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December 16, 1999

Mr. Randy Godfrey
Engineering Manager
U.S. Department of the Army
New England District, Corps of Engineers
696 Virginia Road
Concord, MA 01742-2751

Re: Contract No. DACA31-96-D-0006
St. Albans Veterans Administration Extended Care Facility, Queens, New York
DCN: VAHOSP-121499-AABZ
WESTON W.O. No.: 10971-219-201-0006
Draft Final Decommissioning Plan

Dear Mr. Godfrey:

Roy F. Weston, Inc. (WESTON) is pleased to provide to you three (3) copies of the Draft Final Decommissioning Plan (DP) for the St. Albans Veterans Administration Extended Care Center in Queens, New York. Please do not hesitate to contact me at (516) 873-3814 or Mike Madonia at (847) 918-4087 if you have any questions or comments on this document.

Very truly yours,

ROY F. WESTON, INC.

John Rhyner

John Rhyner, P.G.
Project Manager

Enc.

cc: H. Honerlah, USACE
Todd Jackson, NRC (2 copies)
M. Madonia, WESTON
DCN Files

127749

U.S. Army Corps of Engineers

New England District
Concord, Massachusetts

TECHNICAL SUPPORT SERVICES
ST. ALBANS VETERANS ADMINISTRATION EXTENDED CARE CENTER
QUEENS, NEW YORK

Contract No. DACA31-D-0006

DRAFT FINAL DECOMMISSIONING PLAN

Task Order No. 19
DCN: VAHOSP-121499-AABZ

14 December1999

98P-0892

WESTON
MANAGERS DESIGNERS/CONSULTANTS

DRAFT FINAL DECOMMISSIONING PLAN
ST. ALBANS VETERANS ADMINISTRATION EXTENDED CARE CENTER
QUEENS, NEW YORK

Contract No. DACA31-96-D-0006
Task Order No. 19
DCN: VAHOSP-071399-AABZ

Prepared for:

U.S. ARMY CORPS OF ENGINEERS
NORTH ATLANTIC DIVISION
NEW ENGLAND DISTRICT
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Prepared by:

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14 December 1999

W.O. No. 10971-219-201-0002

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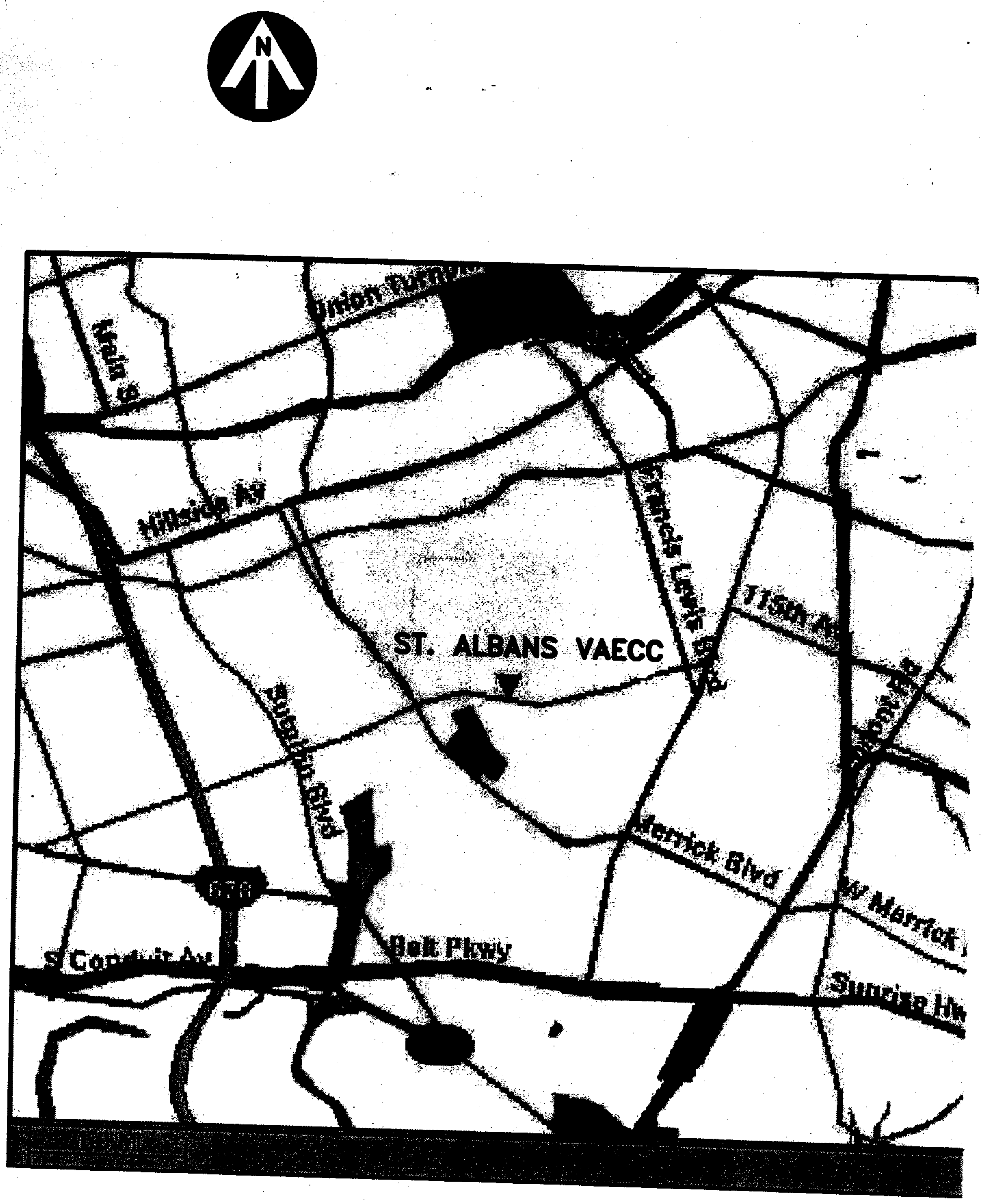
1.0 BACKGROUND INFORMATION

Roy F. Weston, Inc. (WESTON®) has prepared, on behalf of the U.S. Army Corps of Engineers (USACE) North Atlantic Division, New England District (CENAE), this Draft Final Decommissioning Plan (DP) for the St. Albans Veterans Administration Extended Care Center (VAECC), in Queens, New York under contract DACA31-96-D-0006. The U.S. Veterans Administration (VA) currently maintains a U.S. Nuclear Regulatory Commission (NRC) "Possession Only" Byproduct Materials License No. 31-02892-06, Docket 030-34751, Control. No.125705. The license was issued in July 1998 for residual strontium-90 (Sr-90) contamination resulting from laboratory research performed in the early 1960s. Conditions of the license require that several areas of the VAECC be secured from routine access and placed under radiological control due to elevated levels of Sr-90 surface contamination and bulk material concentration. This DP provides a framework to proceed with the license requirement for decontamination and decommissioning (D&D) of the facility.

1.1 Site Description

The VAECC is located on 55 acres at 179th Street and Linden Boulevard, in Queens, New York. The VAECC consists of 15 buildings encompassing approximately 700,000 square feet (ft^2) of the property. The facility employs approximately 500 employees whose work locations are spread throughout the 15 buildings. The facility currently performs extended care services, outpatient services, and a consolidated laundry service. The regional site location is shown in Figure 1-1.

Survey units have been assigned to portions of Buildings 64, 90, and 91 at the VAECC. Building 64 serves as the facility boiler plant, and also houses an inactive incinerator. Building 90 consists of two levels, ground and basement, that have historically housed the majority of the nuclear medicine-related operations at the facility. The basement level of Building 90 was formerly used (until recently) for file storage (uncontrolled access areas only), and includes restricted access areas consisting of a former nuclear medicine laboratory, ejector pit, and bathrooms. The basement level of Building 91 contains the speech pathology department, audiology department, waiting rooms, and a variety of maintenance shops that are currently used by the VAECC. Figure 1-1 also presents the relative locations of Buildings 90 and 91 at the VAECC site.



