SVA DECOMMISSIONING PLAN

April 1990

Introduction and Summary,

Sections 2, 3 & 4

Revised August 1990

INTRODUCTION AND SUMMARY

General Atomics (GA) is submitting this SVA Decommissioning Plan to the Nuclear Regulatory Commission for its review and approval. The facility to be decommissioned is GA's Nuclear Fuel Fabrication Facility (referred to herein as "SVA") operated by GA under USNRC License SNM-696 (Docket 70-734) and State of California License 0145-80 and located at 11220 Flintkote Avenue in San Diego, California.

This Plan is generally organized in accordance with the guidelines provided in USNRC Regulatory Guide 3.65, "Standard Format and Content of Decommissioning Plans for Licensees under 10CFR Parts 30, 40, and 70." Included, to assist the NRC in its review, is a compliance matrix showing where the information required by USNRC Reg. Guide 3.65 is presented in the Plan.

The key aspects of the SVA Decommissioning Project are summarized below.

- The scope of work will include: dismantlement, removal, and off-site disposal of equipment and the facility's inner structures; decontamination of the remaining structure, including, if necessary, removal of portions or all of the roof; and post-decontamination radiation and contamination surveys to verify compliance with the NRC-approved and State of California criteria for release to unrestricted use. Upon conclusion of these activities, GA will invite the NRC and the State of California to perform confirmatory surveys to verify that SVA meets the criteria for release to unrestricted use and will request that NRC License SNM-696 and State of California License 0145-80 be amended.

- Bechtel National, Inc. (BNI) is under contract to perform physical decommissioning activities, however, General Atomics (GA) retains the ultimate responsibility for all aspects of the decommissioning work. GA and BNI will work as a team to accomplish the project.

- Bechtel National, Inc. (BNI) and its subcontractors will perform all decommissioning activities with the exception of those to be performed by GA.

- GA will oversee decommissioning activities to assure, and enforce as necessary, compliance with its NRC and State of California licenses. GA will also be responsible for health physics services and radioactive waste disposal.

- All decommissioning activities will be performed in accordance with written procedures and work instructions which have been reviewed and approved by BNI and GA management in accordance with the document-control methodology described in this Decommissioning Plan.

- SVA decommissioning is expected to take nine months and cost $15 million.

- Funding for the project will be provided and assured by Valley Pines Associates (VPA), a 50/50 general partnership between Chevron U.S.A., Inc., and Shell Oil Company.

- All source material and special nuclear material (S&SNM), with the exception of contamination and hold-up in equipment, will have been removed from the facility prior to the start of decommissioning activities. There is no buried waste at the SVA facility.

- The radioactive contaminants present are U-235, U-238, and thorium and its daughters. Much of the facility is either noncontaminated or only slightly contaminated. Some areas, primarily in and around process equipment, are more highly contaminated. Loose contamination levels range up to approximately 13,000 dpm/100 cm^2 alpha and 7,000 dpm/100 cm^2 beta.

- Radiation levels, with the exception of a few areas, are less than 1 mr/hr. The highest radiation level is approximately 50 mr/hr.

- The U-235 remaining in the facility as hold-up in thousands of linear feet of ventilation ducting and in process equipment may exceed 700 grams. Consequently, it will be conservatively assumed that the potential for a criticality accident exists. Nuclear criticality safety will be of primary concern to GA and BNI management, and decommissioning activities will be conducted in accordance with procedures that provide adequate measures to ensure nuclear criticality safety.

- Decommissioning activities will be conducted in accordance with procedures which maintain radiation exposures and releases of radioactive materials to unrestricted areas ALARA.

- Decommissioning activities will be conducted in accordance with the California Division of Industrial Safety Construction Safety Orders, Title 8, Subchapter 4, of the California Administrative Code.

- All radioactive waste generated during decommissioning is expected to be Class A waste and will be disposed of at an authorized low-level radioactive waste disposal site.

- The only mixed waste which may be present is radioactively contaminated asbestos-bearing floor tiles and lead-based paint residue. GA will meet all EPA and disposal site requirements in disposing of this waste, if any. Compliance with these requirements will not adversely impact decommissioning activities or schedules.

- SVA decommissioning will not require any changes to GA's NRC-approved security plan.

- GA will revise its NRC-approved Fundamental Nuclear Material Control (FNMC) Plan to include the SNM control and accounting procedures to be used during decommissioning. The revised FNMC Plan will be submitted to the NRC by May 1, 1990.

- Both BNI and GA will develop and implement Quality Assurance Programs specifically designed to assure that all decommissioning activities are performed in accordance with approved, written procedures adequate to assure compliance with all Decommissioning Project requirements.

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2.0 DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES

2.1 DECOMMISSIONING OBJECTIVE, ACTIVITIES, TASKS, AND SCHEDULES

2.1.1 Objectives

The objective of the SVA Decommissioning Project is to decontaminate SVA such that it meets the criteria for release to unrestricted use.

2.1.2 Decommissioning Alternative Selection

Four basic decommissioning alternatives were evaluated. They are:

- **Leave in Place (SAFSTOR)**

  Evaluated and not considered as an acceptable option.

- **Entombment (ENTOMB)**

  Evaluated and not considered as an acceptable option.

- **Dismantlement [illegible])**

  In this alternative, the dismantlement of the facility would be performed, including removal of the facility structure, followed by NRC and State of California inspections and release to unrestricted use.

- **Decommission in Place (DECON)**

  In this alternative, decommissioning of the facility would be performed with the structure left in place, followed by NRC and State of California inspections and release to unrestricted use. The facility could then be renovated for non-nuclear utilization.

Decommissioning in place has been chosen as a result of discussions with the NRC in which the NRC indicated a preference to inspect and release the facility in place.

2.1.3 Decommissioning Projections and Methodology

The baseline for initial development of this plan was prepared by Bechtel National, Inc. (BNI) for Valley Pines Associates (VPA) and was issued in November, 1988, as a three-volume decommissioning study and cost estimate. The project prerequisites, schedule, project manpower/equipment/services, and techniques for decommissioning of SVA are described in the following sections. This information is based on site-specific information obtained during site walkdowns, from historical records of facility operations, reviews of facility plans and specifications, and the results of site radiological characterization.

2.1.3.1 Site Characterization Methodology

This section describes the site radiological characterization program that was conducted to provide a basis for the development of this plan. It is intended that the previously used methodology and analytical procedures will be continued throughout the decommissioning process.

A preliminary characterization survey of SVA was performed to determine the extent and nature of hazardous materials and radioactive contamination, to support planned decommissioning activities. This section describes the scope of the characterization effort and the procedures used. Further details regarding the site characterization and the results are located in the Characterization Report, Appendix A of this plan. Additional characterization will be performed, as decommissioning activities occur and on an as-needed basis.

Radiological Characterization

A radiological characterization of SVA was conducted to determine the extent, volume, and nature of contamination. An important secondary objective of the characterization was to identify any routes by which contamination may have migrated into and beneath the concrete pad. Initial sampling was conducted using field-generated grid coordinates. The types of measurements taken and the procedures used are described below. Samples were identified by listing the name of the area from the building floor plan, followed with a code describing the sample. Samples were assigned a number designating the sequence of samples obtained from a given location. For example, the first sample of paint from the east wall in the HEPA room on the first floor of SVA was designated as:

HEPA-F-EW-P-6,6-001

where

HEPA = HEPA Room
F = First Floor
EW = East Wall
P = Paint
6,6 = Sample 6 ft south and 6 ft elevation
001 = First sample depth at this location

Sampling Protocols

The characterization primarily included biased sample locations, selected to estimate worst case conditions. These locations were selected on the basis of suspected contamination as determined from operational and unusual events revealed by historical review.

Samples of media were taken under the direction of GA Health Physics and in accordance with a written plan. In general, samples were taken at or near the surfaces in the facility. A few samples were taken at below-surface depths to assess the extent of contamination penetration, particularly where wet operations were conducted. Since most operations were conducted dry, there were few mechanisms for penetration into structural materials. Consequently, most contamination was found at or near the surface. In many cases, walls and/or floors had been covered or recovered to reduce contamination levels in the working environment. The sampling effort was geared to locating contamination hidden under paint, plaster, tile, concrete, or other coverings.

At sample locations, standard health physics smears and direct alpha and beta-gamma readings were taken prior to sampling.

The overheads were sampled using smear/swipe techniques. Dust that had settled on pipes, ducts, and overhead fixtures was the selected sample media. Surface accumulation was removed down to the original surface of the object being smeared/swiped.

The generally exposed portions of the floors have been surveyed by health physics on a routine basis during operations. This historical information is available from the GA Health Physics Organization. The radiological status of the portions of the floors that had been painted, tiled, or otherwise covered was unknown. Samples were required from these coverings and the media immediately below the coverings. Each layer was removed carefully so that an assessment of the radioactivity in each layer could be made. Sampling techniques included drilling, scraping, cutting, and grinding. A typical floor sample required initial removal of paint or tile as the first sample, followed by removal of successive layers of concrete. Each layer was then segregated, packaged, and labeled.

Based on the historical information discussed above, cores were taken at five locations on the first floor. The cores penetrated the slab and extended 3 to 4 ft into the material under the facility. After the first two layers (1/4 in. each) of concrete were removed, a single core sample was taken of the remainder of the slab. Once the core-borer entered the fill beneath the slab, the first 6 in. of soil was taken as the next sample. After the first soil sample, samples were taken every foot until the core-borer reached approximately 4 ft.

Most exposed wall surfaces were slightly contaminated as demonstrated by routine health physics surveys. The wall samples included areas beneath the exposed surface. Consequently, the sample sequence was significant. Depending on their exact location, these samples included paint, tile, gypsum, or concrete. The procedure used for wall sampling was the same as for floors. The outermost surface was removed first to determine what was underneath. Each successive available layer was sampled to a depth free of contamination. Portable alpha and beta-gamma instrument readings were made on each newly exposed surface as sampling progressed.

Since the floor drains in the process area have been sealed, the only drain that was sampled was the drain in the tunnel entry. The sump located in the north tunnel entry was also sampled. Further sampling of the drains and sumps will be performed as decommissioning activities progress.

The roof of SVA is corrugated metal, covered with roofing felts. Releases on the roof may have resulted in deposition of contamination in the tar. At least a portion of the roof had been replaced. Roof samples were taken in the vicinity of the old fume scrubbers (formerly located on the roof) above the south mezzanine and above the thoria spheres mezzanine. Sampling penetrated the full layer of roofing felts, extending to the corrugated metal on the roof. A single composite sample was taken at each of the two locations.

The soot filter is located north of the main building. Smear samples were taken of the inlet and outlet media.

Sample Analysis

Portable survey meters and radiation counting systems in the GA Health Physics Laboratory used for analysis of the preliminary characterization samples were calibrated using sources traceable to the National Institute of Standard and Technology (NIST). This approach to instrument calibration will continue throughout the decommissioning effort. Background determinations were made for soil, concrete, and tar.

Analyses of characterization samples were and will be performed under an approved Quality Assurance/Quality Control (QA/QC) program.

1) Soil Samples. Soil samples were dried, ground, sieved and transferred to the standard 500ml Marinelli beakers used for soil counting at GA. The samples were counted on the high purity germanium detector to assay U-238, U-235, thorium, and thorium daughters.

2) Concrete Samples. Concrete samples were crushed and ground. A sample of approximately 50g was placed in an appropriate container for a gamma scan on the high purity germanium detector. A 50g standard was developed for this purpose. Analysis included U-238, U-235, thorium, and thorium daughters.

3) Smears/Swipes. All smears/swipes were counted on the low-level alpha/beta counting system for gross alpha and gross beta determination.

4) Tile Samples. All tile samples were analyzed by a direct reading with a portable alpha survey meter. Both sides of the tile were counted and reported separately.

5) Sludge/Drain Samples. Sludge/drain samples were analyzed using the method used for water samples at GA. One of the standard water sample geometries was applied depending on the sample volume. These samples were gamma scanned on the high purity germanium detector. Analysis included U-238, U-235, thorium, and thorium daughters.

6) Paint, Dust and Chips. Thin samples of paint, dust, or chips (e.g., tile, gypsum) were counted on a planchet in the low-level alpha/beta counting system. The results were expressed as gross alpha and gross beta activity.

