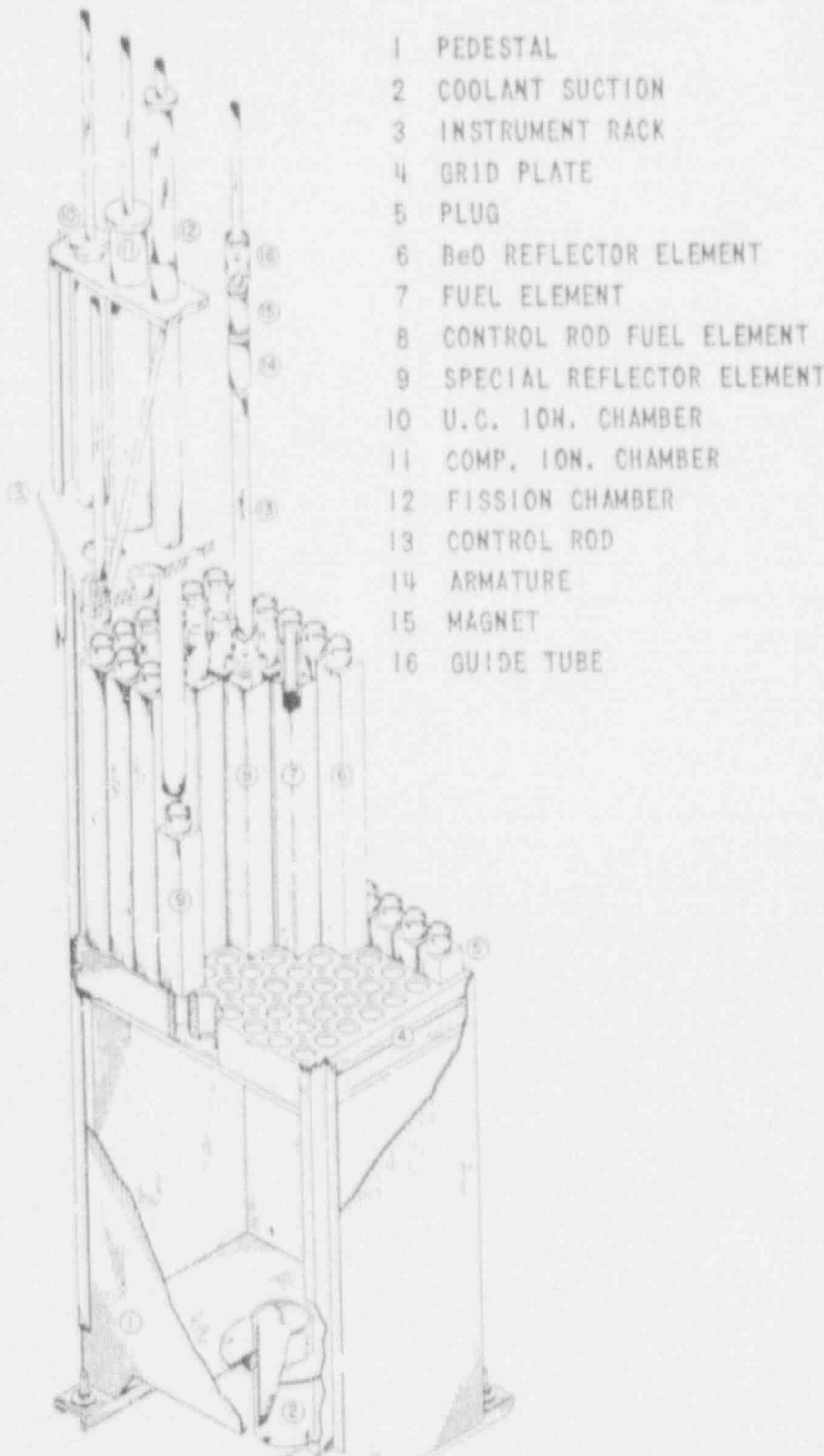


Figure 1-7. Reactor core support.

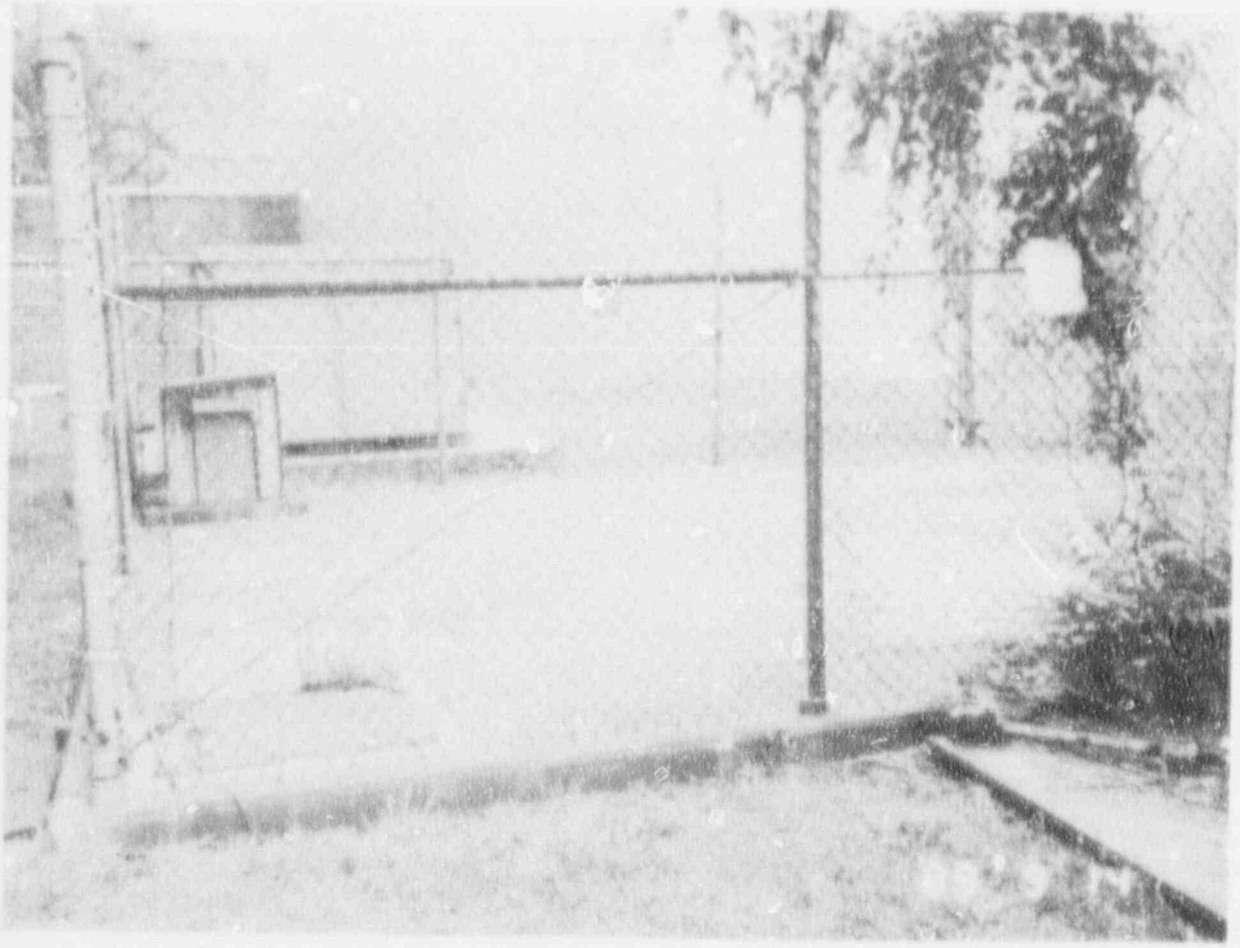


Figure 1-8. View of Cistern 242, looking north. Building 100 is in background.

become seriously contaminated, it could have been pumped to the retention tank. The contaminated water could then be processed in the liquid-waste handling system located in Building 97. In October 1966, a liquid-level-indicating recorder was installed to facilitate waste management and to provide a method of monitoring the retention tank for any appreciable leakage. The water contained in the retention tank was drained and the tank flushed to the sewer system after the reactor was deactivated.

1.1.1.4 Remaining Components of the Secondary Coolant System. The remaining components of the secondary coolant system consist of the following items:

- Secondary coolant pump concrete pad
- Three pumps
- Secondary coolant sump beneath the concrete pad
- Underground secondary coolant piping and conduit between the sump area to the reactor building.

1.1.2 Reactor Usage During Licensed Period

The AMTL Reactor was used to conduct various materials studies, such as experiments in the structure of heavy-metal azides, lattice dynamics studies on explosive-type materials and determinations of vibrational spectra of organic secondary explosives, polycrystalline and single-crystal coherent scattering materials and liquids, activation analysis of samples containing trace impurities, and inducing slight radiation effects in materials. These experiments were conducted during the operating life of the reactor from June 15, 1960 to March 27, 1970.

1.1.3 Comparison of Considered Decommissioning Alternatives

Following shutdown and deactivation of the AMTL Reactor, decommissioning planning was started. The objective of decommissioning the reactor is to obtain release of the facility and site from the NRC for unrestricted use and to terminate the license. The alternatives of safe storage and entombment were not considered because they did not achieve the Army's objective of unrestricted use. The two alternatives determined to be in accordance with the Army's objectives are discussed in detail in the Decision Analysis Report.[1] These considered alternatives were partial dismantlement and total dismantlement.

Partial dismantlement, as described in the Decision Analysis Report, details the removal of those interior and exterior components known to be contaminated. The remainder of the facility would be left intact for possible reuse. The total dismantlement alternative includes all work that makes up partial dismantlement plus the total dismantlement and disposal of Building 100 and the reactor containment shell as clean waste. Both of these alternatives will achieve total decontamination of the facility.

The comparison of the alternatives included estimated costs, project duration, and radiation exposure to personnel; the comparisons are shown in Table 1-1. The basis for the estimated cost for decontamination is given in Section 3.7. The basis for the estimated exposure to personnel is given in Section 1.4.

The Army decided upon total dismantlement as the preferred alternative for reasons explained in the following section. However, this DP describes decontamination through continuous dismantlement as required in order to remove all contaminated materials. This will accomplish decommissioning and lead to the release of the facility for unrestricted use.

The Army will make available to the public a plan which will describe the method by which the reactor building will be dismantled. This plan will be implemented once the NRC has determined the facility meets the unrestricted use criteria.

Table 1-1. Comparison of considered alternatives

Alternative	Estimated Costs[a] ($M)[b]	($M)[c]	Estimated Project Duration[d] (weeks)	Estimated Radioactive Exposure[e] (Man-rems)
1. Partial Dismantlement	4.3	5.1	43	10.0
2. Total Dismantlement	5.3	6.1	54	10.0

a. The estimated costs assume the radioactive waste will be disposed of after January 1, 1992, but prior to January 1, 1993. Disposal of the low-level radioactive waste during that period of time causes a penalty surcharge of $120/ft^3$, which is included in each estimated cost.

b. These cost estimates are based on an estimated volume of packaged radioactive waste of 8,000 ft^3 which includes 10% void volume.

c. These cost estimates are based on an estimated volume of packaged radioactive waste of 12,000 ft^3 which includes 60% void volume.

d. The estimated duration assumes the shipment of radioactive waste will occur in parallel with removal of contaminated components. The estimated duration includes both phases of the termination survey but not the evaluation of the Phase II results.

e. The estimated radioactive exposure is the same for partial and total dismantlement because all of the exposure will occur during decontamination as part of either alternative.

1.1.4 Major Advantages of the Preferred Alternative

The preferred decommissioning alternative was total dismantlement. This alternative was selected because of advantages listed below:

- The building site could be returned to its original condition for unrestricted use

- Total dismantlement will provide, with maximum certainty, that all radioactive contamination has been removed.

1.1.5 Major Tasks and Schedules

The major decommissioning tasks to be performed by the decommissioning contractor and the estimated durations are listed below. The project schedule will depend on the start date (currently unknown) and the number of tasks performed in parallel. A schedule of the entire project is presented in Section 3.8.

Major Tasks	Estimated Duration (weeks)
A. Remove auxiliary structures	8
1. Cistern 242	
2. Piping in Building 97	
3. Secondary coolant system	

Major Tasks	Estimated Duration (weeks)
B. Decontaminate reactor building by removing components	12
1. Reactor pool internals	
2. Reactor pool liner	
3. Platforms	
4. Basement piping	
5. Basement sumps	
6. Gamma facility and storage tubes	
7. Reactor pool	
C. Dispose of radioactive waste	12
D. Perform internal decontamination	2
E. Perform Phase I termination survey	8
F. Evaluate Phase I results	4
G. Backfill and grade	1
H. Perform Phase II termination survey	8
I. Evaluate Phase II results and request release of facility/site	8

1.1.6 Quality Assurance Plan

A Quality Assurance (QA) Program shall be prepared for the decommissioning operation by the decommissioning contractor and approved by the licensee. The program shall comply with American National Standard (ANS) QA Program requirements for research reactors, N 402-1976 (ANS 15.8).

The Quality Assurance Program shall include the following items:

- Review of health and safety training procedures for operating personnel

- Review of radiation monitoring procedures and instrumentation calibration and maintenance practices

- Review and monitor the decommissioning procedures for adequacy in regard to public health and safety, security, maintenance of as-low-as-reasonably-achievable (ALARA) conditions, choice of methods and equipment, and conformance to all applicable state and federal regulations

- Review and comment regarding proposed changes to, or deviations from, the Decommissioning Plan

- Review of procurement documents for equipment and/or services that affect public health and safety

- Monitoring document control system with regard to work instructions and procedures, drawing and information management, radiation survey results, and field changes

- Review of documents released for NRC review/approval.

1.1.7 Contractor Participation

The Department of the Army plans to have the decommissioning performed by one or more outside contractors. The tasks to be performed by contractors and the detailed performance requirements will be specified in a Statement of Work (SOW), which will become part of the contract. The contents of the SOW are described in Section 3.2.1.

Contractor participation will include the following:

- Providing organization and management to implement the SOW

- Preparing plans, including Emergency, QA, Environmental, Health, Safety, Training, Transportation, Sampling and Analysis, and Waste Management Plans

- Preparing work procedures, as specified in the SOW
- Training appropriate personnel, as required
- Preparing progress reports, cost and schedule reports, deviation and/or field change reports, radiation survey reports, and other reports required by the SOW
- Conducting and supervising day-to-day decommissioning
- Procuring services and equipment
- Administering subcontracts and controlling subcontractors.

1.1.8 Termination Radiation Survey Plan

Following completion of decontamination of the facility and removal of equipment surrounding the facility, as specified in this DP, a termination radiation survey shall be performed, following guidance in NUREG/CR-2082, Monitoring for Compliance with Decommissioning Termination Survey Criteria.[2] Details of the survey plan are provided in Chapter 8 of this DP.

The termination surveys shall include direct radiation measurements, analyses of smears and other samples from inside the reactor building, and soil analyses.

1.2 Facility Operating and Post-Operating History

1.2.1 Reactor Power History and Experiments

The AMTL Reactor was the first nuclear research reactor designed to meet the needs of the research programs on materials for the U.S. Army Ordnance Corps., and was constructed at Watertown, Massachusetts, during the late 1950s and 1960.

Initial criticality of the reactor was achieved on June 15, 1960, at a power level of 1 MW. Post-neutron tests consisting of shim rod calibrations, power calibration, temperature and void coefficients of reactivity measurements, and determinations of the worth of experimental facilities were conducted, culminating on September 16, 1960.

Various solid-state physics research programs and experiments were conducted at the 1-MW power level through June 1966 by the Army Materials and Mechanics Research Center (AMMRC).

A number of local institutions (Boston College, Worcester Polytechnic Institute, University of New Hampshire, and Massachusetts Institute of Technology) also made use of the AMTL Reactor for diffraction measurements and irradiations.

The reactor's license was amended in June 1966 to allow the power level to be increased to 2 MW in order to provide higher neutron fluxes for experiments. The approach to 2 MW began on June 6, 1966, and was completed on June 15. The power was increased in steps of 200 kW and all parameters were observed and measured for several hours at each step.

The reactor's license was updated in 1969 from 2 MW to 5 MW. On August 22, 1969, the power-escalation program began and the reactor power was increased in 1-MW steps to the maximum licensed power level of 5 MW. This program was completed with a 79-hour 5-MW run during the week of September 8, 1969, with no abnormal results observed during the power escalation.

Experiments similar to those described above were planned using the higher power level, as were new experiments for advanced material and for research on, development of, and application of composite materials, improved metal alloys, and ceramics. These experiments were performed on an irregular schedule until the reactor was permanently shutdown in March, 1970.

1.2.2 Leaks in Reactor Systems

During the initial criticality, leaks of reactor coolant water through the concrete biological shield were observed. The leaks grew progressively worse and a first attempt to rectify the problem was to drain the annulus and apply glass tape and epoxy resin to all wall and floor joints. All surfaces of the annulus were also finished with epoxy resins. In the annulus there are cavities created by the encasement of concrete over the slant tubes as they pass through the annulus. The concrete encasements had been provided with aluminum drain lines, which passed from the cavities to the main section of the annulus. The drain lines had been provided so that no water remained in the cavities when the annulus was emptied. When these lines were determined to be leaking, they were plugged, which considerably reduced the leakage in the region where the first balcony is tied to the shield. The efforts to stop the leaks met with limited success.

Late in 1961, a second attempt was made to stop the leakage by drilling selected holes (2 in. diameter) and pressure-grouting a lean cement mix into the holes. This was unsuccessful because very little grout was accepted by the holes. A chemical grout (AM-9) that could be pumped as a liquid with a preset jelling time was then tried in place of the cement mix and proved to be quite successful. This method, along with the use of pressure-sensitive tape around each beam tube joint, eliminated approximately 75% of the leaks. Some small leaks continued intermittently but did not hamper reactor operations.

In 1966 a stainless-steel liner was installed in the reactor pool; this eliminated the leaks. In 1968 the volume between the stainless-steel liner and the concrete shield was connected by a drain to the basement sump in order to remove any water leakage. The water that leaked through the biological shield did not cause any major contamination spread outside the shield but is suspected of contaminating approximately 50% of the high-density concrete. The water that did reach the reactor basement drained to the main sump.

In 1961, during an unloading of the reactor core, technicians experienced difficulty in removing some of the beryllium oxide (BeO) reflector elements.

An examination revealed that an element had swollen. Further examination indicated that water was leaking into the elements. Since there is no reaction between BeO and water under reactor operating conditions, it was decided that perforated BeO reflector elements could be used. There were no further difficulties after modifications were made to reflector elements.

In July 1963, the heat exchanger of the reactor coolant system developed a leak. The heads were pulled and the corroded leaking tube was plugged. Leaks developed on four other occasions and in January 1964 the aluminum tube bundle was removed and replaced. After examination of the corroded tube bundle and the well water, two actions were initiated to remedy this problem: (1) replace the aluminum bundle with one made from stainless steel and (2) install a recirculating-water cooling tower to provide secondary coolant. These actions were completed in January 1965.

1.2.3 Unscheduled Shutdowns

Between January 1, 1969, and March 27, 1970, there were 62 unscheduled shutdowns of the reactor. In many instances, no direct cause was readily apparent, as the shutdown would manifest itself as a single rod dropping without any evidence of malfunction or unsafe condition. The majority of these unexplained rod drops were believed to be caused by noise in the period safety channel, which momentarily reduced the magnet current below the drop current.

1.2.4 Unplanned Release from Cistern 242

An unplanned release of radioactive liquid waste occurred from the underground liquid-waste retention tank (Cistern 242) between February 20 and February 27, 1969. The leak was detected after a review of records of the tank level recorder, which indicated that the level of the tank's contents dropped from 15 to 11 ft during this period. This is equivalent to a total loss of 14,000 gallons and a total activity release of 30 μCi. The Reactor Facility Safety Committee concluded that the release was well below the limit set by 10 CFR 20, Appendix C, for burial of radioactive waste in the soil.

1.2.5 Reactor Operation Summary

Based on the information contained in the operations reports of the U.S. Army Materials Research Agency Nuclear Reactor Facility covering the period from June 15, 1960, through March 27, 1970, and a review of the facility safety reports, there are no indications that any fuel was breached during reactor operations or during fuel transfers between the reactor core and the annulus. Further evidence that no fuel was breached are the low levels of radioactivity and contamination found in the reactor vessel and on the reactor internal components.

1.2.6 Post-Operating History

In December 1969, the Department of the Army decided to shut down the operation of the AMTL Reactor. On March 27, 1970, reactor operations were shut down and the reactor was placed in standby mode. A deactivation report was submitted to the NRC, Division of Reactor Licensing and to the Army Reactor Committee for Health and Safety (ARCHS) in December 1970.

The following radioactive materials were removed from the reactor building and disposed of as follows:

- The irradiated and unirradiated fuel elements containing special nuclear material were removed under contract with National Lead Company and returned to the U.S. Atomic Energy Commission

- The beryllium oxide (BeO) reflector elements, shim-safety rods, armatures, and stainless-steel pieces from the guide tubes were disposed of as high-activity radioactive waste

- The fission chambers containing U-235 were transported to another reactor facility and reported under SNM-244

- The ionization chambers were disposed of as low-level radioactive waste

- The radioactive sources used for calibration and check of survey meters were transferred to the Army Radiation and Occupational Safety Branch, Army Material Command (AMC).

The water from the primary and secondary coolant systems, secondary coolant sump, main reactor pool, fuel storage tank (in basement), and Cistern 242 was drained and disposed of. Indications are that the water was monitored for radioactivity and discharged according to standard procedure, which was to either discharge to the sanitary sewer (if found to be below regulatory standards) or to dilute to achieve acceptable release criteria before discharging.

The following liquid-waste system equipment was removed and disposed of from Building 97:

- Three each 3,000-gal liquid waste storage tanks

- Disposable ion-exchange system

- Pool fill, make-up, and laboratory demineralizer system

- Pumps, valves, and piping associated with the above systems.

The reactor stack, and the secondary coolant towers were removed and disposed of at a later date.

The major equipment that remains to be removed and disposed of during decommissioning of the facility is briefly described in the following paragraph. The removal of these components is described in Chapter 3.

The stainless-steel pool liner, reactor pedestal, grid plate, and portions of the beam tubes, control rods, and instrument rack remain in the reactor vessel. The top covers for the reactor vessel are also in place. The primary coolant system (heat exchangers, demineralizer systems, pumps and associated piping) remain in the reactor facility basement. The secondary

coolant sump, three pumps, and piping to the primary coolant system were also left in place outside the containment shell. Cistern 242 and the associated piping going into Building 97 will also be removed during the decommissioning.

1.3 Current Radiological Status of Facility

The current radiological status of the facility is based on results of radiological characterization and neutron activation analysis performed.

1.3.1 Radiological Characterization

The results of radiological characterization performed during 1989 and 1990 are summarized in this section. Details of the AMTL Reactor characterization, including chemical characterization, are contained in the Characterization Report.[3]

1.3.1.1 Building 100, Reactor Containment Building. Areas surveyed inside the building were as follows: the basement, the operations floor, the first and second platforms, the reactor control room (located on the second platform), the top of the reactor, the reactor vessel, the reactor pedestal, and the reactor annulus.

All radiation measurements were made with portable beta-gamma radiation instruments having detection efficiencies of approximately 10%. The smears that were collected were counted at the Idaho National Engineering Laboratory (INEL) with a decade scaler. These instruments were used for all the areas surveyed.

All smears taken and analyzed for the reactor basement area, except smear number 34, revealed less contamination than was called for in the most restrictive part of NRC Regulatory Guide 1.86, which gives acceptable surface contamination levels. The acceptable levels for removable contamination are: less than 200 dpm/100 cm^2 beta gamma and less than 20 dpm/100 cm^2 alpha. Smear number 34 was collected inside the storage tubes of the storage facility

and read 293 dpm/100 cm^2 beta-gamma and less than 20 dpm/100 cm^2 alpha. Figure 1-9 shows in detail the location and number of each smear taken and contact-radiation readings (circled) in areas where readings could be detected in the basement.

Figures 1-10, 1-11, and 1-12 show the location and number of each smear taken and the contact-radiation readings (circled) in areas where readings could be detected for the operating floor and the first and second platforms, respectively. Smears obtained and analyzed for these areas were all less than 200 dpm/100 cm^2 beta-gamma and less than 20 dpm/100 cm^2 alpha.

Figures 1-13, 1-14, and 1-15 provide the locations, numbers, and results of smears taken in the reactor annulus. Contact-radiation readings are also shown for areas with the highest activities. The results of an isotopic gamma scan, performed on one of the more contaminated smears from the reactor annulus, showed europium-152 (Eu-152), Eu-154, and cobalt-60 (Co-60) in the amounts shown in Table 1-2.

Figures 1-16, 1-17, and 1-18 provide the location of contact-radiation readings taken internally in the reactor vessel. Figures 1-18, 1-19, and 1-20 provide the locations, numbers, and results of smears taken in the reactor vessel. Water samples, believed to be primary water, obtained from the reactor beam tubes contained no measurable activity.

Table 1-3 provides a summary of the smear sampling, and Table 1-4 provides a summary of the radiation survey conducted in the reactor building. From these surveys it appears that only the reactor annulus and some of the reactor components are contaminated. Transuranic isotopes were not detected on any of the smears but, as shown in Table 1-2, Eu-152, Eu-154, and Co-60 isotopes were detected. The highest radiation readings were measured on components contained within the reactor annulus and vessel.

1.3.1.2 Building 97, Reactor Facility Laboratory. Building 97 originally contained the liquid-waste handling system for the reactor. This system was removed to make room for a particle accelerator, as discussed in

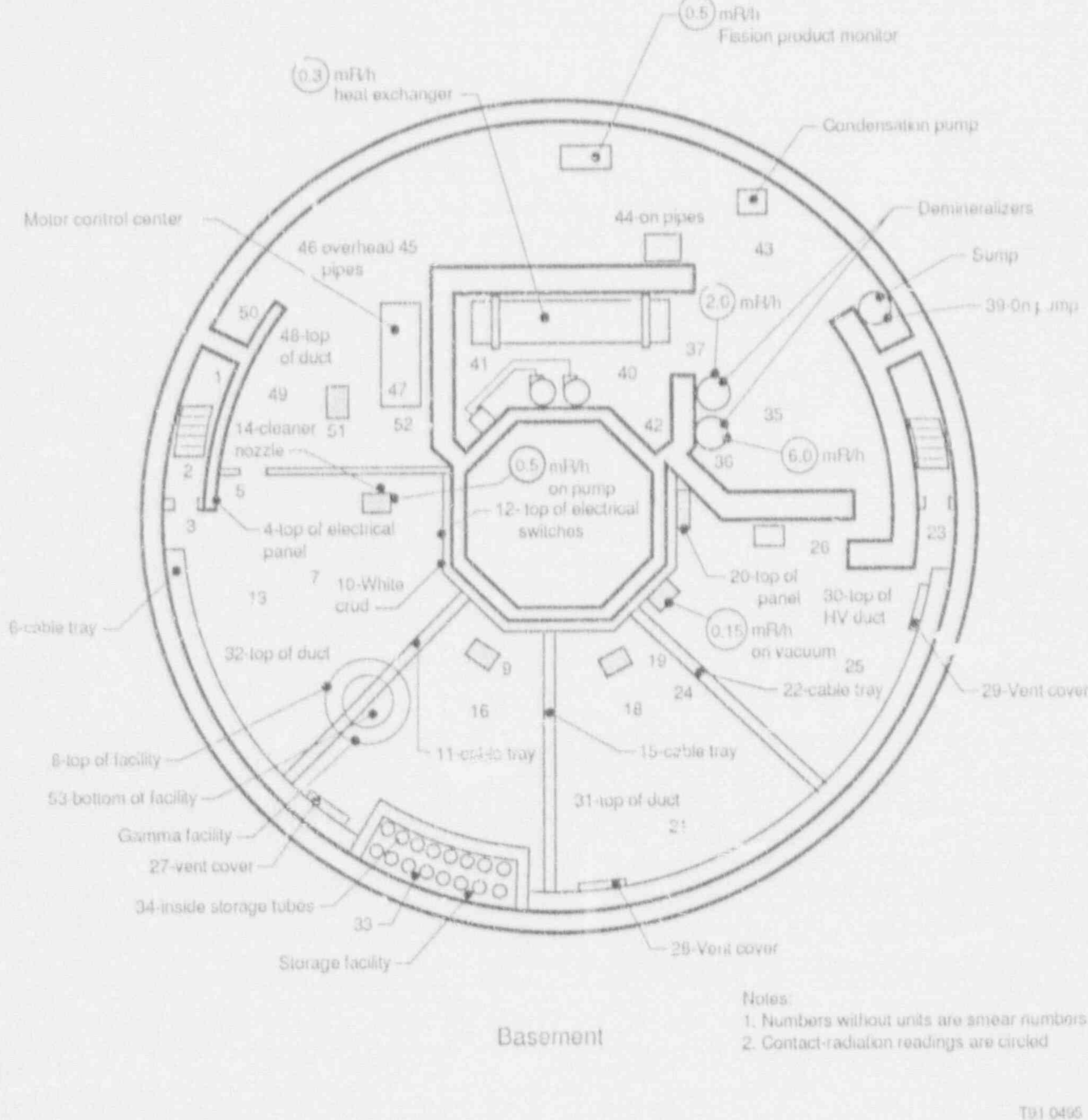


Figure 1-9. Smear survey locations for reactor basement.