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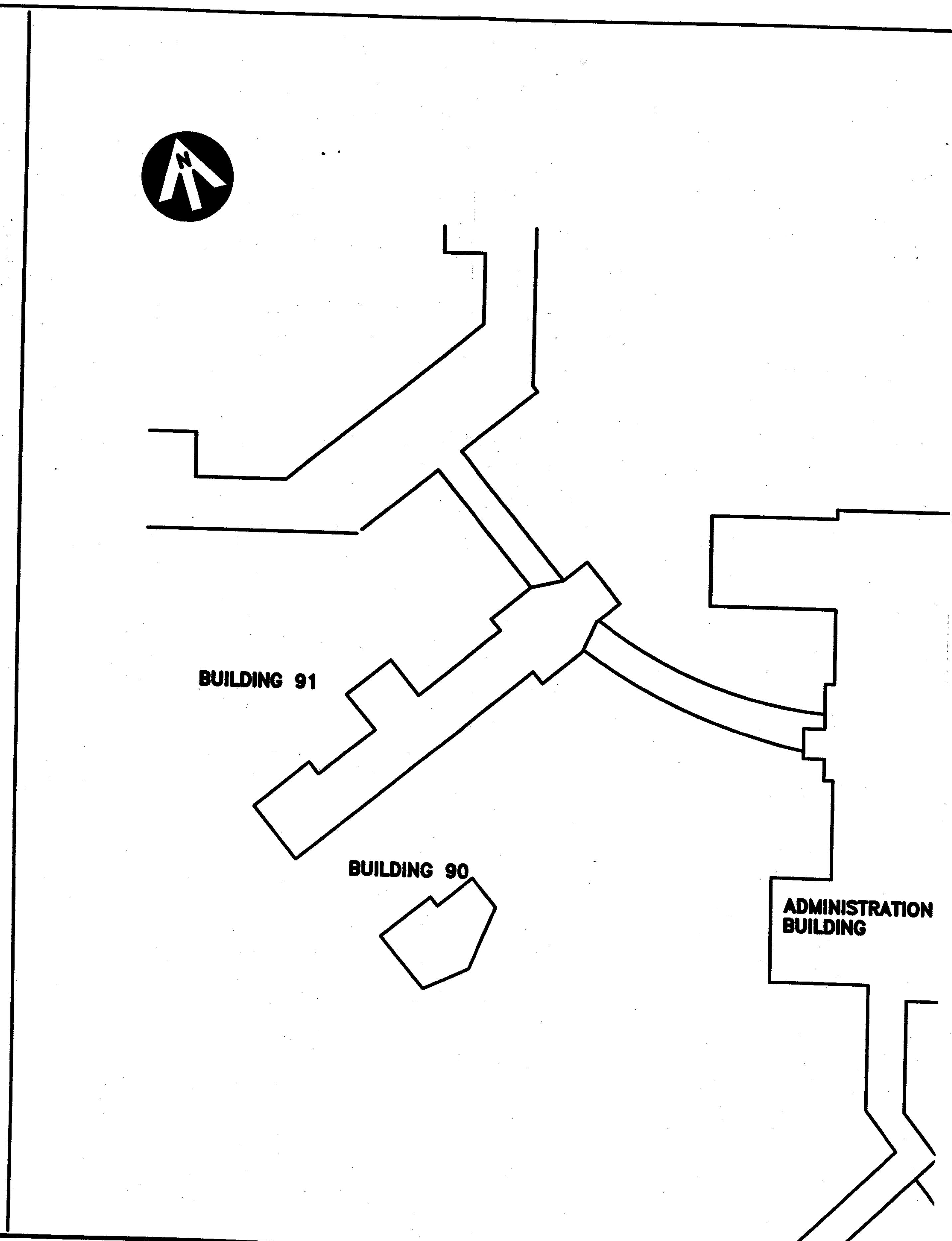


U.S. ARMY CORPS OF ENGINEERS
ST. ALBANS VETERANS ADMINISTRATION EXTENDED CARE CENTER

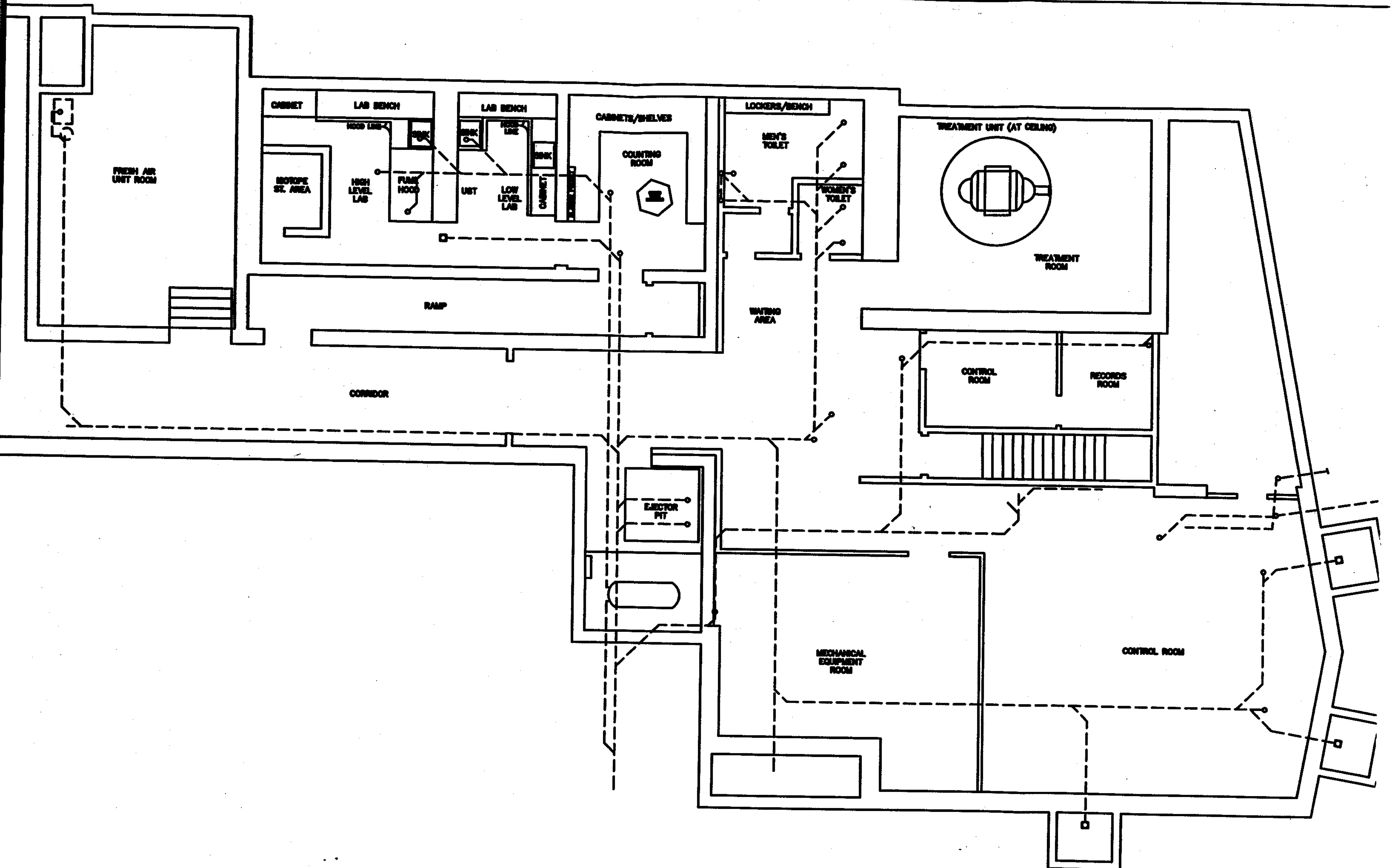
REGIONAL SITE LOCATION AND SITE MAP
FIGURE 1-1

1.2 Survey Unit Descriptions

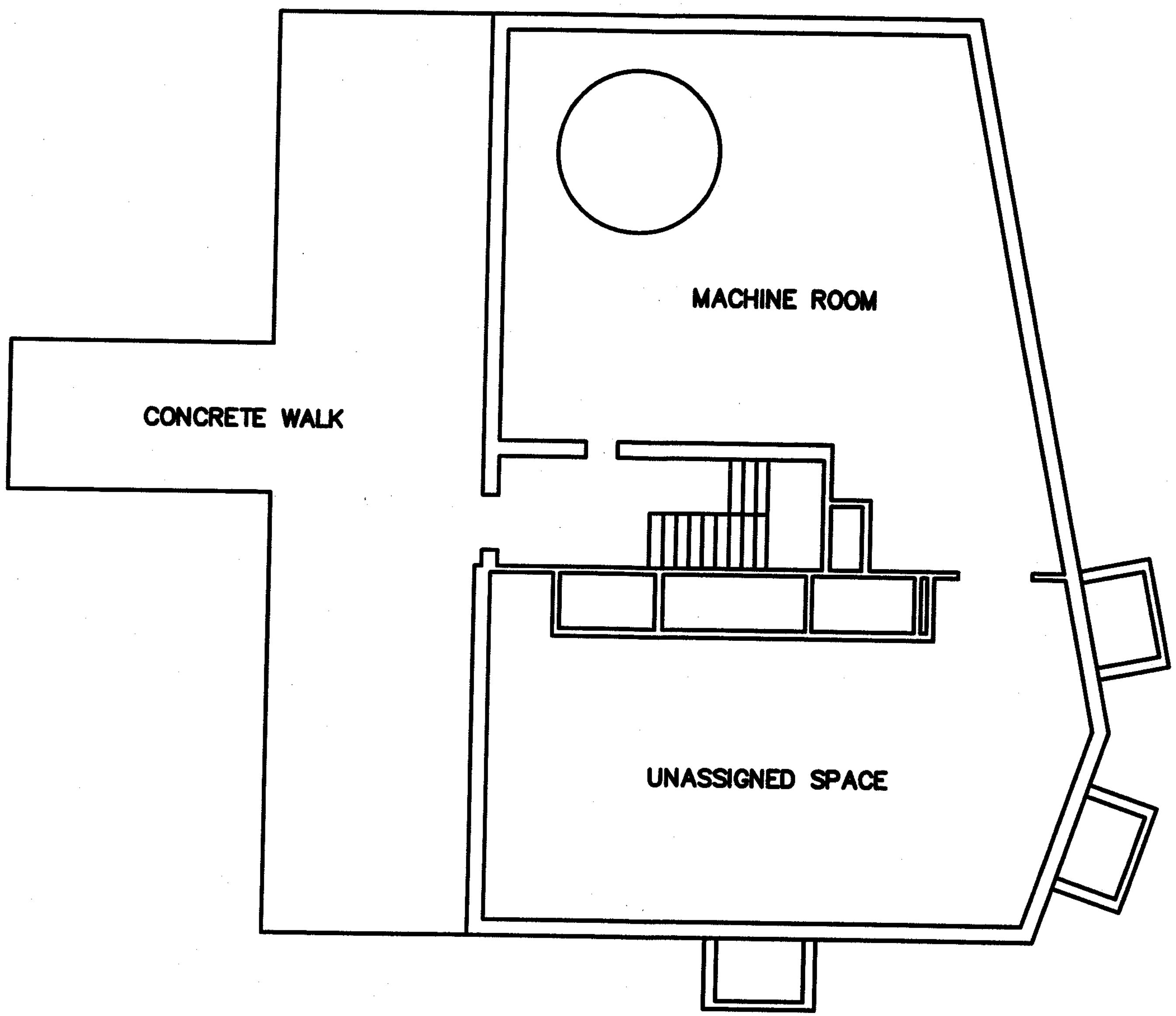
A total of nine survey units have been assigned to portions of the buildings as previously described. Figure 1-2 illustrates the relative location of survey units (SU) 001 through 005 at the basement level of Building 90, while Figure 1-3 shows SU 008 located at the ground level of Building 90. Figure 1-4 illustrates the relationship of the Building 91 basement level SU 006 and SU 007 to those contained in the adjacent Building 90. SU 009 and Building 64 are not pictured.