7) Roofing Felt Samples. Samples of the roofing felt were counted on the germanium detector in an appropriate vessel. Clean roofing felt was counted in an identical vessel to establish background. Only the net counts under photopeaks were reported since calibration standards were not available. Analysis included U-238, U-235, thorium, and thorium daughters.

Asbestos Characterization

A characterization was conducted for suspected asbestos-containing material (ACM). Samples were taken of the potential ACM and were sent to an outside laboratory for analysis by polarized light microscopy. The suspect ACM included:

1) Flashing from the roof

2) Lagging on pipes and vessels in the heating and ventilation equipment room on the second floor

3) Lagging on pipes and ducts in the North Annex (first floor)

4) Certain tiles on the first floor

Two samples of the flashing and one sample of each of the other three materials were obtained. The results for these samples are discussed in Appendix A.

2.1.3.2 Potential Accidents

Since the estimated U-235 hold-up in the HEPA system ducts may be as much as 700 grams and the total amount in the facility may be higher, the potential for a criticality accident, although extremely remote, is conservatively assumed to exist. Decommissioning activities will therefore be performed in accordance with procedures which address nuclear criticality safety. Additionally, nuclear safety training will be provided to all decommissioning employees.

The dismantling of ducts which contain accumulations of uranium and/or thorium could potentially result in the contamination of personnel and equipment. The disassembly and removal of ducting will be accomplished using appropriate precautionary measures (see Section 2.1.9.3).

There is a potential for contamination of decommissioning workers during the removal of fuel fabrication equipment, much of which may be internally contaminated. Contamination control measures will be in place, and workers will be adequately trained to protect themselves against such contamination potential.

The roof of SVA is assumed to be partially contaminated, based upon the need for removal and replacement of previously contaminated roofing and upon recent characterization efforts which detected contamination in the new roofing material (see Section 2.1.3.1). The potential therefore exists for a release of contamination to the environment during roof removal. Radiological controls will be applied during roof removal, to ensure the protection of decommissioning workers and the public from contamination. Similar controls will be implemented during decommissioning of the soot filter enclosure, to prevent the release of contamination to the environment.

Moving heavy equipment, particularly from mezzanine levels, involves a potential for construction-related accidents. Precautions will be taken to ensure that the work will be done in accordance with approved rigging and equipment operating procedures.

A large amount of process and service equipment will have been disconnected prior to the initiation of decommissioning work. Worker training in the proper application of equipment tag-out/lock-out procedures will be implemented as an accident prevention measure.

2.1.3.3 Nuclear Criticality Safety

As previously discussed, the quantity of U-235 present in SVA as holdup in the HEPA system ducting is estimated to be 700 grams (based on survey data). It is likely that only small quantities of U-235 holdup are also present in some process equipment. Since the U-235 in the ducts is spread throughout thousands of linear feet of ducting, the potential for a criticality accident during SVA decommissioning is extremely small. However, a criticality accident is possible in the unlikely event that decommissioning activities result in the concentration of much of the U-235 in the facility in one place.

Nuclear criticality safety will therefore be a primary concern. Decommissioning activities will be conducted in accordance with procedures that limit the accumulation of fissile material. These measures will include, but not be limited to, the following:

- Appropriate surveys and analyses for U-235 will be performed.

- Criticality-safe containers will be used for decontamination liquids and solid residues in areas where U-235 was processed during facility operations or where the presence of U-235 was established by characterization surveys.

- The use of HEPA-filtered vacuum cleaners, which are not criticality-safe, will be administratively controlled to prevent the accumulation of unacceptable quantities of U-235. Work instructions will identify the specific areas and material to be vacuumed and will provide clear direction as to the procedure(s) to be followed if material other than that identified by the work instruction is discovered. The areas and the descriptions of the material to be vacuumed will be based on characterization surveys performed prior to the start of work.

- All dismantled sections of HEPA system ducting will be assayed to establish their U-235 content prior to their placement into waste containers.

- Dismantled HEPA-system ducts will most likely be volume-reduced by compaction. Shredding of the ducting will not be utilized since this technique could result in the accumulation of unacceptably high concentrations of U-235 and the generation of waste packages in which the U-235 content cannot be determined with reasonable accuracy.

- Health Physics will monitor the filter(s) in the shredder/compactor ventilation system daily to detect any buildup of radioactive material in the filter(s). The filter(s) will be changed if conservatively established radiation limits are exceeded.

- All prefilters or HEPA filters removed from existing HEPA systems or from portable HEPA-filtered ventilation units will be assayed for U-235 prior to disposal.

- Criticality alarms will be used in work areas where there is a potential for accumulation of U-235 in significant quantities.

In addition to these measures and others which will be invoked as necessary in particular situations, nuclear criticality safety will be emphasized through nuclear safety training for decommissioning workers.

The Manager of Nuclear Safety will review all procedures. He, and if required, the Criticality and Radiation Safety Committee (CRSC), will approve procedures involving potential nuclear criticality concerns. This review/approval will ensure the adequacy of the measures employed to ensure nuclear safety.
The Nuclear Safety Program described above will continue until decommissioning of the facility progresses to a point at which the GA Manager of Nuclear Safety determines that a criticality accident is no longer a credible event.

2.1.4 Project Prerequisites

The following prerequisites are considered necessary for initiation of the SVA Decommissioning Project:

1) All uranium and thorium in any form other than contamination in process and support equipment will be removed from the facility.

2) The facility HEPA ventilation systems that can be utilized for contamination control during decommissioning operations will be identified and available for use. This includes routine maintenance and filter changeout. The remaining HEPA systems may be deactivated in preparation for their dismantlement.

3) Necessary utilities, such as HVAC, water, electricity, natural gas, and breathing air will be maintained at the site until the decommissioning process allows their termination.

4) Systems that may be shared between the facilities to be decommissioned and the remaining GA-occupied space will have been isolated. Dead legs of potentially hazardous gases and liquid systems which have been disconnected will be drained or purged and will be dried prior to their dismantlement.

5) Process and laboratory chemicals will have been removed from the facilities to be decommissioned.

6) Hydrocarbons such as cutting, hydraulic, and lubricating oil will be drained from tool sumps and storage tanks and placed in labeled drums for analysis, treatment, and disposal. Material Safety Data Sheets (MSDS's) will be

supplied for this material. Unused oils will have been removed from the facilities to be decommissioned, prior to the start of decommissioning work.

7) Salvageable process equipment that GA wishes to retain will have been removed from the facility.

2.1.5 Schedule

The projected decommissioning schedule encompasses an approximate total of nine months. The schedule for accomplishing the inter-related activities and tasks of the decommissioning effort is presented as Figure 2.1-1.

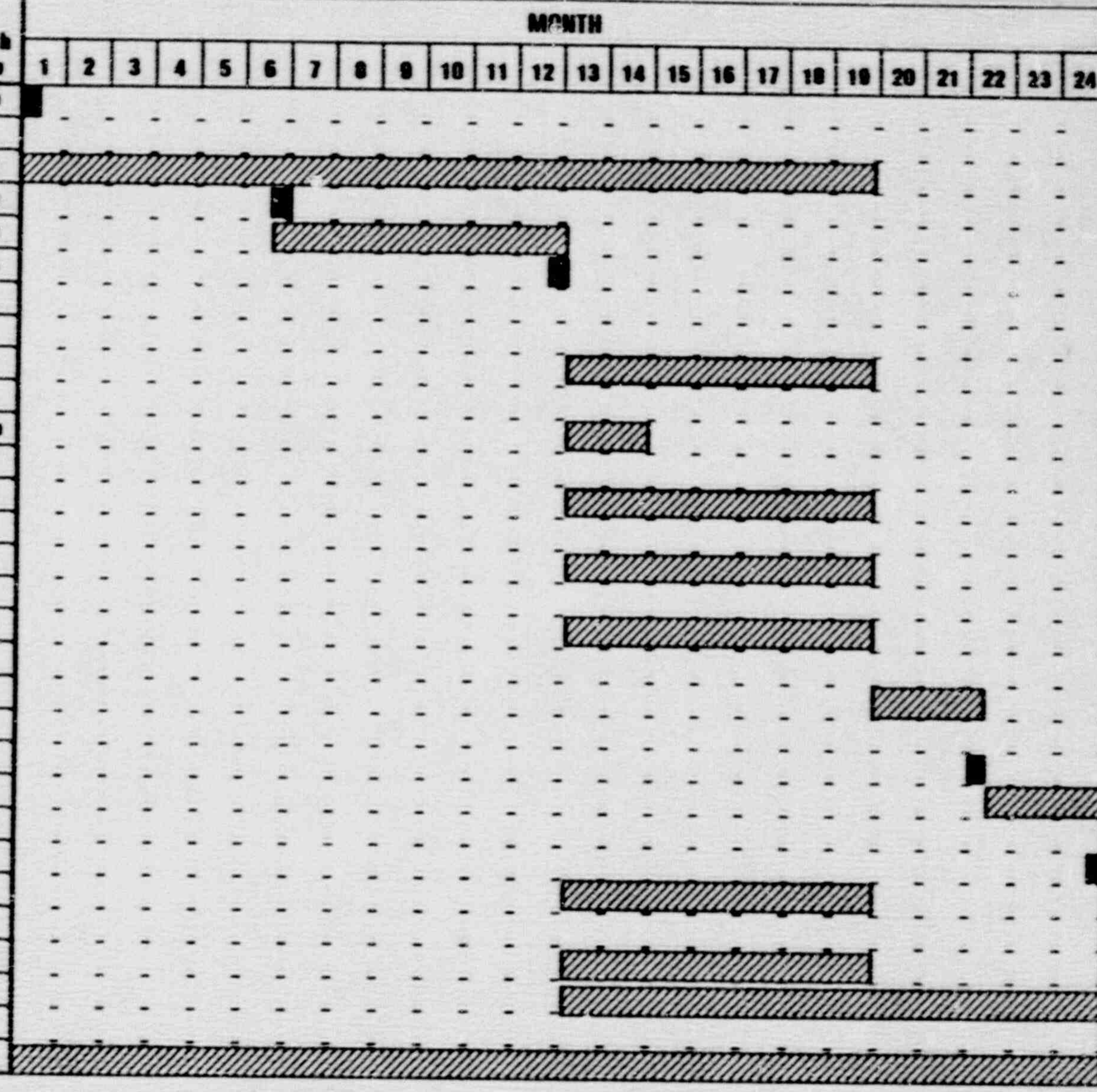
Upon NRC approval of the Decommissioning Plan, site decommissioning operations will commence and will include the decontamination and/or removal of:

1) Extraneous hardware, loose material, miscellaneous residue

2) The process and support equipment

3) Plumbing, electrical wiring and conduit, process, services and utilities piping, HVAC equipment and ducting, and HEPA-filtered ventilation equipment, ducting, and exhaust stacks

		WBS	Start Date	Finish Date
1	BNI Contract Award	J0020446	10/6/89	10/6/89
2	BNI Contract Award			
3	Prepare Plans & Procedures/QA Review	1.0	10/6/89	4/25/91
4	Submit Decommissioning Plan to NRC	1.0	3/30/90	3/30/90
5	NRC Review	1.0	3/30/90	9/28/90
6	NRC Approval		9/28/90	9/28/90
7				
8				
9	Site Characterization	2.0	10/1/90	4/30/91
10				
11	Plant Mods and Temporary Facilities	3.0	10/1/90	11/23/90
12				
13	Decommissioning/Transportation Support	4.0	10/1/90	4/30/91
14				
15	Decon Plant Surfaces, Loose Contamination	5.0	10/1/90	4/30/91
16				
17	Plan Approval Decommissioning Work	6.0/7.0	10/1/90	4/30/91
18				
19	Final Survey - Final Report	10.0	5/1/91	7/1/91
20				
21	Submit Facility for NRC and State Review		7/1/91	7/1/91
22	NRC and State Facility Review		7/2/91	9/27/91
23	NRC Release of Facility		9/27/91	9/27/91
24				
25	Waste Handling & Processing	8.0	10/1/90	4/26/91
26				
27	Transport & Disposal Contaminated	8.1	10/1/90	4/30/91
28	Transport & Disposal - Noncontaminated	8.2	10/1/90	9/27/91
29				
30	Project Management/Technical/HQ Support	11/12/13	10/6/89	9/27/91



Planned Activities

Milestones

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Figure 2.1-1
SVA DECOMMISSIONING PROJECT PLAN SCHEDULE

1 24 1696.4

4) Non-load-bearing interior walls and structures and interior ceiling supports and materials

5) The SVA roof

Scheduled decommissioning activities will also include:

1) NRC and State of California confirmatory radiological surveys, analyses, and evaluation of the remaining structures to be released

2) Preparation of the Decommissioning Project Final Report

2.1.6 Decommissioning Manpower

On-site manpower projections for the approximate nine-month decommissioning schedule are presented in Figure 2.1-2.