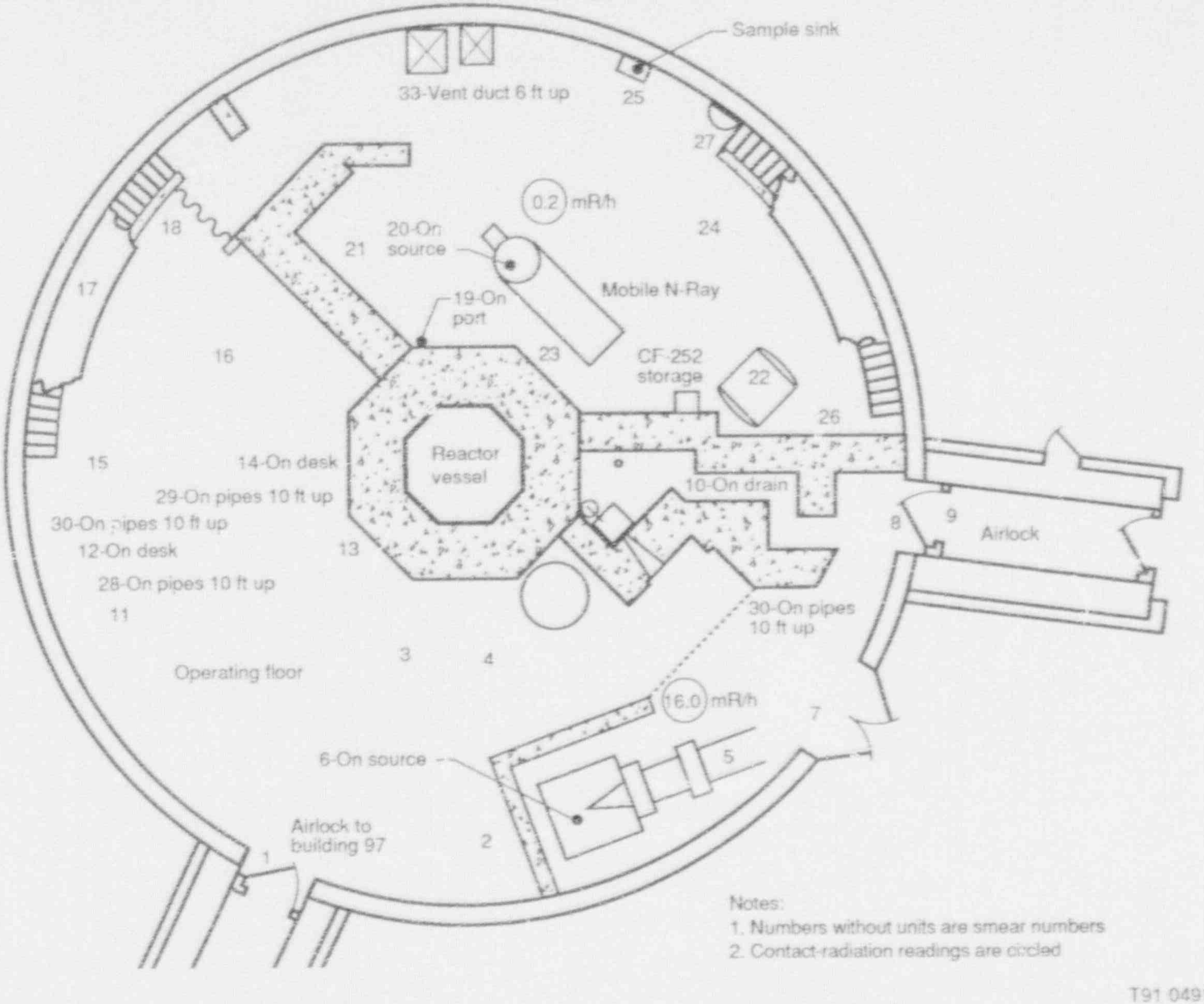


Figure 1-10. Smear survey locations for reactor operating floor.

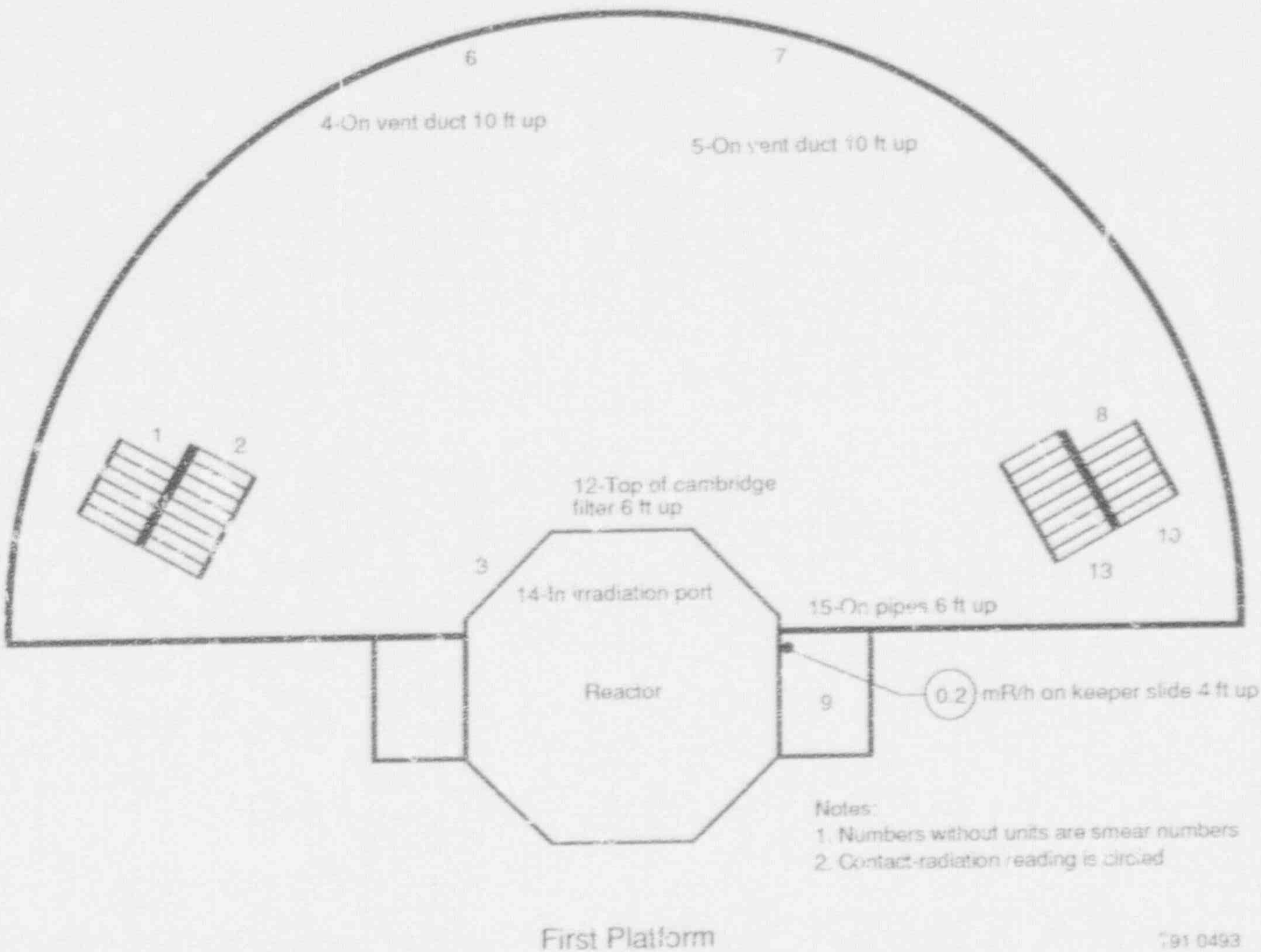


Figure 1-11. Smear survey locations for reactor first platform.

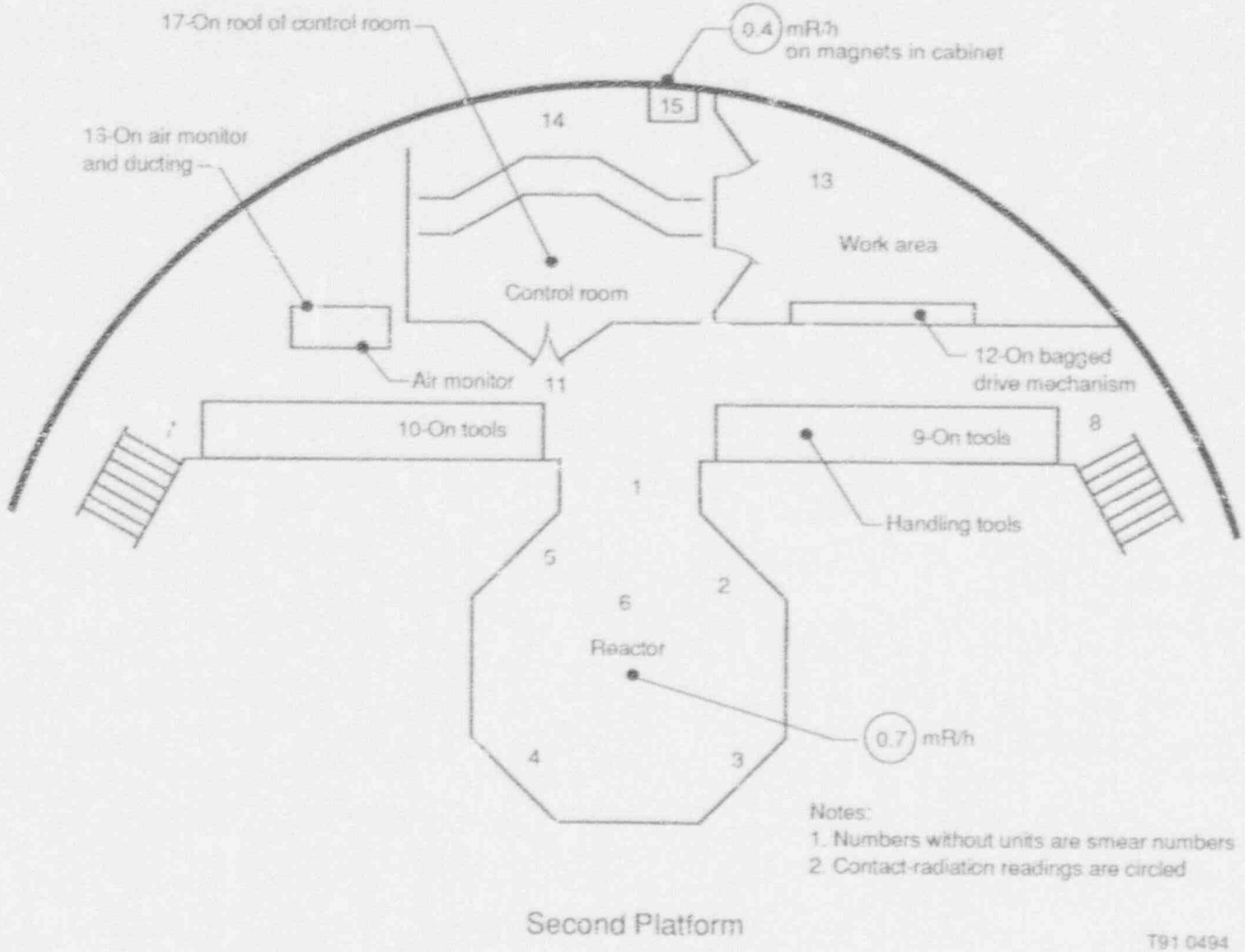


Figure 1-12. Smear survey locations for reactor second platform.

Smear No.	Beta-Gamma (dpm/100 cm²)	Alpha (dpm/100 cm²)
1	200	<20
2	359	<20
3	335	<20
4	293	<20
5	395	<20

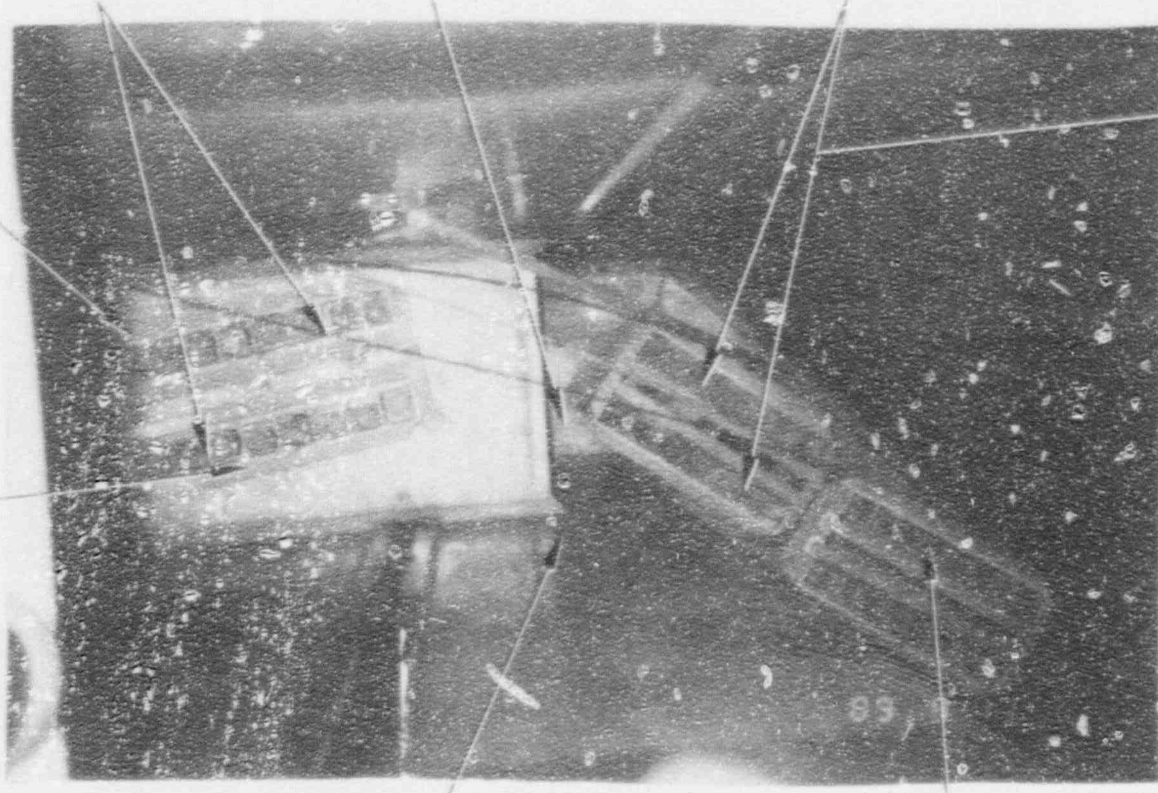


Figure 1-13. Smear survey results and contact-radiation readings for reactor annulus fuel element storage.

Smear No.	Beta-Gamma (dpm/100 cm²)	Alpha (dpm/100 cm²)
6	Lost	Lost
7	749	<20
8	5707	<20
9	755	<20

Figure 1-14. Survey results and contact-radiation readings for reactor annulus components.

Smear No.	Beta-Gamma (dpm/100 cm²)	Alpha (dpm/100 cm²)
10	725	<20
11	240	<20

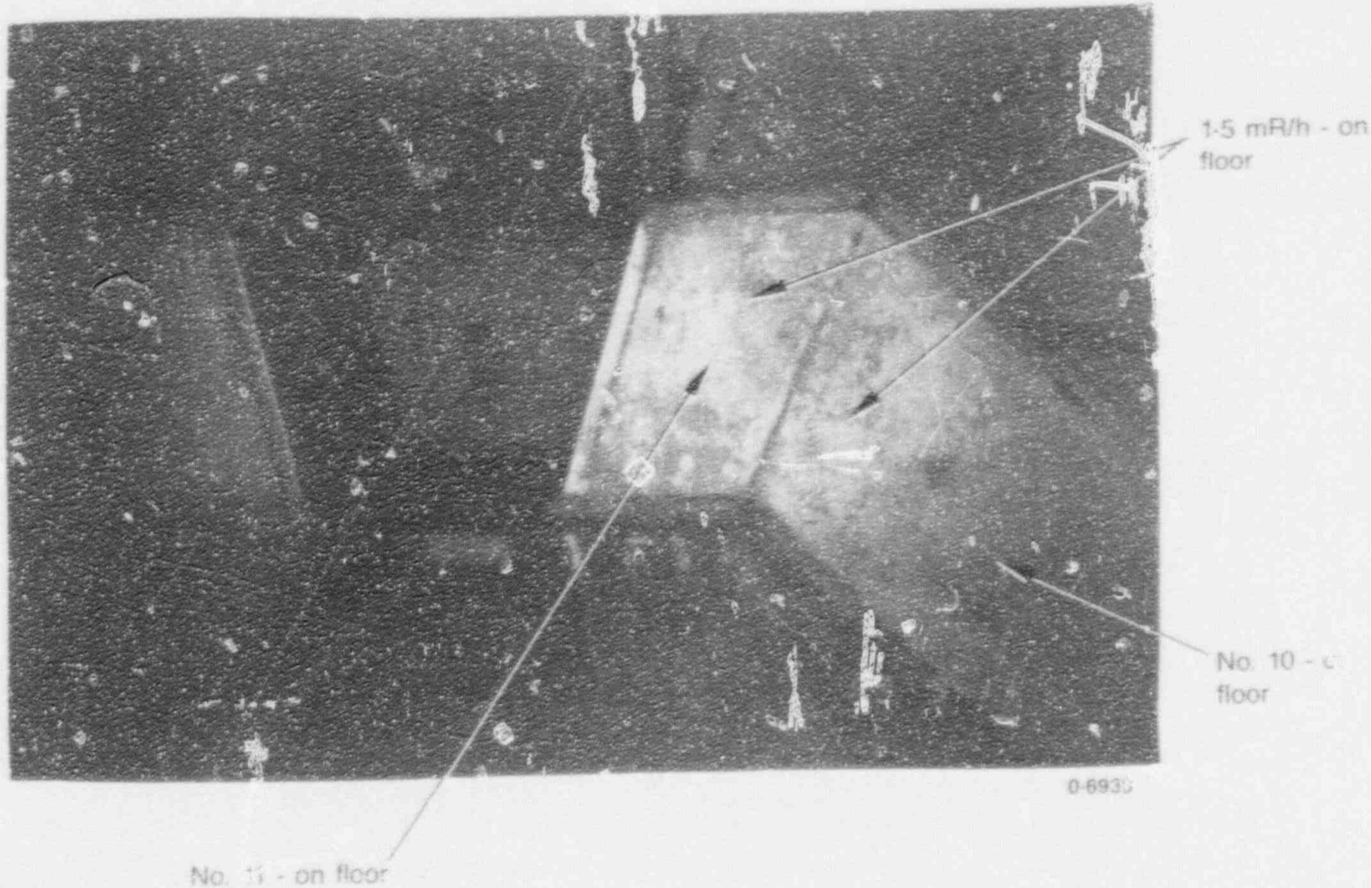


Figure 1-15. Smear survey results and contact radiation readings for reactor annulus floor.

Table 1-2. Results of isotopic gamma scan for annulus smear sample.

Radionuclide	Sample Activity (μCi/100 cm^2)	% Error (1-Σ)[a]
Co 60	1.016E-03	2.52
EU-152	4.848E-04	3.76
EU-154	8.500E-05	22.56

a. The percent error shown for each radionuclide is the statistical/fitting component obtained from weighted results of all photopeaks found for the nuclide. Other error components that should be considered in the total error are detector efficiency (5.0%), geometry/positioning (5.0%), and any other errors that can be quantified (0.0%).

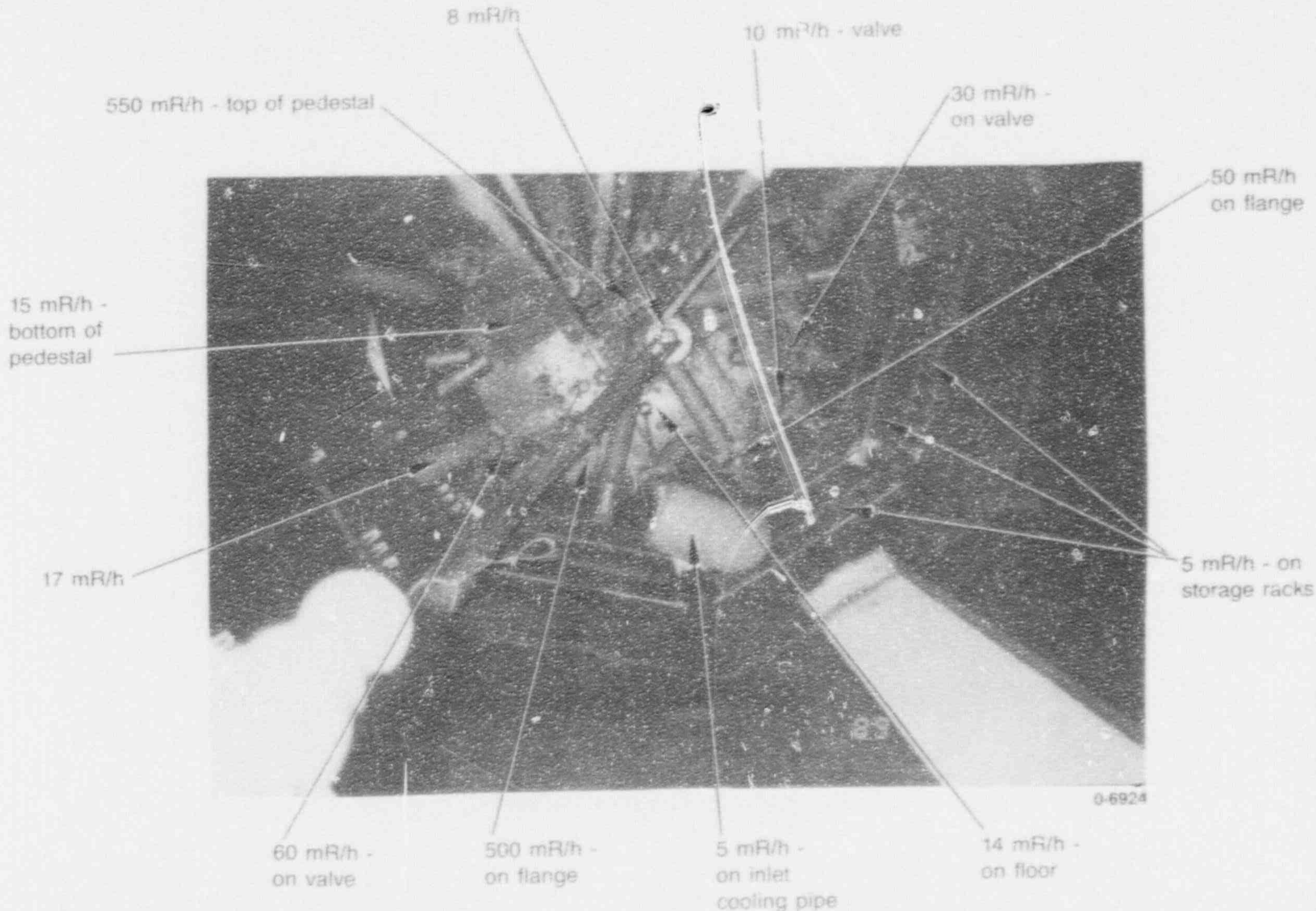


Figure 1-16. Contact-radiation readings for internal components of reactor vessel (View No. 1).

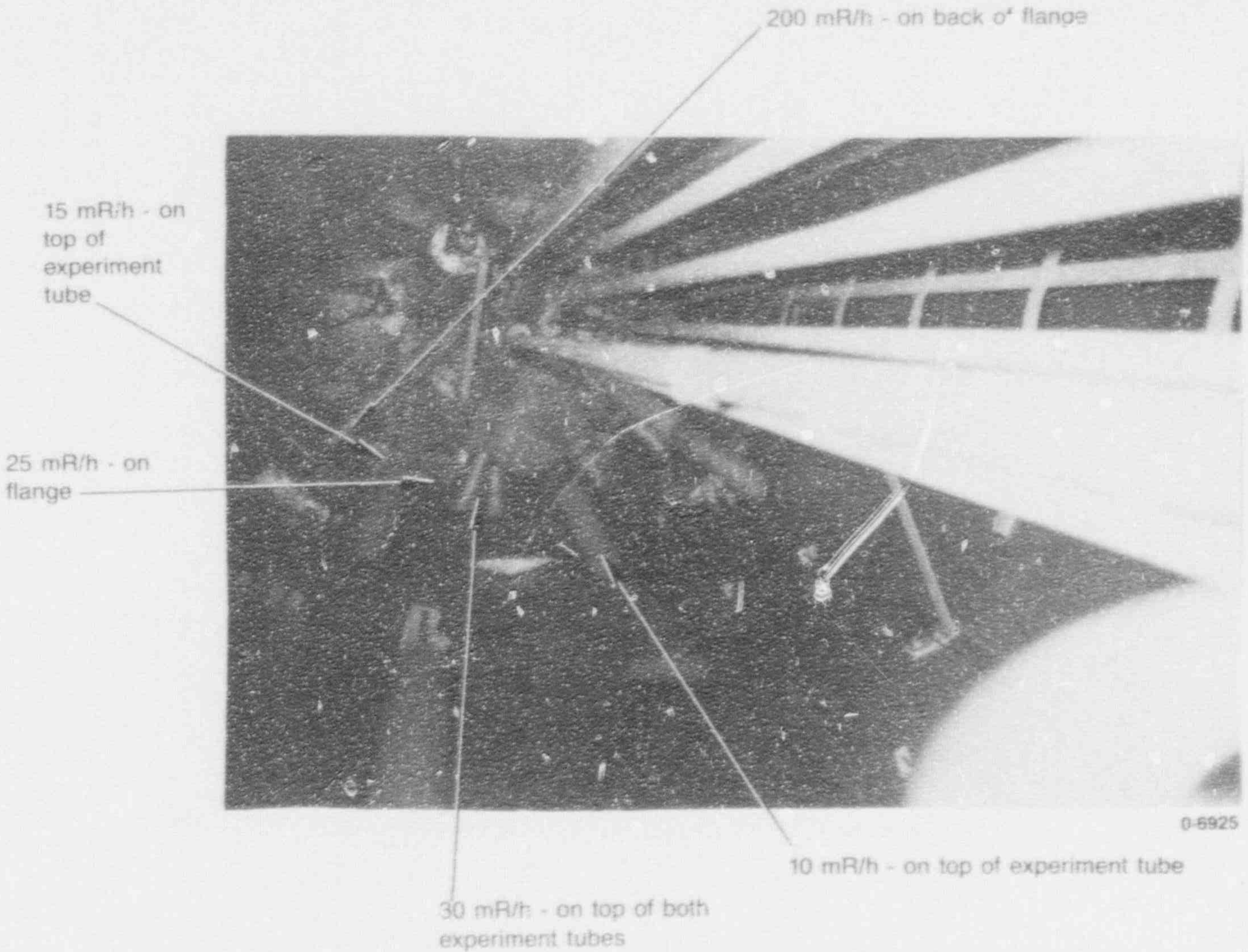


Figure 1-17. Contact-radiation readings for internal components of reactor vessel (View No. 2).

Smear No.	Beta-Gamma (dpm/100 cm^2)	Alpha (dpm/100 cm^2)
1	<200	<20
2	<200	<20
12	<200	<20

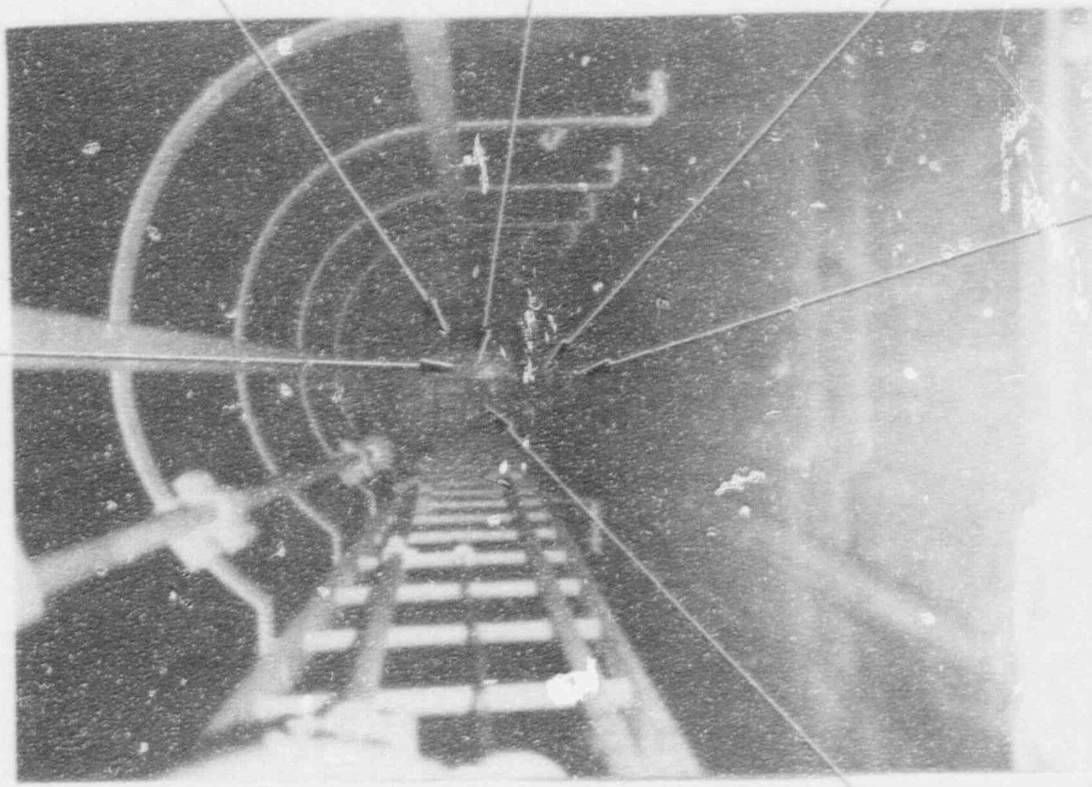


Figure 1-18. Contact-radiation readings and smear survey results for internal components of reactor vessel (View No. 3).