Periodic characterization and decontamination efforts have been performed from 1992 to the present and have added significant knowledge as to the nature and extent of contamination. In some cases, enough data has been collected to support the requirements of a final status survey (FSS) as described in the *Multi-Agency Radiological Site Survey Implementation Manual (MARSSIM)* – NUREG 1575 (NRC 1997). The processing of survey unit data to FSS format is included as a subtask under preparation of a final report following D&D. Survey unit data has been processed to final status survey format by WESTON for survey units 003, 005, 006, 007, 008 and 009. A draft FSS report for these six units will be provided to the Remedial Action Contractor (RAC). The RAC may use this as a template for the final NRC submittal at the close of decontamination activities for all survey units. Survey units 001, 002, and 004 contain extensive contamination that will require remediation followed by a comprehensive FSS. The remedial action contractor (RAC) responsible for D&D in these three survey units will be responsible for processing the survey unit data for FSS format.

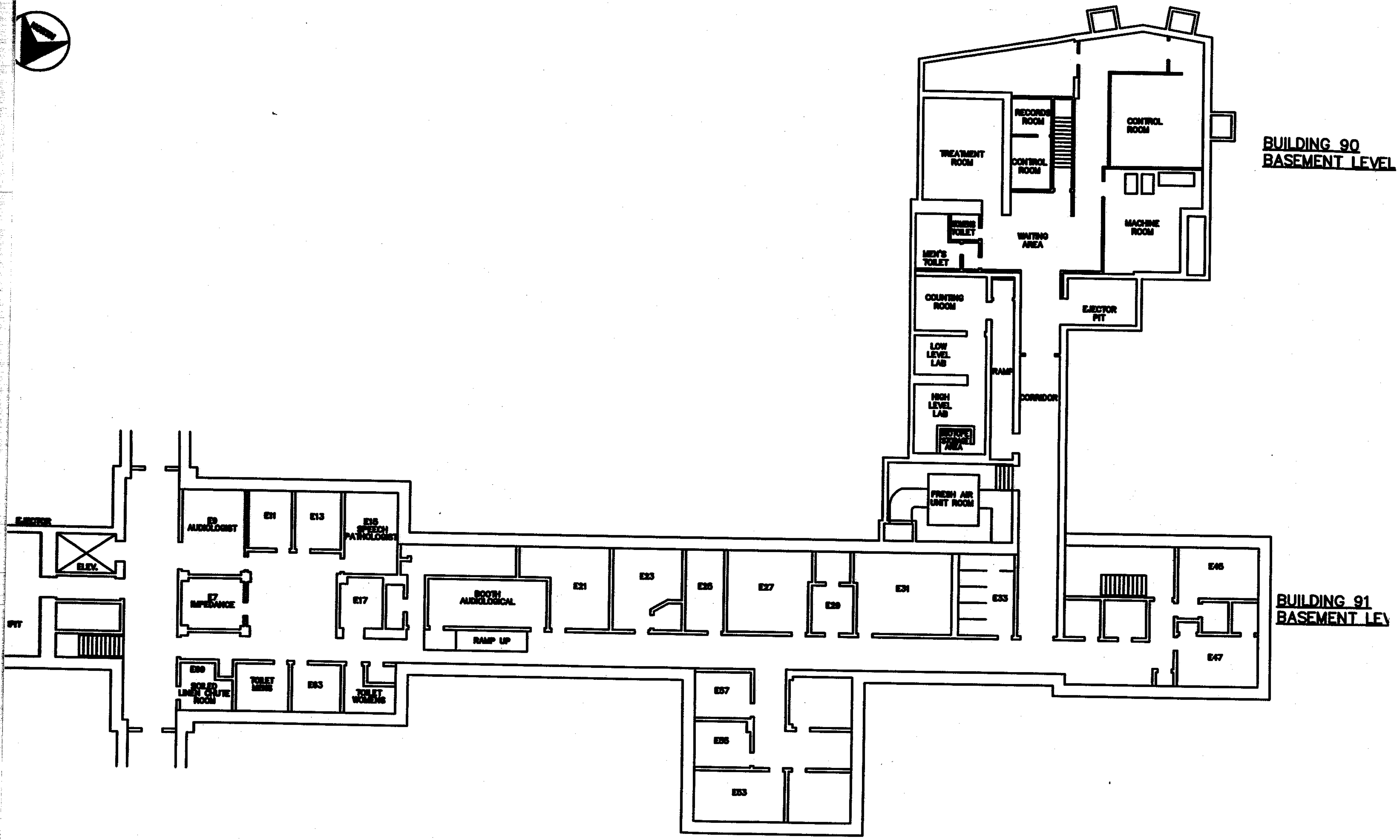
Per WESTON (1999), decontamination operations will be restricted to the contents and surfaces of SU 001 and SU 004. The contents of SU 002 (ejector pit), which include feed piping, hold-up tank and pressurizer, also shall be removed and disposed as radioactive waste. Based on previous characterization efforts, supplemental tritium surveys and small-scale decontamination activities, data demonstrate that SU's 003, 005, 006, 007, 008 and 009 meet the derived concentration guidelines (DCGLs) and requirements for FSS. Table 1-1 presents the status of each survey unit.



WESTON MANAGERS DESIGNERS/CONSULTANTS ROY F. WESTON OF NEW YORK, INC.	DESIGN BY DRAWN BY GENE AMOS/MARIANNE CLUNE CHECK BY MICHAEL MADONIA APPROVED BY	U.S. ARMY CORPS OF ENGINEERS ST. ALBANS VETERANS ADMINISTRATION EXTENDED CARE CENTER	TITLE: BUILDING 90 BASEMENT LEVEL FIGURE 1–2 SCALE: 1/4" = 1'-0"	NO. 1 SHEET 1 OF



WESTON ROY F. WESTON OF NEW YORK, INC.	DESIGN BY DRAWN BY CHECK BY APPROVED BY	U.S. ARMY CORPS OF ENGINEERS ST. ALBANS VETERANS ADMINISTRATION EXTENDED CARE CENTER	BUILDING 90 GROUND LEVEL FIGURE 1-3	1
				1" = 40'	1 of 1



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BUILDINGS 90 AND 91
BASEMENT LEVEL
FIGURE 1-4

NOT TO SCALE

Table 1-1 Current Survey Unit Status and Decontamination Requirements

Survey Unit #	Level of Decontamination Effort	Final Status Survey Required?	Comments
001 – Nuclear Medicine Labs	Major	Yes – following decontamination	
002 – Ejector Pit	Moderate	Yes – following removal of internals	
003 – Women's Toilet	None	Supplemental tritium sampling completed	Data formatted to final status survey report
004 – Men's Toilet	Moderate	Yes – following decontamination	
005 – Lower Level Building 90	None	No – area has been subject to test decontamination	Data formatted to final status survey report
006 – Building 91 Basement: Maintenance Areas	None	No – area has been subject to test decontamination	Data formatted to final status survey report
007 – Building 91 Basement: Laboratory and Maintenance Areas	None	No – area has been subject to test decontamination	Data formatted to final status survey report
008 – Building 90 Ground Level	None	No	Data formatted to final status survey report
009 – Incinerator	None	Supplemental tritium sampling completed	Data formatted to final status survey report

2.0 DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES

2.1 Decommissioning Objective, Activities, Tasks, and Schedules

2.1.1 Decommissioning Objective

The decommissioning objective for activities at the VAECC is to ensure that all cleanup criteria or DCGLs are achieved and that exposures to occupational workers and the public are maintained as low as reasonably achievable (ALARA). The future use of the areas within Building 90 of the VAECC is uncertain. Upon successful remediation, the areas will be restored to a structural stability ensuring physical safety to all workers that must enter or be adjacent to the decontaminated areas.

2.1.2 Description of Activities and Tasks

Activities to be conducted as part of the radiological remediation of the designated survey units at the VAECC will consist of five primary tasks. These tasks are: creation of planning documentation, field mobilization, facility D&D, final status survey and demobilization, and final report preparation.