2.1.7 Decommissioning Equipment and Services

In addition to the use of a large number of standard decommissioning and dismantlement tools, materials, equipment, and services, some specialized equipment and services will be required for decommissioning of the facility. Those considered for this project are described in the following sections:

2.1.7.1 Specialized Equipment

- **HEPA-Filtered Ventilation Systems**

  Decontamination operations and the disassembly/segmentation of radioactively contaminated items will require the application of contamination control devices and methods.

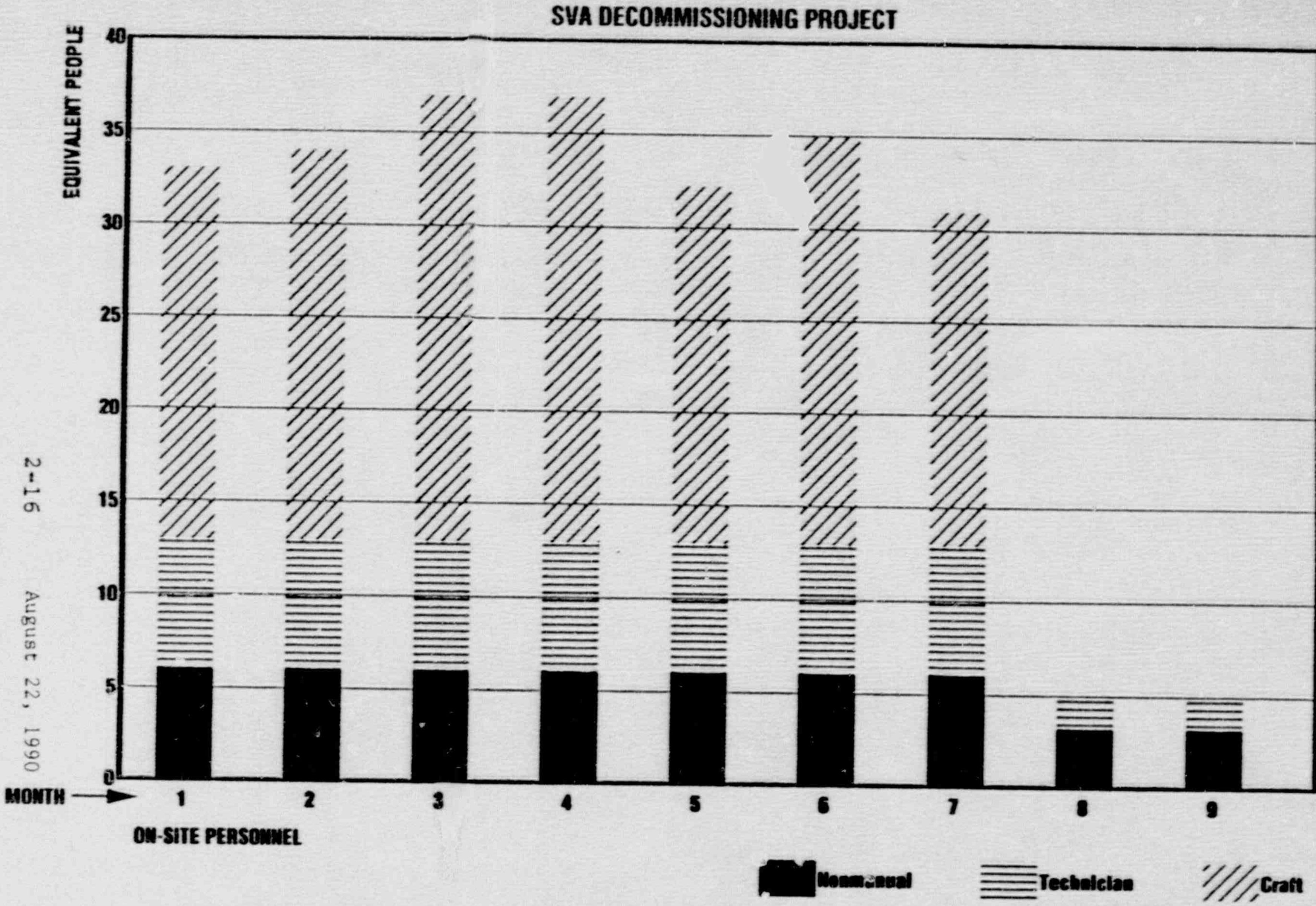

Figure 2.1-2
MANPOWER PROJECTIONS

1 24 1896 3

The existing SVA HEPA-filtered systems will be maintained to provide contamination control coverage. Each system includes rigid ducting directed to specific rooms and/or pieces of equipment. When the system is no longer needed, it will be dismantled, using its own filtering capability to control the spread of contamination during its dismantlement.

The present HEPA system may not always provide adequate contamination control for localized uses. Portable HEPA-filtered ventilation units will be used to provide additional contamination-control ventilation, either in conjunction with the facility systems or as independent systems. Unless monitored for contamination in their exhausts, these units will be exhausted into the existing plant HEPA-filtered ventilation system to ensure against the release of contamination to the SVA interior. High-efficiency HEPA-filtered vacuum cleaners will also be utilized for small volume contamination control, as well as for loose surface decontamination operations. These units will exhaust to the building interior without monitoring of their exhaust.

- **Abrasive Decontamination Equipment**

Surface decontamination operations will be a major part of the overall decommissioning effort. Aggressive decontamination methods will be required to remove existing surface coatings, such as paints, varnishes and similar fixatives, as well as base layers of the surface material, which may also contain embedded contamination. The following decontamination equipment was selected on the basis of efficiency, production rates, and contamination control. Equipment planned for this project has been selected, and modified where necessary to assure that contamination control is fully implemented. The use of decontamination liquids will be strictly controlled to minimize the volume of liquids requiring processing. Used decontamination liquids generated in areas that are contaminated with U-235 will be collected and stored in criticality-safe containers.

1) **Blastrac Machine** - The Blastrac machine is an all-purpose cleaning tool for concrete floors. It can efficiently remove materials such as paint, dirt, grime, and embedded chemical contaminants. The delivery system consists of an enclosed centrifugal blast wheel in the cleaning head. As the wheel spins, metallic abrasive shot are fed into the center and hurled from its blades to blast the floor surface. The abrasive media and contaminants rebound into a separation system which removes the contaminants to an attached dust collector and automatically recycles the abrasive media for re-use. The travel speed of the unit and the shot size both may be adjusted, depending on the required depth of removal. The slower the machine travels, the deeper the etch it creates. Since the shot and dust are collected, a radiological survey may be performed immediately. Any residual metal shot remaining on the floor may be collected quickly with a magnetic broom.

2) **Vacu-blast Machine** - The Vacu-blast machine uses compressed air to convey abrasive media from a pressure generator via an adjustable feed valve through a hose and blast nozzle to discharge against the surface of the item being cleaned. The media is concurrently vacuum-recovered at the point of impact. Air then conveys the media, dust, and debris to the reclaimer where it is air-washed and the media is

returned to the system for recycling. The dust and debris particles are drawn to a secondary cyclone separator and deposited in a collection cylinder. The depth of abrasion is controlled both by adjusting the shot size and travel speed of the unit. This equipment can be used on vertical surfaces.

3) **Scabbling** - Scabbling as a decontamination technique for concrete surfaces has a long history of success. It was used extensively in the Three Mile Island accident recovery program. This technique utilizes tools having 1-, 3-, or 7-bit piston heads, equipped with multipoint tungsten carbide bits. The pneumatically operated tool drives the bits against the concrete surface, which causes the surface to abrade. The pistons for the larger units are mounted in a wheeled chassis to accommodate extensive and unimpeded surfaces. A hand-held unit can be utilized for edging near wall surfaces and other obstructions. Scabbling also has limited application to vertical surfaces.

4) **Abrasive Decontamination of Piping Internals** - For abrasive honing of pipe interior surfaces, a "roto-rooter" device, such as the Flex-hone honing machine, is used in rotationally driving a honing device, which utilizes a choice of abrasive materials and grit sizes and is moved through the pipe to be cleaned. The abraded material can be simultaneously or post-operationally flushed.

5) **High Pressure Water** - Internal pipe surfaces are cleaned by a combination of aggressive abrasion and high-pressure water flushing. Washing and flushing operations are performed with a hydroblaster which delivers up to 10,000 psi water through nozzles specially designed to move themselves through the pipe by virtue of their water jet direction. Nozzle design choice includes rotational capability.

6) **Ultra High Pressure Water** - Ultra-high pressure (UHP) water can be used to scarify concrete and to remove oxidization, paint, and waxes from contaminated surfaces. Water is applied to the contaminated surface with a hand-held lance. The operating pressures can be varied from 10,000 to 33,000 psi. Removal rates depend on the standoff distance from the surface being cleaned, the physical properties of the contaminated material, the roughness of the surface, and the rate of movement. This technique requires the collection and treatment of contaminated water.

7) **Water Treatment Support Equipment** - Water treatment will be used to minimize the accumulation of contaminated liquids from decommissioning tasks, such as decontaminating pipe systems. Particulate filtration will be accomplished with cartridge-matrix multiple filter units, sized to provide water of sufficient quality to permit its reuse in subsequent decontamination tasks. Additional water treatment will be provided by the use of appropriately sized ion exchange resin columns.

- **Sectioning Equipment**

  1) **Plasma Arc Cutting** - Segmentation of large metal pieces, including process equipment, (either in preparation for shredding or as final volume reduction), can be accomplished with plasma arc cutting equipment.

  2) **Mechanical Cutting Equipment** - If modification or removal is required, it can be efficiently accomplished by utilizing powered equipment. The following specialized equipment will be considered:

     - Power tear-off machine

     - Power-bladed covering remover (e.g., tile, roofing, tar paper)

     - Sectioning saw

- **Volume Reduction Equipment**

  Volume reduction of radioactive waste is a major part of the economics of decommissioning projects. The methods of volume reduction which will be used for this project are briefly described below and detailed later in Section 2.1.10.1.

  1) **Shredder** - This equipment will shred an appreciable percentage of the contaminated materials associated with the decommissioning project. The following are typical shreddable materials: electrical conduit, small-diameter piping, desks, chairs, benches, stud-and-plaster walls, sheet metal, and masonry block.

2) **Compactor** - This equipment, a subassembly to the shredder, compacts waste into boxes, obtaining volume reduction factors of up to 10:1, depending on material.

3) **Baler** - This GA-owned equipment compacts waste at a reduction ratio up to 17:1, depending on material, and produces a baled waste form.

2.1.7.2 Specialized Services

Specialized services required in support of SVA decommissioning activities include:

1) Medical coverage to supply routine first-aid services and to perform pre-employment (and, if necessary, closeout) physical examinations.

2) Health physics coverage to monitor all work activities. Analytical capabilities will support the Health Physics technician's daily activities, as well as specific sample evaluation.

3) Analytical services for industrial hygiene and radiological safety.

4) Employee training, instrument calibration, and emergency preparedness.

5) Transportation of nonradioactive waste to a local landfill.

6) Transportation of radioactive waste to an authorized waste disposal site.

7) Segmentation and removal of large concrete and structural members, utilizing abrasive cutting techniques.

2.1.7.3 Installation of Support Equipment

The majority of decommissioning support equipment will be portable, requiring utility connections, (electricity, air, and water). This category includes air compressors, vacuum cleaners, HEPA-filtered ventilation units, water treatment equipment, dry abrasion/vacuum equipment, concrete cutting and rubblizing equipment, and electrical generators. Existing utilities will be utilized when available. When they are no longer available, portable utility sources will be utilized.

2.1.8 SVA Equipment Disposition

This section describes the work methods for dismantling process and process-support equipment and hardware in preparation for the equipment's final disposition. These methods will facilitate both the decontamination operations and contamination surveillance required either to clear the equipment for unrestricted disposal or to identify their contamination status for disposal as radioactive waste.