Smear No.	Beta Gamma (dpm/100 cm²)	Alpha (dpm/100 cm²)
3	204	<20
4	<200	<20
5	<200	<20
6	<200	<20
7	<200	<20

Figure 1-19. Smear survey results for internal components of reactor vessel (View No. 1).

Smear No.	Beta-Gamma (dpm/100 cm²)	Alpha (dpm/100 cm²)
8	<200	<20
9	<200	<20
10	<200	<20
11	<200	<20
13	<200	<20

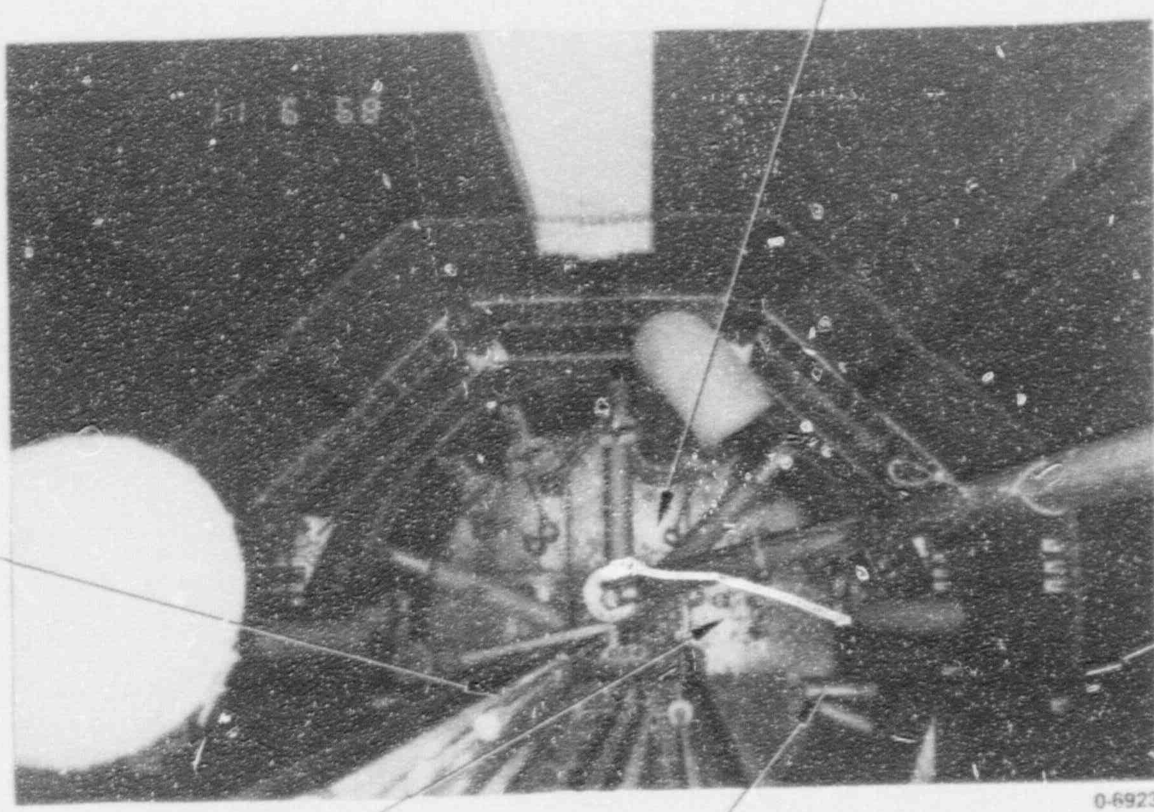


Figure 1-20. Smear survey results for internal components of reactor vessel (View No. 2).

Table 1-3. Summary of the results of the AMTL Building 100 smear sampling

Location	Beta-Gamma (dpm/100 cm^2)	Alpha (dpm/100 cm^2)
<u>Basement</u>		
Inside tubes of the storage facility	293	<20
All other basement smears	293	<20
<u>Main Floor</u>		
All main-floor smears	<200	<20
<u>First Platform</u>		
All first-platform smears	<200	<20
<u>Second Platform</u>		
All second-platform smears	<200	<20
<u>Reactor Vessel Internals</u>		
Floor by the access ladder	204	<20
All other reactor-vessel internal smears	<200	<20
<u>Reactor Annulus</u>		
Stainless-steel racks	293-295	<20
Annulus floor	200-725	<20
Stainless-steel piping below reactor gate	749-5707	<20

Table 1-4. Summary of the results of the AMTL Building 100 radiation survey.

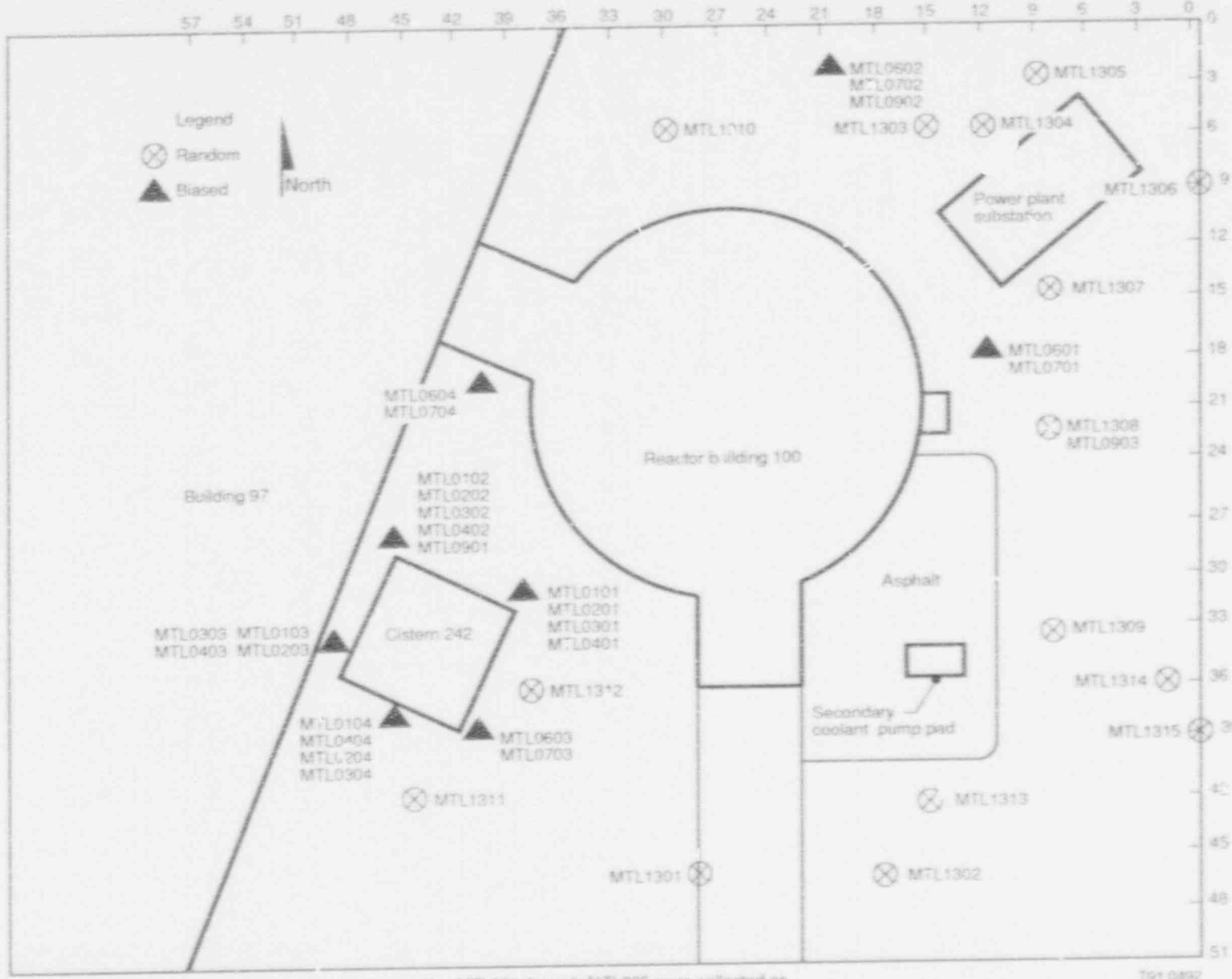
Location	Contact-Radiation Reading (mR/h)
Basement	
Demineralizer	2.0-6.0
Heat exchangers	0.3
Fission-product monitor	0.05
Main Floor	
Californium-252 source	16.0
Mobile N-Ray	0.2
First Platform	
Reactor keeper slide	0.2
Second Platform	
Reactor top	0.7
Magnets in cabinet	0.4
Reactor Vessel Internals	
Blind flanges	50.0-550.0
Slant tubes	8.0-30.0
Valves	10.0-60.0
Pedestal (top)	550.0
Pedestal (bottom)	15.0
Reactor Annulus	
Stainless-steel racks and stainless-steel pipe	55.0
Below reactor gate	1,300.0
General body field by stainless-steel pipe at 3 ft	18.0

Section 3.2. All that presently remains is the piping (4-in. iron coolant transfer line, 2-1/2-in. iron sump pump drain line, 2-in. aluminum demineralized water line, and 1-in. iron test connection line) connecting Buildings 97 and 100. Smears taken from the piping inside Building 97 indicate that the radioactivity levels are less than 200 dpm/100 cm^2 beta-gamma and less than 20 dpm/100 cm^2 alpha. The piping (2-1/2-in. iron outlet line, 4-in. iron inlet line, and 4-in. cast iron overflow line) to Cistern 242 is mostly inaccessible since it is underground and will be surveyed during the decommissioning of the facility. The inlet and outlet pipes are buried 5-1/2 ft deep and the overflow pipe is buried 3-1/2 ft deep. The overflow pipe originally went from a catch basin in Building 97 into the Cistern.

1.3.1.3 Cistern 242. Cistern 242 was filled with water during the radiological surveys conducted at AMTL during September 1989 and was not surveyed at that time. Smears and sediment samples were also postponed until the tank is pumped. The water in the cistern was sampled during February 1990 and found to be well below the radioactivity levels of drinking-water standards. There was no measurable gamma activity, other than the natural potassium-40 and radon/theron daughters from the natural uranium and thorium decay chains. The gross alpha and beta activity levels for the sample in $\mu Ci/mL$ were $(2 \pm 5) \times 10^{-10}$ and $(1.3 \pm 0.17) \times 10^{-8}$, respectively.

Samples were not collected from the interior of Cistern 242 when characterization was performed during the week of March 26, 1990, because the water had not been removed. Based on the above results of the analysis of the water sample taken in February 1990 and on the low levels of contamination found in the reactor vessel, it is assumed that the cistern does not contain any substantial amount of radioactively contaminated materials in the form of sediment or sludge.

1.3.1.4 Soil from Around Building 100. Soil samples including background samples were collected in March 1990 at locations shown in Figure 1-21. These samples were analyzed for gamma-emitting nuclides by gamma spectroscopy. Table 1-5 summarizes the results from data that were found to



Note: In addition to the sample locations shown, samples MTL801 through MTL805 were collected as background samples. The background samples were collected 25 ft south of the fence surrounding the residence of the Base Commander.

Figure 1-21. Building 100 and Cistern 242 area sediment and soil sample locations.

Table 1-5. Analytical results of manmade true positive gamma-emitting nuclides for March 1990 sediment and soil samples. (See Figure 1-21 for location of each sample).

Sample ID No.	Depth (ft)	Radionuclide	Activity Concentration (S)[a] (pCi/g)	Uncertainties (%) Statistical	Geo	Eff	Activity Concentration (T)[b] (pCi/g)
MTL0103	2-3	Co-60	(+6.32 +/- .30)E-01	4.8	5.0	5.0	(+6.32 +/- .54)E-01
		Cs-137	(+1.89 +/- .22)E-01	11.5	5.0	5.0	(+1.89 +/- .26)E-01
MTL0104	2-3	Cs-137	(+3.18 +/- .34)E-01	10.6	5.0	5.0	(+3.18 +/- .40)E-01
MTL0203	5-6	Co-60	(+2.45 +/- .29)E-01	11.6	5.0	5.0	(+2.45 +/- .33)E-01
MTL0204	5-6	Cs-137	(+1.62 +/- .19)E-01	11.6	5.0	5.0	(+1.62 +/- .22)E-01
MTL0304	9-10	Cs-137	(+1.72 +/- .28)E-01	16.2	5.0	5.0	(+1.72 +/- .30)E-01
MTL0601	Surface	Cs-137	(+2.12 +/- .31)E-01	14.6	5.0	5.0	(+2.12 +/- .34)E-01
MTL0602	Surface	Cs-137	(+1.83 +/- .31)E-01	16.8	5.0	5.0	(+1.83 +/- .33)E-01
MTL0603	Surface	Cs-137	(+4.89 +/- .42)E-01	8.5	5.0	5.0	(+4.89 +/- .54)E-01
MTL0604	Surface	Cs-137	(+2.83 +/- .34)E-01	12.0	5.0	5.0	(+2.83 +/- .40)E-01
MTL0801[c]	Surface	Cs-137	(+2.50 +/- .34)E-01	13.5	5.0	5.0	(+2.50 +/- .38)E-01
MTL0802[c]	Surface	Cs-137	(+3.79 +/- .46)E-01	12.0	5.0	5.0	(+3.79 +/- .53)E-01
MTL0803[c]	Surface	Cs-137	(+4.07 +/- .60)E-01	14.6	5.0	5.0	(+4.07 +/- .66)E-01
MTL0804[c]	Surface	Cs-137	(+1.91 +/- .41)E-01	24.5	5.0	5.0	(+1.91 +/- .43)E-01
MTL0805[c]	Surface	Cs-137	(+3.24 +/- .42)E-01	13.0	5.0	5.0	(+3.24 +/- .48)E-01

Table 1-5. (continued)

Sample ID No.	Depth (ft)	Radionuclide	Activity Concentration (S)[a] (pCi/g)	Uncertainties (%) Statistical	Geo	Eff	Activity Concentration (T)[b] (pCi/g)
MTL0903	Surface	Cs-137	(+1.81 +/- .28)E-01	15.4	5.0	5.0	(+1.81 +/- .31)E-01
MTL1301	Surface	Cs-137	(+1.21 +/- .37)E-01	30.8	5.0	5.0	(+1.21 +/- .38)E-01
MTL1302	Surface	Cs-137	(+1.26 +/- .30)E-01	23.8	5.0	5.0	(+1.26 +/- .31)E-01
MTL1303	Surface	Cs-137	(+2.63 +/- .40)E-01	15.1	5.0	5.0	(+2.63 +/- .44)E-01
MTL1304	Surface	Cs-137	(+4.89 +/- .53)E-01	10.8	5.0	5.0	(+4.89 +/- .63)E-01
MTL1305	Surface	Cs-137	(+2.07 +/- .30)E-01	14.6	5.0	5.0	(+2.01 +/- .34)E-01
MTL1306	Surface	Cs-137	(+2.19 +/- .29)E-01	13.2	5.0	5.0	(+2.19 +/- .33)E-01
MTL1308	Surface	Cs-137	(+2.41 +/- .33)E-01	13.5	5.0	5.0	(+2.41 +/- .37)E-01
MTL1310	Surface	Cs-137	(+4.83 +/- .39)E-01	8.0	5.0	5.0	(+4.83 +/- .52)E-01
MTL1311	Surface	Cs-137	(+4.12 +/- .39)E-01	9.4	5.0	5.0	(+4.12 +/- .49)E-01
MTL1312	Surface	Cs-137	(+3.64 +/- .38)E-01	10.5	5.0	5.0	(+3.64 +/- .46)E-01
MTL1313	Surface	Cs-137	(+2.10 +/- .32)E-01	15.3	5.0	5.0	(+2.10 +/- .35)E-01
MTL1314	Surface	Cs-137	(+2.03 +/- .27)E-01	13.4	5.0	5.0	(+2.03 +/- .31)E-01
MTL1315	Surface	Cs-137	(+1.17 +/- .39)E-01	33.1	5.0	5.0	(+1.17 +/- .40)E-01

a. Activity Concentration (S) includes the statistical uncertainty, from counting statistics and photopeak fitting--expressed as 1 standard deviation.

b. Activity Concentration (T) includes the total uncertainty resulting from the statistical, sample/detector geometry, and efficiency. These uncertainties have been propagated in quadrature--expressed as 1 standard deviation.

c. Samples MTL 0801 through MTL 0805 were background samples collected 25 feet south of the fence surrounding the residence of the Base Commander.

give true positive. True positive results are defined as values that have a measured activity >2 measured standard deviations.

Table 1-5 includes the activity concentration with the associated statistical uncertainty, Activity Concentration (S), and the activity concentration with total uncertainty, Activity Concentration (T). The statistical uncertainty includes the statistics associated with counting, backgrounds, and photopeak fitting. The total uncertainty includes statistical uncertainty, estimates of the uncertainty in the sample geometry (5%), and detector efficiency (5%). These uncertainties have been propagated in quadrature and are expressed as one estimated standard deviation. It is recommended that the activity concentration in the right hand column of Table 1-5, Activity Concentration (T), be used.

The two samples containing positive indications of Co-60 (MTL0103 and MTL0203) were taken from the area between Building 97 and Cistern 242, where elevated radiation readings were detected during the preliminary radiological surveys performed in September 1989. The highest concentration of Cs-137 and Co-60 detected in the samples collected from around the reactor building are 4.8 pCi/g and 6.3 pCi/g respectively.

The NRC has no published release criteria for radioisotopes in soil. The NRC determines whether or not a site can be released after an NRC site-specific assessment on a case-by-case basis. For comparison, the Department of Energy has published release criteria for Idaho National Engineering Laboratory (INEL) soils having radioisotopic concentrations for Cs-137 and Co-60 in releasable soil of 10 pCi/g and 4 pCi/g respectively. The DOE criteria are based on extensive pathways analysis. While the characteristics of the two sites are different, it is assumed through comparison that the AMTL soil concentrations have a high likelihood of being approved for release. However, the NRC will have final approval as to whether or not the site can be released for unrestricted use.

The soil samples were also analyzed for alpha-emitters; sample number MTL 0102 was the only sample that contained statistically positive Am-241 and Pu-238. The activity concentration of this sample is 0.21 pCi/g. As stated previously, the NRC has no published release criteria for soil. For

comparison, however, the DOE INEL soil release criteria allow soil to be released with a Pu-238 concentration of 300 pCi/g and an Am-241 concentration of 80 pCi/g. It, therefore, appears that the AMTL soil has a very high probability of meeting NRC requirements for alpha-emitters, because the AMTL alpha-emitting radioisotopic concentration is orders of magnitude lower than the DOE criteria, which are based on very extensive pathways analysis.

1.3.2. Neutron Activation Analysis

1.3.2.1 Introduction. During the course of operating a nuclear reactor, reactor construction materials can become activated through transformations caused by materials absorbing neutrons and possibly further transforming through various radioactive decay schemes. In order to determine the amount of activation products present in reactor construction materials, it is necessary to know the neutron energies and corresponding flux intensity to which the materials were subjected, the exact nature of the materials present, and the material exposure times. A method of verifying the activation product theoretical calculations is to obtain samples of reactor construction materials and analyze them for the concentration of activation products that are present.

Due to the AMTL Reactor being nonfunctional for the past 20 years, some operational records have been difficult to locate. To support the decommissioning planning effort, the available data were analyzed and assumptions made to facilitate an estimate of the gamma emitting activation products present in the remaining reactor construction materials. The only verification available is the radiological characterization discussed in Section 1.3.1.

Rough order of magnitude estimates of the gamma emitting activation products present in the reactor support structure and biological shield were conducted to support the estimates of wastes to be generated and worker radiation exposures. Assumptions were made in several key parameters in the absence of reliable data so that the estimates could be quantified. In the absence of reliable records of plant construction materials and operating history data, assumptions were made regarding the actual materials used in the reactor and biological shield, as well as the value of the flux to which the

materials were exposed. Samples were not taken of reactor materials to support the activation calculations. These samples would have been of assistance in calculations involving the support structure and the pool liner.

The number of assumptions that were necessary to allow any estimate of activation products, resulted in rough order of magnitude estimates that can be used for planning purposes only. Actual measurements of radiation fields in the reactor pool and annulus were used to form the basis for the worker exposure estimates. The presence of the unknown amounts of contamination in the biological shield as a result of leakage from the pool makes the use of activation for estimating the shield waste volumes of little value. Consequently, the waste volume estimates of concrete contained in Section 1.5 are based on assumptions of the extent of pool leakage contamination.

1.3.2.2 Materials. The activation calculations that were performed were based on descriptions of plant conditions contained in plant drawings from about 1965. Not all materials were identified on the available drawings. When materials were not specified, they were assumed to be of a particular type to facilitate the estimate of activation. Based on an interpretation of the plant drawings it was estimated that there is 142,800 cm^3 of stainless steel liner, 4,540 cm^3 of carbon steel piping in the shield wall, 182,000 cm^3 of carbon steel rebar, 1,020 cm^3 stainless steel in slant tubes, 12,600 cm^3 stainless steel in horizontal beam tubes, 18,480 cm^3 stainless steel in through tubes, and 9,439 cm^3 of stainless steel bolts. Stainless steel materials were all assumed to be Type 316 with a density of 7.8 g/cm^3 and composed of 11% nickel; 0.1% carbon; 2.5% molybdenum; and the balance iron. The carbon steels were assumed to be AISI-SAE-1945 with a density of 7.9 g/cm^3 and composed of 0.45% carbon, 0.75% manganese; and the balance iron. The total amounts of materials present for activation are 1.44 Mg of stainless steel and 1.47 Mg of carbon steel.

The activation of aluminum was neglected in the estimate as all activation products that result from aluminum have decayed in the 20 years that have elapsed since the reactor was shut down. The field survey data support this assumption.

1.3.2.3 **Neutron Flux.** Data documenting the actual radial neutron flux within the reactor pool and shield were also lacking. To allow the activation estimate to proceed, a radial core flux profile was assumed and core external flux values estimated based on provided values of average neutron fluxes. In the absence of the actual power history and fuel loadings, the activation of materials was conservatively assumed to be at saturation at the end of the operations.

An average value of fast neutron flux of 2×10^{11} n/cm^2-sec was utilized in the activation calculations. This value was based on 2 MW operation and estimated radial neutron flux distribution values and pool shielding between the core and the liner.

1.3.2.4 **Activated Isotopes.** An analysis of the gamma emitting nuclides from neutron activation that result from the assumed construction materials is summarized in Table 1-6. Examining the number of half-lives elapsed in 20 years indicates that the only isotopes that are present in any significant amounts after 20 years are Co-60 and Mn-54. The cobalt is present due to production from an n,p reaction with Ni-60. The Manganese is produced by an n,p reaction with Fe-54. It should be noted that although the Mn-54 half-life is relatively short, the intermediate precursor, Fe-55, has a 2.73 year half life. This makes the Mn-54 contribution to the residual activity more important than its half-life alone would indicate.

These isotopes emit gamma radiation with energies of 0.835 MeV(100%) for Mn-54 and 1.17 (100%) and 1.33 (100%) MeV for Co-60.

In addition to the gamma emitting radionuclides that are products of neutron activation, there are beta emitting radionuclides to consider. These radionuclides, such as C-14, Nb-94, and Ni-63 are not important from the standpoint of personnel exposure considerations, but may be important for the classification and characterization of the waste generated during decommissioning. No attempt was made to calculate the beta emitting radionuclides because of the uncertainties previously mentioned, and because these activation components will be assessed during decommissioning activities to accurately characterize their identity and quantities.

Table 1-6. Principal gamma emitting nuclides from neutron activation

Target Element	Activation Product	Half-Life	No. of Half-Lives (20 years)
Nickel	Ni-65	2.5 h	70,880
	Co-58m	9.1 h	19,250
	Co-58	70.9 h	103
	Co-60m	10.5 m	1,000,000
	Co-60	5.3 y	3.8
	Co-61	1.6 h	109,500
	Co-62	13.9 m	756,000
	Fe-59	44.5 d	164
Manganese	Mn-56	2.6 h	67,400
Iron	Fe-59	44.5 d	164
	Mn-54	312.2 d	23.4
	Mn-56	2.6 h	67,400

By utilizing the equation above, the quantities of nickel and iron atoms were estimated, and are shown in Table 1-7.

Table 1-7. Atoms present for activation

	Subject Material	
Isotope	Stainless Steel	Carbon Steel
Ni-60	4.14×10^{26}	N/A
Fe-54	6.37×10^{26}	9.12×10^{26}

1.3.2.5 Calculations. The number of target atoms for each isotope of concern was estimated using the equation

$$N_0 = (f)\ (g)\ (a)\ (A)\ (aw)$$

where

N_0	=	the number of target atoms present at time (0)
f	=	the fractional amount of element present
g	=	the total gram quantity of the parent material
a	=	the atom percent of the target radionuclide
A	=	Avogadro's number
aw	=	gram atomic weight of the element of concern

Given the number of target atoms and estimating the fast neutron flux present for activation, the activity of the resulting radionuclide can be determined by using the following equation.

$$A=(\sigma)\ (\phi)\ (N_0)\ (e^{-\lambda t})$$

where

A	=	Curie amount of the product radionuclide
σ	=	reaction cross-section
ϕ	=	fast neutron flux
N_0		number of target atoms present at time (0)
$e^{-\lambda t}$	=	decay factor, t equals 20 yrs

Performing this calculation for the two isotopes of concern resulted in an estimate of 23 Ci of Co-60 and 0.26 mCi of Mn-54.

These activities would result in stainless steel having an approximate average specific activity of 33 µCi/g of Co-60 and 76 pCi/g of Mn-54 and carbon steel with 102 pCi/g of Mn-54.

1.3.2.6 Conclusion. It should be noted that many assumptions were made to arrive at virtually all of the key input factors to the activation calculations. This has resulted in the values listed above being rough estimates. The values presented here are felt to be conservatively higher than are expected to be encountered during decommissioning. Samples of the wastes will be analyzed as they are generated for the purpose of waste classification.

1.4 Estimated Personnel Dose

The estimated dose to workers during decommissioning activities is calculated from estimated task duration, estimated crew size, and estimated average radiation field. The estimated task duration is the duration of the task during which workers would be exposed to radiological hazards. The estimated crew size only includes personnel working in the estimated field. The estimated average radiation field is based on measured radiation fields.

The estimated radiation dose for each task required to perform decommissioning of the AMTL Reactor is given in Table 1-8. The sum of the estimated exposure for all the tasks is 10.0 man-rems as shown in Table 1-8.

The actual radiation dose to personnel, however, will be kept to levels ALARA by utilizing engineering controls during decommissioning activities.

1.5 Estimated Volume of Radioactive Material to be Removed

The estimated volume of radioactive waste generated during decommissioning of the AMTL Reactor is summarized in this section. The waste volume estimate is based on the following assumptions:

- The secondary coolant sump and associated piping is not contaminated.
- The waste holdup tank beneath the reactor is empty but contaminated.
- The concrete shield is 50% radioactive waste due to contamination from pool leakage. (The activation of the concrete is insignificant compared to the assumed contamination due to pool leakage).
- Cistern 242 is uncontaminated.

Table 1-8. Estimated Radiation Dose for Decommissioning of the AMTL Reactor

Task Description	Task Duration (hr)	Estimated Crew Size	Estimated Average Radiation Field (mrem/hr)	Man-rems
Remove components from reactor annulus	2	3	18	0.1
Remove reactor pedestal and associated hardware	35	4	10	1.4
Remove reactor vessel stainless-steel liner	20	4	5	0.4
Remove heat exchangers, piping, and demineralizers	34	6	2	0.4
Remove reactor inner concrete shield	84	10	4	3.4
Remove reactor high density concrete shield	156	10	2	3.1
Remove reactor outer concrete shield	32	10	1	0.3
Remove reactor vessel base in basement	88	10	1	0.9
			Total man-rems	10.0

In the absence of concrete coring data, the amount of concrete shield contamination can only be estimated because of unknown amount of contamination as a result of pool leakage. The concrete shield material will be surveyed during removal of the concrete shield. Only material meeting NRC release criteria will be disposed of as uncontaminated waste. All other waste will be disposed of as radioactive waste after it is analyzed to identify radioisotopes and their specific activity.