Task 1 - Planning Documentation

Primary planning documentation which will be generated prior to commencement of any field remediation activities will include a Quality Assurance Program Plan (QAPP), Health and Safety Plan (HASP), and Radiological Protection Program (RPP). Descriptions of the planning documents are discussed below. The contents of the RPP are described in detail in Section 3.3 of this DP.

The QAPP will be included as a section of the Work Plan and will include quality assurance and quality control objectives associated with data collection for the project. Specific information that will be included in the QAPP will be; sampling procedures, laboratory and field analytical procedures, calibration procedures, sample custody, internal QC checks and corrective action plans.

The HASP will contain descriptions of safety risks associated with each of the tasks that will be performed as part of the remediation at the St. Albans facility. These descriptions will include the roles and responsibilities of each individual working on specific tasks with respect to safety. Personal protective equipment required, decontamination procedures, air monitoring, work zone delineation, and engineering controls for tasks will also be established in the HASP. Possible incident contingencies will be evaluated and responses outlined in the HASP as well.

Task 2 - Mobilization

The first phase of activity that will be conducted at the facility will involve mobilization activities and will include the following tasks:

- Establish a method of removal of contaminated materials from the basement. Two suggested methods are as follows.
  - Radioactive waste removed from the controlled zone is containerized at the buffer zone. Containers are sealed and a surface contamination survey is performed before release to non-controlled areas. Containers would need to be small which may not be as efficient for waste disposal.
  - Another approach would be to construct a lift or similar transfer system to raise materials through the 1,000 kVp Treatment Room that is open above to the ground floor. This area would need to be covered in polyethylene sheeting to prevent extensive spread of contamination during removal operations. This method of removal of contaminated materials is suggested because it avoids the transfer of waste through continuously occupied spaces. It should be noted that any pathway used for contaminated material removal will require re-survey following completion of removal activities, regardless of results of previous surveys.

Establish a grade-level location where materials being removed from the basement can be surveyed and segregated. This would involve establishing a radiological control zone on the ground floor level and covering all surfaces with polyethylene sheeting to prevent the spread of contamination. Additionally this area would need to be protected from the elements. Temporary buildings are available which would suit this purpose and could be assembled in the building courtyard. Contaminated items can be placed directly in larger shipping containers to provide efficient waste disposal.

- Establish a decontamination line for remediation workers to doff personal protective equipment, screen themselves for contamination and decontaminate themselves as necessary to prevent the spread of contamination outside of existing areas.

Task 3 – Facility D&D

Each contaminated unit at the site will have several D&D tasks required to meet project objectives. Primary objective of the removal activities will be: minimizing the spread of contamination to other areas, minimizing the volume of contaminated waste to be disposed of, minimizing the exposure to workers, and minimizing the amount of residual radioactive materials left in place at the site.

Following completion of the mobilization activities described above, remedial activities would commence. These activities will include the following tasks. A final status survey is performed by the RAC after the completion of the tasks and before demobilization.

- SU 004 Remedial Activities

  1. Remove all internal contents from SU 004. SU 004 contains materials such as toilet partitions that can be easily disassembled and removed.

2. Remove contaminated tile from SU 004. Following removal of internal contents of SU 004, the floor tile will be removed from the floor and placed into sealable containers. This activity will be done in a manner that minimizes the breaking of tiles and generation of airborne particulates to avoid spread of contamination.

3. Remove contaminated concrete from SU 004. It is anticipated that surface contamination may be present on the concrete under the floor tile in some locations of SU 004. Removal of this contamination will be accomplished using scarification or similar methods. Dust generation will be controlled using a vacuum and HEPA filtration system.

- SU 001 and SU 002 Remedial Activities

1. Remove loose debris from SU 001. Some loose equipment, glassware, chairs, miscellaneous debris, and two drums of tiles from the nuclear medicine laboratories are still present in SU 001. These materials will be removed prior to further work being conducted in this area in order to clear space for other removal activities.

2. Assess the hazard potential of residual floor tile mastic in nuclear medicine laboratories. Asbestos-containing floor tiles have been removed in this area exposing the underlying mastic. This material should be analyzed and appropriate abatement actions should be taken if the mastic is found to be an asbestos-containing material (ACM).

3. Apply a fixative to the floors of SU 001. Previous attempts at removing contamination on concrete surfaces in this area by scarification have produced large amounts of contaminated dust in laboratory areas. A fixing compound will be applied to these dusts to prevent them from causing an airborne contamination hazard and to prevent them from causing spread of contamination to other areas during remediation activities.

4. Remove contaminated appliances and furniture from the laboratory and counting rooms of SU 001. Items such as laboratory benches, shelves and fume hoods will be disassembled to the extent practicable and removed via the previously established route. After removal to the surface level structure, all surfaces of the removed items will be surveyed and materials will be segregated accordingly.

5. Remove remaining ductwork in SU 001. Most of the ductwork servicing SU 001 has previously been removed; however, some smaller sections are still present in the walls. This ductwork will be removed, screened and decontaminated or disposed of, as necessary.

6. Remove interior, non-load bearing walls from SU 001. Many of the walls in SU 001 are not load bearing and may be sectioned and removed for screening and decontamination or disposal, as necessary.

7. Cut floor slab in radiation therapy rooms. Contamination has penetrated several inches into the floor in this area making decontamination techniques such as scarification impractical. Therefore, the floor slab will need to be removed. This can be done by sectioning the slab using a concrete saw and drilling holes where anchor bolts could be attached to individual sections for lifting. A HEPA filtration system can be used to control airborne dust hazards.

8. Remove hardware from SU 002. SU 002 contains a 50-100 gallon hold-up tank, pressurizer and feed lines to and from the ejector pit. This hardware will be disconnected, removed, screened and decontaminated or disposed of, as appropriate.

9. Remove sections of concrete floor slab from nuclear medicine laboratory. Sections of the floor will be removed and lifted from attached anchor bolts to the survey area by the previously established route.

10. Remove drain lines in nuclear medicine laboratory in SU 001. With the floor slab removed, the drain lines for the nuclear medicine laboratory can be accessed. Drain lines will be sectioned, capped, and moved to the above ground screening area, where detailed examination of contamination deposition can be performed. Attempts should be made to section the drain line so that more extensively contaminated sections such as elbows, unions, and cleanouts can be segregated from less contaminated sections, to minimize contaminated waste volumes generated.

11. Remove tile from ramp of SU 001. The tile on the ramp in SU 001 is suspected to be asbestos-containing and should be handled and disposed of according to prescribed methods for ACM and radioactive materials. Tile removal operations may be combined with other asbestos removal and handling operations.

12. Perform surface scarification of exposed concrete of ramp in SU 001. Some surface contamination may be present on the underlying concrete of the SU 001 ramp. Surface scarification techniques will be employed to remove any areas of surface contamination identified on the concrete.

13. Remove drain lines under ramp in SU 001, under tunnel and waiting room in SU 005. Characterize drain line under restroom floor in SU 004 after removal of floor and remove line if appropriate. Determine whether drain line in SU 003 needs to be characterized and removed. Access to underlying soil will be created by cutting the floor slabs above drains lines with a concrete saw or similar technique. With the floor slab removed, the drain lines from the nuclear medicine laboratory to the ejector pit can be accessed. Drain lines will be sectioned, capped, and immediately placed

in waste containers or moved to the above ground screening area, where detailed examination of contamination deposition can be performed. As in the case of other drain lines, attempts should be made to section the drain line so that more extensively contaminated sections such as elbows, unions, and cleanouts can be segregated from less contaminated sections to minimize contaminated waste volumes generated.

14. Backfill removed drain line trenches. Following completion of the final status survey, clean fill material will be placed in the trenches and the floor slabs replaced.

15. Backfill nuclear medicine laboratory. Following completion of the final status survey, the void space created by the removal of the nuclear medicine laboratory will be backfilled and compacted, and the finish ground level brought to specifications as provided.

Task 4 – Final Status Survey and Demobilization

A final status survey is performed (see section 4.0) before demobilization to ensure all areas are adequately remediated. Demobilization is anticipated to occur after the completion of all remediation and include removal of heavy equipment and support facilities. Prior to release, all equipment that has been used in radiological control areas (RCA) will be surveyed and decontaminated as necessary to demonstrate compliance with the surface contamination criteria shown in Table 2-1.

Task 5 - Final Report

The final report will contain detailed descriptions of the site condition prior to D&D operations, all activities performed as part of the removal and decontamination operations, major deviations from the work plan that were necessary to complete the work, and conditions at the site following remediation activities. A final status survey report is performed by the RAC to include the draft final status survey report performed by WESTON for survey units 003 and 005 through 009. The final status survey report is presented as an attachment to the final report.

2.1.3 <u>Cleanup Levels</u>

WESTON (1999a) and its addendum WESTON (1999b) presented the rationale for modified DCGLs, and recommended that the tritium and strontium-90 generic screening levels as presented in NRC (1998a) be applied as surface contamination DCGLs for building surfaces left in place at the VAECC. Based on pathway analyses performed, a strontium-90 soil and concrete DCGL of 35 pCi/g will be applied. Rationale for concrete volumetric DCGL is presented in Appendix A. Concrete is also subject to building surface contamination DCGLs. Volumetric contamination is not expected to exist in other bulk material. Equipment and material released from an RCA will be subject to the traditional surface contamination requirements of NRC (1998b). In all cases, D&D efforts will be performed to reduce surface contamination levels as well as soil and concrete concentration to as low as reasonably achievable (ALARA). Table 2-1 summarizes all DCGLs for the D&D operations.