2.1.8.1 Dismantling Techniques

Facility process equipment will not be salvaged for reuse. Consequently, the dismantlement objectives exclude re-assembly concerns and include only efficiency of decontamination, volume reduction, and final handling based on safety and cost-effectiveness considerations. Despite dismantlement efforts, many of the resultant pieces will be unwieldy and heavy, requiring approved special handling and rigging techniques.

Applicable task-specific and generic procedures in conjunction with training will assure the proper uses of these techniques.

Generally, equipment dismantlement will require standard disassembly and segmenting methods, which include powered and manual tools (e.g., portable band saws and abrasive cut-off wheels). Pneumatically operated tools (jack hammers) and flame operated tools (torches) will be used only with the application of adequate contamination control measures. More specialized segmentation methods, such as plasma arc cutting, may be necessary for certain equipment items, particularly if their original assembly required welding.

2.1.8.2 Decontamination Methods

The standard pattern of decontamination operations will be to utilize the simpler and more passive methods first, advancing to more aggressive methods as the need dictates. The majority of contamination removed from the equipment will be loose or semi-fixed particulate material contained either on interior surfaces or in confined exterior points. In some instances, contamination fixatives, such as varnish and paint, have been used on exposed surfaces. These fixatives will require the more aggressive techniques for removal.

The generic decontamination methods for equipment treatment will be described in detail in operational procedures and will consist of those techniques discussed later in Section 2.1.9.

The decontamination of equipment leading to nonradioactive waste will be conducted until the cost of further decontamination is judged to override the cost benefit. The decontamination program will then be terminated and the item prepared for radioactive waste disposal.

2.1.8.3 Contamination Control Methods

Good radiological control practices require the containment of loose contamination. During decommissioning, certain work evolutions will require the segmentation of components, the demolition of concrete, and the opening of process systems, which may disturb residual contamination. Typical contamination control measures, including containment envelopes, surface fixatives, and localized ventilation, are described below.

1) Tents - Containment tents are effective for controlling the spread of airborne and surface contamination with larger work pieces and equipment. The enclosures are usually fabricated from reinforced plastic fabric attached to externally framed aluminum pipes and scaffolding clamps or to structural framework. Personnel and equipment entrances are controlled with self-latching fabrics or zippers. Clear plastic panels can be added for surveillance purposes or to reduce the need for internal lighting. These enclosures will be maintained under negative pressure. Standard practice is to evacuate the enclosure through HEPA filters, using either the building's ventilation system or an appropriately sized auxiliary system.

2) Glove Bags - Glove bags will be either selected from vendor catalogues or fabricated at the jobsite from Herculite or similar materials. These bags can be equipped with glove ports and sleeves for HEPA-filtered ventilation and for pouch transfer. A variation of the glove bag, plastic sleeving, or tubing can be used for segmenting contaminated piping. Immediately after segmentation, the sleeving is extended over the cut surfaces and sealed. The bags may also be equipped to utilize drains and other external devices or functions.

Their use has consistently proven to be both cost effective and in accordance with ALARA principles. Their utilization establishes a contamination control barrier between the worker and the work environment, as well as limiting that environment to a minimum volume. The manufacturers of these products provide detailed instructions for their use. In addition, their proper utilization will be ensured through training programs and procedures.

3) Surface Fixatives - It is sometimes prudent to protect clean work area surfaces before radiological work is initiated, or to fix otherwise transferable contamination on surfaces to be handled. A good quality fixative on porous surfaces is helpful in contamination control. Wrapping items with plastic sheeting and applying strippable coatings are additional means of adequate surface protection which may be utilized.

4) Use of Local Ventilation - Portions of the existing HEPA-filtered ventilation system will be maintained and adapted to serve project needs. Temporary ducting will be attached to the existing ductwork and positioned to supply localized suction at the work area or to create negative pressure within a tent or glove bag. The installed HEPA-filtered system will be maintained for use throughout the phases of work that might generate loose surface or airborne contamination. The use of local ventilation will be administered through training programs and approved procedures.

Portable HEPA-filtered ventilation units will be available to augment the permanent system's capabilities and to serve as an independent means of contamination control.

2.1.9 Facility Disposition

After the majority of the process and support equipment has been removed, the remaining interior structures (housings, walls, floors, etc.) will be evaluated for compliance with release criteria. Those interior structures which are contaminated will be subjected to volume-reduction measures and prepared for final packaging/transportation as radioactive waste.

2.1.9.1 Decontamination Methods

When feasible, passive decontamination techniques will be applied, as dictated by radioactive surface characterizations and previous work experience. These techniques include standard vacuuming, damp cloth wiping and, to a limited degree, hand washing/scrubbing operations.

When these passive methods fail to reduce surface contamination to releasable levels, more aggressive decontamination methods, such as those previously discussed in Section 2.1.7.1, may be applied. The following aggressive techniques are identified in order of their preference as a decontamination method:

1) Dry abrasive blasting, with vacuum

2) Scabbling/scarification

3) Ultra-high-pressure water

2.1.9.2 Building 37 North Roof Removal

The roof of Building 37 North, consisting of weather coat and steel deck, may require partial or complete removal to verify the absence of radiological contamination within the roof zone. The roofing is assumed to be partially contaminated based upon (1) a previous history of radioactive releases within the building, and (2) recent characterization efforts which detected some minimal radioactive contamination in the roofing material.

If the roof cannot be decontaminated, removing the roof from Building 37 North will require additional measures. They include:

1) Preventing the spread of radiological contamination during removal

2) Providing an interim weather cover in place of the existing roof, from the time the roof is removed until the building is dismantled

Description of Roof

The roof of Building 37 North is a steel deck roof consisting of a ribbed steel deck topped with a layer of rigid insulation, multiple layers of roofing felt, and a bituminous top dressing. The steel decking consists of sheets of steel which are 24 in. wide and have 1-1/2 in. deep longitudinal ribs spaced about 6 in. on centers. The sheets of steel decking span the supporting roof trusses and purlins, to which they are welded. The layer of rigid insulation is 1 in. thick and appears to be glued to the underlying steel decking.

Removal of Roofing Weather Coat

The removal process for the weather coating layers (i.e., bituminous top dressing, roofing felts, and rigid insulation) includes cutting the layers into sections; freeing the cut sections from the roof deck; gathering the material into rolls or stacks; and conveying it to the ground via chutes or lifting devices. This will be accomplished using both motorized and manual equipment. Motorized equipment, which is generally preferred on larger areas, includes power tearoff machines, roofing saws, and power-bladed roof removers for loosening the cut sections from the deck. Manual methods will be used on smaller areas and to supplement the motorized equipment. These methods include sectioning with long-handled chisels and knife-bladed splitters and loosening the sections from the deck with long-handled scrapers.

In the case of Building 37 North, the weather coating will be removed to the top of the metal decking; however, since the rigid insulation board might be glued to the underlying metal, small amounts of the weather coating may remain on the decking. Radiological controls will be implemented during the operations. The weather coat is relatively new and is not expected to be contaminated. However, radiological contamination was detected in one small area by a recent characterization survey. As a precaution, all the removed material will be monitored for radiological contamination.

Removal of Steel Deck

The typical removal process for steel roof decking welded to its supporting structure (such as that found in Building 37 North) includes cutting the welds; removing the sheets of decking; grouping them into bundles; and transferring them from the roof to the ground. The welds may be broken with impact chisels, saws, cutting torches, or nibblers.

Removing the steel roof deck will require an increased level of radiological controls since some contamination could exist on the inner surfaces within the interstices of the decking (e.g., seams and joints). If loose contamination is identified, appropriate measures (decontamination, applying a fixative, wrappings) will be implemented to prevent its spread.

Until proven otherwise, the removed steel decking will be considered contaminated. The radiological controls will focus on preventing the spread of contamination during removal and handling. The contaminated decking will be decontaminated or prepared for containerization and shipment as radioactive waste. Radiological controls during this phase will focus on controlling the spread of contamination during the cutting and containerization operations.

If economical to decontaminate the decking, or if contamination in the decking is not widespread, decontamination techniques will be utilized to prepare the material for release as nonradioactive waste to a conventional landfill. Radiological controls during this phase will focus on surveys of the decking.

Maintaining Building Containment

Contamination controls will be implemented during and following roof removal operations. These controls will focus on 1) preventing any residual contamination inside the building from spreading to the building's exterior, and 2) preventing any contamination within the roof zone from spreading to the building's interior. These measures will also provide weather protection for the building's interior.

During roof removal operations, the roof zone will be covered by plastic sheeting inside the building near roof level. This membrane will be draped from the roof trusses and purlins. The function of this membrane will be to prevent particulates within the building from escaping to the exterior.

A more durable interim cover will be provided to maintain building containment from the time the existing roof is removed until the building is dismantled. The interim cover will be built using general construction materials and practices and will be comprised of a decking for structural support and an outer membrane for weather protection. The decking will be built of timber (2x4s, 2x6s, etc.) and roof sheathing (plywood, waferboard, oriented strand bo[illegible], etc.) and will be secured to the existing roof trusses an[illegible] purlins. The outer membrane will be waterproof, tear resistant, and resistant to sunlight for its design life. The membrane will also be secured against wind uplift. Provisions will be made to route rain water to the building roof drains.

2.1.9.3 HEPA-Filtered Exhaust Ductwork and HVAC

HVAC duct removal will be accomplished using standard contamination control methodology until its interior surfaces can be checked for contaminants. The ducts will be surveyed for U-235. When significant U-235 contamination is detected, the portion of the ducting branch will be removed as contaminated and transferred to a staging area for packaging as radioactive waste.

The HEPA ductwork will be dismantled starting at the furthermost points from the system fans. Negative pressure will constantly be applied through HEPA filters as disassembly proceeds. The open ends of the ducting will be secured with plastic sheeting as they are exposed. The ducting will be segmented into manageable lengths at joints or flanges, if possible, and will be transferred to a staging area for U-235 assay prior to packaging. This procedure will be repeated until only the filter housing, blower housing, and discharge piping remain. The blower will then be de-energized at the breaker which will be removed if possible, and electrical leads will be lifted both at the source and at the fan motor. The prefilters and HEPA filters will be removed from the filter housing, bagged, assayed for U-235, and delivered to a staging area for volume-reduction and packaging. The filter housing and blower will then be similarly processed. The exhaust ducts (downstream side of blowers) will be dismantled and surveyed. If certified clean, they will be discarded at the local landfill. If radiologically contaminated, they will be subjected to volume reduction, packaged, and disposed of as radioactive waste.

2.1.9.4 Sanitary Sewer and Radioactive Waste Systems

Sanitary Sewer System

Exposed sanitary system pipe contained within the facility will be surveyed, decontaminated if necessary, and segmented for transport to a local landfill. If releasable levels cannot be achieved, the pipe will be disposed of as radioactive waste.

Exposed/Above Ground Radioactive Waste System

Exposed radioactive piping and equipment, such as contaminated sinks and traps, will be isolated, cleaned to meet release levels, or subjected to volume-reduction methods and packaged for radioactive waste disposal. The piping and equipment will include (1) associated plumbing fixtures (2) drain lines, and (3) vent lines.

Embedded/Underground Radioactive Waste Sub-System

The radioactive waste lateral drain lines buried below the main-level floor may be internally decontaminated by applying high-pressure water through hydro-driven nozzles. The nozzles will be directed into the main-floor-level opening of each lateral until the lateral and its appendages are cleared to the radioactive waste header located in the tunnel. The water utilized in the decontamination operation will be treated with the original filtration/holding system located in the tunnel and/or the water treatment equipment discussed in Section 2.1.7.1 and will be recycled for re-use in subsequent operations. When this decontamination operation is complete, the clean condition of the piping interior will be verified with a radiation detection instrument remotely manipulated through the piping. If necessary, a flex-hone will be inserted into the pipe (at each trap including vent-lines) and the pipe will be further decontaminated. The pipe will again be examined for radioactivity with a remotely-manipulated detector (as above). Any residual contamination present at this time will be assumed to be fixed. The buried laterals will then be disconnected from the radioactive waste header located in the tunnel. If the lines have no detectable contamination after the above two operations, they will be plugged and abandoned in-place. If fixed contamination is still present, the lines will be excavated and removed for disposal as radioactive waste.