Activated material quantities were estimated as explained in Section 1.3.2. Analyses of radioactive waste will accurately characterize the waste to identify gamma emitting and beta emitting radionuclides that are products of neutron activation. The estimated contamination waste volume, activity, and principal radionuclides expected are shown in Table 1-9.

The potentially contaminated lead listed in Table 1-9 is from the lining of the floor storage pits, boxes in the annulus, and sleeves around the outer portion of the beam tubes. This potentially contaminated lead will be removed and surveyed. If contaminated with fixed contamination, it will be packaged and given to the licensee for storage as mixed waste. If it is contaminated with removable contamination it will be decontaminated and recycled. If the lead is uncontaminated, it will be recycled.

Table 1-9. Estimated radioactive waste

Material	Estimated Volume (ft^3)	Activity	Principal Radionuclide(s)[a]
Stainless steel	130	23 Ci	Co-60
Steel	200	0.15 milli Ci	Mn-54
Lead[b]	51	unknown	unknown
Concrete	6,750	unknown	Cs-137, Sr-90, Co-60

a. Beta emitting radionuclides produced by neutron activation are not included. These activation components will be identified by analysis of waste during decommissioning.

b. If the lead is contaminated with fixed contamination, it will be packaged and given to the licensee for storage as mixed waste.

1.6 Decommissioning Alternative

The proposed decommissioning alternative, decontamination, will achieve the decommissioning objective of releasing the site for unrestricted use and termination of the license.

1.7 Decommissioning Organization and Responsibilities

The decommissioning operations of the AMTL Reactor facility will be performed by a qualified contractor under a contract awarded and administered by the U.S. Army. A qualified decommissioning contractor shall have decommissioning experience with a facility similar to the AMTL Reactor facility. Decommissioning requirements are generally specified in this Decommissioning Plan. Specific requirements for the decommissioning contractor, including safety requirements, will be specified in a Statement of Work to be finalized after approval of this DP.

The Statement of Work (SOW) will include appropriate parts of this DP and will become the major specification for the AMTL Reactor decommissioning operations. The appropriate parts of the DP to be included in the SOW are all parts constituting decommissioning requirements or information useful to prospective contractors during preparation of their proposals. However, some information, such as estimated cost, would not be included in the SOW.

The AMTL Reactor decommissioning project organization relative to safety is shown in Figure 1-22. A brief description of the key positions in Figure 1-22 and the responsibility of each are given below.

As the licensee, the MTL Commander is responsible for the overall decommissioning project and has authority in all associated matters, including project safety. His representative for the decommissioning project, including safety, is the Project Technical Monitor, an MTL employee. He has the responsibility and authority to monitor the decommissioning of the reactor. The MTL Technical Monitor will have direct access to the MTL Commander in order to recommend work be stopped.

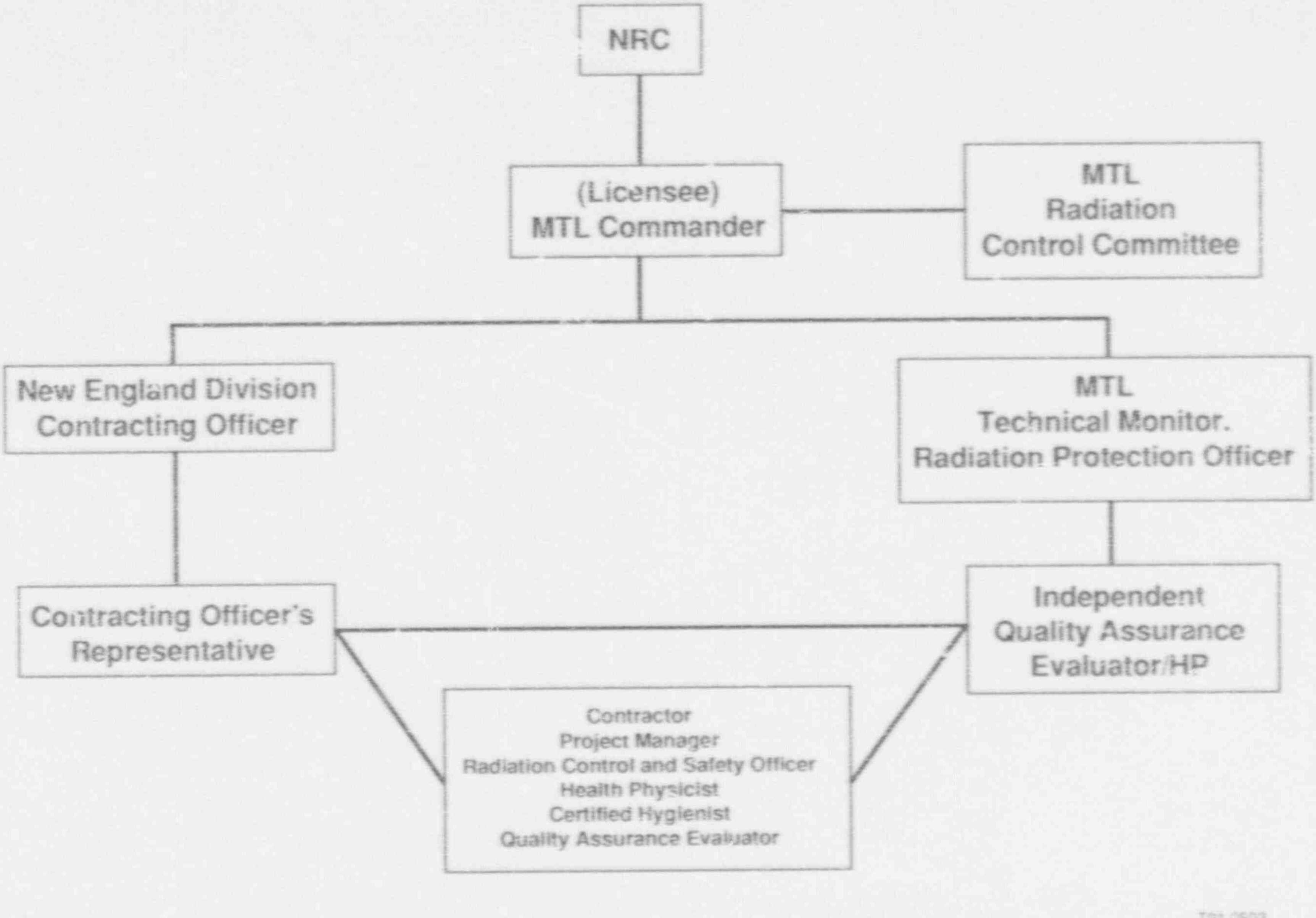


Figure 1-22. AMTL Reactor decommissioning organization relative to safety.

The Corps of Engineers (COE) New England Division (NED) shall act as MTL's Contracting Officer. MTL's Contracting Officer can legally stop the work based on personal knowledge or on input from the Project Technical Monitor, if appropriate. The Contracting Officer's Representative (COR) will have full authority of administering the decommissioning contract. The COR shall also be provided by the COE, NED.

An Independent Quality Assurance Evaluator/Health Physicist (QAE/HP) is responsible for the continuous monitoring of all decommissioning activities and will report regularly to the Licensee, Project Technical Monitor, and the Contracting Officer (KO). The QAE/HP will be experienced in decommissioning projects and will ensure contractor compliance with all provisions of the contract including safety requirements.

The Radiation Protection Officer (RPO), an MTL employee, is responsible for reporting any safety violations and technical inconsistencies with installation procedures. The RPO has the authority to directly stop work on any operation which he has determined to be in violation with 10 CFR and the licensing order for decommissioning.

The AMTL Radiation Control Committee (RCC) will advise the licensee, as needed, in matters related to radiation safety during decommissioning. Organization and conduct of the RCC shall be in compliance with Army Regulations 385-11, Ionizing Radiation Protection.

The decommissioning contractors' Project Manager is the person assigned to the project by the contractor. The Project Manager will be experienced in decommissioning projects and will have contractor responsibility for all aspects of the project including safety.

The contractor's Radiological Control and Safety Officer (RC&SO) will report to the Project Manager and will have responsibility for safety during decommissioning operations. The RC&SO will be a health physicist and will also have received extensive training in industrial safety and industrial hygiene. Reporting to the RC&SO will be a health physicist and a certified industrial hygienist (CIH). Either the CIH or the RC&SO will be an OSHA certified safety professional.

The contractor shall also provide a QAE to monitor the project and ensure contractor compliance with all provisions of the contract including safety.

1.8 Regulations, Regulatory Guides, and Standards

The terms "regulation," "guideline," "standard," and "criteria" are often used interchangeably, but there are distinctions. Regulations are rules having the force of law and are issued by an executive authority or a government. A guideline is a recommended practice or guiding information supplied by an agent with implied intimate technical knowledge. A standard is established by "authority" as a rule to follow. In general, standards set forth limits or definitive ways of accomplishing an objective, whereas criteria provide a yardstick for comparison as a basis for judging the acceptability of a practice.

This section identifies and discusses the regulations, guides, and standards applicable to decommissioning of the AMTL Reactor.

1.8.1 Applicable Regulations

Federal regulations that are applicable to decommissioning research reactors appear in the Code of Federal Regulations (CFR). While all the federal government regulations are contained in the CFR, different titles are associated with various government agencies, commissions, and administrations. For example, Title 10--Energy, pertains to the NRC; Title 29--Labor, includes worker health and safety; Title 40--Protection of Environment, includes regulations of the Environmental Protection Agency (EPA); and Title 49--Transportation, deals with transportation of hazardous materials. Some of the regulations under these titles have immediate applications in decommissioning, and some have application by implication of related subject matters.

1.8.1.1 Code of Massachusetts Regulations (CMR).

CMR Title 310	Massachusetts Air Pollution Control Regulations

CMR Title 310, Chapter 30	Massachusetts Hazardous Waste Management Rule
CMR Title 310, Chapter 40	Massachusetts Oil and Hazardous Release Regulations
CMR Title 310, Chapter 7.10	Noise.

1.8.1.2 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 (Note: The termination of the reactor license must be in accordance with 10 CFR 50)
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	Safety and Health Regulation for Construction
40 CFR Part 260	Hazardous Waste Management System General
40 CFR Part 261	Identification and Listing of Hazardous Wastes
40 CFR Part 262	Standards Applicable to Transporters of Hazardous Waste
40 CFR Part 61	National Emission Standards for Hazardous Air Pollutants
40 CFR Part 141	National Primary Drinking Water Regulations

49 CFR Parts 170-199	Department of Transportation Hazardous Material Regulations.

1.8.2 Regulatory Guides

In addition to regulations that carry the force of law, regulatory bodies such as the NRC and EPA prepare regulatory guides that, among other things, suggest agency-approved methodology and solutions to problems. They generally provide the most effective method of obtaining approval for a particular course of action.

The NRC Regulatory Guides applicable to this project are as follows:

NRC Regulatory Guide Number	Title
1.8	Personnel Qualification and Training
1.16	Reporting of Operating Information
1.86	Termination of Operating Licenses for Nuclear Reactors
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants
3.X	Draft Standard Format and Content of Decommissioning Plans for 10 CFR 30, 40, and 70 Licenses.
8.2	Guide for Administrative Practices in Radiation Monitoring
8.3	Film Badge Performance Criteria
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters
8.6	Standard Test Procedures for Geiger-Muller Counters
8.7	Occupational Radiation Exposure Records Systems
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposure As Low As Reasonably Achievable

8.15	Acceptable Programs for Respiratory Protection
N/A	Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct Source, or Special Nuclear Material, August 1987.

1.8.3 Standards

A number of institutions or technical societies, such as the American Society for Testing and Materials (ASTM), the International Commission on Radiological Protection (ICRP), the National Committee on Radiation Protection and Measurement (NCRP), and the American National Standards Institute (ANSI), publish standards. While not carrying the force of law, they do represent the formal statement of technical opinion of the bodies issuing them.

<u>ANSI and ASTM Standards</u>

ANSI N13.13	Control of Radioactive Surface Contamination of Material, Equipment, and Facilities to be Released for Uncontrolled Use (Draft)
ANSI Z88.2-1980	Practices for Respiratory Protection
ANSI N13.1	Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities
ANSI N323-1977	Radiation Protection Instrumentation Test and Calibration
ANSI/ANS-15.10-1981	Decommissioning of Research Reactors
ASTM E 1281	Standard Guide for Nuclear Facility Decommissioning Plans.

1.8.4 Applicable Army Requirements and Regulations

U.S. Army Corps of Engineers Safety and Health Requirements Manual, EM 385-1-1

Army Regulation No. 385-11 (AR 385-11), Ionizing Radiation Protection

Army Regulation No. 40-14 (AR 40-14), Medical Services Control and Recording Procedures for Exposure to Ionizing Radiation and Radioactive Materials.

1.8.5 Informal Guidance and Technical Reports

Informal guidelines published by the NRC can be found in NUREG documents, Branch Technical Position papers, Inspection and Enforcement Branch notices, and other external or internal documents.

There are numerous technical reports published by the NRC and the DOE that support the subject of decommissioning research reactors.

The following NRC documents are directly applicable:

- NUREG/CR 1756 "Technology, Safety, and Costs of Decommissioning Reference Nuclear Research and Test Reactors" and addenda
- NUREG/CR 2082 "Monitoring for Compliance with Decommissioning Termination Survey Criteria"
- NUREG-0586 "Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities"
- NRC's "Guidance and Discussion of Requirements For an Application to Terminate a Non-power Reactor Facility Operating License"
- NUREG/CR-2241 "Technology and Cost of Termination Surveys Associated with Decommissioning of Nuclear Facilities."

1.8.6 Permits/Licenses Covering the AMTL Reactor

Agency	License/Permit	Number
NRC	Facility Operating License	R65

1.9 TRAINING AND QUALIFICATIONS

The following is a summary of the training program. Details are discussed in Chapter 2.

The training and qualifications of personnel will depend on the individual task assignments and the experience of the personnel assigned to the decommissioning activities of the AMTL Reactor. Training and qualifications of personnel shall comply with ANSI 3.1. Only qualified personnel will be assigned to the AMTL Reactor decommissioning activities.

1.9.1 Training Program Descriptions

Training topics will depend upon:

- The health and environmental impacts of planned operations
- Applicable regulations, standards, and guidelines pertinent to operations involving radiologically or chemically hazardous materials/waste
- The purpose of the training
- The personnel to be trained (e.g., their education, training, and experience).

Documentation of training shall be by appropriate Environmental Health and Safety (EH&S) form, currently "Training Record Sheet."

Personnel having received substantial radiation safety training within the past year may, upon demonstration of their knowledge to the satisfaction of the Project Health Physicist, be exempt from general employee training.

The anticipated training programs include:

- Hazardous Waste Operations Training: The contractor shall comply with the safety and health requirements for hazardous waste operation as prescribed in 29 CFR 1910.120. In addition, alternate workplace standards recommended in publications related to workplace exposure criteria, such as the Threshold Limit Values and Biological Exposure Indices by the American Conference of Government Industrial Hygienist, shall be used in lieu of OSHA standards, where OSHA standards are less stringent or do not exist.

- General Employee Training: General employee training in compliance with Title 10 CFR Part 19.12 will be required for all personnel involved with radioactive materials or working in the vicinity of radioactive materials.

- Respiratory Protection: Respiratory protection training will be implemented to meet project requirements in compliance with ANSI Z-88.2, NRC Reg Guide 8.15, NRC NUREG-0041 "Manual of Respiratory Protection" and 29 CFR 1910.134.

- Hearing Conservation: A hearing conservation training program will be conducted to implement 29 CFR 1910.95.

- Hazard Communications: Hazard communications training in compliance with 29 CFR 1910.1200 will be conducted as applicable.

- Technical Training: Job activity simulations or briefings shall be conducted daily to address proper handling and use of equipment, health and safety issues, and ALARA considerations. These will depend upon the task and personnel and will be documented in the training record.

1.9.2 Administration and Recordkeeping

The Project Health Physicist will be responsible for the training program and maintenance of personnel training, qualification, and exposure records. The Contractor's responsibilities for training are specified in Section 2.1.2 and 2.2.2.

Complete up-to-date training, qualification, and exposure records will be maintained on all personnel. The records will include:

- Bioassay analysis
- Personnel exposure records
- Individual dosimeter readings as related to daily tasks and work procedures
- Respiratory protection qualifications (medical clearance and fit test)
- Audiogram results
- Training records
- Visitor logs and exposure information

CHAPTER 2

OCCUPATIONAL AND RADIATION PROTECTION PROGRAMS

2. INTRODUCTION

The Occupational and Radiation Protection Programs (ORPPs) for the AMTL reactor decommissioning project consist of a set of policies, procedures, and instructions to protect workers, the general public, and the environment. Objectives of the ORPPs include:

- Ensuring the health and safety of personnel by providing protection programs that include a commitment to the principles of maintaining exposures to ALARA levels

- Minimizing the exposure of the general public and the environment to the radioactive and/or hazardous chemical effluents that may be released during decommissioning activities

- Identifying and separating contaminated structures, surfaces, systems, and components from those which are not contaminated

- Disposing of contaminated and noncontaminated components and materials properly and safely

- Ensuring that the facility and site meets all radiological decommissioning requirements and is ready to be released for unrestricted use.

The ORPPs provide integrated occupational health, health physics, industrial hygiene, and safety elements. To meet NRC reporting guidance, these elements are discussed in Sections 2.1, Radiation Protection Program, and 2.2, Industrial Safety and Hygiene Program. This format creates repetition in the text because normally a respiratory protection program is provided for all airborne hazards for both radiologically and chemically

hazardous substances, not in separate programs. Additionally, personnel training is an integral facet of both portions of the ORPPs, and training is discussed in Section 2.2.2. However, a brief discussion of certain training items was presented in Section 1.9, Training and Qualifications.

2.1 Radiation Protection Program

The Radiation Protection Program (RPP) for the decommissioning project includes requirements to monitor radiation and radioactive materials, to control distribution and releases of radioactive materials, and to keep radiation exposure for individuals and the collective radiation exposure within the limits of 10 CFR 20, 29 CFR 570.57, and at ALARA levels.

2.1.1 Personnel

The selection, training, and qualification of decommissioning personnel shall comply with the criteria specified in ANS 3.1.

The decommissioning contractor's personnel will become familiar with the location and magnitude of sources of radiation to which personnel may be exposed during the course of work. In addition, health physics personnel will become familiar with the use of approved Radiation/Hazardous Work Permits (RHWPs), Standard Work Permits, and detailed Work Procedures.

The decommissioning contractor's health physics personnel will include:

- Radiological Control and Safety Officer (RC&SO)
- Radiological Engineer/Health Physics Supervisor (RE/HPS)
- Health physics technicians.

ORPPs procedures will be prepared by the decommissioning contractor in accordance with ANSI 15.10. The licensee will require they be reviewed by the

AMTL Radiation Protection Officer and other appropriate AMTL safety personnel. Implementation guidance will be provided by the RC&SO.

2.1.2 Notices, Instructions, and Reports to Workers, Inspections

Notices, instructions, and reports shall be given to individuals participating in the decommissioning of the AMTL Reactor. These notices, instructions, and reports shall be in compliance with 10 CFR 19.

In addition, NRC inspectors may consult privately with decommissioning workers during inspections. The conduct of the inspections and the rights of the decommissioning workers relative to radiological working conditions are specified in 10 CFR 19 and shall be communicated to decommissioning workers.

2.1.3 Training

Although all persons will receive training, not all persons should receive the same type of training. To prevent duplication and to make efficient use of time, project personnel will be grouped into three categories and will be given training commensurate with potential radiological problems to be encountered in their scope of work. The three groups formed are:

- Nonradiation workers
- Radiation workers directly involved in handling radioactive/contaminated materials and entering radiation areas
- Radiation protection technicians.

The ORPPs shall include a training program for all persons who will be involved in the decommissioning project. All decommissioning personnel shall receive instruction concerning radiation protection through orientation/training. Each worker will attend one of these orientations and will be evaluated by examination upon the conclusion of the training. A passing score is required before these personnel may be occupationally exposed

to radiation. The training program will adhere to the guidelines of the Institute for Nuclear Power Operations (INPO) General Employee Training Program.

The RC&SO, or a designee, shall administer comprehensive radiation worker training, OSHA hazardous communication training, and an indoctrination session for all workers and to all on-site management personnel. Additional briefings and practical-factors training shall be performed on a routine basis to familiarize personnel with work procedures, equipment, radiation control requirements, and hazards associated with the various work elements. Documentation of individual training and qualifications shall be filed and maintained in the health physics office at the work site. Records of training shall be maintained; these will include trainee's name, training date, subjects covered during training, equipment in which training was received, results of written tests, and the instructor's name.

Specialized training shall be provided to employees before they are allowed to undertake jobs with high exposure potential. Employees and supervisors shall receive this training as part of the crew. Mockups and other training aids may be used to train the workers so that time spent in high-radiation fields is minimized.

Objectives of the training program are to accomplish the following:

- Provide involved personnel with information about radiologically and chemically hazardous substances, sources and types, exposure routes, and effects

- Provide information on the ORPPs for the decommissioning project in order to enable each person to comply with health and safety rules and to respond properly to all conditions

- Provide instruction in the fundamentals of radiation and chemical protection to enable individuals to maintain their own exposure and collective exposure ALARA

- Provide information and training on personal protection equipment, monitoring instruments, and equipment available, and how to use them

- Inform each person about NRC, EPA, Occupational Safety and Health Administration (OSHA), state and local regulations and requirements, and other applicable rules and regulations concerning health and safety.

2.1.4 Administrative and Radiological Controls

Administrative and radiological controls comprise the measures taken to limit radiation exposure to personnel and the spread of contamination.

2.1.4.1 Exposure Limits. Limits on the radiation exposure of individual workers involved in radiation-related work have been set for the nuclear industry by the NRC and are applicable here. These limits are stated in 10 CFR 20, "Standards for Protection Against Radiation." However, in order to ensure that individual and collective doses are kept ALARA, the contractor shall establish work procedures and a radiation/hazardous work permit system to ensure that all work performed is evaluated with respect to the ALARA philosophy during the decommissioning of the AMTL reactor.

Personnel 18 years of age or older classified as radiation workers shall have their whole body doses administratively controlled according to the guidelines listed in Table 2-1. Occupation whole body dose limits may be permitted to exceed Table 2-1 administrative guidelines, provided that an approval has been signed by the RC&SO. Under no circumstances will the limits in 10 CFR 20, listed in Table 2-2, be exceeded.

Radiation exposure limits to any individual who is under the age of 18 years are specified in 10 CFR 20.104.

Visitors and nonradiation workers will generally not have access to the radiation control area (RCA). If access is required, exposure to radiation

Table 2-1. Administrative guidelines for radiation whole-body doses during decommissioning

	Administrative Guidelines (mrem)	
	Nonradiation Workers and Visitors	Radiation Workers
Hourly	0.2	--
Daily	2	100
Weekly	10	300
Calendar quarter	13	500
Calendar year	50	1,000

Table 2-2. Regulatory limits for radiation doses during decommissioning for a calendar quarter (mrem)

	Regulatory Limits for Radiation Workers[a] (mrem)
Whole body, gonads, blood-forming organs Lens of eye	1,250
Hands and forearms, feet and ankles	18,750
Skin of the whole body	7,500

a. In accordance with an agreement between the Army Material Command (AMC) and the NRC, 10 CFR 20 requirements will be kept until January, 1993.

will be kept below Table 2.1 values. Visitors and nonradiation workers who are allowed access to the RCA shall be escorted by a radiation worker, according to requirements specified in standard procedures, whenever they enter the RCA. These nonradiation workers must be required to wear radiation dosimetry and will have exposure levels documented.

The dosimeter type and frequency of exchange shall be specified in procedures to be prepared by the decommissioning contractor and approved by the AMTL RPO. A direct reading radiation dosimeter will be worn in conjunction with personal dosimeter badges in high radiation areas. All dosimetry shall be National Voluntary Accreditation Program (NVLAP) accredited.

To ensure compliance with ALARA principles, the contractor's RE/HPS will be available to review the work permits and assist in preparation of work permits. An ALARA checklist shall be prepared by the decommissioning contractor and approved by the AMTL RPO. The checklist shall be used to ensure that all work is preplanned to minimize radiation exposure. The checklist shall include the physical and administrative implementation of the radiation exposure controls.

The RC&SO implements the ALARA philosophy. He/she also reviews and approves procedures concerning work that has the potential for occupational exposure. Appropriate health physics procedures shall be referenced in the Work Procedures to ensure that any occupational exposure is maintained ALARA. Entrance to the restricted areas of the facility shall be controlled by the RE/HPS and requires the issuance and approval of Radiation/Hazardous Work Permits (RHWPs). The description of the RHWP system is discussed below. When the Permit is initiated, the work assignment and applicable procedures are developed and listed. As determined on a case-by-case basis, additional health physics procedures can be implemented at that time, if needed.

Public radiation exposure resulting from decommissioning the AMTL reactor must comply with 10 CFR 20. The maximum public exposure limits for external exposure are specified in 10 CFR 20.105, "Permissible Levels of Radiation in

Unrestricted Areas." Limits for internal exposure pathways are given in 10 CFR 20.106, "Radioactivity in Effluents to Unrestricted Areas." As in the case of occupational exposure, 10 CFR 20.1(c) requires application of the ALARA principle to the control of public radiation exposures and releases of radioactive materials to the environment. During the decommissioning of the AMTL Reactor, no measurable public radiation exposures or releases of radioactive materials to the environment are expected because of engineering controls during decommission. However, an environmental monitoring program shall be in place, and it will be in accordance with ANSI 15.10 to demonstrate compliance with 10 CFR 20.

2.1.4.2 Radiation/Hazardous Work Permits. RHWPs shall be established to ensure that hazardous conditions and protection measures are identified and communicated to those who will perform work in potentially hazardous areas or who may work with material that may be radioactive or radiologically contaminated. The procedures shall also provide the mechanism for exposure accountability and ALARA assessment. A RHWP shall be issued for a specific task, job, or series of tasks to be performed within restricted areas; with material which is radioactive, radioactively contaminated, or chemically hazardous; or with high-hazard operation (confined space, platforms, crane operation, etc.). RHWPs shall be used for areas where hazards are significant and may change. RHWPs shall establish:

- Mapping radiation and contamination based upon radiological surveys, analytical results, and calculations

- Segregating the available work area into sections (e.g., contaminated, clean, working area, examination area)

- Limiting access to control the spread of contamination from contaminated to clean areas and limiting access to all personnel who are not directly involved in the specific task

- Describing the methods to identify and mark all removed items, and noting their place of origin and any other pertinent radiological information

- Packaging contaminated wastes in appropriate containers (as prescribed by NRC and DOT regulations and radwaste disposal site criteria)

- Maintaining accurate shipping records throughout the operation

- Listing work area monitoring requirements, which warn of any unexpected changes in the radiological conditions

- Listing personnel monitoring and protective devices

- Listing requirements for maintaining accurate and updated records of personnel exposure, surveys, and lessons learned in order to improve and revise procedures as necessary.

The supervisor of workers who perform work under RHWPs shall be responsible for ensuring that the workers have been properly prepared prior to entry into a restricted area. Proper preparation includes:

- Successfully completing Radiation Worker Training and OSHA Hazards Communication Training and Respiratory Protection Training (if applicable)

- Checking workers' dose records to ensure entry and/or work without exceeding established limits (administrative and regulatory)

- Providing pre-job ALARA briefings, training, or instruction, if recommended or required

- Ensuring that approved detailed procedures covering the total radioactive work aspects have been prepared prior to the start of work

- Ensuring that appropriate procedures, tools, and equipment are available to perform the job

- Ascertaining that all workers and supervisors have read and understand the RHWP and its requirements, as well as the work conditions and special controls necessary.

The RC&SO shall be responsible for ensuring that personnel have been properly prepared for entry before approving their entry to the restricted area. The RE/HPS, the work supervisor, and the workers responsible for performing the work shall ensure that all radiation/hazard controls are properly implemented throughout the job cycle.

RHWPs will be valid only for the period of the task(s) to be performed and only for the specific task(s) indicated on the RHWPs. RHWPs are provided for entry and work in areas where radiological conditions are subject to significant or unexpected change; therefore, additional instructions or requirements shall be incorporated in the RHWPs, as changes warrant, through the following procedures:

- Any supervisor responsible for completion of a task to be performed within the RCA requiring a RHWP may request one by completing the applicable portions of the Radiation Hazardous Work Permit Request Form and forwarding it to the RC&SO and RE/HPS. Requests for RHWPs should be submitted a minimum of 24 hours prior to scheduled job initiation.

- The RC&SO and the RE/HPS will review the request and prepare the RHWP, after determining the following:

  - The radiological status of the work area, through the appropriate contractor monitoring and surveys.

  - The necessary precautions, based on radiological status, including protective equipment, special control measures, and actions to reduce exposure to ALARA levels.