Table 2-1 DCGLs to be Applied at the VAECC

Isotope	Equipment and Material Release DCGL Total ($dpm/100cm^2$)	Equipment and Material Release DCGL Removable ($dpm/100cm^2$)	Equipment and Material Release DCGL Maximum ($dpm/100cm^2$)	Building Surface Contamination DCGL[1] ($dpm/100cm^2$)	Soil and Concrete Contamination Limit (pCi/g)
Tritium	--	1000	--	1.2×10^8	---
Sr-90	1000	200	3000	8.7×10^3	35

[1] Removable contamination levels not to exceed 10% of value.

Application of these DCGLs indicate that significant D&D activities will be required in Survey Units 001 and 004. Limited D&D is required in Survey Unit 002.

2.1.4 Procedures

Standard operating procedures will be utilized for radiological survey and protection, construction safety, and occupational health and safety. Radiological survey and protection procedures are included in the Radiological Protection Program described in Section 3.3.

2.1.5 Schedules

The proposed D&D project schedule is shown in Figure 2-1. Five tasks including planning, mobilization, remediation, final status survey and demobilization, and final report preparation are shown with a start date of 15 February 1999 and a completion date of 31 July 1999.

2.2 Decommissioning Organization and Responsibilities

D&D operations will be administered and managed through CENAE and its RAC. The CENAE Project Manager (PM) will facilitate communication with the designated NRC PM, manage the preparation of all planning documents, and facilitate the D&D operation. Compliance with this DP and the VAECC license will be managed by the VA Radiation Safety Officer (RSO), who may designate a qualified field RSO from the RAC to oversee field operations. An organization chart for D&D operations is shown in Figure 2-2.

Figure 2-1 Proposed VAECC Decommissioning Project Schedule

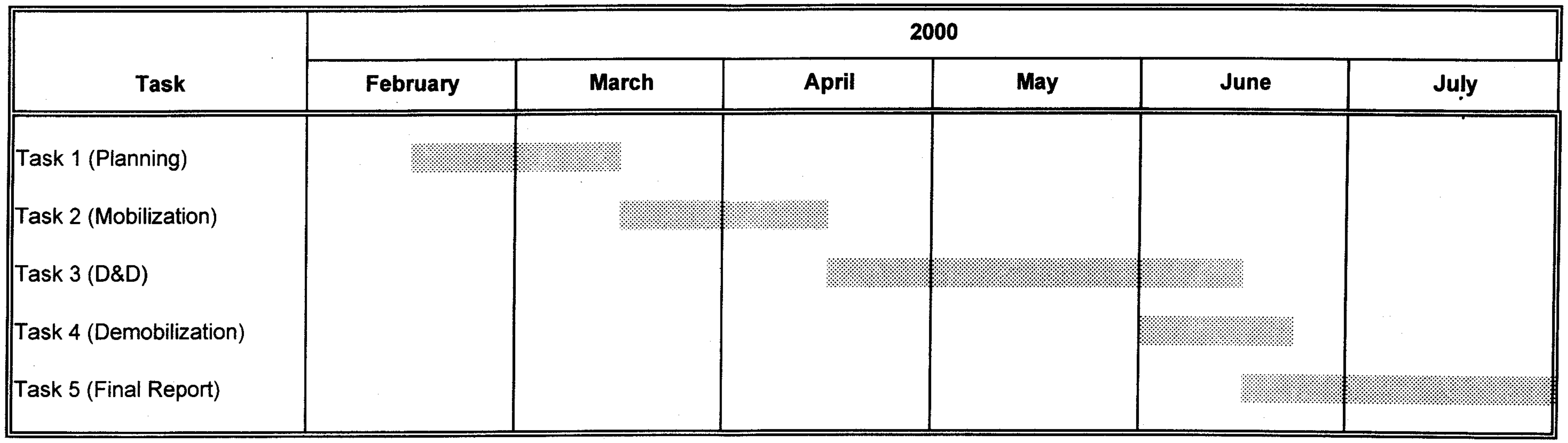
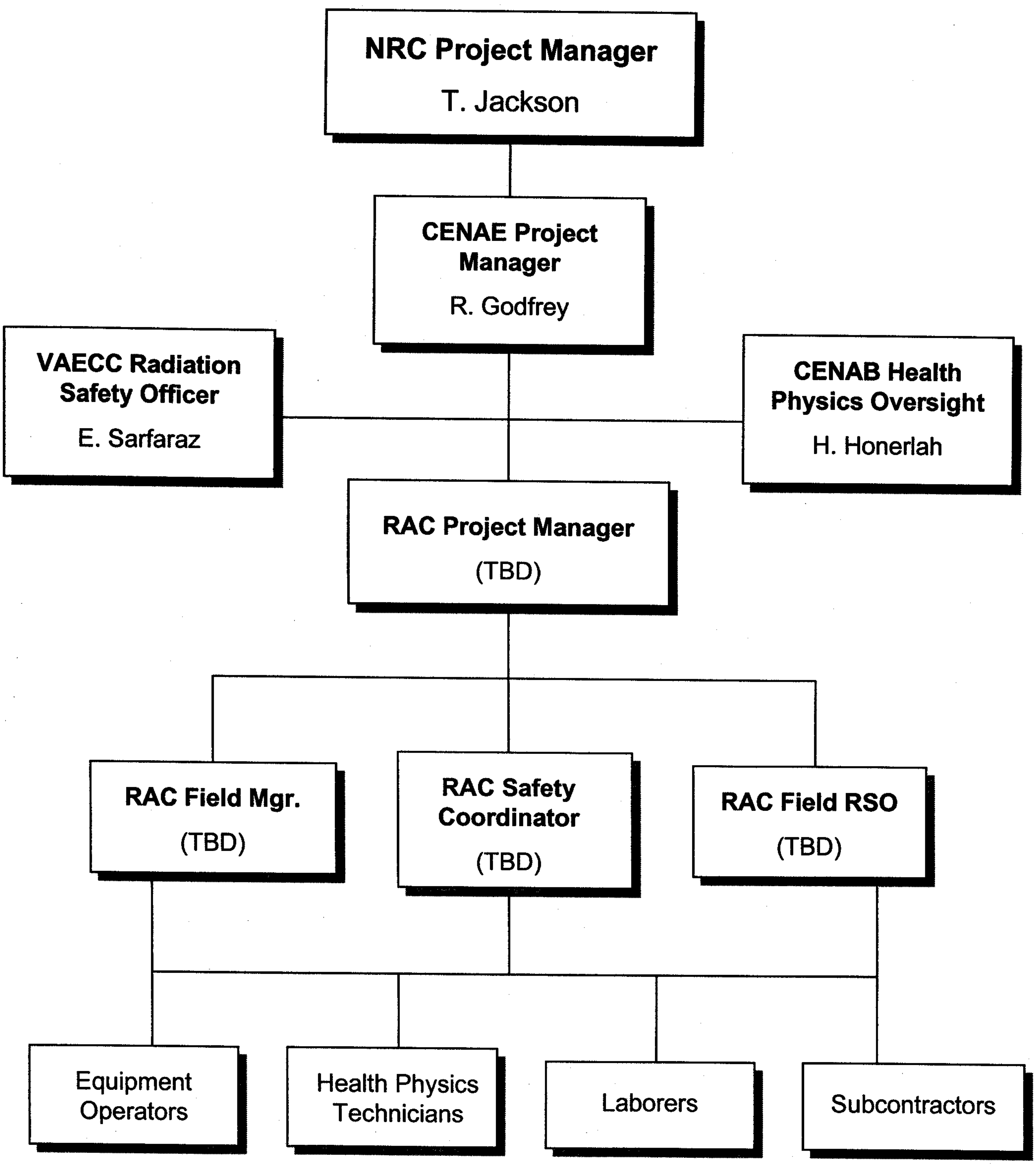


Figure 2-2 VAECC Decommissioning Project Organization Chart



2.3 Training

All occupational workers (including contractors and vendors) performing remediation activities will receive a training course in the fundamentals of radiation and radiation protection, and site-specific radiological hazards. Site specific topics for personnel entering an RCA will include site description and background, contaminant characterization, action levels and instrumentation, personal protective equipment (PPE) selection, radiation work permit implementation, dosimetry, personnel and equipment exit from an RCA, decontamination of workers and equipment, respiratory protection, pregnant workers, ALARA policy and emergency procedures. Occupational worker training for Veterans Hospital staff will be provided by USACE. The RAC shall provide trained workers. Training documentation will be maintained by the RAC for all personnel.

Prior to participating in site activities, occupational workers will obtain medical clearance per 29 CFR 1910.120 and 10 CFR 20.1703.

2.4 Contractor Assistance

The CENAE RAC, Stone & Webster, will manage all D&D operations. In the execution of D&D operations, the RAC may utilize a variety of subcontractors for specialty tasks such as construction/demolition, health physics, and waste transportation and processing. The actions of all RAC subcontractors will be managed to be consistent with the requirements of this DP and related documents.