Tunnel-Contained Radioactive Sub-System

This portion of the radioactive waste system includes waste storage (a serpentine pipe and four storage tanks) and a mixing/transfer/treatment system comprised of circulating pumps, a compressor for sparging air, a bank of cartridge filters, and the necessary interconnecting piping.

The system will be drained and all valves closed to isolate individual components. Interconnecting lines will be removed and segmented for disposal. Pumps, compressors, a wall-mounted service sink, and the cartridge filter array will be volume-reduced (if possible) and packaged for disposal as radioactive waste.

2.1.9.5 Conduit and Piping

The majority of the unpainted conduit is assumed to be externally noncontaminated or capable of cost-effective decontamination. Likewise, all internal surfaces and contents are assumed to be noncontaminated. If feasible, the wiring will be removed and cleared for release as nonradioactive waste. The conduit sections will be decontaminated as necessary and cleared for release as nonradioactive waste. Conduit runs that do not permit utilization of the decontamination, dismantlement, and removal procedure described will be treated as contaminated and boxed for disposal as radioactive waste.

Piping that supplies gas, oil, water, air, and vacuum service, in addition to more specialized process services, ranges in size from 1/2 in. to 6 in. in diameter. As with the facility conduit, the unpainted pipe is expected to be noncontaminated and can be released as nonradioactive waste. Otherwise, the piping will be volume-reduced and disposed of as radioactive waste.

2.1.9.6 Vaults and Liquid Waste Storage Room

The west and south vaults and the liquid waste storage room will be decontaminated using the techniques previously described and will be released for subsequent dismantlement.

2.1.9.7 Concrete Flooring, Main, and Mezzanine Levels

The main floor of Building 37 North is nominally 8 in. thick and the floors of ancillary structures are 4 in. thick, both constructed of reinforced concrete. The floors may have surface coatings, such as tile which may contain asbestos (see Section 3.1.2), paint which may contain lead, or varnish. Decontamination of floor surfaces will require removal of these coatings, and/or sufficient removal of concrete surface to permit release of the areas to unrestricted use. During removal of floor tile and paint, measures will be taken to protect workers from exposure to asbestos and/or lead-bearing materials.

Localized floor removal may be required if contamination has penetrated the floor into the underlying soil. The technique for segmentation and removal of these localized areas is described as follows. Concrete saws will be used to cut the slabs into squares sized to facilitate their handling for removal. The cuts will be aligned to take advantage of the slab's expansion joints. The depth of cut will be adjusted to leave approximately 1 in. of concrete intact. The cutting coolant will be retained with wet-vacuuming equipment. Coolant which could transport previously undetected contamination into the soil below the slab will be collected and routinely sampled for radioactivity. If the presence of contamination is noted, mitigating actions will be initiated to control its spread.

2.1.9.8 Precast Concrete Wall Panels (Tilt-ups)

The exterior walls of Building 37 are constructed of 7.5 in.-thick, 10-ft wide precast concrete wall panels varying in height from 24 to 35 ft. Concrete surfaces will be subjected to radiological characterization and decontaminated as necessary to meet the criteria for release to unrestricted use, using both passive and aggressive techniques previously described.

2.1.9.9 Sub-Floor Soil

Where the substructure is exposed, subfloor soil samples will be taken and analyzed for radioactive contamination, with specific emphasis on soil below slab cracks and expansion joints and on soil volumes adjacent to penetrations such as floor drains and clean-out ports. If contaminated soil is detected, the extent of contamination will be determined. The contaminated soil will be removed until the area meets the criteria for release to unrestricted use. The contaminated soil will be packaged for disposal as radioactive waste. Soil samples from site characterization surveys indicate that the sub-floor soil is not contaminated.

2.1.9.10 Building 37 North Structural Steel, Columns, and Roof Trusses

A basic assumption is made that inaccessible, repainted, or recoated surfaces are radioactively contaminated. Structural steel will be decontaminated using passive hands-on techniques. Follow-up surveys will either clear the items for release to unrestricted use or identify those surfaces requiring further decontamination.

Recoated steel surfaces and contaminated surfaces may be subjected to more aggressive cleaning techniques, to the extent that surface coatings are removed to the base metal. Subsequent surveys will identify those structural members that can be released to unrestricted use. If isolated surfaces cannot be cost-effectively cleaned, the affected volumes will be sectioned and packaged for disposal as radioactive waste.

2.1.9.11 Interior Walls and Ceilings

The majority of the SVA interior partitions and walls are standard metal stud and two-surface drywall construction, described as "stud-and-plaster" (S&P). It is assumed that they are contaminated and will be packaged for disposal as radioactive waste.

Sheet metal was utilized to form some wall and ceiling surfaces of interior areas of Building 37 North and its appurtenant buildings. These surfaces and surface coatings will be subjected to decontamination operations necessary to clear the steel for release as nonradioactive waste. Surfaces which cannot be easily decontaminated will be volume-reduced if possible and will be packaged for disposal as radioactive waste.

2.1.9.12 Concrete Block Construction

Interior building surfaces that are constructed of cinder/concrete block and exhibit surface contamination will be vacuumed or coated with fixative and disposed of as radioactive waste.

2.1.9.13 West Shed, Service Building, and North Annex Structures

The decommissioning procedures described here will apply to all ancillary structures. They are similarly constructed with corrugated metal walls and roofs, and supported by standard structural steel with 4 in. thick concrete slab floors poured on-grade.

The structural shells will be emptied of equipment, hardware and extraneous material, and surfaces decontaminated to meet the criteria for release to unrestricted use. Passive decontamination of the walls, roofs, and structural steel will be initiated, followed with abrasive techniques as necessary. If these methods are unsuccessful, the structures will be disposed of as radioactive waste.

The concrete floor will be stripped of paint, wax, and tile coatings. Contaminated concrete will be subjected to abrasive scarification operations until all surfaces qualify for release to unrestricted use. It is conceivable that localized contamination could have migrated into expansion joints, cracks, or other types of penetrations. Special attention will be given to clearing these areas, including fracturing and removing the concrete.

2.1.9.14 Interface at Column 25, A through P

A double walled isolation zone currently exists between Building 37 North and the remaining southern portion of Building 37. See Figures 1.3-1 and 1.3-2. Prior to the initiation of decommissioning operations, this isolation zone will be modified to provide an engineered barrier at the interface between the two facilities. The purpose of this barrier will be to provide personnel access control and to prevent the migration of contamination (and casual debris) between the two facilities. The design of the personnel access will permit emergency exit only.

2.1.9.15 Soot Filter Enclosure

The enclosure is a free standing structure, external to Building 37 North, with nominal dimensions of a 10 ft square base and a 26 ft height. Its external construction is corrugated steel siding and steel deck plate roofing, supported by steel beam-and-channel mounted on a 6 in. concrete slab. It is designed to house a large bag filter and miscellaneous ducting and is ventilated by the plant's HEPA system.

Prior to decommissioning the enclosure, the plant HEPA-ventilation system will be operationally verified and will be augmented or replaced as necessary with a portable HEPA-ventilation system. Appropriate contamination control measures, as discussed in Section 2.1.8.3, will be established and maintained throughout the operation.

The decontamination and decommissioning of the structure's interior will be accomplished utilizing anticontamination clothing and respiratory protection for the work crews, as well as contamination control measures for protection of the general environment.

Various decontamination techniques, discussed previously, will be selected. These techniques will be applied to the internal surfaces of the structure and to the miscellaneous ducting and equipment housed within the structure.

Parts, materials, and components which cannot be decontaminated will be disassembled and bagged (or contaminants otherwise fixed) for transport to the volume reduction equipment discussed in Section 2.1.10.1. The interior surfaces of the structure will be decontaminated to meet the limits for release as nonradioactive waste or will have a fixative applied as a contamination control measure.

The exterior structure will be verified free of contamination or will have appropriate fixatives applied to the contaminated areas. The previously established contamination control measures will be maintained throughout the dismantlement of the structure.

The roofing, exterior siding, and structural support steel will be dismantled and segmented as necessary for processing as nonradioactive or radioactive waste.

2.1.10 Waste Processing of Radioactive and Nonradioactive Materials

One of the most cost-sensitive activities associated with decommissioning of the facility will be the management and disposal of resultant wastes. An efficient waste disposal program includes the minimization of disposable volumes, as well as the reconciliation of waste density and volume with transport regulations and limits.

Alternative decontamination methods have been considered to achieve two objectives: (1) to reduce the contamination levels throughout the facility to minimize personnel exposure during disassembly; and (2) to attempt to decontaminate as much material as possible to meet the criteria for release as nonradioactive waste, thereby reducing the quantities of material that must be packaged and shipped to a radioactive waste disposal site.

Liquid Decontamination Wastes

Decontamination methods can involve the use of both chemicals and water to remove radioactive materials. Chemical decontamination (Nutech 316 or equivalent suitable for solidification and disposal) may be performed on systems and building surfaces that have measurable residual contamination.

In the few cases where chemical and water decontamination methods are used, the liquid wastes will be processed prior to disposal. The generation of liquids will be held to a minimum. In areas contaminated by U-235, liquid waste will be collected in criticality-safe containers. After analysis for U-235 content, the liquid will be either decontaminated to unrestricted release levels or solidified.

Contaminated Equipment and Materials

Much of the equipment in the various systems in the building has been contaminated during the operation of the facility. This equipment will be either decontaminated, released, and disposed of as nonradioactive waste at a local landfill or disposed of as radioactive waste. The choice between the two options depends on the activity levels, achievable decontamination, personnel exposure, and the decontamination costs.

Non-Contaminated Equipment and Materials

Major equipment expected to be in this category of waste will be control equipment, instruments, electrical conduits, piping, supports, HVAC system, and motors. It is also expected that significant quantities of noncontaminated structural concrete and steel will be removed as part of the decommissioning process.

The release procedure for disposal of noncontaminated equipment and materials as nonradioactive waste is described in Section 3.5.8.2. Nonradioactive wastes can be shipped without packaging. Therefore, volume-reduction of these wastes is required only as necessary to meet the size and weight capabilities of loading and hauling vehicles.

2.1.10.1 Volume Reduction

Waste volume reduction will be accomplished through the three techniques described below.

- **Baler/Compactor**

  The equipment to be utilized for this process is an existing GA-owned and licensed baler located at the Nuclear Waste Processing Facility adjacent to SVA. The baler process compresses solid waste into approximately 37 in. x 21 in. x 18 in. bales which are subsequently placed into strong-tight shipping containers. This technique requires the waste to be removed from the machine and placed into a shipping container under strict ventilation control to prevent the spread of contamination. The baling technique provides a potential volume reduction of 17:1.

- **Shredder/Compactor**

The shredder/compactor process uses a commercially available high-torque shredder/compactor. The process shreds the waste into small pieces, so that it will occupy less packaging/shipping space than the original item. The shredding process can be augmented by partial disassembly and/or segmentation prior to shredding. The shredded waste is then compressed by the compactor section of the assembly directly into a strong tight shipping container placed below the compactor assembly. This operation will be performed under strict ventilation control to prevent the spread of contamination.

The following list is representative of items that can be volume-reduced through shredding:

- Steel panels

- Small diameter piping and conduit

- Corrugated steel

- Furniture

- Stud & plaster walls

- Ceiling tile

- Floor tile

- Concrete block

- Steel containers

- Miscellaneous glass columns

- **Nesting**

  Contaminated piping and large components containing internal void volumes are logical candidates for volume-reduction by nesting. For example, successively smaller-diameter piping and tubing may be segmented to identical lengths and nested together, one into the other, to form a reliable, minimum-void-volume package for disposition. Similarly, an internally contaminated tank that is not easily volume-reduced may be filled with shredded or loose contaminated material and sealed, facilitating efficient waste disposal.

2.1.10.2 Solidification, Absorption and Filtration

Small amounts of liquid will be generated during decommissioning operations. Most liquid wastes resulting from decommissioning activities will contain relatively small quantities of radioactive and/or chemical contaminants.

Low-level aqueous wastes from which it would be too expensive to remove the radioactive contaminants will be solidified prior to disposal. These liquids will be primarily water that was utilized for decontamination of piping and surfaces.

Solidification will be performed in GA's Nuclear Waste Processing Facility (NWPF) located adjacent to SVA. Approximately 30 gallons of waste are placed in a 55-gallon drum and neutralized with an acid or base, if necessary. The contents of the drum are mixed while adding cement and expanded mica (filler) until the mix level is within a few inches of the top of the drum. Each drum requires approximately 100 pounds of concrete and 50 pounds of filler. The mix is allowed to set up for one to two days. The contents of the drum will become a monolithic mass containing a negligible amount of liquid.