- Each RHWP must be assigned a number indicating the year and month of issuance and the order in which the RHWP was issued that month. Example: The first RHWP issued in November 1991, would be numbered 91-11-01.

- All RHWPs shall be reviewed and approved by the RC&SO and RE/HPS. The work supervisor will sign all RHWPs, indicating cognizance of the work to be performed, the work location and conditions/restrictions, and approval to enter and perform the work.

- Copy 1 (original) of approved RHWPs shall be posted at the entrance to the job location area to allow review immediately prior to entry to the worksite. Copy 2 shall be posted near the entrance to the RCA, and Copy 3 shall be retained by the RC&SO. Copy 4 of the RHWP shall be retained by the Work Supervisor.

- Exposure time sheets shall be provided for each RHWP, and all persons shall provide the appropriate information requested on the time sheet upon entry and exit from the work location.

- All information entered on exposure time sheets shall be printed legibly in ink. Information entered on RHWP time sheets indicates that the individual for whom information is entered has read, understands, and will comply with the requirements of the RHWP, and that entry and work will be in accordance with established radiation protection rules and policy.

- Upon job completion, the individual(s) responsible for performing the work shall inspect the work area to ensure that it is clear of materials, tools, equipment, or other items used or produced by performance of the job. The individual(s) responsible for the work shall ensure that the work area is in a condition equal to, or better than, that at the commencement of the job, and the responsible supervisor shall sign the back of Copy 1 of the RHWP posted at the jobsite, requesting termination of the permit.

- The RHWP termination request shall be forwarded to the RC&SO and RE/HPS, who will verify the condition of the job site and sign approval to terminate the RHWP.

- All copies of the RHWP shall then be collected. The original (Copy 1) and Copy 2 shall be retained as file copies by the RE/HPS and RC&SO, respectively. The copies shall have the termination date recorded on them. After retaining a copy for the licensee, all other copies should then be discarded. The licensee shall receive all required ANSI 15.10 documentation which will be specified in the SOW.

2.1.4.3 Controlled Surface Contamination Area. Contaminated or potentially contaminated items, materials, and surfaces shall be handled, dismantled, and decontaminated within a Controlled Surface Contamination Area (CSCA). Radioactive waste material shall be placed in designated containers and stored in radioactive material storage zones. To minimize areas designated as CSCAs and the potential that contamination will be spread throughout these areas, small CSCAs shall be established to promote work efficiency.

These CSCAs may correspond to locations where cutting, dismantling, and decontamination operations are performed. When materials with loose surface contamination are properly wrapped and carefully handled to prevent breaking the wrapping, they may be carried through or handled in areas that are not controlled for surface contamination.

All work involving contaminated material shall be performed inside the boundaries of a CSCA. A Contamination Control Point (CCP), through which all entries and exits will be made, shall be located on the perimeter of each CSCA. The floor of the CCP shall be covered with paper, plastic sheet, or other material. This is to provide an easily removable surface within the CCP to prevent the spread of contamination from the area. A step-off pad shall be placed at the exit of the CCP. This shall be used when removing clothing during exit from the area. Receptacles for waste and contaminated clothing shall be maintained at the CCP.

Instruments for monitoring personnel and equipment shall be on hand. All equipment, parts, materials, surfaces, and wastes that have been exposed to radioactive contamination or to neutrons from the reactor will be handled as radioactive and shall not be released for unrestricted handling until they are surveyed and show results in compliance with NRC Regulatory Guide 1.86. If loose contamination is suspected to be in excess of regulatory limits on surfaces not accessible for measurement, the material shall be handled as radioactive. Actual frisking shall be performed whenever possible in low radiation background areas where audible response of the frisker can be distinguished more easily. Adequately trained personnel will be permitted to frisk themselves.

Material that is neutron activated and is to be retained shall be evaluated on a case-by-case basis as required by ANSI 15.10 and NRC guidance[4].

Radiation tags and labels will be available at the CCP to identify the contaminated or activated items being removed from the area. The entrance to the contaminated area shall be posted with:

- Approved RHWP specific to the operation
- Information concerning radiation and contamination levels
- Precautions for entry
- Precautions for exit
- Step-off points
- Frisking instructions
- A sign that prohibits eating, drinking, smoking, or chewing gum or tobacco.

The RE/HPS or a designee shall be responsible for the CCP and will ensure that personnel and equipment are surveyed and logged as required. If work involving contaminated material inside the CSCA requires the use of glove boxes and enclosures, the installation, use, and dismantlement of the glove boxes will be supervised by the health physics personnel.

2.1.5 Radiation Protection Facilities, Instrumentation, and Personal Protective Equipment

Radiation protection facilities, instrumentation, and personal protective equipment used during dismantlement activities are discussed below.

2.1.5.1 Facilities. Facilities provided to enhance the effectiveness of the ORPPs will include the following:

- Facilities and equipment to clean, repair, and decontaminate personal protective equipment, monitoring instruments, tools, and other material

- An emergency shower/personnel decontamination facility

- Change areas to allow changing into anti-contamination clothing. Anti-contamination clothing will be worn during work in the RCA, and the anti-contamination clothing will be disposed of as contaminated waste. There will be no contaminated clothing laundry.

- Control stations for entrance or exit of personnel into radiation or contaminated areas, for movement of radioactive waste material, and for movement of potentially contaminated equipment and instruments

- Equipment to facilitate communication between workers and supervisory personnel between radiation and nonradiation areas

- Calibration source check facilities for the instruments that will be used during decommissioning.

Coordination shall be maintained with AMTL services, such as those provided by the local Fire Department, in accordance with the Memorandum of Agreement between the AMTL and the town of Watertown, Massachusetts. Procedures shall be developed and included in an Emergency Response Plan prior to any decommissioning activities.

2.1.5.2 Instrumentation. A wide range of portable and nonportable instruments and lab-counting equipment will be supplied by the decommissioning contractor and used during decommissioning for radiation surveys, radioactive contamination surveys, personnel monitoring, area monitoring, air monitoring, and sample analysis.

All instruments shall be calibrated in accordance with the specifications contained in ANSI N323-1977 or the most recent revision. Detailed calibration records (including date, method, source description, results, and person) shall be kept as quality assurance records and will be auditable under a quality assurance program.

On a daily basis, or as frequently as required, each type of instrumentation shall be checked and source checked to verify that it is functioning properly and is in calibration.

Table 2-3 lists typical types of instruments required for a decommissioning project.

2.1.5.3 Personal Protective Equipment. Other personal protective equipment shall be provided by the contractor for use as needed. Typically, such equipment includes:

- Anti-contamination clothing

- Contamination control equipment, such as hoods, plastic containers, bags, filters

- Signs, labels, tags

- Special tools

- Decontamination equipment

- Mobile or temporary shields

- Respiratory protection devices

- Hard hats, steel-toed boots, and gloves.

Donning and removal of anti-contamination clothing shall be limited to designated change areas and RCA exit area.

Respiratory Protection Program: A respiratory protection program in compliance with 10 CFR 20.103, ANSI Z-88.2, NRC Regulatory Guide 8.15, and OSHA shall be developed by the decommissioning contractor to provide protection against airborne radioactive and/or chemically hazardous substances. The following elements are included in the program:

- Written standard operating procedures governing selection and use of respirators

- Assignment of responsibilities

- Types of records

- Training of employees and supervisors

- Quantitative and qualitative testing

- Work area surveillance

- Medical surveillance

Table 2-3. Typical radiation survey and monitoring instrumentation and equipment to be provided by the Decommissioning Contractor

- Portable ion chamber rate meters
- Portable GM survey meters
- Alpha survey meters
- Pocket ion chamber dosimeters
- Area monitors including periphery area monitors
- Air sampling equipment
- Windowless gas flow GM counting systems
- Liquid scintillation counter system
- Hand and foot monitor
- Pressurized ion chamber
- Gamma spectrum analyzer
- Permanent personnel dosimeters, either film badges or thermoluminescent dosimeters[a]

a. All radiation workers, in accordance with AR 40-14, shall wear Army radiation dosimetry badges at all times during decommissioning. This will be in addition to the contractor required badges. All dosimetry shall be NVLAP accredited.

- Special respirator use problems and limitations
- Maintenance and repair of respirators.

The contractor shall supply only respiratory protection equipment approved by the Mine Safety and Health Administration and the National Institute for Occupational Safety and Health. Potential exposure hazards, radioactive particulates and vapors, and airborne toxic chemicals shall be monitored and evaluated. Workers shall be instructed and supervised to ensure that all respiratory protection equipment provided is used in accordance with the training and instructions received. The contractor shall routinely inspect respiratory equipment, and protect it from outside contamination and damage. The RC&SO shall delegate responsibility to the Certified Industrial Hygienist (CIH) to direct, evaluate, and provide guidance on all aspects of the facility respiratory protection program and ensure that employees required to wear respiratory protection are physically and medically able to do so.

The Respiratory Protection Program shall ensure that exposure of individuals to concentrations or radioactive materials in the air is in compliance with 10 CFR 20.103.

The respiratory protection program administration shall be accomplished under the guidance and direction of the CIH in coordination with the RC&SO. The AMTL Technical Monitor will provide oversight for the respiratory protection program. In addition, the CIH shall be responsible for maintaining an adequate supply of respirators and cartridges on-site for personnel use. All purchasing of respiratory equipment will be under the guidance of the CIH.

Respiratory selection shall be based on the following criteria:

- Nature of the hazard
- Physical properties of the contaminants involved
- Contamination on surfaces and airborne contamination

- Location of the hazard

- Time frame for which respiratory protection will be required

- Operational activities of personnel required to wear respiratory equipment

- Functional capabilities and limitations of respiratory equipment

- Potential for the presence of conditions immediately dangerous to life and health

- Potential for the presence of oxygen deficient atmospheres.

All decommissioning personnel required to use respiratory protection shall receive periodic training pertaining to all respiratory protection. This training shall be given under the guidance of the CIH. Training shall include but not be limited to the following:

- Proper use of all available respiratory protection including hands-on training

- Reasons for the selection of a particular type of respiratory equipment based on potential hazards

- Functional capabilities and limitations of all available respiratory equipment

- Identification of respirator malfunction and how to correct the malfunction

- Proper methods of donning respiratory protection equipment

- Reasons for determining respiratory fit, methods to be used, and factors affecting respiratory fit

- Proper care and maintenance of respiratory protection

- Training in the use of respiratory protection as it relates to the recognition and handling of emergency situations

- Discussions of potential contaminants against which the wearer is to be protected, including physical properties, physiological action, toxicity, and means of detection

- Discussion of the application of various cartridges and canisters available for air respiration

- Instruction in emergency action to be taken in the event of malfunction of the respiratory protection devices.

All personnel required to use respiratory protection shall be advised that they may leave the work area at any time for relief from physical or psychological distress, procedural or communication failure, significant deterioration of operational conditions, or any other condition that might require such relief.

The CIH shall designate which personnel are qualified to give instructions on respiratory protection. Qualifications shall be based on the designee's knowledge of the application and use of respiratory protection and the hazards associated with potential chemical and radioactive contaminants.

All decommissioning personnel required to wear respiratory protection and who have demonstrated that they are physically and medically able to do so shall receive a qualitative respirator fit test to be administered under the guidance and direction of the CIH.

The program shall include employee orientation, employee sensitivity tests, performance of the fit test in a test enclosure, and respirator assignment. The qualitative fit test protocol will use Isoamyl Acetate or Irritant Smoke tubes as appropriate. Records of respirator fit test results shall be provided to the licensee.

In addition, all personnel shall be instructed in the proper procedure for the performance of the positive and negative pressure tests. These quick respirator fit checks shall be performed by all personnel immediately after donning approved respirators and prior to entering an area designated for respirator use.

Personnel shall only wear air purifying respiratory equipment for which they have successfully passed the qualitative fit test protocol. No respirators shall be worn by personnel who have facial hair such as beards or long sideburns which interfere with the sealing periphery of the respirator face piece or with respirator valve function.

Contact lenses may be worn by any employee while in an area designated for respirator use. Prescription glasses may be worn as long as the seal of the respiratory face piece to face is not directly affected.

Only the CIH or a properly trained designee shall be permitted to issue respiratory protection to decommissioning personnel or outside contractors and visitors.

All subcontractor personnel and visitors shall be required to adhere to the same respiratory protection procedures as regular decommissioning personnel.

For unusual operational instances or special projects, respirator issuance shall be made under the guidance and direction of the CIH.

All workers for whom the potential of contact with hazardous materials exists shall participate in a medical surveillance program. As a minimum, this program must provide baseline health assessments to investigate existing conditions that may predispose a worker to illness following exposure to hazardous substances or to the physical demands of using protective equipment. In addition, periodic health assessments shall be provided to screen workers for signs of occupational exposure to toxic agents and to determine their subsequent assignments.

2.2 Industrial Safety and Hygiene Program

The Industrial Safety and Hygiene Program (ISHP) for the decommissioning project is concerned with the protection of all personnel from potential nonradioactive exposures and hazards. The ISHP shall be developed by the decommissioning contractor, and the ISHP shall include a hazard analysis. In addition, the ISHP shall specify the applicable regulations, standards, and other requirements to be followed including the U.S. Army Corps of Engineers Safety and Health Requirements Manual, EM 385-1-1. The ISHP shall be administered in accordance with OSHA regulations. In the absence of a particular regulation, guidance shall be obtained from the NIOSH or the American Conference of Governmental Industrial Hygienists (ACGIH).

2.2.1 Personnel

The ISHP shall be administered by the contractor's CIH in coordination with the RC&SO. In addition to responsibilities previously discussed as aspects of the RPP, responsibilities shall include:

- Inspections and audits
- Occupational health
    - Medical surveillance
    - Hearing conservation
    - First aid
- Emergency services
- Operational activities.

2.2.2 Training

To supplement the comprehensive training program described in Section 2.1.2, the contractor shall also supply all workers with instruction concerning the project safety program through orientation/training prior to being assigned to project activities. Each new-hire or transferee shall attend one of these orientations, which consists of instruction in job safety action plans, hazard recognition and correction, fire extinguisher training, and safety awareness films. Specialized training applicable to specific conditions shall be given as the progress of decommissioning activities mandates.

Supervisory safety training is an integral part of the safety training program. Supervisors shall receive a safety orientation detailing the safety responsibilities of their positions.

Training courses and a qualified staff roster shall be documented and updated, with follow-up training conducted as needed. Topics for presentation at these ISHP training sessions shall include:

- Specific project safety procedures
- Fire protection and prevention
- Work practice procedures and tool-box safety
- Confined-space entry
- Special housekeeping requirements
- Material-handling techniques
- Safety and warning devices
- Hazard identification and reduction
- Hazard communication

- Enforcement policy

- First aid and emergency procedures and equipment.

Records shall be kept of all personnel attending, level of accomplishment, follow-up sessions, etc., as necessary, to ensure that the appropriate awareness and competency have been demonstrated.

Individuals performing asbestos sampling shall have completed the Asbestos Hazardous Emergency Response Act (AHERA) training course for asbestos inspectors

2.2.3 Administrative and Work Practice Controls

These controls comprise the measures taken to limit chemical exposure and safety hazards and to reduce the risks associated with the decommissioning activities. Essential to the ISHP are the RHWPs which are discussed in Section 2.1.3.2 and access controls discussed in Section 2.1.3.3.

2.2.3.1 Exposure Limits. Personnel exposures to toxic/hazardous materials shall not exceed limits established by OSHA or those recommended by ACGIH in "Threshold Limit Values and Biological Exposure Indices."

Evaluation of potential toxic/hazardous risks to workers shall be performed by the CIH using appropriate methods. These methods may include the use of Dragger tubes, air sampling, analyses for hazardous materials, and visual observation based on experience and training.

In the event that the evaluation by the CIH reveals potential toxic/hazardous risks to workers, the CIH will require appropriate personal protective equipment.

2.2.3.2 Inspection and Audit Programs. Inspections shall be conducted routinely by the contractor's CIH and RC&SO during active work phases and on at least a weekly basis. Safety violations shall be recorded, identified, and

corrective actions taken immediately. Copies of any infraction notices shall be maintained by the RC&SO and documented in a Weekly Safety Report to the Army's KOR. The following items shall be specifically addressed during these routine inspections:

- Barricades
- Safety signs
- Scaffolds
- Hoisting and rigging
- Confined-space entry
- Excavations
- Torch cutting
- Hearing protection
- Any other industrial hazard
- Radiation/Hazardous Work Permits
- Personal protective equipment (PPE).

In addition to inspections conducted by the contractor, the Army will provide an independent Quality Assurance Evaluator/Health Physicist (QAE/HP) to continuously monitor the project to ensure that decommissioning is performed in compliance with all specifications and requirements. The independent QAE/HP will report directly to the licensee.

Supervisors shall be required to participate actively in the investigation of any accident occurring in their areas and which result in any

personal injury to employees under their direction, equipment or property damage, and near misses with the potential for serious injury or loss. The investigation will be aimed at determining facts, not fault, so that recurrences can be prevented.

In order to provide verification of the program, an audit procedure of the ISHP will be developed by the licensee, incorporating approved evaluation criteria. Audits will be conducted by the independent QAE/HP. Audits are conducted for:

- Compliance with all safety requirements
- Implementation of health and safety procedures
- Health and safety organization
- Job descriptions and tasks
- Review of records and documentation relating to health and safety
- Site layout and inventory
- Training materials.

The audit criteria include, at a minimum, the evaluation of:

- Written procedures
- Qualifications, education, and training of management and staff
- Communications and coordination of the various health, safety, and medical elements of the program
- Environmental surveillance

- Facilities, apparatus, and monitoring equipment
- Medical surveillance
- Emergency planning.

Using this procedure, an assessment will be completed and the results properly communicated to the RC&SO and the licensee.

Written recommendations shall be prepared to improve deficient areas. It is the responsibility of the contractor's Project Manager to ensure correction of deficiencies and documentation of actions taken.

2.2.3.3 Accident Reporting. Accidents resulting in a fatality, lost-time injury or illness, hospitalization of 5 or more personnel, or property damage to government or contractor property (which occurred during the performance of the contract) equal to or exceeding $2,000.00 shall be telephonically reported to the NED Safety Office and COR, as soon as possible, but not later than 2 hours after occurrence and reported in writing within 5 days of occurrence on ENG Form 3394 (encl)3. All other accidents/incidents shall be telephonically reported within 8 hours of occurrence. All accidents that occur during decommissioning of the AMTL Reactor shall be reported in compliance with EM 385-1-1. The COR will provide all information regarding accidents to the MTL Safety Office.

2.2.3.4 Medical Surveillance Program. A medical surveillance program shall be established by the contractor for all workers who may be occupationally exposed to radiological or hazardous chemical agents. The program shall comply with applicable state and Federal requirements including DOD 6055.8, Occupation/Radiation Protection Program and may include the following items as appropriate in the baseline health assessment:

- Occupational history

- Medical history
- Family history
- Physical examination
- Pulmonary function testing
- Audiometric testing
- Baseline bioassay.

The regulatory guide for the baseline bioassay is NUREG 8.9. ANSI standards shall be used to determine maximum internal doses.

Criteria for the health assessment shall be developed to identify the need for pre-employment and periodic health assessments, termination examinations, and return-to-work and other special examinations.

The decommissioning contractor shall determine available clinics and hospitals for routine and emergency medical needs.

Medical records shall be maintained, with attention given to federal requirements, inclusion of exposure data, appropriate update frequency, and access privileges.

The work site shall be monitored for health hazards associated with the work environment, including chemicals that may be present in liquid, dust, fume, mist, vapor, or gaseous forms. Physical hazards such as noise, pressure, vibration, and illumination will also be monitored and controlled.

2.2.3.5 Hearing Conservation Program. A hearing conservation program shall be established by the contractor for all workers who are exposed to noise levels in compliance with U.S. Army Corps of Engineers Safety and Health Requirements Manual, EM 385-1-1. This program shall include:

- Noise monitoring in areas where the levels exceed 80 dBA

- Audiometric testing for workers (as part of the medical surveillance program) to determine baseline hearing performance before exposure and test results after exposure

- Personnel training and education

- Recordkeeping

- Hearing protection devices.

Personnel who are assigned tasks in known noise-hazardous areas as defined in EM 385-1-1 shall be enrolled in the hearing conservation program prior to beginning their work.

Noise control measures in compliance with EM 385-1-1 shall be determined by the CIH after appropriate noise monitoring is completed in the work area.

The contractor shall maintain records that document all noise monitoring conducted, employee training done, control measures implemented, and protective equipment issued.

2.2.3.6 Fitness for Duty Program. The decommissioning contractor shall implement an appropriate Fitness for Duty Program to comply with NRC Notice Number 90-81. The requirement and procedures to implement a Fitness for Duty Program will be incorporated into the Statement of Work. The program will assist the AMTL Commander in providing a drug-free work place.

2.2.4 Operational Activities

The requirements for fire protection, emergency response, and equipment/tools are discussed below.

2.2.4.1 **Fire Protection and Prevention.** Fire protection devices shall be made available by the contractor in appropriate locations during decommissioning tasks. Portable Type A and B/C fire extinguishers shall be strategically located to serve areas partitioned by the various decommissioning activities.

Fire prevention measures shall be implemented to avoid ignition hazards from electrical wiring and equipment and from combustible materials. Smoking shall not be permitted in areas where a potential fire hazard is present.

2.2.4.2 **Emergency Response.** Prior to start of decommissioning activities, the decommissioning contractor shall prepare an Emergency Response Plan in compliance with EM 385-1-1. The Emergency Response Plan will be approved by the licensee and COR.

2.2.4.3 **Hand and Power Tools and Cutting Equipment.** The condition of the hand and power tools used during decommissioning shall be routinely checked for proper operation and for compliance with applicable requirements.

2.2.4.4 **Lifting Equipment.** Lifting equipment (such as hoist, fork lift, and bridge crane) used in decommissioning shall comply with applicable provisions of OSHA. These provisions include:

- Compliance with the manufacturer's specifications and limitations applicable to equipment operation
- Posting of rated load capacities, operating speeds, and special hazard warnings or instructions
- Inspection of equipment by competent personnel prior to each use and during use to make sure it is in safe operating condition
- Limiting the travel of rail-mounted equipment with limit stops
- Removing from service any equipment which has damaged wire ropes, chains, or other components.

2.2.5 Personal Protective Measures

To minimize the effects of industrial and radiological hazards associated with decommissioning, specific health and safety measures shall be implemented by the decommissioning contractor. Anticipated hazards and mitigation measures are listed in Table 2-4. The decommissioning contractor shall specify task-specific personal protective equipment in the work procedures for each task and in the RHWP covering each task.

2.2.6 Excavations

Excavations required during decommissioning activities shall comply with applicable provisions of OSHA and EM 385-1-1. These provisions include:

- Sides of excavations tapered in compliance with OSHA requirements
- Protection of workers with personal protective devices as discussed in Section 2.2 of this document.
- Provisions to prevent workers from standing under loads handled by lifting equipment
- Daily inspection of excavations by contractor industrial safety personnel for evidence of potential or actual cave-ins or slides
- Supporting systems (e.g., underpinning, etc.) designed by qualified contractor personnel and inspected daily
- Excavated materials and other material stored at least 2 ft from the edge of the excavation
- When using heavy equipment in the vicinity of excavation, the sides of the excavation braced to resist extra pressure by superimposed loads

Table 2-4. Mitigation and monitoring of hazards during decommissioning

Anticipated Hazards[a]	Mitigation	Monitoring
Airborne radionuclides	• Respirators (full face) • HEPA filtration units	• Whole-body counts • Continuous air sampling • Grab air sampling
Falling and flying debris	• Safety glasses with side shields • Limited access, safety shoes • Hard hats	• Incident Reports
High sound levels	• Ear protectors	• Physical exams • dB measurements
Beta-gamma exposure rates	• Limited access • Portable shielding	• Personnel dosimetry • Daily surveys
High heat and humidity	• Ventilation • Work breaks • Cooled protective clothing	• Temperature measurements • Stay time limits
Loose surface contamination	• Anti-contamination clothing • Cleanup/decontamination	• Frisking • Daily surveys
Airborne hazardous chemicals/vapors	• Respirators and supplied air • Cleanup/decontamination	• Air sampling
Airborne dust	• Water fogger • Safety goggles	• Air sampling

a. The listed anticipated hazards may be encountered during all decommissioning tasks discussed in Chapter 3.

- Adequate barrier physical protection provided around the excavated area

- Safety harnesses/belts provided for use in confined spaces

- Ladders and scaffolding used in excavation provided around the excavated area

- Ladders and scaffolding used in excavation complying with the applicable provisions of OSHA.

2.3 Contractor Assistance

A decommissioning contractor will be used to perform the decommissioning of the AMTL Reactor. The activities to be performed by the contractor include the following:

- Performing all decommissioning operations

- Supervising day-to-day decommissioning activities including directing craft supervisors and crew leaders

- Providing health physics support including installing, calibrating, and testing equipment; and conducting radiological surveys

- Providing health and safety support, including a CIH

- Furnishing quality assurance support, including preparing procedures and complying with quality during decommissioning

- Providing crafts and labor for temporary construction work, performing decontamination and demolition tasks, and processing, packaging, and shipping radioactive materials.

These will be ongoing activities during the entire decommissioning period, with the Army's personnel overseeing and reviewing the work as it takes place. However, the contractor will be delegated responsibility for health and safety via the contract between the Army and the contractor.

CHAPTER 3

DECOMMISSIONING TASKS, SCHEDULE, COST, AND FUNDING

3. INTRODUCTION

Major tasks to be performed during decommissioning of the AMTL reactor facility are shown in Figure 3-1, Work Breakdown Structure (WBS) for Decommissioning the AMTL reactor. Each element in the WBS will be discussed in Sections 3.1 through 3.6. The estimated schedule and cost for the project are given in Section 3.7. The source of funding for the project is discussed in Section 3.8.

Descriptions of decontamination tasks include general procedures for how the facility should be decontaminated; the descriptions address health and safety considerations if applicable. Task descriptions in this chapter are general, but detailed work instructions will be specified in work procedures to be prepared by the decommissioning contractor and be available for review by the AMTL staff.

All operations and tasks that involve equipment and materials that produce ionizing radiation shall be conducted in such a manner as to maintain radiation exposure to personnel as low as reasonably achievable (ALARA). Operations and tasks involving ionizing radiation shall be planned so that the limits established by the NRC, DA, AMC, OSHA, and MTL regulations are not exceeded.

Work procedures will also specify radiological surveying procedures to be followed during removal and disposal of equipment, components, material, and other items. These surveying procedures are required to ensure that all radioactively contaminated equipment, components, material, and other items are disposed of appropriately. It is essential that no radioactive waste above releasable levels, as defined in Reference 4, be disposed of as domestic waste.

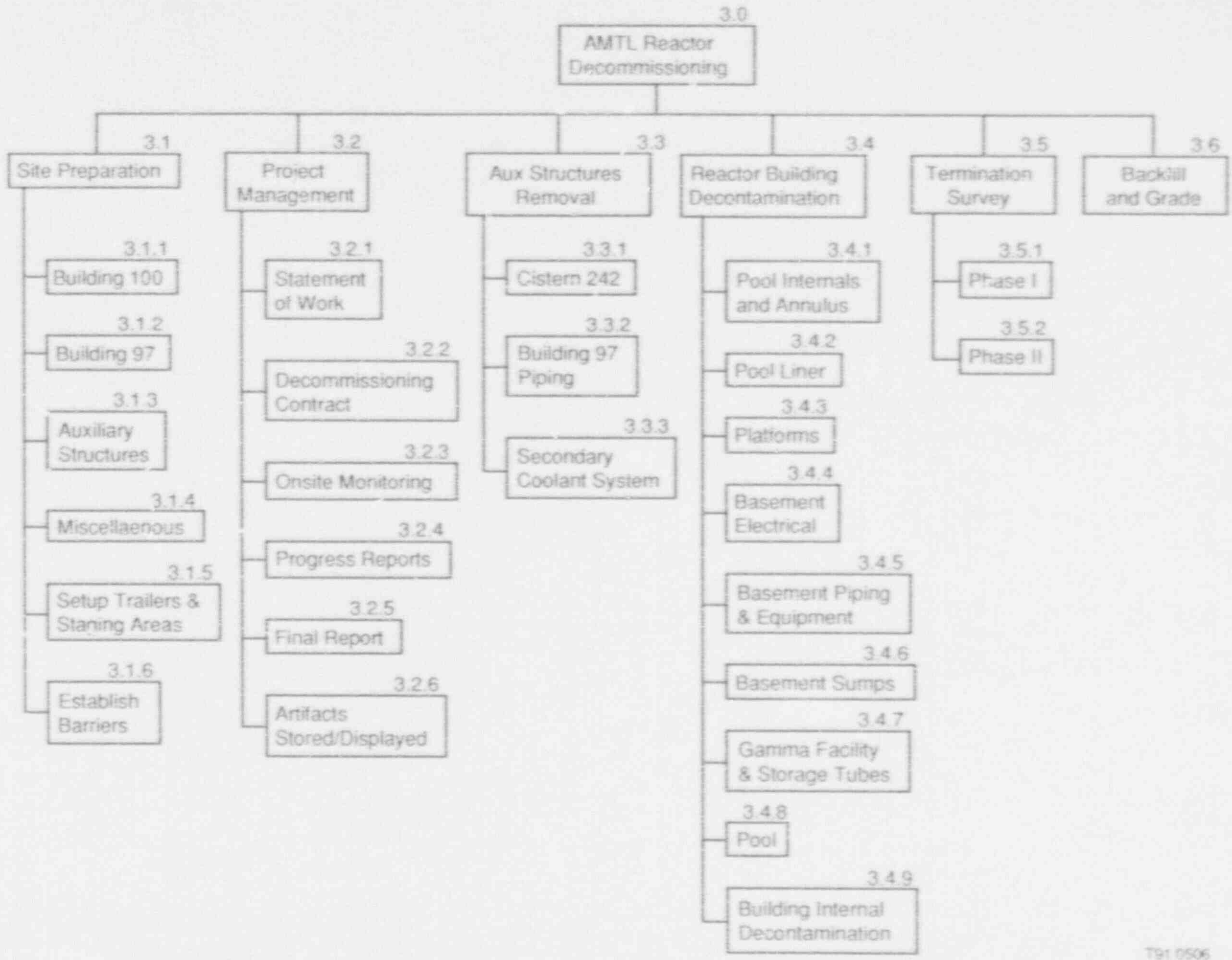


Figure 3-1. Work breakdown structure (WBS) for decommissioning the AMTL Reactor.