3.0 DESCRIPTION OF METHODS USED FOR PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

3.1 Facility Radiological History Information

The VAECC was operated as a Naval Hospital prior to its acquisition by the VA in 1976. The Naval Hospital provided nuclear medicine services under several NRC licenses. NRC-licensed activities ended with the termination of NRC License #31-0007606 on 31 December 1973. In 1976 St. Albans was transferred from the Navy to the VA. The VA did not hold a radioactive materials license at St. Albans. In May 1992, the USACE performed a review of former U.S. Department of Defense (DOD) sites that had handled radioactive materials. While performing follow-on field surveys, the USACE and its contractors identified areas of elevated radiological contamination. In August 1992, Teledyne Isotopes partially remediated the VAECC and recommended an expanded survey and decontamination of other rooms. In September 1997, Ogden Environmental and Energy Services, Inc. (Ogden) surveyed the nuclear medicine labs and ejector pit located adjacent to tunnel 45 and identified radioactive contamination in excess of NRC release criteria (Ogden/NAE 1998). The Ogden report concluded that the scope of the survey needed to be expanded.

WESTON performed a supplemental radiological characterization at the VAECC from January-February 1999. The characterization survey focused on areas adjacent to the nuclear medicine laboratories and ejector pits, with the intent to gather data adequate to support a final status survey (FSS) where possible. Several isolated hot spots of up to 20,000 dpm/100 cm^2 (fixed contamination only) were noted in SU 005, 006 and 007. These spots appeared to be the result of foot transfer from the laboratories and were later decontaminated using physical methods. A final surface contamination survey was performed on the areas after decontamination. The objective of this activity was to limit future decontamination to SU 001, 002 and 004. Table 3-1 presents the range of and average Sr-90 surface contamination levels of survey units requiring remediation or supplemental sampling as presented in WESTON (1999).

Table 3-1. Sr-90 Surface Contamination Levels in Survey Units Requiring Remediation

Survey Unit	Description	Total Beta-Gamma Range (dpm/100 cm^2)	Total Beta-Gamma Average (dpm/100 cm^2)	Removable Beta-Gamma Range (dpm/100 cm^2)	Removable Beta-Gamma Average (dpm/100 cm^2)
001	Nuclear Medicine Lab	-250 to 2.8 E6	1.1 E5	-4 to 2.7 E3	2.7E2
002	Ejector Pit	67 to 7.5 E3	3.2 E3	11 to 76	36
004	Men's Toilet	-67 to 1.5 E5	3.0 E4	-4 to 83	36

A concrete slab and underlying soil sampling program was instituted in SU 001 and 002. Sampling results indicated Sr-90 contamination exceeding the DCGLs at depths up to two inches in the concrete slab of SU 001. The only soil contamination levels exceeding the DCGL were noted in the vicinity of the cleanouts/floor drain system serving SU 001. Soil concentrations as high as 2,054 picoCuries per gram (pCi/g) were noted.

3.2 Ensuring that Occupation Radiation Exposures Are As Low As Reasonably Achievable (ALARA)

The intent of the Radiation Safety Program is to ensure that occupational radiation exposures are maintained to ALARA. Implementation will be facilitated through USACE/CENAE/VAECC interaction with the site RSO. USACE utilizes a three-tiered approach for its projects in accordance with USACE ER 385-1-80. The program includes a written VAECC policy, qualified personnel, periodic audits, radiation protection training and appropriate monitoring equipment, and adequate supplies and manpower for radiation protection. During D&D operations, the RSO shall facilitate a meeting of the ALARA committee at a minimum of once per month or in emergency situations. The ALARA Committee shall consist of the RSO, the CENAE PM, the USACE HP and the RAC PM. Site specific dose limits will be established by the ALARA committee. The purpose of the meeting is to identify improved means to reduce worker and public exposures to ALARA and ensure dose limits are not approached or exceeded. Meeting minutes will be documented and retained in the project files.

3.3 Radiation Protection Program

All site activities will be performed to the requirements of a Radiation Protection Program (RPP) administered by the site RSO and/or their designee. The RPP shall contain specifications to ensure that exposures to workers and the public are maintained ALARA. These specifications include environmental monitoring requirements, access control and posting, PPE requirements, radiological release and personnel monitoring, respiratory protection, decontamination, and emergency response. These RPP elements will be implemented through the use of SOPs that must be reviewed and understood by all personnel participating in D&D activities. Base procedures include, but are not limited to:

Environmental Monitoring – Describes the set up and use of particulate air samplers at the locations designated in Section 3.6. Also describes the use of portable/lapel sampling equipment when necessary. The procedure specifies the types of particulate air samplers, which may include Eberline RAS-1, MSA lapel, Staplex High Volume, or reasonable equivalents, and will provide a means to calculate the airborne Sr-90 concentration, uncertainty and measurement lower limit of detection (LLD).

Site Access Control and Posting – All RCAs will be posted per the requirements of 10 CFR 20.1902. Boundaries will be clearly marked and where feasible, physical barriers will be used to restrict inadvertent access. Changes in boundaries will occur at the discretion of the RSO and will be based on current radiological survey data. It is anticipated that all work areas in survey units 001, 002 and 004 will receive posting as contaminated areas. No "radiation area" postings are anticipated. Workers entering an RCA will be required to sign an access control log upon entry and exit. Appropriate contamination monitoring equipment will be maintained at each RAC access control location. All notices to workers as required in 10 CFR 19 will be posted in a conspicuous work area.

Issuance of Radiation Work Permits (RWP) – All work in RCAs will be controlled through the use of RWPs. RWPs will be issued and/or approved by the RSO or their designee, and will include a description of radiological conditions such as gamma and beta exposure rates, surface contamination levels and airborne radionuclide concentrations. The RWP will specify the necessary personnel dosimetry and PPE required to enter the RCA, and work restrictions as applicable. Decontamination, release survey protocol, and emergency notification procedures to include estimates of internal committed effective dose equivalent (CEDE) and ALARA dose limit analysis also will be presented in the RWP.

Equipment Function Check and Maintenance – This procedure describes the daily preparation and troubleshooting of portable and laboratory counting equipment that may be used during D&D. Critical elements of the procedure include methods to perform background and efficiency checks, trend charting and chi-square distribution analyses, minimum detectable activity (MDA) calculation, and electronic adjustments. This procedure will be applicable to a wide variety of equipment including but not limited to gas-flow proportional detectors, Geiger-Mueller (GM) detectors, plastic scintillation detectors, phoswhich smear counters, and ionization chambers. Table 3-2 illustrates equipment potentially used to support D&D activities. All equipment maintenance and use procedures will be consistent with the recommendations of the American National Standards Institute (ANSI) and American Society for Testing of Materials (ASTM) recommendations. Equipment application will be consistent with the methods described in *"Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions"*, Draft NUREG-1507 (NRC 1995). Radioactive check sources will be of activity consistent with the measurements to be performed. All sources will be NIST-traceable and maintained in a secure area.

Table 3-2. Example Radiological Survey Equipment to be Used to Support D&D Activities

Manufacturer	Model	Purpose	Estimated Sensitivity	Limitations
Ludlum	44-116/2221	Recordable Surface Contamination Measurements (beta-gamma)	500-1200 dpm/100 cm^2	Fragility
Ludlum	44-9/12	Informational Surface Contamination Measurements, Surface beta dose rates	3000-4000 dpm/100 cm^2	Poor sensitivity
Ludlum	44-68/2221 or 2350	Recordable Surface Contamination Measurements (beta-gamma)	500-1200 dpm/100 cm^2	Fragility, gas supply
Ludlum	19	Area Gamma Exposure Rates	N/A	Mode of calibration
Ludlum	43-10-1/ 2929	Removable Contamination Measurements/Gross Beta-Gamma Particulate Air Samples	100-200 dpm/100 cm^2 (smear) 10^{-13} uCi/ml (air sample)	

Routine Radiological Surveys of Work and Support Areas – Work and support areas will be surveyed at a minimum frequency of once per week or more often based on changing radiological conditions. Surveys will be performed for total and removable contamination levels and beta/gamma dose rates, and documented on SOP forms/diagrams.

Respiratory Protection Program – A respiratory protection program will be implemented if airborne radionuclide concentrations exceed 10% of the Derived Air Concentration (DAC) for Sr-90 (assumed to be Class W – 2 E-9 uCi/ml). The respiratory protection program will be consistent with the requirements of 10 CFR 20, Subpart H and include a cleaning, testing, survey and maintenance component. Prior to initiation of the program, all participating workers will be subject to baseline bioassay analyses for Sr-90. Routine bioassay samples will be utilized in conjunction with the air sampling program to assess internal dose commitments. Should routine contamination surveys of respirators indicate contamination, the RSO may initiate nasal smear analyses for potentially exposed workers.

Internal and External Dose Recording Program – Each occupational worker will have their radiation doses recorded in accordance with 10 CFR 20, Subpart C. Where necessary, internal committed effective dose equivalent (CEDE) will be calculated based on air sample, bioassay, and work area stay time data. Committed dose equivalents (CDE) to target organs also will be calculated where necessary. External doses will be monitored through the use of thermoluminescent dosimeters (TLDs), and may include reporting components of total penetrating gamma (effective dose equivalent – EDE), penetrating beta, and beta skin dose. A final report will be issued to workers containing a summary total effective dose equivalent (TEDE) that is comprised of the CEDE and EDE components.

Personnel Decontamination – The implementation of RWP, access control and contamination reduction procedures will limit the potential for contamination of personnel leaving RCAs. In the event that detectable contamination is noted on personnel leaving the RCA, basic decontamination procedures using soap and water will be applied only under supervision of a person qualified in personnel decontamination procedures. All decontamination fluid and media will be retained and subject to appropriate disposition. Following decontamination, individuals will be subject to repeat surveys to ensure that the contamination has been removed. Repeated contamination events will be reviewed as a potential system failure, with appropriate corrective actions taken.