As an alternative to solidification, some low-level aqueous wastes may be decontaminated by treatment with the original SVA filtration/holding system or with the water treatment equipment discussed in Section 2.1.7.1. After certification for unrestricted release, the treated water will be discharged into the sanitary sewer system in accordance with local, state, and federal (NRC) regulations. The captured residue from the water treatment will be solidified as described above.

Contaminated organic liquids such as pump oil, solvents, etc. will be absorbed with appropriate solidification agents which are acceptable for disposal at radioactive waste disposal sites. The resultant solid material will be packaged in 55-gallon barrels. This operation will be performed at GA's NWPF with a motor-driven cement mixer with an approximate 100-gallon capacity.

2.1.11 Management of Decommissioning Activities

Several key factors comprise the management of decommissioning activities. These factors include: the management plan, work instructions and procedures, and readiness reviews.

2.1.11.1 Management Plan

A management plan prepared early in the planning and preparation stage has provided a format within which the programmatic aspects of the decommissioning have been established. This management plan includes the following components:

1) This Decommissioning Plan

2) The Project Safety & Health Physics Program (See Section 3.3)

3) The Project Organization Charts (See Section 2.2)

4) Project Procedures (See Section 2.1.11.2)

5) The Project QA Plan (See Section 7.0)

6) Project Schedules Including Project Milestones (See Section 2.1.5)

7) Project Budgets and Funding (See Section 5.0)

Each of these elements is discussed in the referenced sections of this plan.

2.1.11.2 Unit Work Instructions (UWIs) and Procedures

Written work instructions and procedures will be used to perform and control decommissioning activities. The procedures will establish the methodology for performing specific tasks. BNI procedures will be reviewed and approved by BNI management. The BNI review/approval process is covered by an approved BNI document control procedure. GA approval of BNI procedures involving, or potentially involving, work with radioactive material will be in accordance with the Work Authorization approval process described in Section 3.2.3.1.

New or revised GA procedures will be reviewed/approved in accordance with the existing GA document control procedures for the GA organizations involved.

BNI will use Unit Work Instructions (UWIs) in the performance of daily activities of the decommissioning work force. The UWIs will invoke GA/BNI approved procedures to perform specific tasks or activities. These UWIs will be reviewed/approved by the BNI Health Physicist, BNI QA Engineer, and the BNI Project Engineer or their designees.

Approved BNI and GA procedures covering work with radioactive material will be incorporated into one or more GA Work Authorizations (WAs). These WAs will be reviewed/approved by GA in accordance with the WA approval process (see Section 3.2.3.1). UWIs will not require WA-type approval. UWIs involving work with radioactive material will invoke the requirements of Radiological Work Permits.

Work instructions and procedures are based on licensing, permitting and specific technical/engineering considerations and shall be in compliance with this decommissioning plan. The work instructions and procedures shall contain all the information required by the field staff to perform the work. The work instructions and procedures will include information on the following:

- Disassembly, dismantling, and demolition methods
- Detailed sequences

- o Radioactive waste collection, processing, temporary storage, and shipment
- o Contamination control techniques
- o Equipment operation and maintenance
- o Methods for ensuring industrial safety
- o Security procedures
- o Activity specifications
- o Equipment operations
- o Training procedures
- o Systems operation
- o Environmental monitoring procedures
- o Health Physics procedures
- o Reporting requirements
- o Emergency procedures
- o Procurement procedures
- o Work instructions
- o Toxic and hazardous materials control, if any

All of the individual work instructions and procedures will provide for feedback/correction to ensure proper safety, adequacy, and efficiency of operations.

Procedures and work instructions will be controlled so that worker qualifications are defined, quality assurance hold points are an integral part of the program, and management reviews occur throughout the entire process.

2.1.11.3 Readiness Review

A readiness review will be accomplished prior to commencement of daily site activities. When determined by the BNI Project Manager/designee, a more detailed assessment may be performed by conducting a Quality Assurance assessment. The purpose of the readiness review is to evaluate the status of the prerequisites for starting field activities and to formulate an action plan to complete the prerequisites prior to starting work.

A Quality Assurance assessment is used to evaluate a specific field activity when the BNI Project Manager/designee determines that there is a potential for the occurrence of significant quality failures or health and safety failures.

2.1.12 NRC Review and Approval (Clean Waste Surveys - Confirmatory Survey)

During the decommissioning of SVA, much of the debris originating from the facility's structures (e.g., concrete floors, walls, roof, etc.) will be disposed of as non-radioactive waste at a local landfill. Such disposal will be accomplished in accordance with Annex C to GA's license SNM-696, "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," dated July 1982. GA will notify NRC Region V authorities at least 30 days in advance of projected shipments to provide the opportunity for them to survey the waste prior to its disposal.

Waste will be shipped to a local landfill for disposal, subsequent to; 1) the completion of the NRC survey confirming that the material meets the criteria for unrestricted release, or 2) the expiration of the 30-day notification period, at which time the material will be considered released, based upon locally performed and documented surveys.

When SVA has been decontaminated, GA will perform a comprehensive radiation and contamination survey to demonstrate compliance with NRC-approved criteria for its release to unrestricted use. The results of these surveys will be submitted, as part of the final decommissioning report, to the Fuel Cycle Safety Branch, USNRC, Washington, D.C., with a copy to the Administrator, USNRC Region V.

With the submission of the final report, GA will invite the NRC to perform confirmatory surveys to verify that the facility meets the criteria for release to unrestricted use. GA will also request that its NRC special nuclear material license SNM-696 be amended to indicate that the subject facility is released for unrestricted use and is deleted from license SNM-696.

2.1.13 State of California Review and Approval

California is an agreement state. The thorium and depleted uranium used in SVA is licensed under GA's California Radioactive Materials License No. 0145-80.

As noted in Section 2.1.12, during the decommissioning of the facility a large amount of decommissioning debris will be disposed of as nonradioactive waste at a local landfill. Such disposal shall be performed in accordance with the State of California's Department of Health Service's criteria, "DHS Criteria for Release of Facilities and Equipment to Unrestricted Use," also known as "DECON-1." For the radionuclides of concern in the decommissioning of SVA, the criteria in DECON-1 are the same as those of the USNRC.

When SVA has been decontaminated, GA shall perform a comprehensive radiation and contamination survey to demonstrate compliance with the State of California approved criteria for its release to unrestricted use. GA will submit the results of this survey as part of the final survey report to the Radiological Health Branch of the Department of Health Services, with a copy to the Environmental Management Branch.

With the submission of the final report, GA will invite the Department of Health Services to perform confirmatory surveys to verify that the facility meets the criteria for its release to unrestricted use. The Department of Health Services will be advised that the results of the USNRC confirmatory survey will be made available for their use.

With the submittal of the final report, GA will request that its California Radioactive Material License (No. 0145-80) be amended to indicate that the subject facility is released for unrestricted use.

2.2 Decommissioning Organization and Responsibilities

2.2.1 Contractual Relationships

The organizations with major interests in the decommissioning are:

- General Atomics (GA) - Owns the equipment within SVA, operated the fuel manufacturing business and is the holder of the NRC license (SNM-696)

- Valley Pines Associates (VPA) - A 50/50 general partnership between Chevron U.S.A., Inc., and Shell Oil Company

- Bechtel National, Inc. (BNI) - Is responsible for and will perform decommissioning activities

- U. S. Department of Energy (DOE) - Most of the radioactive waste from the decommissioning project will be disposed of at a DOE burial site as the majority of the uranium processed in SVA was from AEC contracts

- Public Service Company of Colorado (PSC) - Procured nuclear fuel fabricated at SVA prior to the decommissioning project

- Lindbergh Properties, Inc. - Has title to the land where SVA is located

- Sorrento Valley Real Estate, Inc. - Owns the SVA building

- Subcontractors to GA and BNI

Lindbergh Properties, Inc., Sorrento Valley Real Estate, Inc., and General Atomics are all affiliated companies, under the common control of Neal Blue and Linden Blue.

VPA is one of the former owners of GA. Through various agreements, VPA has retained the financial responsibility for the SVA Decommissioning Project.

All PSC uranium, materials, and equipment have been removed from SVA.

For the decommissioning project, GA, as the licensee, will be responsible for ensuring compliance with and enforcement of the NRC and State of California license requirements and government regulations and for interfacing with the NRC and other regulatory agencies.

GA will be responsible for disposal of radioactive waste at authorized burial sites and for providing the primary health physics service. VPA, with GA's consent, has hired BNI to provide the organization for the decommissioning effort.

Figure 2.2-1 shows the relationship of the organizations from a business standpoint for funding source, administration of the decommissioning project, and performance of the work. Figure 2.2-2A shows the functional relationship with respect to regulatory requirements. Figure 2.2-2B shows the GA/BNI functional management interfaces.

2.2.2 GA Responsibilities and Organization

2.2.2.1 Responsibilities

This plan was developed by GA and BNI. GA is submitting this plan for the decommissioning of SVA. In furtherance of this plan, GA shall: (1) assist BNI in developing a detailed implementation schedule for the decommissioning project, (2) prepare specifications and furnish procedures for the decommissioning project, and (3) perform decommissioning activities in compliance with GA's assigned responsibilities.

GA will have the primary responsibility for health physics and related services to ensure compliance with and enforcement of NRC