Surveying will include smearing for removable contamination, and direct contact radiation measurements for alpha and beta-gamma radiation. For direct radiation measurements, the scanning speed will be slow enough to ensure detection of the most restrictive release levels specified in Table 1 of Reference 4. The scanning speed will depend upon specifications of the instrument being used. The scanning speeds will be calculated and included in the procedures.

Prior to decommissioning, the items listed below shall be performed:

Tasks to be Performed Prior to Decommissioning

1. Ensure that all experiment equipment and materials are removed from the reactor building, including all concrete and phenolic blocks, cabinets, desks, etc.

2. See that all concrete blocks and other materials stored around the secondary coolant pumps and sump are removed prior to contractor mobilization.

3. Ensure that Cistern 242 and the secondary coolant sump are empty and that no controllable inflow of water exists.

4. Provide utility hookups for the contractor. These must include water, power, telephone, and sanitary sewer lines.

5. Provide office space for the Quality Assurance Evaluator (QAE) in the immediate vicinity of the reactor building. This office will include a desk, chair, file cabinets, lighting, power, and telephone service.

6. Designate sufficient area for the contractor to place his equipment and portable facilities, and to stage materials awaiting use or disposal.

7. Prepare the reactor building and immediate area for decommissioning operations by performing the actions listed below.

   a. The areas in Building 97 adjacent to the airlock entry into Building 100 and the pipe alley containing reactor-related piping will be free of equipment and materials to allow the contractor free access.

   b. Remove two cabinets adjacent to Building 100 in the area of the secondary coolant pumps.

   c. Remove the yellow steel structure adjacent to Cistern 242.

3.1 Site Preparation

This section briefly describes the activities and tasks required to prepare the AMTL reactor facility and site for decommissioning. Tasks described in Sections 3.1.1 through 3.1.4 are not part of the decommissioning contractor's scope of work and will be performed by the licensee prior to start of decommissioning. Site-preparation tasks described in Sections 3.1.5 and 3.1.6 shall be performed by the decommissioning contractor.

3.1.1 Building 100

All experimental equipment, office furniture, file cabinets, concrete blocks, nonradiologically contaminated lead bricks, and all other nonreactor related materials will be removed from the building and the surrounding area.

Reactor control equipment that is reusable will be sent to other reactor facilities prior to initiation of decommissioning operations.

3.1.2 Building 97

The brass plaque above the airlock door will be removed and placed in the artifact repository.

The areas adjacent to the airlock and piping corridor will be cleared of all loose materials.

3.1.3 Auxiliary Structures

Cistern 242 and the secondary coolant sump will be emptied and isolated from all sources of liquid influx other than groundwater. The sources of any fluids that are controllable will be permanently isolated as close to the source system as possible and identified.

A yellow metal structure adjacent to Cistern 242, the two metal cabinets and pallets of concrete bricks near the secondary coolant pumps, and any other loose items in the area of the reactor will be removed from the vicinity.

3.1.4 Miscellaneous

Areas within AMTL and adjacent to the reactor building will be designated for the contractor to place temporary facilities and to stage materials and wastes. A buffer zone will be established by erecting a fence around the reactor building to prevent decommissioning and other base activities from impacting each other. The approximate location of the fenced area is shown in Figure 1-2.

A protected storage area will be designated by the licensee and prepared for the temporary storage of containerized low level radioactive waste until the waste can be shipped.

A temporary office for the independent QAE/HP will be established as close to the reactor building as possible.

Provisions for temporary utilities such as telephone, power, water, and sewer for use during decommissioning activities will be identified.

3.1.5 Setup Trailers and Staging Areas

Following the award of the contract for the conduct of decommissioning, the decommissioning contractor will mobilize the equipment and facilities needed to support the decommissioning activities and locate them at AMTL. It is anticipated that the contractor will bring office trailers, change room/shower facilities, decontamination facilities, radiological analysis laboratories, various pieces of construction/demolition equipment, barricades, signs, and other items to support their activities.

The contractor shall also establish areas within the fenced area for the staging of materials prior to utilization. Packaged waste shall be temporarily stored (if necessary) in a protected storage area near the reactor facility prior to shipment to a disposal site. In each case, the amount of materials staged or stored will be kept at the lowest level possible.

The decontamination and change facilities shall be located as close as practicable to the containment building to minimize the potential for the spread of contamination during access and egress.

At the completion of the contracted work and following acceptance by the licensee, the contractor shall ensure that all materials and equipment belonging to the contractor are removed from the AMTL.

3.1.6 Establish Barriers

In order to maintain separation of decommissioning activities from the other activities at AMTL, it will be necessary for the contractor to establish barriers. The barriers shall be established to include the minimum area needed by the contractor for isolation of the decommissioning activities. Barrier locations will be specified in the SOW. To minimize impacts of restricting access to areas of AMTL it may be necessary to relocate the

barriers to accommodate the various stages of decommissioning. Any relocation must be approved by the AMTL Commander. At the completion of portions of the work scope requiring barriers, the barriers shall be removed from AMTL promptly.

3.2 Project Management

The decommissioning operations will be performed by a decommissioning contractor. The overall project management, however, will be accomplished by the U.S. Army Corps of Engineers (COE), New England Division (NED). This section describes the tasks required of the COE, NED.

3.2.1 Statement of Work

A Statement of Work (SOW) will be prepared for and approved by the Army. The SOW will be the basis for which the decommissioning contract is written and selection of the decommissioning contractor is made.

The SOW will specify the work to be performed by the decommissioning contractor, standards of performance, and all deliverables to be supplied by the contractor. The SOW must also include a detailed description of the decommissioning Contractor's responsibilities for health, safety, and environmental implications and effects of the work performed. The SOW will also state the items to be furnished by the Army during decommissioning. The required work specified in the SOW will include all decommissioning tasks described in this DP, except those tasks specified to be performed by the Army or an independent contractor.

3.2.2 Decommissioning Contract

Following completion of the SOW, a Request for Proposal (RFP) will be prepared, proposals requested and evaluated, and a contract awarded for the performance of decommissioning as described and specified in the SOW. The RFP

will include the SOW, which will be the basis for the proposals submitted by prospective contractors. The contract will be for the work described in the SOW.

Prior experience with decommissioning of an NRC-licensed facility, as well as health physics and safety experience, shall be a requirement for any contractor chosen. A complete list of qualifications for the decommissioning contractor will be included in the SOW.

3.2.3 Onsite Monitoring

The Army COE will provide a QAE/HP to be onsite at all times while the contractor is performing work. The QAE/HP will ensure that the contractor is performing work in accordance with the terms of the contract. In addition, the QAE/HP will be immediately available for technical consultation when unforeseen situations are encountered. The QAE/HP must have decommissioning experience.

The QAE/HP will provide written daily status reports to the designated Army point of contact as to the status of work progress as compared to the plan, problems, corrective actions, and any other pertinent information. Should any high-concern incidents occur, such as injuries, contamination releases, etc., the QAE/HP will immediately inform the Army contact of the situation.

The QAE/HP will also maintain a photographic record of the work progress for inclusion in the status reports and final report.

3.2.4 Progress Reports

Written daily progress reports will be made by the QAE/HP to give the status of the work with respect to the plan, report on any unusual occurrences, and status of any corrective actions. These reports will be brief and concise descriptions of the work performed each day.

Monthly reports will also be prepared by the QAE/HP. These reports will be detailed status reports of the technical, budgetary, and schedule progress of the activities. This report will assess how work is going as compared to the contract schedule and will provide variance analyses, impacts of variances, and status of corrective actions.

Copies of all progress reports will be distributed to the COE, [illegible], COR, and the AMTL Commander, as a minimum.

3.2.5 Final Report

A Final Report will be prepared by the QAE/HP to document decommissioning of the AMTL Reactor. This report will include descriptions of the tasks completed, the actual schedule of accomplishments, photographs of the work and final conditions, final radiological conditions, a summary of worker exposure, waste volumes generated and repositories of the wastes, and lessons learned.

3.2.6 Artifacts Stored/Displayed

Artifacts from the reactor facility that meet radiation release criteria specified in Reference 4 will be collected by the Army prior to start of decommissioning. The Army may choose to display some of the artifacts to help describe activities that were conducted at AMTL throughout its history.

Examples of artifacts that will be preserved are the model of the facility, the plaque over the airlock in Building 97, photographs of construction and operations, and operating logbooks.

3.3 Auxiliary Structures Removal

3.3.1 Cistern 242

This below-grade concrete cistern (Figure 3-2) is to be removed by demolition and any contaminated soils surrounding the cistern also removed.

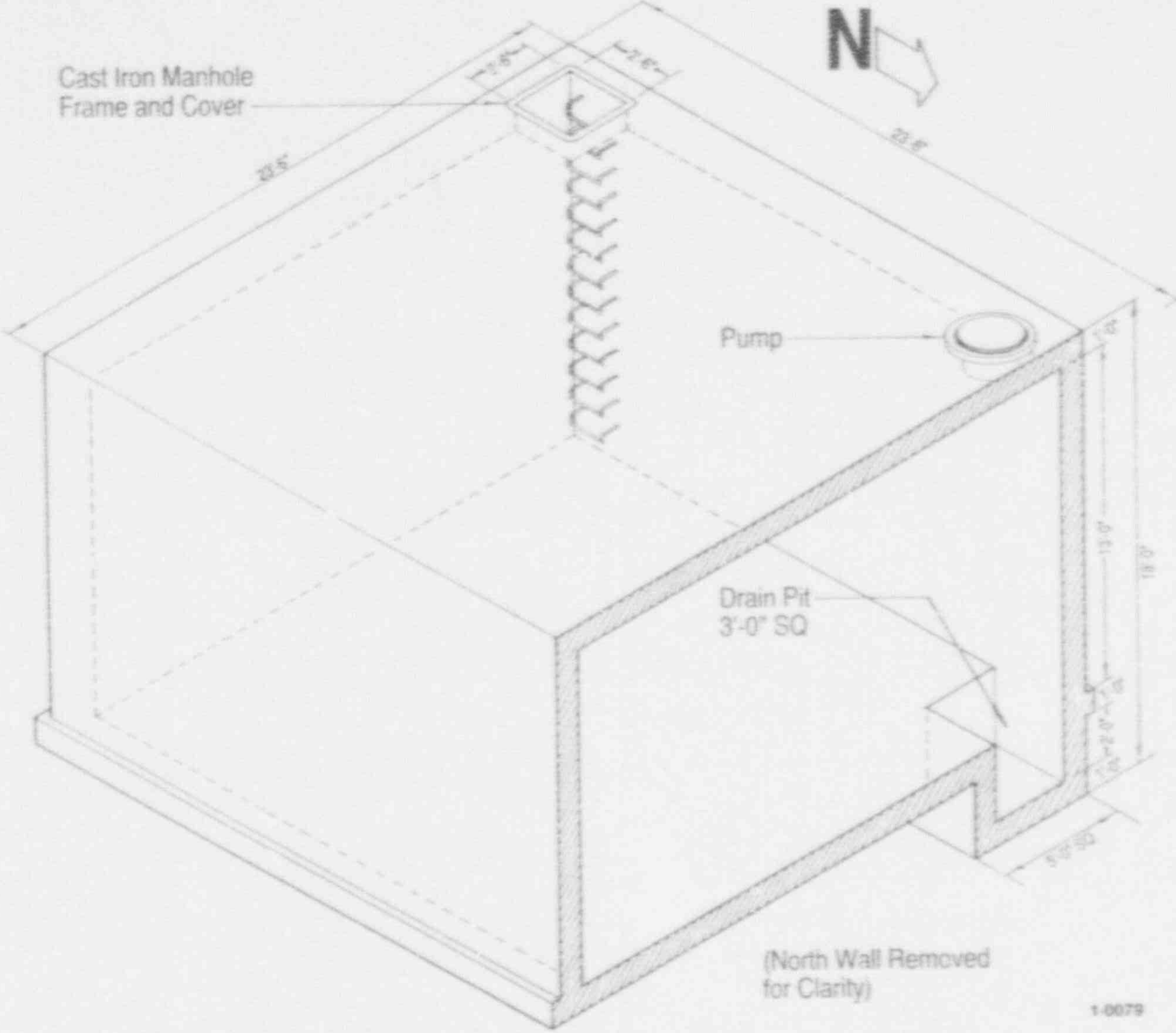


Figure 3-2. Isometric view of Cistern 242 at the AMTL Reactor site.

The removal of the cistern is to be one of the first activities performed under the decommissioning order so that information concerning the immediate subsurface conditions at the cistern may be applied to any future excavation and removal activities. Prior to demolition, the cistern is to be sampled for radioactive and hazardous materials. In the unlikely event that radiological contamination is found inside the cistern, the inside of the cistern will be decontaminated to releasable levels before the cistern is demolished. Following receipt of sample analyses data, the cistern will be demolished and the materials disposed of in a manner approved by the AMTL as determined by the analyses results. It is anticipated that the cistern will contain little or no contamination above releasable levels.

The cistern demolition will include removal of the chainlink fencing, pump, and electrical supply equipment.

Following removal of the cistern structure, the soils surrounding the cistern will be radiologically surveyed, and any contaminated soils shall be immediately removed and packaged to prevent contamination spread. If immediate removal of contaminated soil is not feasible, the soil will be covered to prevent contamination spread. When the contractor is reasonably ensured that the soil is at releasable levels, Phase 1 of the final survey will be performed as specified in Chapter 8.

Removing the Cistern will also include removal of piping connecting the cistern to Building 97. The piping removed as part of this task will be radiologically surveyed, results compared with Reference 4 and disposed of as clean waste or radiological waste.

During entry into and while conducting work in enclosed spaces, special precautions specified in Section 2.2 will be implemented to ensure the safe conduct of the work. A permit for confined space entry must be obtained and continuous monitoring conducted for oxygen-deficient/explosive atmospheres.

Adequate sloping and/or shoring of the sides of the excavation will be implemented to minimize the potential for cave-ins. Due to the nonuniform nature of the fill materials in the area of the reactor building, shoring will be the preferred method of cave-in prevention.

3.3.2 Building 97 Piping

There are several piping runs remaining in Building 97. These are liquid waste lines leading to the previously removed waste plant, demineralized water lines, steam and condensate lines, and water supply lines to the cistern. Prior to removal, these lines will be checked to ensure that the systems are not pressurized, the lines drained of all materials, and verified to be free of asbestos-containing insulation materials. Should asbestos-containing materials be encountered, the decommissioning contractor will follow appropriate procedures prepared by the decommissioning contractor and approved by the Army. Asbestos sampling operations shall be performed in accordance with U.S. Army Technical Manual 5-612. If asbestos work is performed, only qualified asbestos workers will perform the work.

When the piping has been determined to be empty and free of external contamination, the piping will be removed, sectioned, and the interior surveyed to determine if the individual sections can be disposed of in accordance with the unrestricted use criteria specified in Reference 4.

3.3.3 Secondary Coolant System

The secondary coolant pumps remaining atop the sump are shown in Figure 3-3. They will be removed, surveyed for radioactive materials, labelled, and staged for disposal.

The sump will be surveyed, excavated, and removed in the same manner as described for Cistern 242. During the removal of the secondary coolant sump, underground piping and electrical conduit between this region and the reactor building will be excavated, surveyed, and appropriately disposed of. The sump contents, sump, and underground piping are not expected to be contaminated.

Figure 3-3. Photograph of the secondary coolant pumps and pad (The secondary sump is beneath the pad).

3.4 Reactor Building Decontamination

During reactor building decontamination, components will be removed and contaminated sections separated from uncontaminated sections. In no case will the facility containment be broken until it has been demonstrated that the facility has been decontaminated to releasable levels.

3.4.1 Pool Internals and Annulus

To minimize the radiation fields to be experienced by decommissioning workers in the containment building, the first step to be taken should be the removal of loose materials in the annulus, the core support structure, and beam tube ends in the reactor pool area (in that order). Because of the high radiation fields associated with the annulus materials and core support structure, the contractor shall use remote cutting and handling equipment during this removal to minimize worker radiation exposure. Immediately after removal, the items with high radiation fields shall be appropriately shielded during packaging for disposal.

The deluge system piping will be removed from the reactor pool area to provide access to the equipment within the pool area.

The beam tube ends will be removed by cutting off the portions that protrude into the pool. These pieces will be placed in waste containers and staged for disposal.

Instrumentation, remote operating equipment, access ladder, and manway cover to the lower pool access opening will also be removed as part of this task.

The last components to be removed from the pool will be the doors that connect the annulus to the reactor pool. These will also be placed in waste containers.

3.4.2 Pool Liner

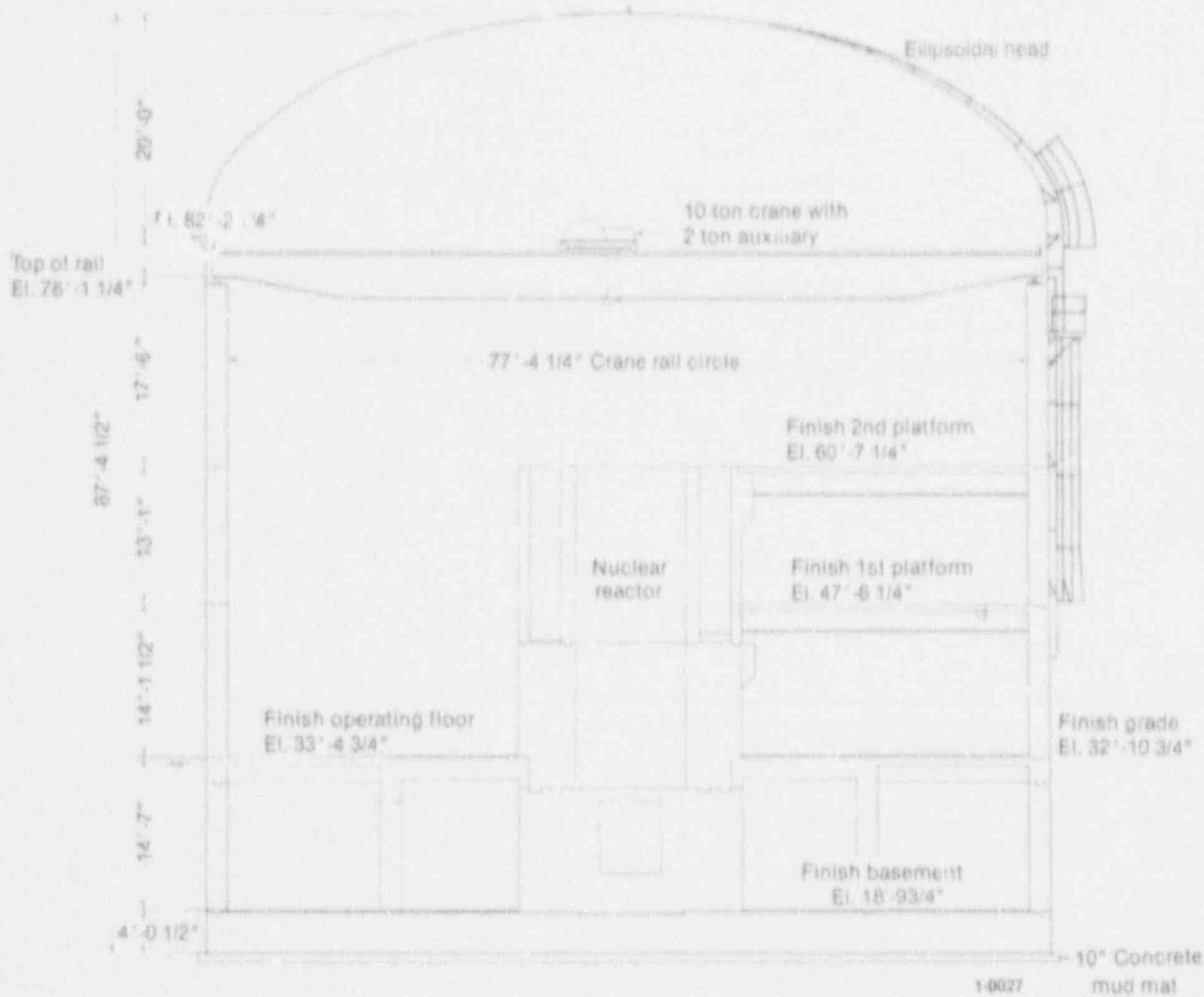
The stainless steel pool liner shall be removed following the removal of the pool internals. The liner will be torch cut or mechanically cut and sized for decontamination and/or disposal. If stainless steel is torch cut, air supplied respirators will be required because of generation of nickel carbonyl.

During the early efforts at alleviating leakage from the reactor pool, fiberglass materials were used on the inside walls of the pool. No reference is made in any of the available documentation to removing the fiberglass materials prior to installing the stainless steel liner. Therefore, it should be assumed that the fiberglass liner material remains between the stainless steel and the concrete walls of the pool. Extra respirator requirements will be enforced during removal of the stainless steel liner to prevent the inhalation of fibers during cutting operations. Should the fiberglass be on the pool walls, this task will also include removal and disposal of such fiberglass materials.

3.4.3 Platforms

The platforms surrounding the reactor pool monolith shall be removed after removing the pool internals. A cross section of the reactor building is shown in Figure 3-4.

The first step in this phase shall be the removal of insulating materials and lighting fixtures from the interior of the ellipsoidal head of the containment. The removals will be accomplished by constructing scaffolding on the bridge crane to gain access to the underside of the head. Access to the crane is by way of the roof of the control room. The next step would be the removal of the control room structure and related equipment on the second platform shown in Figure 3-5. Care must be taken to ensure that electrical power supplying the areas being worked on have been deenergized prior to removing electrical equipment and fixtures. Remote operating tools, stored on



Containment Cross Section

Not to scale

Figure 3-4. Cross section of the AMTL Reactor building.

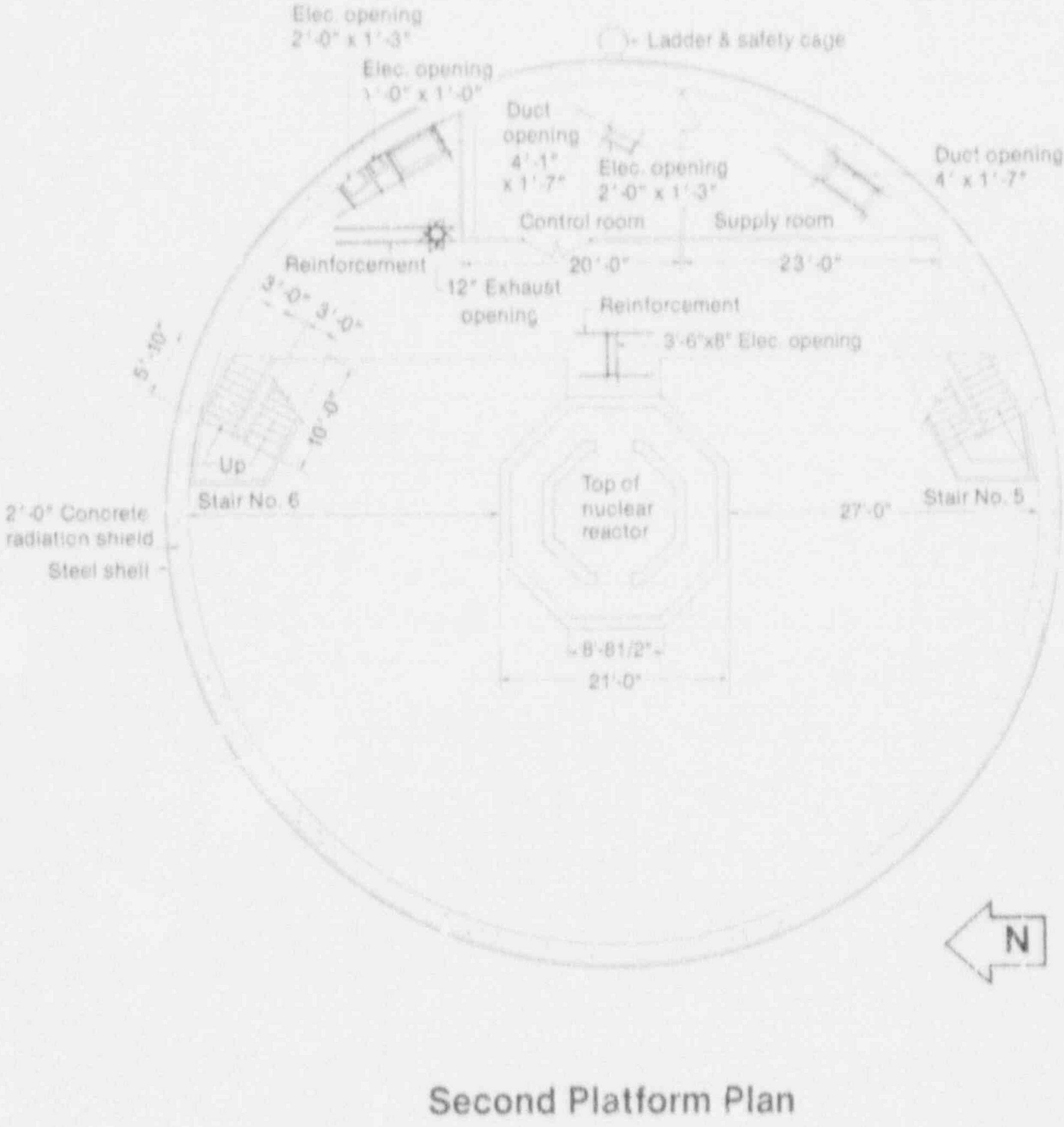


Figure 3-5. Second platform in the AMTL Reactor building.

the second platform railing, and the HVAC equipment and ducting will be removed as part of this task. The stairway connecting the first and second platforms will be removed prior to removing the second platform decking.

As HVAC and electrical equipment are removed, temporary ventilation, electrical power, and lighting must be provided to ensure a safe working environment during dismantling. The temporary services will also include emergency lighting in the event of a power failure.