Emergency Procedure – The nature of contamination is such that limited doses would be expected even under the most severe emergency conditions. In all cases, workers will immediately be treated with regard to physical injuries after basic decontamination. The requirements for medical emergencies will be described in detail in the HASP.

3.4 Contractor Personnel

Contractor personnel participating in decommissioning activities will report to USACE/CENAE/VAECC management and will be subject to the conditions of NRC License No. 31-02892-06 and this decommissioning plan.

3.5 Radioactive Waste Management

Handling/Processing

Radioactive wastes generated from D&D at the VAECC are expected to include concrete, plywood, laboratory fixtures, drain lines, soil and other building materials. The estimated volume of radioactive waste requiring disposal is estimated at 1600 ft^3. Treatment methods are directed at reducing the volume of material necessary for disposal and include abrasion, scarification, physical segregation and separation, and application of non-hazardous cleaning solutions. Materials that do not meet the applicable DCGLs for unrestricted release will be prepared for transportation and disposal at a licensed facility.

In an effort to reduce cross-contamination and the generation of additional radioactive waste, non-essential tools and machinery will not be allowed into an RCA. Likewise, removable contamination areas will be treated with fixing agents prior to major demolition activities.

Transportation

Based on the projected volume of radioactive waste, truck shipment of waste containers including B-25 boxes and 55-gallon type H drums or other appropriate, certified shipping containers will be utilized. The transportation program will be implemented in accordance with U.S. Department of Transportation (DOT) requirements for manifesting, labeling, contamination and radiation levels, and transport. Exclusive use vehicles will be employed.

Disposal

All radioactive waste will be shipped to a USACE-approved disposal facility. All waste profile and acceptance criteria will be ensured through sampling, processing and required documentation.

3.6 Effluent Releases and Monitoring

The primary effluents that are postulated from decommissioning activities include airborne particulates and decontamination liquids/solutions generated from dust suppression and cooling activities.

Airborne Effluents

Airborne effluents may be generated during construction/demolition activities. The highest potential for release is noted within SU 001. Prior to demolition activities in this survey unit, a fixative will be applied to the concrete floors to retain Sr-90-containing dusts. Particulates generated during concrete cutting and dismantlement will be controlled through the use of negative pressure containment when working in contaminated areas of SU 001 and SU 004. Ambient air will be filtered through two-stage HEPA filtration units. The excavation of soil underlying the SU 001 concrete slabs poses some minor potential for the resuspension of soil particulates. Where necessary, small quantities of water may be applied as a dust suppression measure.

Airborne radionuclide concentrations will be monitored at multiple locations about the VAECC. Locations will include the immediate work area, adjacent areas occupied by non-radiological worker VAECC staff, and the work area boundary at ground level outside of Building 90. Background stations may be placed indoors and outdoors at locations not impacted by D&D operations. The typical means of sampling will be gross-beta sampling of particulate air filters. All activity will conservatively be attributed to Sr-90. Samples will be archived and made available for radiochemical laboratory analyses as necessary. An administrative limit of 2 E-10 uCi/ml (10% of Derived Air Concentration of 2 E-9 uCi/ml) will be utilized as an action level for occupational workers to upgrade to respiratory protection.

Airborne effluent concentrations in unrestricted areas of the VAECC will be limited to less than 6 E-12 uCi/ml. The aforementioned sampling network will be used to assess airborne Sr-90 concentrations in unrestricted areas.

Liquid Effluents

Potentially contaminated liquids may be developed from the use of water-cooled concrete cutting equipment and decontamination washes. Liquid effluents will be retained in hold-up tanks and allowed to separate phases. The liquid phase will be sampled and analyzed for Sr-90 content prior to discharge to waterways or sewers per the concentration limits specified in 10 CFR 20, Appendix B Tables 2 and 3, respectively (5 E-7 and 5 E-6 uCi/ml). The sludge phase of any fluids will be solidified and disposed as radioactive waste as necessary.

4.0 FINAL STATUS SURVEYS

It is anticipated that an FSS will be performed in areas that have undergone remediation and/or have the potential to have been impacted by D&D operations.

4.1 Survey Areas Description

Survey areas fall into two categories: no remediation required and adequate data exist to support FSS; and substantial remediation to be followed by FSS. These designations are shown in Table 1-1.

Prior to the characterization summarized in WESTON (1999), all survey units were assigned a MARSSIM survey classification I, II, or III in S&W (1998). A survey classification was applied to floors and lower walls (<6') and upper walls and ceilings. Based on the classification, each survey received the following scanning percentage:

Class I – 100% for lower walls and floors; 25% for upper walls and ceilings,
Class II – 50% for lower walls and floors; 25% for upper walls and ceilings, and
Class III – 25-50% for all floor, wall and ceiling components in survey unit.

4.2 Survey Design

Category 1 – No Remediation Required and Adequate Data Exist to Support FSS

Category 1 includes SU 003, SU 005, SU 006, SU 007, SU 008, and SU 009. As part of the characterization described in WESTON (1999), these areas were subject to large-area surface contamination surveys performed with the SCM-SIMS™ and discrete total, maximum and removable contamination measurements. Under the original MARSSIM classification assigned in S&W (1998), all survey units were Class III, with the exception of SU 003 and SU 005, which had received a Class I assignment. Survey results identified several isolated "hot spots" in SU 005, SU 006 and SU 007. These hot spots were mostly restricted to doorway thresholds, and were decontaminated using physical removal methods. Decontaminated areas were then resurveyed to demonstrate compliance with DCGLs. No hot spots were noted in SU 008.

Based on the measurement saturation obtained with the SCM-SIMS system in these six survey units, and the thorough follow-up survey performed following spot decontamination, it is anticipated that the data may be combined to create a FSS report demonstrating compliance with the surface contamination DCGLs. This activity will be undertaken independent of field D&D and survey activities. It should be noted that, due to the nature of the hot spots (door thresholds), the Class III areas were not reclassified. This action was taken because, in most cases, the Class III areas received far more survey coverage than required (approaching 100% of lower walls and floors).

Category 2 – Substantial Remediation Followed by FSS

SU 001, SU 002 and SU 004 are expected to require substantial D&D operations that will alter the existing building material configuration and radiological characteristics. Activities in SU 001 are expected to leave bare soil where the floor previously was located, and bare structural walls that have had all internal plywood covering removed. Core sampling indicates that any soil contamination will be localized to areas where floor drain systems have been routed. As all building surfaces and soils will have been remediated to substantially less than the DCGLs (after ALARA provisions have been incorporated), the anticipated standard deviation of the survey unit Sr-90 surface contamination levels and soil concentrations is anticipated to be extremely low in comparison to the DCGL and the lower bound of the gray region (LBGR).

In developing the LBGR and subsequent delta (between the DCGL and LBGR) as discussed in MARSSIM, it is evident that these parameters are large with respect to the anticipated sample population standard deviation for Sr-90. Using MARSSIM methods to calculate the required number of survey unit tritium smear samples will result in a Δ/σ (delta/standard deviation) value of much greater than 3 and a Sign p value of 1. Using formula 5-2 from NRC (1997) with type I and II error rates set at 0.05, the calculated number of samples for the sign test is 11 discrete samples from the exposed soil parcel and the wall and ceiling surfaces.

It should be noted that all building and soil surfaces in SU 001 will be subject to a 100% scan using gas-flow proportional counting equipment (or equivalent) and beta scintillators, respectively. The soil requirement is anticipated to be overly restrictive given the non-homogeneous nature of the excavated soil contamination and given the relatively small exposed soil surface area (300-400 ft^2). It is recommended that three discrete soil samples be collected in SU 001 at the approximate location of the floor drain penetration in each laboratory room or locations of the highest measurement as determined by surface scanning the soil with beta scintillation detectors. Additionally, three random soil samples will be collected from the exposed soils of SU 001 and submitted for Sr-90 analyses. Some soil areas may be exposed in SU 005 while removing drain line sections. Should these soils appear to have been impacted, similar sampling protocol will be incorporated.

The surface survey protocol of 100% scan Class I and 11 discrete measurements will be applied to the surfaces of SU 002 and SU 004.

4.2.1 Survey Layout

The survey layout will be placed using a standard grid system following D&D activities. Grid lines will be placed with appropriate survey and marking equipment.

4.2.2 Radiological Monitoring Techniques

Surface Contamination Monitoring

Total surface contamination monitoring will be performed with gas-flow proportional counting or plastic scintillation-based detectors (or equivalent systems as appropriate) as described in Table 3-2. Removable contamination smears will be analyzed with similar systems that are configured for smear analyses.

Portable detectors will be utilized in either scanning or discrete measurement mode. The primary variable (that may be controlled) impacting the portable detector sensitivity during measurement of residual Sr-90 on surfaces is the scanning speed. A separate scanning speed will be established for the final survey of building surfaces versus the unrestricted release of items from an RCA. The target detection sensitivity for scanning of building surfaces is 700-800 dpm/100 cm^2, while the target detection sensitivity for release of material from an RCA is 200-300 dpm/100 cm^2. Detector sensitivities will be calculated using the methods described in NRC (1995) and will incorporate detector background and efficiency at the time of survey. Given the removable contamination DCGLs for building surfaces and unrestricted release of items shown in Table 2-1, attainment of detection sensitivities of 870 dpm/100 cm^2 and 200 dpm/100 cm^2, respectively for these types of measurements will eliminate the requirement for removable contamination measurements. Where such measurements are required, the detection sensitivity will be controlled primarily through variation of counting time.