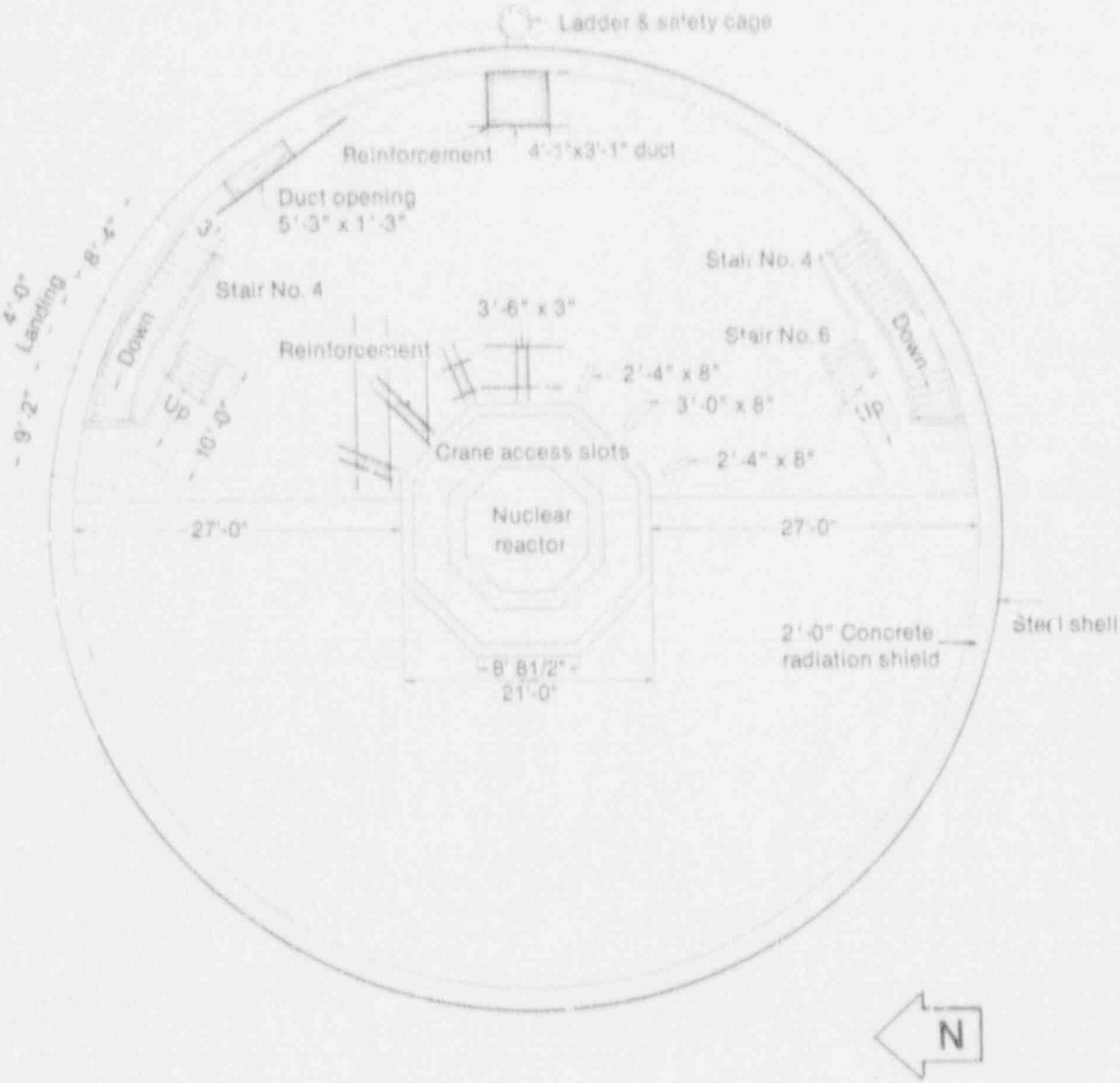
The decking of the second platform will be demolished using standard industry-accepted techniques. Care must be taken when working in the area adjacent to the pool monolith and the containment wall so that these structures are not damaged until they are scheduled for removal. The supports for the second platform will also be removed at this time.

<u>NOTE:</u> At this point, the contractor may elect to integrate the demolition of the pool with the removal of the platforms. That is, after the removal of the second platform, the contractor could initiate the demolition of the reactor pool structure down to the level of the first platform then remove the first platform and another section of pool structure continuing down to the basement level in a similar manner.

The first platform, shown in Figure 3-6, will be removed in a similar manner as the second platform, as explained above. The conduits, piping, ductwork, and stairway will be removed first, followed by removal of the decking and supports down to the level of the main floor.

3.4.4 Basement Electrical

At the point in the project where all planned usage of the installed electrical system has passed, the electrical distribution system located in the basement can be deenergized and the equipment removed. This removal will include the motor control centers, distribution boxes, conduit, and other installed equipment.



First Platform Plan

Not to scale

1-0022

Figure 3-6. First platform in the AMTL Reactor building.

Caution should be exercised to ensure that all circuits are deenergized, including shorting out all capacitors prior to starting removal activities.

3.4.5 Basement Piping and Equipment

The basement contains HVAC, water conditioning equipment, coolant piping and components, barricades, and other miscellaneous equipment. These items will be removed from the facility in a convenient order, with the coolant related equipment being removed last to minimize the potential for contamination of materials. As in dismantling other piping systems, steps must be taken to ensure that systems are empty and not pressurized.

The basement equipment can be hoisted to the main floor through either of the 7- by 7-ft removable hatches in the main floor (see Figure 3-7).

The coolant equipment enclosure walls (Figure 3-8) should be removed prior to removing equipment in order to facilitate access to the equipment.

3.4.6 Basement Sumps

The two basement dry sumps (Figure 3-8) shall be removed by cutting the steel liners away from the concrete. The concrete will then be surveyed and chipped away as needed to remove contaminated materials.

The floor drains and connecting lines shall be chipped out from the drain inlet to where the lines enter the sump.

The sump pump shall be removed, the sump liner cut away from the concrete, and concrete removed, as needed, and materials disposed of as surveys and analyses dictate.

Openings made into the floor system shall be covered and/or barricaded to prevent personnel injury from falls.

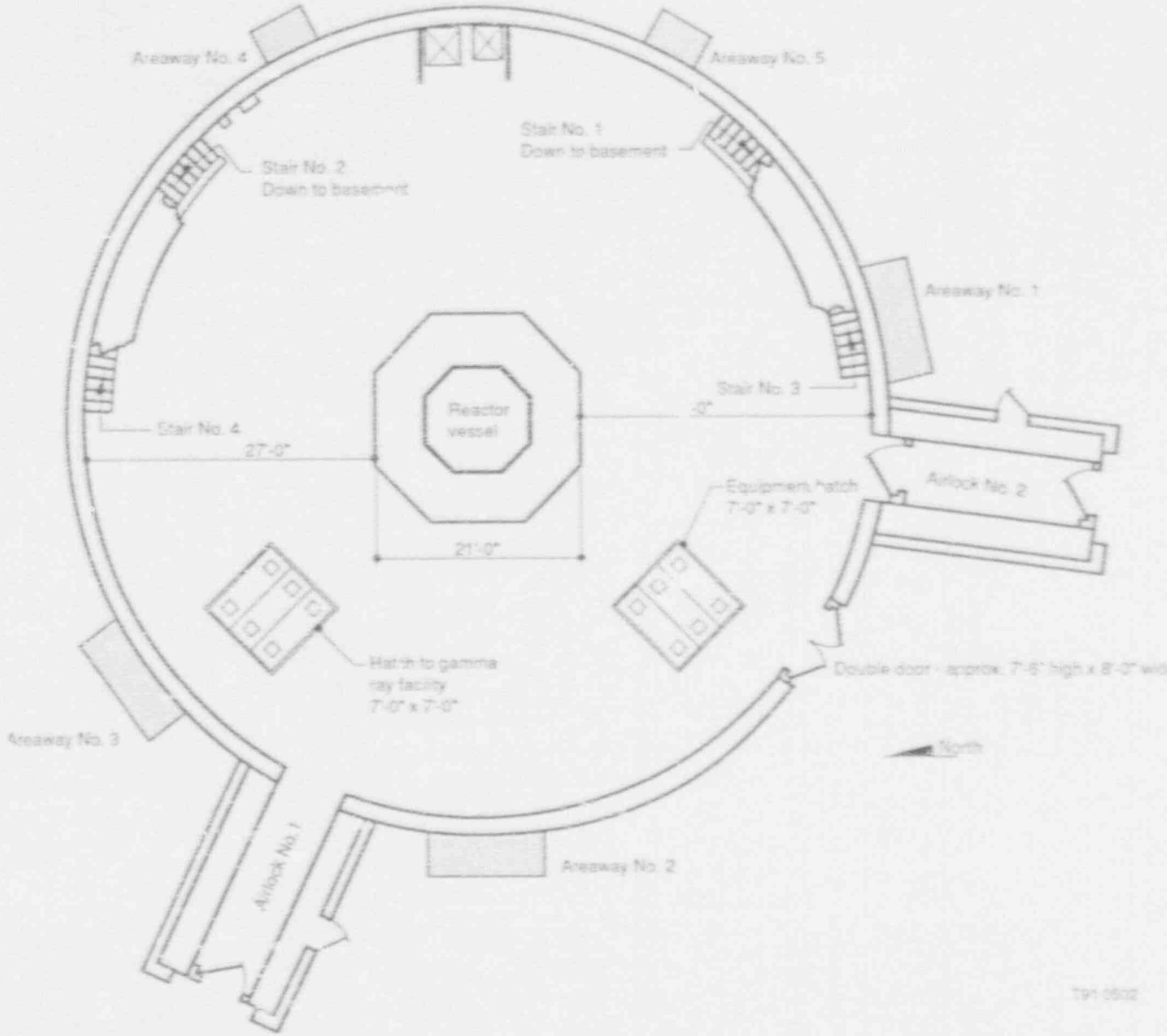
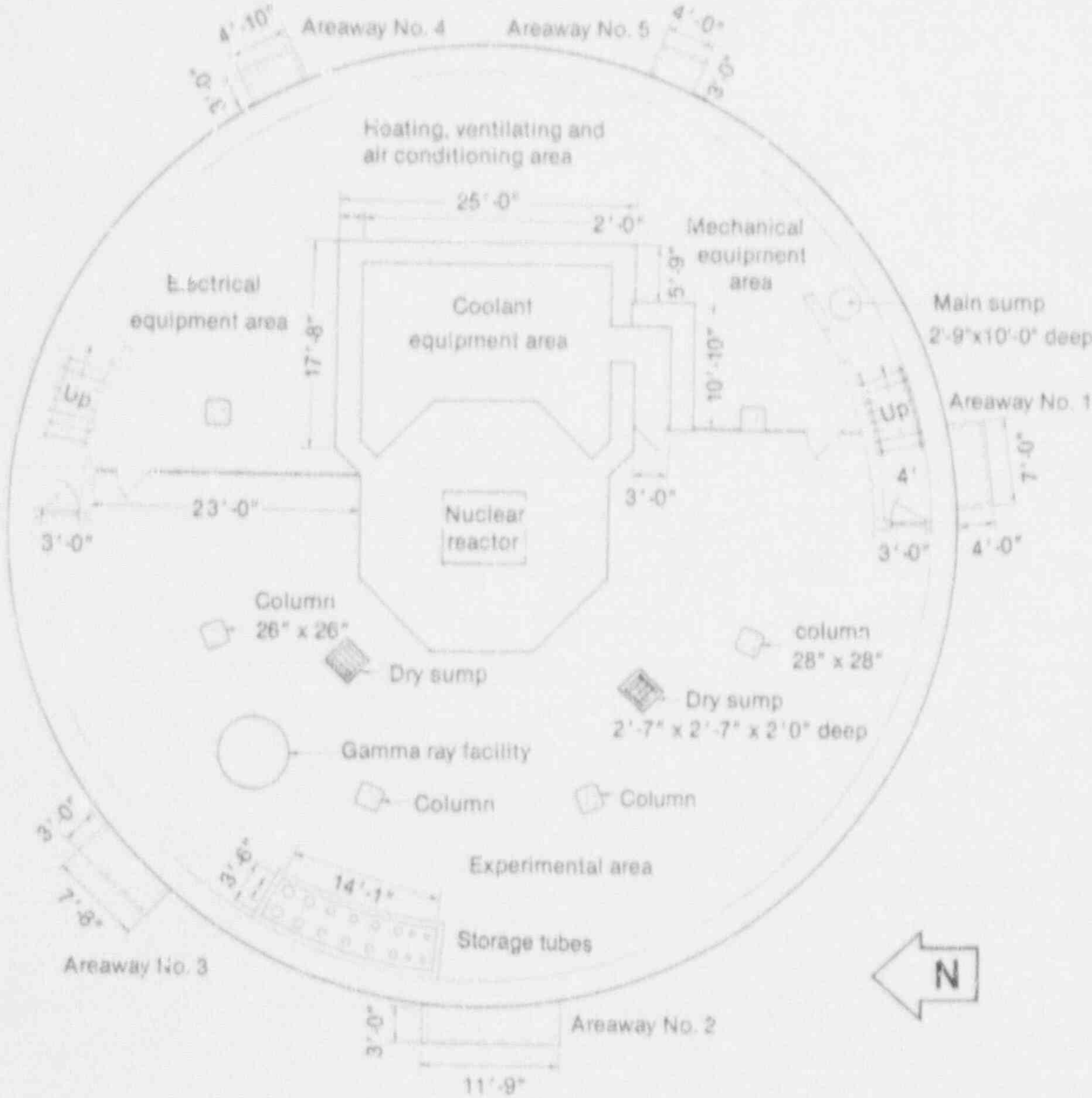


Figure 3-7. Plan view of the AMTL Reactor main floor.



Basement Floor Plan

Not to scale

1-0025

Figure 3-8. Plan view of the AMTL Reactor basement floor.

3.4.7 Gamma Facility and Storage Tubes

The gamma-ray facility (Figure 3-8) steel liner and storage rack shall be cut away from the concrete and the concrete surveyed and scabbled as needed to remove contamination. The storage tubes (Figure 3-8) shall be removed using coring, cutting, or other methods chosen by the contractor to separate the tubes individually or in groups from the basement floor.

The openings resulting from removal of the storage tubes shall be temporarily covered to prevent personnel injuries.

3.4.8 Pool

If not phased in earlier, the pool demolition is the next step after the basement has been decontaminated and all materials removed. If pool demolition was begun earlier, progress can now be made beyond the main floor.

During pool demolition, the operating floor shall also be removed back to the supporting columns.

Pool structural materials are very likely to be contaminated in the area above the reactor core and for a short distance below the core area. There is also a high probability of contamination in the hold-up tank below the reactor pool. Extra care, such as the use of temporary work enclosures, must be taken during this phase to avoid unnecessary exposure to workers and to prevent the spread of contamination.

Demolition of the pool monolith can be accomplished utilizing chipping techniques, sawing, drilling and wedging or other methods proposed by the contractor and approved by the Army. In any case, the methodology to be used in this phase must be approved by the Army before implementation to ensure that the method will minimize the potential for spread of airborne contamination.

Approximately 51 ft^3 of lead is present in the lining of the floor storage pits and boxes in the annulus as well as between the horizontal beam tubes and the pool wall. This lead will be surveyed for radioactive contamination. Contaminated lead will be packaged and delivered to the licensee for storage as mixed waste. Uncontaminated lead will be salvaged for reuse. If the lead has removable contamination, the lead shall be decontaminated and salvaged for reuse.

Specific attention shall be given to checking the rebar removed from the pool structure for the presence of contamination and/or activation.

3.4.9 Building Internal Decontamination

Following removal of building internal components as previously described, all that will remain will be the building shell, the concrete wall, basement floor, operating floor from the wall to the supporting columns, and the airlock between the two buildings. A survey of the building interior shall be performed, and if radioactivity above releasable limits as defined in Reference 4 is detected during this survey, decontamination efforts, including scabbling, will be used to remove the contamination. This survey and subsequent decontamination (if decontamination is required) are necessary to ensure that there is no residual radioactive contamination above releasable limits. The survey of the building interior will be performed in accordance with a written procedure prepared by the decommissioning contractor and approved by the licensee. The procedure will be based on the radiological contamination encountered during the performance of tasks described in Chapter 3, Section 3.4, and the procedure will reflect the applicable parts of Reference 2.

3.5 Termination Survey

The termination survey/characterization methodology is described in detail in Chapter 8 but is briefly summarized here. The termination survey will consist of two phases, separated by backfilling and grading the trenches

and pits formed when piping and tanks are excavated. Each phase will be performed by the decommissioning contractor with a follow-on verification survey to be performed by the U.S. Army Environmental Hygiene Agency (USAEHA).

3.5.1 Phase I

At the conclusion of removing Cistern 242, the secondary coolant system, and associated/piping, there will be several trenches and holes in the terrain around the reactor building. These excavated areas shall be radiologically characterized to demonstrate to the NRC that the residual contamination levels meet NRC release criteria. This phase of the termination survey must be performed prior to backfilling the holes created by excavation in order to prevent covering potential contamination with several feet of soil.

In addition to soil characterization during Phase I of the termination survey, the entire inside surface of the reactor building shall be gridded and characterized. This characterization will consist of direct radiation measurements and analysis of smears and other samples collected after completion of decontamination as described in Section 3.4.

3.5.2 Phase II

After backfilling the excavated trenches and holes and grading the site, Phase II of the termination survey shall be performed as specified in Chapter 8.

3.6 BACKFILL AND GRADE

Following approval of Phase I of the final survey, the trenches and holes caused by excavation will be backfilled and graded. Backfill soil will consist of excavated soil.

3.7 Estimated Schedule and Cost

Figure 3-9 is a critical path method (CPM) network showing the major project activities to be performed. These include activities to be performed by the Army as well as activities to be performed by the decommissioning contractor. The CPM network shows required sequencing of activities and also shows activities that may be performed in parallel. Estimated duration for activities is shown in weeks above each activity line.

The estimated cost of the project is $5.1 million assuming a radioactive waste void volume of 60%, and $4.3 million assuming a void volume of 10%. The actual void volume will depend on the efforts by the decommissioning contractor to minimize void volume. A breakdown of this estimate is given in Table 3-1. The cost estimates are based on partial dismantlement, variations 1-B and 1-D, described in Reference 1. These variations assume a void volume of 10% and 60% respectively in the packaged radioactive waste. Also, it is assumed that the radioactive waste would be disposed of after 1 January 1992, but before 1 January 1993. Disposal of the waste during that period requires a state penalty surcharge of $120 per cubic foot for low-level radioactive waste.

3.8 Funding

Funding for decommissioning the AMTL Reactor will come from appropriate U.S. Army sources. The closure action at AMTL and the funds required for this action are mandated under Public Law 100-528.

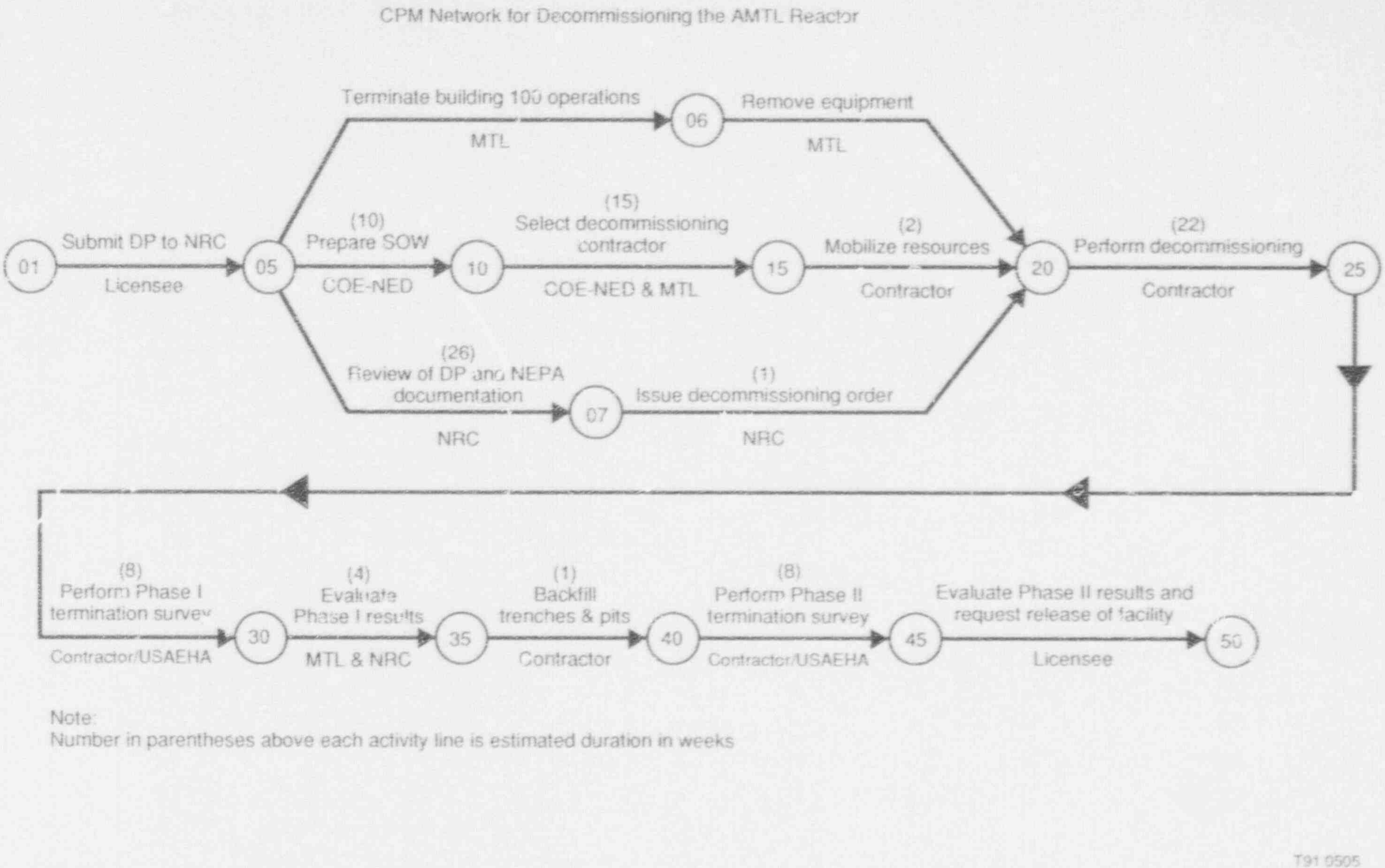


Figure 3-9. Critical Path Method (CPM) network for the AMTL Reactor decommissioning.

Table 3-1. Cost estimate summary for decommissioning the AMTL Reactor.

Project Activity and WBS Number		Estimated Costs	
		($K)[a]	($K)[b]
Site Preparation	3.1	6[c]	6[c]
Project Management	3.2	378	378
Auxiliary Structures Removal	3.3	25	25
Reactor Bldg. Decontamination	3.4	3300[d]	3994[e]
Termination Survey	3.5	108	108
Backfill and Grade	3.6	22	22
	Subtotal	3839[f]	4533[f]
Contingency (13%)		499	589
	Total	4338	5122

a. This cost estimate is based on 10% void volume for radioactive waste.

b. This cost estimate is based on 60% void volume for radioactive waste.

c. The cost estimate for site preparation covers only the tasks to be performed by the decommissioning contractor described in Section 3.1, and does not cover tasks to be performed by the Army.

d. This cost estimate includes $960K for the state penalty surcharge for the disposal of 8,000 cubic feet of low-level radioactive waste generated in Massachusetts.

e. This cost estimate includes $1440K for the state penalty surcharge for the disposal of 12,000 cubic feet of low-level radioactive waste generated in Massachusetts.

f. Subtotal includes $422K of escalation to an activity midpoint of September 1992.

CHAPTER 4

SECURITY

4. INTRODUCTION

The nuclear fuel was previously removed from the AMTL Reactor and shipped off site. Therefore, there is no requirement for safeguarding special nuclear material.

4.1 PHYSICAL SECURITY

During decontamination of the AMTL Reactor as specified in Section 3.4, access to the inside of the reactor building will be controlled, using the personnel entrance between Building 97 and the reactor building. An appropriate barrier will be installed in the airlock. Equipment and material will be moved in and out of the reactor building using the double door shown in Figure 3-7.

During decommissioning as specified in Section 3, access to the site will be controlled using standard construction practices such as warning signs, fencing off the area, and installing physical barricades around excavations. Access will be limited to the construction site as required by the NRC for radiation areas, high radiation areas, and areas containing radioactive material as specified in 10 CFR 20.

CHAPTER 5

RADIOLOGICAL ACCIDENT ANALYSIS

A Radiological Accident Analysis is not required for this DP because there is no nuclear fuel at the AMTL Reactor site. Potential radiological accidents, during decommissioning activities of the facility, are discussed in Sections 3.3 and 3.4.

CHAPTER 6
RADIOACTIVE MATERIALS AND WASTE MANAGEMENT

6. INTRODUCTION

During decommissioning activities, radioactive wastes in liquid, solid, and particulate forms are expected to be generated. Planning the management of these wastes is an integral part of the DP. Provisions for minimizing the amount of waste generated, and waste collection, treatment, packaging, and shipment off site for disposal are discussed in the following sections.

6.1 FUEL DISPOSAL

The fuel was removed from the AMTL Reactor during deactivation in the early 1970s.

6.2 LIQUID RADIOACTIVE WASTE

Liquid radioactive wastes generated during decommissioning activities will be collected, monitored, and released to the sanitary sewage system if the conditions of 10 CFR 20.303 can be met. The project manager will contact the Massachusetts Department of Environmental Protection (DEP), Massachusetts Water Resource Authority (MWRA), and the NRC in order to obtain all permits and certificates required to discharge these wastes into the local sanitary or storm sewer systems. Contaminated water will be treated to remove radioactive contamination, or processed for disposal as low-level radioactive waste. Discharge shall be in accordance with 10 CFR 20 and the DEP and MWRA for waste water. Discharge must be approved by HQ AMCCOM.

Possible sources of liquid radioactive waste are:

- Decontamination of components and parts

- Personnel decontamination liquids

- Decontamination of structures and floors

- Residual liquid in piping and other components.

Efforts will be made throughout decommissioning activities to minimize generation of liquid waste. Whenever possible, scrubbing with swabs will be used instead of spraying. The water mist used during the demolition of activated concrete will be closely controlled, and approved liquid absorbers will be used around the floor to absorb any runoffs.

Liquid waste will be absorbed or solidified using absorbent materials or solidification agents specified in the license requirements of the radioactive waste disposal site. Absorbent material will be provided to absorb at least twice the volume of radioactive liquid contents in all radioactive waste packages.

6.3 Solid Radioactive Waste

The solid radioactive wastes generated during decommissioning activities will be packaged on site in containers suitable for shipping and disposal at an NRC approved disposal site in the United States. Packaging requirements shall be in accordance with the latest Federal Regulations, and the packaging requirements shall also comply with the requirements of the NRC approved disposal site chosen.

6.3.1 Packaging

The types of solid radioactive waste to be packaged include the following:

- Demolition Materials--These include all contaminated and/or activated systems, components, and equipment removed during

dismantlement of the AMTL Reactor facility described in Section 1.1.1. Packaging these materials will be specified in work procedures written by the contractor and approved by the Army.

- Equipment and Tools--This includes such items as saws, jackhammers, forklift, shovels, pumps, tanks, ventilation system components, filters, and piping. Not all of this equipment is expected to be discarded as radioactive waste. A determination of volumes of solid radioactive waste generated from this category will be possible only during cleanup, when measurement of contamination level and evaluation for decontamination will be made.

- Auxiliary Materials and Clothing--This includes confinement barrier plastic sheets, protective mats, rags, work platforms, and protective clothing. It is assumed that these materials will be compacted and packaged in 55-gal drums if contaminated.

6.3.2 Temporary Storage of Radioactive Waste

Containerized waste shall be transported to the chosen disposal site in a timely manner to avoid accumulation of waste. However, in the event that packaged waste must be temporarily stored, it will be stored in a protected area near the reactor facility.

6.4 VENTILATION SYSTEM

Ventilation exhaust systems will be equipped with roughing filters to capture large particles, and with high-efficiency particulate absorption (HEPA) filters to provide up to 99.99% particulate retention. HEPA filters will be changed in the event of high radiation level readings/alarm in the exhaust duct, or based on maximum pressure differential readings indicating that the filters are filled with dust. Radiation monitoring will be provided for the exhaust air to atmosphere; readings above prescribed limits will shut off the exhaust fans. Ventilation units serving confined enclosures will be

mobile type and connected with flexible ducts to the enclosure and to the exhaust duct. The exhaust duct will be provided with gravity louvers, which will automatically close in the event of failure of the ventilation exhaust unit.

6.5 Waste Classification

The criteria for waste classification for low-level waste disposal are contained in 10 CFR 61.

It is concluded that most all the radioactive wastes from this project can be classified as Class A unstable waste, and will be segregated at the disposal facility. Items that can be compacted shall be sent to the Defense Consolidation Facility.

6.6 Shipping Radioactive Wastes

Unless otherwise specified, it is assumed that most shipments will be low-specific activity (LSA) and will be shipped in exclusive use vehicles.

Department of Transportation (DOT) Regulation 49 CFR 173 provides radiation level limitations for transportation of packages of radioactive materials in closed, exclusive-use transport vehicles as follows:

- External radiation levels must not exceed 1000 mrem/h on the accessible surface of the package, if the shipment is made in a closed transport vehicle, the package are secured, and there is no loading or unloading operations during transit (49 CFR 173.441)

- The radiation level must not exceed 200 mrem/h at any point on the outer surface of the transport vehicle, 10 mrem/h two meters from the vehicle sides and 2 mrem/h in the tractor cab (49 CFR 173.441).

A quality control program shall be established by the decommissioning contractor to ensure that radioactive waste shipping regulations are enforced. The decommissioning contractor shall have the appropriate DOT certification. The decommissioning contractor's quality control program shall include the following requirements:

- Waste Containers

    - All Type A containers or Type B containers if the A quantity is exceeded are DOT specification 7A (49 CFR 173.425, 178.350, and 173.24). Exemptions to this must be specified. Most containers for this project will be strong, tight containers, not Type A or B. This assumes LSA exclusive use vehicle shipments.

    - All containers are in good physical condition, with no evidence of damage, corrosion, or leakage (49 CFR 173.475 and 173.24).

    - All metal drums with a capacity of 55-gal or greater will have a 5/8-in. or larger bolt for securing the closure device (ring assembly). All metal containers will have an intact heavy-duty closure device with gasket when presented for disposal.

    - Ring bolt is torqued to approximately 45 ft pounds (recommended).

    - Drum lids are a proper fit, and bungs (if any) are tight (49 CFR 173.475).

    - Radiation levels at the package surface are limited to those specified in 49 CFR 172.403 for packages requiring Yellow-II or White-I labels and to those specified in 49 CFR 173.441 for all other packages.

- Surface contamination levels are below DOT limits (49 CFR 173.443) 220 alpha dpm/100 cm^2 and 2200 beta-gamma dpm/100 cm^2.

• Labels and Markings (if not excluded because of exclusive-use vehicle exceptions)

- Each container ha [illegible] tive labels affixed to it (49 CFR 172.403 a [illegible] 05).

- Each container has b [illegible] tive Material, LSA, n.o.s. UN 2912" in grea [illegible] high letters (49 CFR 172.101, 172.301).

- Each container has been marked "USA DOT 7A, Type A" or "DOT Type B" in 1/2-in.-high letters (49 CFR 178.350 and 172.310).

- The waste class has been marked on each container using greater than 1/2-in.-high letters (10 CFR 61.57).

<u>Waste Class Example</u>

Class A Unstable
Class A Stable
Class B
Class C

- For each container in excess of 110 lbs, the weight and unit of measurement have been marked on the container (49 CFR 172.310).

- Markings must be durable and legible, and displayed on a background of sharply contrasting color, unobscured, and located away from any other marking, such as advertising, (49 CFR 172.304).

- The name and address of the shipper has been attached to each container if vehicular transfers are involved (49 CFR 172.306).

- Transport Vehicle

  - The total transport index number does not exceed 50 (49 CFR 177.842). This is not applicable to exclusive-use vehicle shipments.

  - The containers have been loaded, blocked, and braced so that they cannot change position during conditions normally incident to transportation (49 CFR 177.425 and 177.842d).

- Exclusive-Use Vehicle Exceptions

  - Specific instructions for maintenance of exclusive-use shipment controls must be provided by the shipper to the carrier and be included with the shipping paper information (49 CFR 173.425 b9).

  - An exclusive-use vehicle shipment of Type A Low Specific Activity (LSA) radioactive material (RAM) is exempt from Type A packaging specifications if it meets a strong, tight packaging criterion (49 CFR 173.24).

  - Shipments must be loaded by consignor and unloaded by consignee. The consignor shall be the U.S. Army Armament, Munitions, and Chemical Command (AMCCOM) representative who has the authority to approve packaging and shipping radioactive waste from this project. The AMCCOM representative will be onsite full time.

  - There must be no loose radioactive material in the conveyance.

- Shipment must be braced so as to prevent shifting of lading under conditions normally incident to transportation (49 CFR 173.425).

- For shipments of LSA radioactive materials or shipments containing packages bearing Radioactive Yellow-III labels, the transport vehicle shall be placarded in accordance with Table I of 49 CFR 172.504.

- Exclusive-use vehicle shipments of LSA RAM are exempt from specified markings and labeling if the exterior of each package is stenciled or otherwise marked "Radioactive - LSA."

  All DOT regulations listed in this section are current regulations. The most current, applicable regulations shall be used when the shipments are made.

• Documentation

The generator has a valid disposal site and (where required) state User Permit.

- All required documents are fully completed and are legible.

- The Radioactive Shipment Manifest (RSM), formerly RSR/Manifest, is complete in all details. (49 CFR 172.200 Subpart C, 10 CFR 20.311, and radioactive waste disposal site requirements.)

- The number of containers listed on the RSM agrees with the physical count of containers loaded.

- All required certifications, RSMs, and other documents as appropriate are signed.

- The use of abbreviations must conform to DOT and NRC specifications.

- Shipping Routes

The route out . AMTL to the NRC approved disposal facility shall be chosen using the PC Miler Program, which calculates the shortest highway routes. Local, state, and federal ordinances may impact the routes selected to the chosen disposal site. Currently, the cost estimate is based on shipping the radioactive waste.

If shipments are made to Barnwell, prior notification must be sent to the State of South Carolina for each shipment. The prior notification must be received at least 72 hours before the shipment arrives at the Barnwell disposal facility.

CHAPTER 7

TECHNICAL AND ENVIRONMENTAL SPECIFICATIONS

7. INTRODUCTION

Technical and environmental specifications will be implemented by the decommissioning contractor to control conditions, parameters, and variables so that during decommissioning activities the radiation exposure to workers and the public shall be maintained as low as reasonably achievable.

The technical and environmental specifications will include items in the following categories:

- Health and safety limits
- Surveillance requirements
- Administrative controls
- Design features.

7.1 Health and Safety Limits

Radiation and industrial exposures to workers and the public must be ALARA and less than regulatory limits. Under no circumstances will the exposures exceed regulatory limits as specified in 10 CFR 20.

During decommissioning activities the following limits shall be enforced.

7.1.1 External Exposure

External radiation exposure for individuals in restricted areas during decommissioning shall not exceed the limits specified in 10 CFR 20.101.

7.1.2 Internal Exposure

Internal radiation exposure from inhalation of radioactive material in air in restricted areas shall not exceed that which would result from the inhalation of the limiting quantities specified in 10 CFR 20.103. This will be verified by performing a baseline bioassay prior to assigning contract employees to the restricted area. After the decommissioning has been completed, a termination bioassay will be performed for all individuals potentially exposed to airborne radioactivity.

7.1.3 Concentration of Airborne Radioactive Material in Restricted Areas

Concentration of airborne radioactive material in restricted areas shall not exceed the limits set in 10 CFR 20.103.

7.1.4 Concentration of Airborne Radioactive Material in Unrestricted Areas

Concentration of airborne radioactive material in unrestricted areas shall not exceed the limits specified in 10 CFR 20.106.

7.1.5 Concentration of Nonradioactive Substances in Restricted Areas

Concentrations of nonradioactive substances in restricted areas shall not exceed limits established by the American Conference of Governmental Industrial Hygienists (ACGIH) as listed in "Threshold Limit Values (TLV) and Biological Exposure Indices for 1987-1988."

7.1.6 Concentration of Nonradioactive Substances in Unrestricted Areas

Concentration of nonradioactive substances in unrestricted areas shall not exceed 1/30 the ACGIH TLV limits.

7.1.7 Noise Levels

Noise emissions shall be in compliance with Title 310 of Code of Massachusetts Regulation 310 CMR 7.10 and Town of WaterTown Ordinances

Chapter VI.I, Section 32. Protection against the effects of noise exposure to workers shall be in compliance with U.S. Army Corps of Engineers Safety and Health Manual, EM 385-1-1.

7.1.8 As Low as Reasonably Achievable (ALARA)

The AMTL Commander is committed to the concept of ALARA in terms of both individual and collective doses. Administrative controls, training, protective devices, and measures will be used to achieve this commitment to ALARA and minimize the hazards to health and safety. The ALARA program shall comply with ASTM E 1167, Standard Guide for Radiation Protection Program for Decommissioning Operations.

7.2 SURVEILLANCE REQUIREMENTS

Surveillance activities to be conducted during decommissioning are described below.

7.2.1 The Dosimeter Program

A dosimetry program will be developed and implemented by the decommissioning contractor in compliance with state and Federal requirements. Vendors supplying dosimeters shall have NAVLAP accreditation in compliance with the 10 CFR 20.

7.2.2 The Routine Swipe Program

All material, components, and equipment removed during decontamination will be swiped to determine their disposition relative to removable contamination. The swipe program will be specified in procedures prepared by the contractor, and must meet measurement requirements to detect removable surface contamination at the most restrictive levels specified in Reference 4.

7.2.3 The Routine Instrument Survey Program

Direct radiation measurements will also be performed on all material, components, and equipment removed during decontamination to determine their disposition relative to fixed contamination or activation. Details of the survey program will be specified in procedures prepared by the contractor and must meet measurement requirements to detect surface contamination at the most restrictive levels specified in Reference 4.

7.2.4 The Air Sampling and Monitoring Program

Continuous air monitors will be located outside the reactor enclosure. Monitoring inside the enclosure will be conducted for personnel exposure under the Health and Safety Program. Details of this program will also be specified in procedures prepared by the contractor. The procedures shall comply with ASTM E 1167, Standard Guide for Radiation Protection Program for Decommissioning Operations.

All instruments/systems used for monitoring purposes shall be properly calibrated prior to use in compliance with ANSI N 323, Radiation Protection Instrumentation Test and Calibration.

7.3 Administrative Controls

The administrative controls that will be used during decommissioning operations are briefly discussed below as are the responsible organization and documentation requirements.

7.3.1 Administrative Controls During Decontamination

Administrative controls during decontamination are the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to ensure completion of the decommissioning in a safe manner. Administrative controls shall be in Compliance with ASTM E 1167.

7.3.2 Responsibility

The licensee shall have overall responsibility for the completion of decommissioning of the AMTL Reactor. However, the contractor's Project Manager will be delegated the responsibility for decommissioning via the contract.

7.3.3 Organization

The organizational structure for management and performance of the decommissioning activities is shown in Figure 1-22. The functions and responsibilities, and minimum required qualifications and experience of each key position are detailed in Section 1.9. Qualifications of decommissioning personnel shall comply with ANSI 3.1, Standard for Selection, Qualification, and Training of Personnel for Nuclear Power Plants.

7.3.4 Records and Reports

Accurate and complete records and reports shall be maintained by the contractor in compliance with 10 CFR 20 and other applicable regulations. The records and reports will cover performance and completion of all activities that may have resulted in exposure of workers or the public to radiation or other hazardous/toxic materials.

- Records

  Records shall be maintained in compliance with ANSI 15.10. Records to be maintained during decommissioning shall include those listed below.

  Health and Safety Related Activities:

  - Work permits
  - Work procedures
  - Radiation survey reports
  - Contamination survey reports

- Airborne survey reports
- Environmental survey reports
- Counting data on air samples, smears, and gamma spectrum analysis
- Instrument calibrations
- Source inventory and storage
- Radioactive material inventory and storage
- Shipment records
- Waste disposal - surveys and records
- Package certifications/records
- Incidents and accidents
- Confined space entry permits
- Continuous monitoring records for oxygen deficient/explosive atmospheres.

Personnel Records

- Bioassay analysis
- Personnel exposure records
- Individual dosimeter readings as related to daily tasks and work procedures
- Respiratory protection qualifications (medical clearance and fit test)
- Audiogram results
- Training records
- Visitor logs and exposure information.

• Reports

Reports prepared by the decommissioning contractor shall be written and submitted to the US Army COE-NED Contracting Officers representative.

- Review

  Responsibility for review of procedures, practices, and performance shall rest with the appropriate individuals and/or committees specified in Section 1.5.

7.4 Engineering Controls

Confinement barriers and HEPA filtered ventilation systems will be employed during decontamination in Compliance with ASTM E 1167. These systems will provide:

- Maintenance of a negative pressure in confined areas. This will be accomplished by a HEPA filter system in conjunction with the installed confinement barrier.

- Provisions for dust control during decontamination to minimize airborne particulates.

- Provisions for air monitors at outlet of ventilation systems to turn off ventilation system if excessive levels are measured.

CHAPTER 8

PROPOSED TERMINATION RADIATION SURVEY PLAN

8. INTRODUCTION

The termination radiation survey (characterization) of the AMTL Reactor and site will be performed in two phases. Each phase will require both the decommissioning contractor and the U.S. Army Environmental Hygiene Agency (USAEHA) to conduct independent surveys. The contractor's survey will be conducted to ensure that no residual radioactivity above releasable levels remains at the facility or site. The USAEHA survey, which will be conducted subsequent to the contractor's survey, will serve as the means of validating the conclusion of the contractor's results. The first phase of the termination survey will include the reactor building interior and the trenches and pits formed during required excavation outside the reactor building. The required excavation is described in Chapter 3. The second phase of characterization will consist of the soil area outside the reactor building and will be performed following backfilling and grading the excavated regions of the site.

This chapter addresses background soil and the two phases of characterization to be performed after partial decontamination. The surveys performed on removed components and material for disposal were described in Chapter 3 and are not included in this chapter.

The determination of radiological background for this project and the performance of each phase of the termination radiation survey will be in accordance with written procedures. The decommissioning contractor and the USAEHA will use acceptable procedures which will be submitted to the licensee for approval.

The decommisioning contractor's procedures will be in accordance with the general criteria of Reference 2 and the specific requirements specified in this chapter. The USAEHA procedures will be based on the verification

inspection criteria of Reference 2. In addition, both sets of procedures will establish data quality objectives to be met by each performing organization.

8.1 Background Soil

Results of soil measurements and analyses during both phases of the termination radiation survey shall be compared with background radiation levels to determine subsequent actions relative to the soil areas. Reliable background soil data are essential and must be collected prior to performing the measurements and analyses of the soil.

Features that contribute to good background data are (1) survey design to provide representative, unbiased sampling, (2) proper allocation of sampling, (3) selection of areas least likely to have been affected by the reactor or other AMTL operations, (4) appropriate instrumentation, and (5) quality ensured analyses.

Soil background locations for this project will be determined by drawing twelve radius lines from the center of the reactor, and extending each radius line out to 3.5 km from the reactor center. Each of the twelve radii will be 30 degrees apart. Next, draw concentric circles with radii of 0.5, 1.5, and 3.0 km, with the center of each circle at the reactor center. The background sample locations will be selected at the intersection of each radius line and concentric circle. If there are obstructions or other reasons preventing accessing soil at those locations, adjust the distance or angle in order to find accessible soil.

Perform a soil contact radiation measurement of alpha, beta, and beta-gamma at each of the 36 locations. Record each measurement in dpm/100 cm^2. Recording the measurement in dpm/100 cm^2 requires converting to dpm/100 cm^2 by knowing the instrument efficiency and the active area of the detector.

In addition, collect a sample from the top 15 cm of soil at each location for analyses. The analyses shall include gross alpha and beta concentrations. Also, gamma spectrometry shall be performed for each sample, and the concentration determined for each detectable radionuclide.

8.2 PHASE I

8.2.1 Reactor Building Interior Survey

All interior surfaces remaining in the reactor building after completion of decommissioning described in Chapter 3 shall be surveyed for fixed and removable alpha and beta-gamma residual contamination.

The interior of the reactor building following decommissioning will consist of the basement floor, the basement walls, the operating floor from the reactor walls out to the supporting columns (the center portion of the operating floor will have been removed during decommissioning), the reactor building walls, and the ellipsoidal ceiling. No neutron activated material will remain in the reactor building following decommissioning. Therefore, no concern need be given to subsurface activation products during the termination survey.

Two degrees of survey intensity shall be used, Level I and Level II. Level I will be the more survey-intensive level and will be performed in the areas most likely to contain residual contamination such as the basement floor, sumps, and pits. Level II intensity surveys will be performed in areas less likely to contain residual radioactivity such as the reactor walls above the operating floor and the ceiling. The areas to be designated Level I and Level II will be determined after completion of decommissioning activities and will be based on the contamination discovered during decommissioning. Level I grid size shall be 1 meter X 1 meter and Level II grid size shall be 3 meter X 3 meter.

The details of the Level I and II surveys will be specified in procedures prepared by the decommissioning contractor. The procedures prepared by the

USAEHA will use the same Level I and II designated areas as the decommissioning contractor. Both sets of procedures will be approved by the licensee.

The objective of the surveys of the AMTL Reactor building interior is to show that residual fixed or removable alpha or beta-gamma contamination is below acceptable levels specified in Reference 4.

The survey procedures shall incorporate principles described in NUREG/CR-2082 and include the following requirements:

- The Level I and II surveys shall include direct radiation measurements to determine alpha and beta-gamma surface contamination in disintegrations per minute (dpm) per 100 cm^2.

- Smears shall be collected and analyzed to determine removable alpha and beta-gamma surface contamination in dpm per 100 cm^2.

- Measurements must be performed using instruments capable of measuring alpha and beta-gamma radioactivity below releasable levels specified in Reference 4 for applicable radionuclides. If radionuclides are not known, the most restrictive levels in Table I of Reference 4 shall be used.

- In addition to measurements to determine surface fixed and removable contamination, exposure rate measurements shall be performed in accordance with Reference 4.

8.2.2 Radiological Characterization of Trenches and Pits

Figure 8-1 is a sketch of the AMTL Reactor site after excavation but prior to backfilling and grading. The excavated areas with potential contamination are the cistern pit, the pit where the secondary coolant system sump was previously located, and the piping trenches. In addition, the berms of excavated soil may contain contamination.

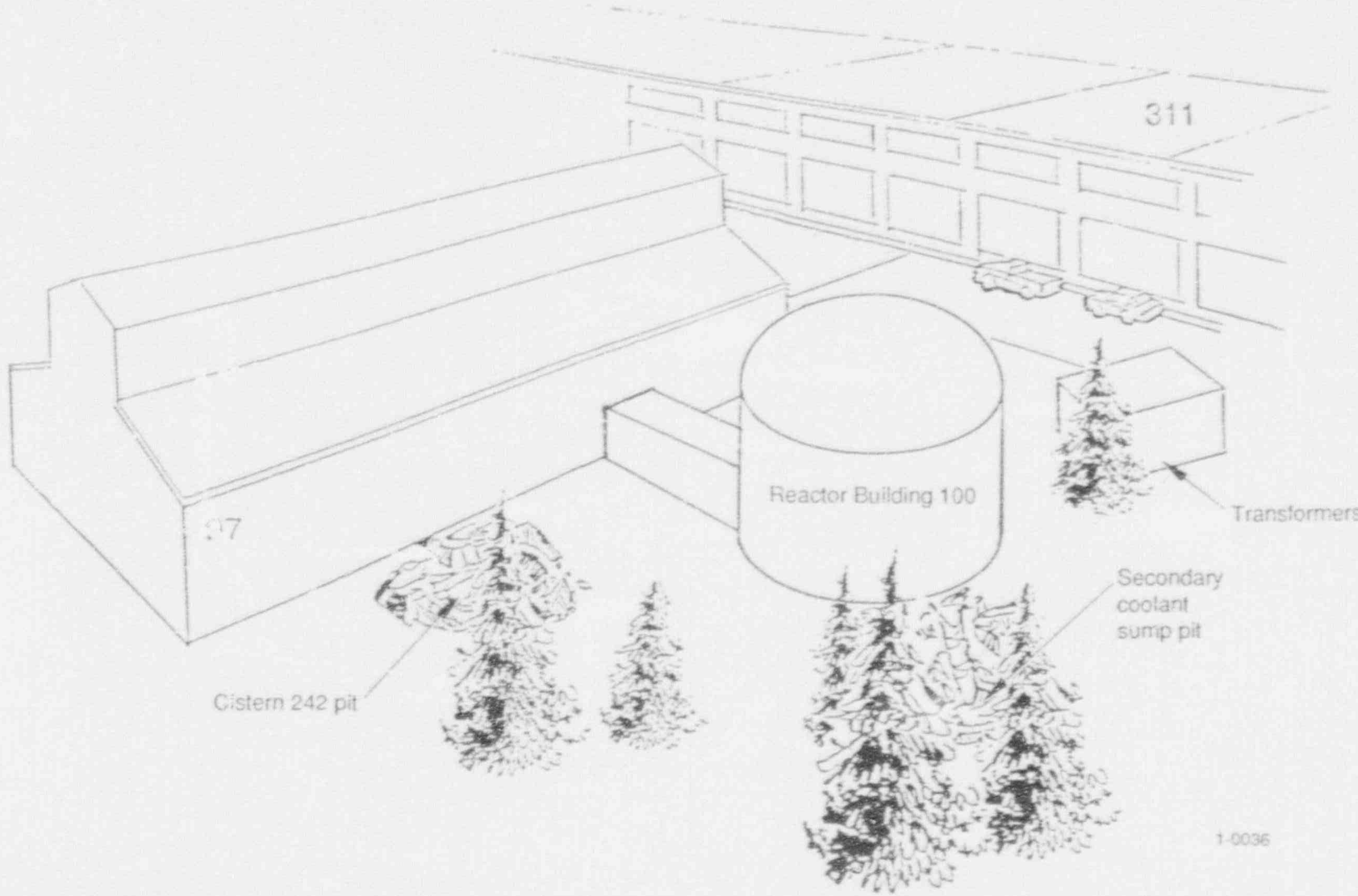


Figure 8-1. Sketch of AMTL Reactor site after completion of decommissioning.

The bottoms of the two excavated pits and trenches will be grided into 1- x 1-m grid squares. The berms of excavated soil will also be grided into 1- x 1-m grid squares.

The Phase I characterization will include soil contact radiation measurements for beta-gamma at each grid intersection and at the center of each grid square. The measurement will be performed by integrating counts for at least 30 seconds, converting the measurement, and recording in dpm/100 cm^2.

Phase I characterization will also include collection of surface soil samples from 50% of all grid squares (randomly selected) in the bottom of each pit and trench. In addition, a soil sample will be collected at depths of 1 ft and 3 ft from 50% of all grid squares (randomly selected) over the berms of excavated soil. These soil samples should adequately represent the excavated soil, pits, and trenches because of the homogenization that takes place during excavation.

Each soil sample will be analyzed for gross alpha and beta concentration and gamma spectrometry. If gross alpha or beta is detected statistically significant above background, the analytical results of that sample will be verified. If results are positively verified, an appropriate alpha or beta isotopic concentration analysis will be performed for each sample exceeding gross alpha or beta in background soil.

8.2.3 Evaluation of Phase I Results

The results of the decommissioning contractor's Phase I survey, including soil analyses, will be summarized in a Phase I Characterization Report. This report will be submitted in draft form to the USAEHA for review. After completing their review, the USAEHA will attach the results of their own Phase I survey as an addendum to the contractor's results and the report will be submitted to the licensee. Backfilling will not be performed until authorized by the licensee. The licensee's decision will be based on the evaluation of the Phase I soil survey results. If either evaluation reveals contaminated soil above releasable levels, the contaminated soil will be removed and that area resurveyed, using the same methodology as specified above. If either

evaluation reveals surface contamination inside the reactor building above acceptable levels, additional decontamination shall be performed and those areas resurveyed.

During the time excavations are open, the licensee will notify the NRC and invite them to perform a confirmatory, independent radiological characterization of the soil prior to backfilling. During the period of time between the completion of excavation and the finalization of the survey results, the excavations will be protected from the intrusion of precipitation and the dispersion of soil particles due to wind erosion.

8.3 Phase II

Phase II characterization will begin after the pits and trenches have been backfilled and graded.

The backfilled and graded areas will be grided into 1- x 1-m grid squares. Beta-gamma soil contact radiation measurements shall be performed at each grid intersection and at the center of each grid square. Each contact measurement shall be made by integrating counts for at least 30 seconds, converting the measurement and recording in dpm/100 cm^2.

In addition, a gamma measurement will be made at 1 meter above the soil surface at each grid intersection and the center of each grid square. This measurement will be recorded in $\mu R/h$.

Surface and subsurface soil samples shall be collected in the center of 50% of all the grid squares (randomly selected). Each surface sample shall be collected from the top 15 cm of soil. Each subsurface sample shall be collected at depths of five, ten, and fifteen feet.

Each soil sample will be analyzed for gross alpha and beta concentration and gamma spectrometry. If statistically significant alpha or beta radioactivity is detected above background and is verified, that soil sample will be appropriately analyzed to identify the existing alpha or beta radioisotopes and determine alpha or beta radioisotopic concentration of each identified radioisotope.

The results of the decommissioning contractor's Phase II survey will be summarized in a Phase II Characterization Report. This report will be submitted in draft form to the USAEHA for review. After completing their review, the USAEHA will attach the results of their own Phase II survey as an addendum to the contractor's results and the report will be submitted to the licensee for evaluation. If either evaluation indicates no residual radioactivity above releasable limits, a Termination Radiation Survey Report shall be prepared as discussed in Section 8.5. If not, additional decontamination shall be performed and the site characterization process repeated for those areas requiring decontamination.

8.4 INSTRUMENTATION

The field instruments to be used during the final survey of the AMTL Reactor site must be capable of measuring alpha and beta-gamma radioactivity at the most restricti[illegible] release levels specified in Reference 4. Instruments will be tested and calibrated in accordance with the specifications contained in the "[illegible]rican National Standard, Radiation Protection Instrumentation Test and Cali[illegible]ion," ANSI N323-1977, or the most recent revision.

Table 8-1 lists a few types of instruments that could be used during final surveys. The detection capabil[illegible]y of each instrument is also shown in Table 8-1.

8.5 DATA DOCUMENTATION

Radiation measurements and analytical results shall include the following data:

- Location of the measurement or sample
- Date or dates of measurements or sample collection
- Measured concentration of the specific nuclides in pCi/g for soil samples

Table 8-1. Typical survey instruments and detection capabilities for measuring soil surfaces.

Nuclide	Instrument or Method	Detection Levels
Gross Alpha	Thin-walled, shielded Pancake G-M	$\geq 10^3$ pCi/cm², very slow scan
	ZnS scintillator	$\geq 10^2$ pCi/cm², source dependent
	Thin NaI or CaF_2	$\geq$ 100 pCi/cm², source dependent
Gross Beta-gamma	Thin-walled, shielded Pancake G-M	500-5,000 pCi/cm², Emax $\geq$ 0.26 MeV
	ZnS scintillator	300 pCi/m², source dependent
	NaI spectrometer	500-5 x 10^5 pCi/cm²
	Ion chamber	$\geq 10^3$ pCi/cm²
	Phoswich	$\geq$ 200 pCi/cm²
	Intrinsic Germanium	$\geq$ 200 pCi/cm²
	Portable scintillator	5,000-50,000 pCi/cm²

- Measurements of radioactivity should be reported as follows: alpha and beta-gamma in dpm/100 cm^2 and gamma at one meter above surface in $\mu R/h$

- Analytical error at 95% confidence level should be reported for all analyses

- Name of surveyor, sampler, or analyst

- Analysis date

- Instrument specifications and calibration data

- Confidence level, standard error, etc. attached to analytical results

- Name of person verifying results

The actual net measured values (including negative values) and their associated errors shall be reported. For values lower than the lower limit of detection (LLD), the LLD will be provided. Whenever possible, values lower than the LLD will be reported in the following manner: 7.4 ± 18.5 pCi/g. The following supplemental information shall be included:

- Description of survey and sampling equipment

- Survey and sampling procedures, including sampling times, rates, and volumes

- Analytical procedures

- Calculation methods

- Calculation of the lower limit of detection

- Calibration procedures

Table 8-1. Typical survey instruments and detection capabilities for measuring soil surfaces.

Nuclide	Instrument or Method	Detection Levels
Gross Alpha	Thin-walled, shielded Pancake G-M	$\geq 10^3$ pCi/cm^2, very slow scan
	ZnS scintillator	$\geq 10^2$ pCi/cm^2, source dependent
	Thin NaI or CaF_2	$\geq$ 100 pCi/cm^2, source dependent
Gross Beta-gamma	Thin-walled, shielded Pancake G-M	500-5,000 pCi/cm^2, Emax $\geq$ 0.26 MeV
	ZnS scintillator	300 pCi/m^2, source dependent
	NaI spectrometer	500-5 x 10^5 pCi/cm^2
	Ion chamber	$\geq 10^3$ pCi/cm^2
	Phoswich	$\geq$ 200 pCi/cm^2
	Intrinsic Germanium	$\geq$ 200 pCi/cm^2
	Portable scintillator	5,000-50,000 pCi/cm^2

- Measurements of radioactivity should be reported as follows: alpha and beta-gamma in dpm/100 cm^2 and gamma at one meter above surface in $\mu R/h$

- Analytical error at 95% confidence level should be reported for all analyses

- Name of surveyor, sampler, or analyst

- Analysis date

- Instrument specifications and calibration data

- Confidence level, standard error, etc. attached to analytical results

- Name of person verifying results

The actual net measured values (including negative values) and their associated errors shall be reported. For values lower than the lower limit of detection (LLD), the LLD will be provided. Whenever possible, values lower than the LLD will be reported in the following manner: 7.4 ± 18.5 pCi/g. The following supplemental information shall be included:

- Description of survey and sampling equipment

- Survey and sampling procedures, including sampling times, rates, and volumes

- Analytical procedures

- Calculation methods

- Calculation of the lower limit of detection

- Calibration procedures

- Discussion of the quality assurance program for ensuring the quality of results.

The data shall be presented so that the radiological condition of the facility and site is completely and accurately depicted and the radiological condition can be ascertained without further analysis and manipulation of the data.

Based on the results and conclusions of the Phase I and II Characterization Reports, the decommissioning contractor will prepare a Termination Radiation Survey Report. The report will be reviewed by the USAEHA and provided to the licensee. After the licensee approves the termination Radiation Survey Report, it shall be submitted to the NRC as required by NRC Regulatory Guide 1.86. The report will include a description of the survey methods, instruments, analyses, and an evaluation of the results. The report is expected to conclude that the AMTL Reactor Facility and site are suitable for release for unrestricted use.

The Army will make available to the public a plan that will describe the methods by which the reactor building will be dismantled. This plan will be implemented once the NRC has determined the facility meets unrestricted use criteria.

REFERENCES FOR ENTIRE DOCUMENT

1. Decision Analysis Report for US Army Materials Technology Laboratory Research Reactor, EGG-WM-8979, September 1990.

2. U.S. Nuclear Regulatory Commission, Monitoring for Compliance with Decommissioning Termination Survey Criteria, NUREG/CR-2082.

3. Characterization Report for U.S. Army Materials Technology Laboratory Research Reactor, EGG-WM-8978, August 1990.

4. U.S. Nuclear Regulatory Commission, Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-product, Source, or Special Nuclear Material, August 1987.