All raw and calculated surface contamination levels will be recorded electronically or on data forms associated with standard operating procedures.

Soil Parcel Survey and Sampling/Concrete Material Analyses

Section 5.5.2.4 of the MARSSIM describes methods for addressing survey/sampling of small, elevated areas of contamination. This discussion is applicable primarily to pure soil sampling efforts. The implementation of continuous recycle field surveys for total beta and/or gamma radiation over a survey unit will reduce or eliminate the need for special sampling patterns, as verification samples may be collected from biased areas of elevated count rate.

A Utah-certified analytical laboratory will perform Sr-90 analyses of soil and concrete. The analytical technique will be U.S. Environmental Protection Agency (EPA) 901.0 and the required detection sensitivity will be 7 pCi/g or approximately 20% of the DCGL. All samples will be logged, packaged and shipped according to QA/QC protocol. Note that the concrete must also pass surface release criteria.

4.2.3 Final Status Survey Reporting

Upon completion of field remediation activities, a final decommissioning report will be prepared. The report completion is dependent upon receipt of final laboratory data and disposal verification. Upon receipt of this information, a final report discussing the following topical areas will be prepared:

- Survey Procedures.
- Instrumentation Used.
- Coordinate Systems.
- Findings – Residual Contamination/Hazardous Material.
- Results.
- Suggestions.
- QA/QC practices.

The final report will be prepared under the direction of the RAC Project Manager, and will be completed and submitted within 30 working days of receipt of all laboratory results and disposal certifications. The FSS package for all site survey units will be included as an attachment or appendix to the final decommissioning report. Variances from the prescribed DCGLs will be documented and compared to hot spot analysis criteria described in Section 5.3 of MARSSIM.

4.3 Quality Assurance and Control

Quality assurance and control will be ensured through adherence to data quality objectives (DQOs), standard operating procedures, and all requirements specified in the QAPP. Additional QA/QC requirements will include the collection of 20% duplicate discrete survey measurements for remediated survey units, and the collection of one duplicate bulk material sample (to be analyzed for Sr-90) for soil and concrete and each survey unit. The quality of radiological scan surveys performed in remediated survey units will be ensured through the calculation and recording of all parameters affecting the detection sensitivity. These parameters include detector background, scanning speed, and efficiency. Based on equipment preparation/function check procedures, variations of these parameters in excess of +/- 20% will be investigated.

5.0 FUNDING

Remediation activities will be funded through the Defense Environmental Restoration - Formerly Utilized Defense Sites (DERFUDS) Program. Administration will be facilitated through the USACE New York District.

6.0 PHYSICAL SECURITY PLAN AND MATERIAL CONTROL AND ACCOUNTING PLAN PROVISIONS IN PLACE DURING DECOMMISSIONING

All radioactive waste generated during D&D operations will be packaged for transport and disposal. These wastes will be stored in a locked, fenced area prior to shipment, and assigned tracking numbers as part of the waste manifest preparation procedure. Radioactive check sources will be logged and stored in locked containers at a designated equipment storage/maintenance area. The VAECC is a fenced facility with 24-hour security. As part of the security and accountability program, VAECC security personnel will routinely inspect the integrity of the waste storage area. Any theft or vandalism will be reported immediately to the RAC Project Manager and VAECC RSO.

7.0 REFERENCES

NRC 1998a. Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination. Federal Register Notice FR18NO98-121, November 21, 1998.

NRC 1998b. Guidelines for the Decommissioning of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Byproduct, Source, or Special Nuclear Material Licenses. Policy and Guidance Directive FC83-23.

NRC 1997. Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM). U.S. Nuclear Regulatory Commission Report NUREG-1575, December 1997.

NRC 1995. Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions. Draft U.S. Nuclear Regulatory Commission Report NUREG-1507, August 1995.

Ogden/NAE 1998. Data Collection at Defense Environmental Restoration Program – Formerly Used Defense Sites (DERP-FUDS), St. Albans Extended Care Center, Queens, New York (DERP-FUDS Site C02NY0763). Report prepared for Northern Ecological Associates by Ogden Environmental Services, Inc.

S&W 1998. Draft Final Work Plan for the Radiological Characterization Survey of the St. Albans Veterans Extended Care Center Queens, New York. Prepared for the Department of Army, New England District, Corps of Engineers by Stone and Webster Environmental Technology and Services, Boston, Massachusetts. September 1998.

Weston 1999. Radiological Characterization Survey Report – St. Albans Veterans Administration Extended Care Center, Queens, New York. Weston report prepared under Task Order 19, Contract DACA31-96-D-0006. April 1999.

Weston 1999a. Justification for Modified Derived Concentration Guidelines (DCGL). Weston report prepared under Task Order 19, Contract DACA31-96-D-0006. June 1999.

Weston 1999b. Addendum to Justification for Modified Derived Concentration Guidelines for Strontium-90 Concentrations in Soil. Weston report prepared under Task Order 19, Contract DACA31-96-D-0006. October 1999.

8.0 ABBREVIATIONS

Δ/σ	delta/standard deviation
ACM	asbestos-containing material
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
ASTM	American Society for Testing of Materials
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CENAE	North Atlantic Division, New England District
CFR	Code of Federal Regulations
D&D	decontamination and decommissioning
DAC	Derived Air Concentration
DCGLs	derived concentration guidelines
DERFUDS	Defense Environmental Restoration Formerly Utilized Defense Sites
DOD	Department of Defense
DOT	U.S. Department of Transportation
DP	Decommissioning Plan
$dpm/100cm^2$	disintegrations per minute per 100 square centimeters
DQO	data quality objectives
EDE	effective dose equivalent
EPA	United States Environmental Protection Agency
FSS	final status survey
ft^2	square feet
ft^3	cubic feet
GM	Geiger-Mueller (detector)
HASP	Health and Safety Plan
HEPA	High Efficiency Particulate Air
LBGR	lower bound of the gray region
LLD	lower limit of detection
MARSSIM	Multi-Agency Radiological Site Survey Implementation Manual
MDA	minimum detectable activity
NRC	U.S. Nuclear Regulatory Commission
pCi/g	picoCuries per gram
PM	Project Manager
PPE	personnel protective equipment
QAPP	Quality Assurance Program Plan
QC	Quality Control
RAC	Remedial Action Contractor
RCA	radiological control areas
RPP	Radiological Protection Program
RSO	Radiation Safety Officer
RWP	Radiation Work Permits
SOPs	Standard Operating Practices
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeters

uCi/ml	microCuries/milliliter
SCM-SIMS™	Surface Contamination Monitor/Survey Information Management System
QA/QC	Quality Assurance/Quality Control
QAPP	Quality Assurance Project Plan
Sr-90	strontium-90
SU	survey unit
USACE	U.S. Army Corps of Engineers
VAECC	St. Albans Veterans Administration Extended Care Center
VA	U.S. Veterans Administration

APPENDIX A – CONCRETE VOLUMETRIC CONTAMINATION DCGL

This appendix presents the rationale for the volumetric contamination DCGL in concrete and how it is applied. Conceptually, strontium in concrete is in a more stable configuration than strontium in soil. Concrete can be used for water barriers, consequently, water penetration and subsequent contaminant removal is not expected. Postulated St. Albans soil exposure pathways include plant-human intake pathways associated with agricultural scenarios. Direct exposure is not a pathway since strontium-90 or its daughter does not emit gamma radiation. Exposure to a slab of concrete that is not degraded should theoretically result in a lower potential dose than an equal volume of soil with the same strontium-90 concentration. There is no direct exposure, plants do not grow in concrete, and ingestion or inhalation is not possible by definition.

However, degraded concrete could produce a source of surface contamination. A worst case scenario assumes that the entire slab (uniform concentration of 35 pCi/g) was degraded to a depth of 0.5 mm and this was ground to a fine powder. In any 100 cm^2 of surface area, if 0.5 mm is converted to surface contamination (0.5 cm^3) the strontium concentration is 35 pCi/g. The calculation applies a concrete density 2.25 g/cm^3 (Radiological Health Handbook).The resulting surface contamination level is 874 dpm/100 cm^2 as calculated below. It is conservatively postulated that use of larger depths would result in concrete fragments that are not readily available for introduction to the environmental transport pathways.

$$\text{Surface Contamination}\left[\frac{\text{dpm}}{100\text{cm}^2}\right] = 35\ \frac{\text{pCi}}{\text{g}} \times 2.25\frac{\text{g}}{\text{cm}^3} \times 100\text{cm}^2 \times 0.5\text{mm} \times \frac{\text{cm}}{10\text{ mm}} \times 2.22\frac{\text{dpm}}{\text{pCi}}$$

This value represents removable contamination and is at 10 percent of the recommended strontium-90 surface contamination level listed in the Justification for Modified Derived Concentration Guidelines, 24 June 1999.

The concrete DCGL of 35 pCi/g will be applied to intact slabs that are left in survey units after decontamination activities. Concrete with residual contamination removed for decommissioning purposes is disposed as radioactive waste. All samples taken of the concrete must be below the DCGL level of 35 pCi/g, averaging will not be applied. Concrete slabs must also pass the removable and total surface contamination DCGL values.