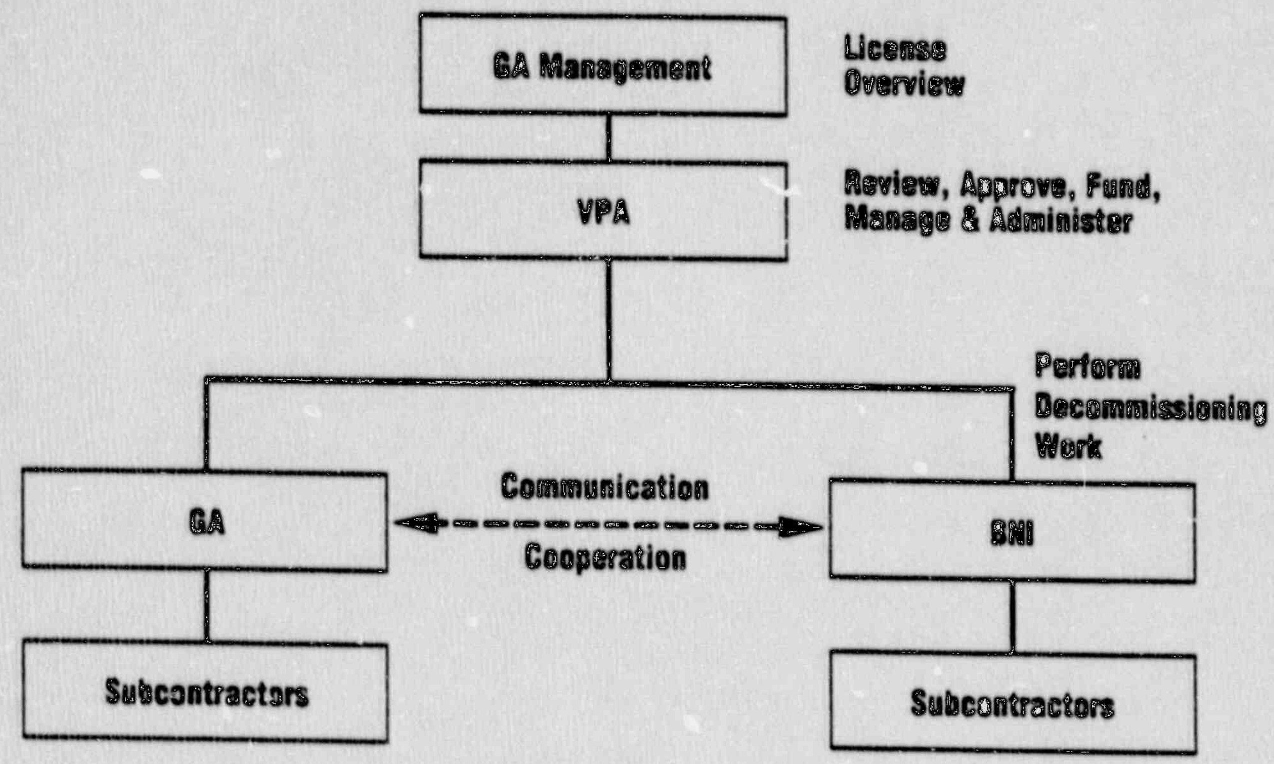

Figure 2.2-1
BUSINESS RELATIONSHIPS

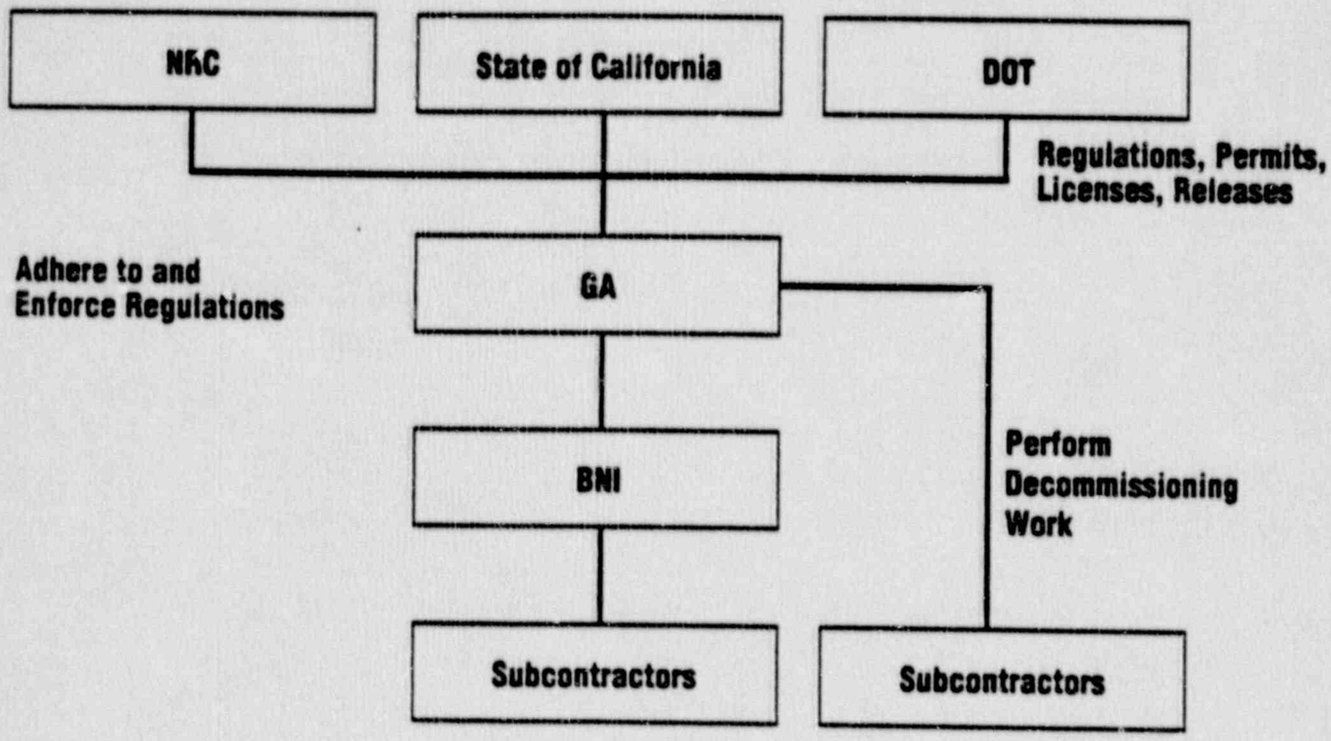

Figure 2.2-2A
REGULATORY REPONSIBILITY RELATIONSHIPS

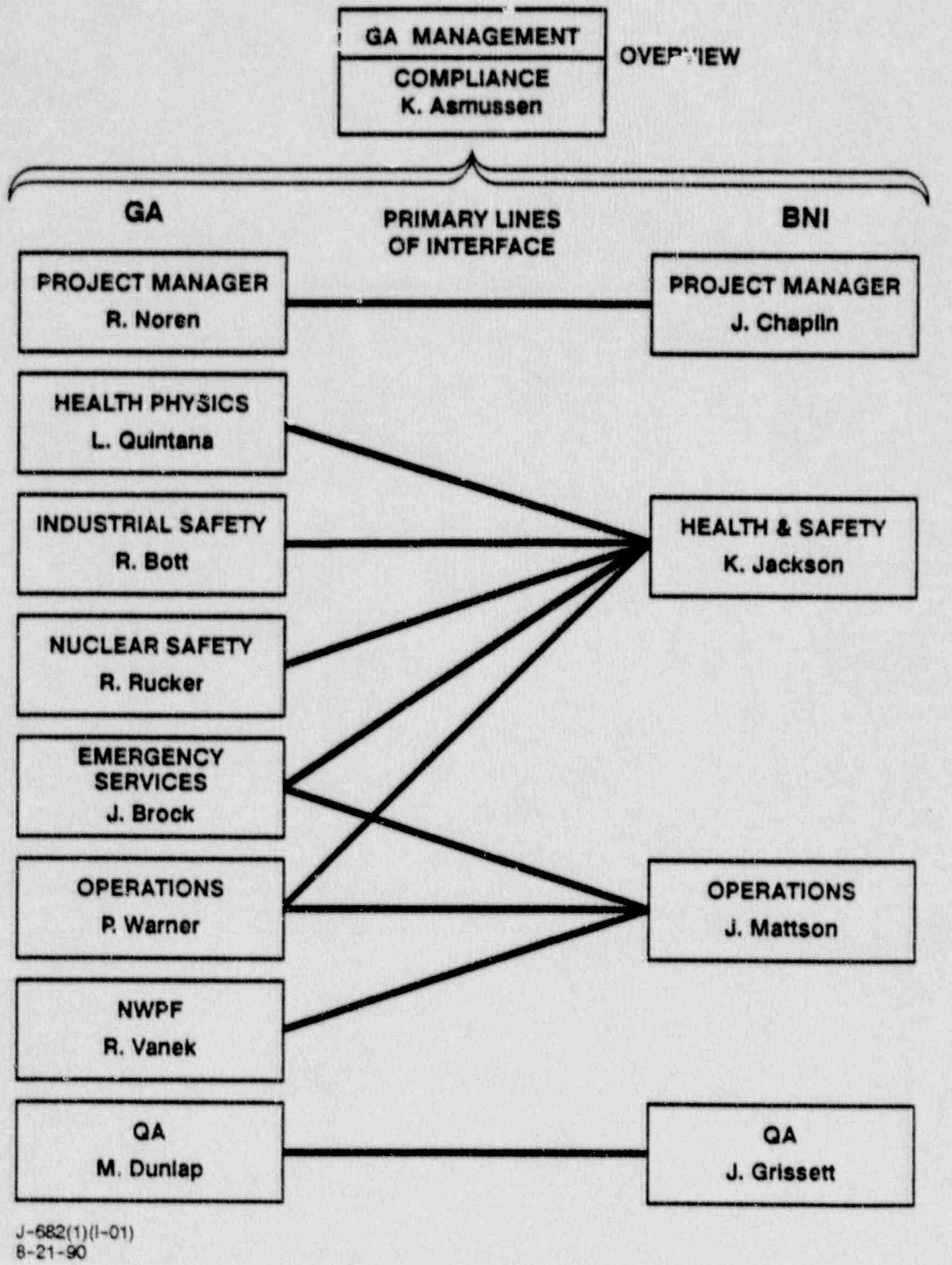

Fig. 2.2-2 B. GA/BNI functional management interfaces for SVA decommissioning project

license requirements and will interface with the NRC and other regulatory agencies. Submittals to and substantive communications with the NRC and other governmental agencies concerning the decommissioning will be the responsibility of GA.

The major responsibilities for GA during the project are to:

1) Ensure compliance with and enforcement of NRC and other governmental regulations

2) Ensure compliance with and enforcement of GA's NRC License SNM-696 and State of California License 0145-80

3) Provide the security force

4) Provide medical support

5) Provide operational health physics

6) Provide radiological training

7) Provide personnel monitoring

8) Provide environmental monitoring

9) Dispose of radioactive waste, including:

   - measurement

   - documentation

   - certification

- transportation

- disposal

10) Perform license-related QA/QC

11) Perform the final radiation survey

12) Prepare the final report to the NRC

13) Remove hazardous materials (nonradioactive)

14) Inspect and release waste for clean landfill disposal

15) Interface with governmental agencies

16) Conduct all public relations

17) Perform other services as agreed between VPA and BNI

2.2.2.2 Organization

The GA organizations which will be participating in the SVA Decommissioning Project are shown in Figure 2.2-3. These organizations are managed by employees who are well qualified to perform their decommissioning project responsibilities by virtue of their past and continuing experience in performing similar or related responsibilities. These responsibilities include project management, nuclear waste processing, quality control, quality assurance, industrial safety, licensing, health physics, nuclear safety, nuclear material accountability, statistics and measurement control, and security.

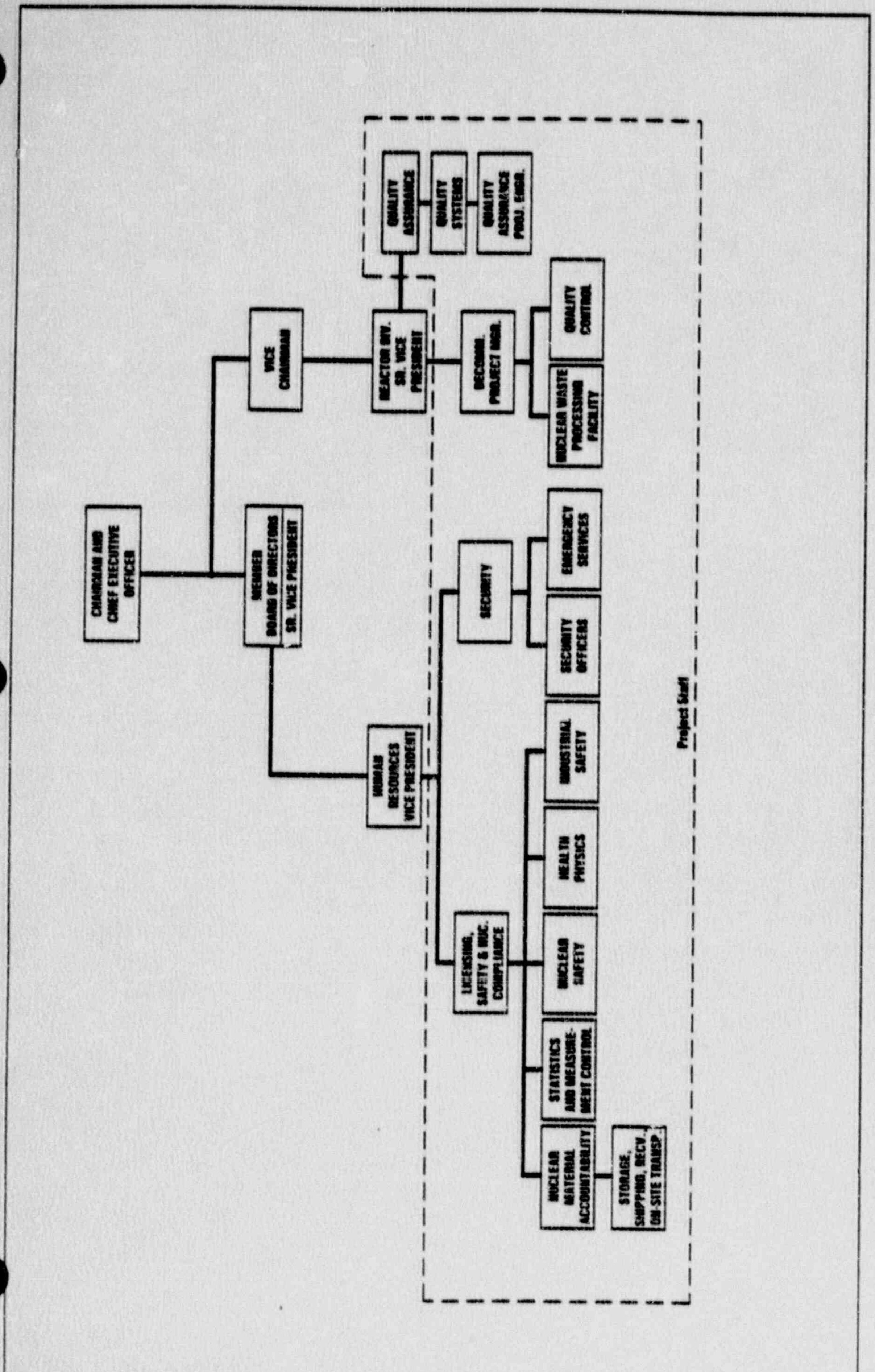

Figure 2.2-3
GA ORGANIZATION FOR SVA DECOMMISSIONING PROJECT

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The responsibilities and minimum qualifications* for each position are discussed below:

GA Project Manager

The Project Manager is responsible to GA's management for the successful completion of the project (i.e., release of the facility to unrestricted use). He has the overall responsibility for the control and safety of GA decommissioning activities and for protection of the environment against potential radiation/contamination exposures from these activities. The specific responsibilities of the Project Manager include:

- Approving procedures required for implementation of the Decommissioning Plan
- Approving overall project plans and GA budget
- Coordinating the activities of the GA staff to carry out GA's responsibilities for the decommissioning project
- Approving GA subcontracts
- Interfacing with BNI and VPA management
- Ensuring, with the assistance of the GA Licensing, Safety and Nuclear Compliance organization (see Section 3.2.2) and BNI management, that the conduct of decommissioning

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* A two year degree plus five years of directly applicable experience or a high school diploma (or equivalent) plus 10 years of directly applicable experience may be considered equivalent to an accredited four-year college degree and appropriate work experience, subject to a determination by the Vice President of Human Resources that proficiency has been demonstrated.

activities is in compliance with all criteria, rules, and practices as set forth in the decommissioning project procedures approved in accordance with GA's license

The minimum qualifications for the Project Manager include:

- An accredited four-year college degree in science or engineering

- At least five years of nuclear industry project management experience

- At least two (of the five) years of project management experience on decommissioning or decontamination projects.

- Substantial knowledge of the SVA facility including its history and the operations conducted therein

- Knowledge of GA's radiation protection, nuclear safety, and industrial safety programs and experience in interfacing with the GA Compliance organization

<u>Manager, Nuclear Waste Processing Facility (NWPF)</u>

The NWPF Manager is responsible for the treatment and packaging of radioactive waste generated during decommissioning and for its shipment to authorized low-level radioactive waste disposal sites. The specific decommissioning responsibilities of the NWPF Manager include:

- Maintaining a qualified NWPF work force sufficient to meet GA's radioactive waste disposal responsibilities

- Reviewing, approving, and maintaining the procedures required for operation of the NWPF

- Reviewing NWPF operations for industrial, radiological, and nuclear safety

- Maintaining accurate and current nuclear material inventories

- Attending annual refresher courses on nuclear safety, health physics, and DOT regulations

The qualifications for the NWPF Manager include:

- An accredited four-year college degree

- Two years previous experience in nuclear waste processing and shipping

- Knowledge of waste treatment and packaging methods, NRC and DOT regulations, and waste disposal site requirements

- Knowledge of GA's radiation protection, nuclear safety, and industrial safety programs and past experience in interfacing with the GA Compliance organization

Manager, Quality Control (QC)

The Quality Control Manager is responsible for inspection of radioactive waste packaging and shipping activities and for inspection of nonradioactive waste to verify conformance to release criteria. The specific decommissioning responsibilities of the QC Manager include:

- Developing and implementing the procedures for quality control inspection of radioactive waste packaging and shipping activities

- Developing and implementing the procedures for QC verification that wastes classified as nonradioactive waste meet the criteria for release as nonradioactive waste

- Maintaining a qualified staff of QC inspectors sufficient to meet GA's waste inspection responsibilities

The qualifications for the QC Manager include:

- An accredited four-year college degree

- Two years of previous experience in quality control inspection of radioactive waste packaging and shipping activities

- Knowledge of waste treatment and packaging methods, NRC and DOT regulations, and waste disposal site requirements

- Knowledge of uranium and thorium assay/survey techniques, including measurement theory

Manager, Project Quality Assurance (QA)

The Project QA Manager is responsible for developing and maintaining the GA QA Program for the SVA Decommissioning Project and coordinating its implementation with the GA Project Manager and the BNI Project QA Manager. This function will be independent of the project manager, the NWPF, QC, and Compliance organizations, and will have direct access to upper management on matters related to quality. The Project QA Manager will have the authority to stop work which is in noncompliance with approved procedures. The specific decommissioning responsibilities of the Project QA Manager include:

- Developing, implementing, and maintaining the GA Quality Assurance Program Document (QAPD, see Section 7.2)

- Reviewing and approving decommissioning procedures and changes thereto for conformance with quality related requirements, and providing quality-related input in accordance with the GA WA process

- Providing quality assurance indoctrination and training to GA project personnel

- Supervising/performing audits and surveillances to verify conformance of GA organization activities to written operating procedures and decommissioning project requirements and to verify the effectiveness of the BNI Quality Assurance Program

- Advising project management on quality-related inquiries from internal and external sources and on the results of QA audits and surveillances

- Reviewing and approving completed Corrective Action Requests (CARs) and monitoring decommissioning activities to verify that approved corrective actions are effectively implemented

The qualifications for the Project QA Manager include:

- An accredited four-year college degree

- Two years experience as lead QA representative on related projects

- Knowledge of GA's radiation protection, nuclear safety, and industrial safety programs and past experience in interfacing with the GA Compliance organization

- Certified Lead Auditor, ANSI/ASME N45.2.23

<u>**Quality Assurance Project Engineer**</u>

The Quality Assurance Project Engineer reports directly to and receives direction from the Project QA Manager. The specific decommissioning responsibilities of the Quality Assurance Project Engineer include:

- Performing overview surveillances of decommissioning activities

- Assisting the Project QA Manager in audits of GA decommissioning activities and of the BNI Quality Program

- Assisting the Project QA Manager in the quality assurance indoctrination and training of GA project personnel

- Reviewing project procedures for quality requirements and providing quality-related input in accordance with the GA WA process

- Initiating Surveillance Discrepancy Reports (SDRs), Nonconformance Reports (NCRs), and Corrective Action Requests (CARs) as necessary to identify and correct conditions adverse to quality

The qualifications for the Quality Assurance Project Engineer include:

- o An accredited four-year college degree
- o Certified Auditor, ANSI/ASME N45.2.23
- o Two years experience as a project quality engineer on related projects

Manager, Industrial Safety

The Industrial Safety Manager has the responsibility to ensure compliance with the Occupational Safety and Health Act of 1970 and the California Occupational Safety and Health Act of 1973 as implemented by California Labor Code Section 6400 and the General Industry Safety Orders (GISO 3203). The specific decommissioning responsibilities of the Industrial Safety Manager include:

- o Conducting a training program to instruct employees in general safe work practices

- o Reviewing BNI procedures to verify adequate coverage of industrial safety concerns and requirements

- o Performing periodic inspections of decommissioning work areas and activities to identify and correct any unsafe conditions and work practices

- o Providing industrial hygiene services as required

- o Advising project management on industrial-safety matters and on the results of periodic safety inspections

The qualifications for the Industrial Safety Manager include:

- o An accredited four-year college degree

- Two years of directly applicable experience as a safety engineer
- Knowledge of and demonstrated proficiency in the technical aspects of industrial safety programs, worker's compensation insurance, industrial hygiene, and environmental laws

Manager, Licensing, Safety, and Nuclear Compliance (LSNC)

This function administers licenses and reviews and approves all Work Authorizations (WA) for compliance with applicable regulations and license conditions. It also provides interpretation of licenses and regulations, determines the need for licensing actions, coordinates the preparation and processing of applications, disseminates license requirements to the operating organization, and maintains or oversees maintenance of master license records for NRC or GA audit or review. This function is responsible for the overall planning, coordination, and administration of the special nuclear material (SNM) measurement control and accounting, nuclear safety, health physics, and industrial safety functions.

The Manager, Licensing, Safety and Nuclear Compliance shall have demonstrated his/her proficiency in activities relevant to the functions assigned. Demonstration of this proficiency shall be based on the manager's qualifications which shall include an accredited four-year college degree in science, engineering, or other related field, a minimum of two years applicable work experience and/or training in nuclear industry management, and a knowledge of nuclear safety and health physics.

Manager, Health Physics

This function ensures compliance with Title 10 Code of Federal Regulations, Parts 19 and 20 and with license-imposed radiological safety requirements. It is responsible for review and approval of proposed activities involving radioactive material, monitoring/auditing of such activities, and providing services such as personnel and environmental monitoring, dose rate measurement, radioactive material detection, and assay. The Manager, Health Physics, or designee, will be responsible for ensuring that decommissioning activities are conducted in compliance with GA's license and the Health Physics program described in this Plan.

The GA Health Physics Manager will have demonstrated his/her proficiency in nuclear health physics activities and the evaluation of potential radiological hazards. Demonstration of this proficiency shall be based on the manager's qualifications, which shall include an accredited four-year college degree in the physical sciences, biological sciences, or other related field and a minimum of two years of experience in phases of nuclear health physics related to GA licensed activities and the evaluation of potential radiological hazards therefrom, or equivalent work experience or training.

The GA Health Physics Manager and the BNI HSM shall each have the authority to stop any decommissioning activity which threatens the safety and health of workers, visitors, or the public or the quality of the environment.

Manager, Nuclear Safety

This function reviews and approves proposed activities and changes in processes, equipment, and procedures involving SNM for nuclear safety considerations, performs inspections, and monitors operations to ensure implementation of the required nuclear safety controls. This function approves all determinations of criticality limits, provides and obtains independent verifications thereof, and approves the content of training programs for nuclear safety.

The Manager, Nuclear Safety shall have demonstrated his/her proficiency in activities of nuclear safety and outside reactor criticality safety. Demonstration of this proficiency shall be based on the manager's qualifications which shall include an accredited four-year college degree in science, engineering, or other related field and a minimum of two years experience and/or training in nuclear safety activities related to nuclear fuel fabrication operations, or one year of such experience and/or training plus two years of other nuclear safety experience and/or training.

Manager, Nuclear Material Accountability (NMA)

This function ensures compliance with source material and SNM (S&SNM) custody and control regulations and practices. It implements the program for accountability, custody, and control of S&SNM. NMA maintains a manual of S&SNM accounting control procedures to ensure compliance with safeguards and material control and accountability regulations and license requirements.

The Manager, Nuclear Materials Accountability shall have demonstrated his/her proficiency in activities relevant to the functions assigned. Demonstration of this proficiency shall be based on the manager's qualifications, which shall include an accredited four-year college degree in a relevant field and a minimum of two years applicable work experience and/or training in nuclear material accountability and control activities.

Manager, Statistics and Measurement Control

This function implements the measurement control program as it relates to SNM control and accountability. It ensures adequate

SNM measurement quality by monitoring and evaluating applicable information, practices, and procedures. This function is also responsible for maintaining a SNM Measurements & Statistical Control Manual, which contains the procedures for performing measurements, meeting measurement control and statistical control requirements, and implementing a program to provide measurement and statistical information for inventories, receipts, and shipment for SNM.

The Manager, Statistics and Measurement Control shall have demonstrated his/her proficiency in activities relevant to the functions assigned. Demonstration of this proficiency shall be based on the manager's qualifications which shall include an accredited four-year college degree in science, engineering or other related field and a minimum of two years applicable work experience and/or training in quality assurance/control, statistics, nuclear measurements, or related activities.

Manager, Security

This function provides security measures to prevent unauthorized access to SNM and to provide appropriate industrial security. The security function is responsible for ensuring compliance with federal and state security regulations and license conditions, including maintenance of a security force. The security responsibilities related to SVA decommissioning will include the following: 1) development and implementation of a security plan for SVA decommissioning, 2) providing for physical protection of SVA during decommissioning, including maintenance of an Entry/Exit Control Station to control access into the facility, and 3) monitoring the fire and intrusion alarms and related communication systems.

The Manager, Security, shall have demonstrated his/her proficiency in activities relevant to the functions assigned.

Demonstration of this proficiency shall be based on the manager's qualifications, which shall include an accredited four-year degree and a minimum of two years applicable work experience and/or training in managing or implementing security programs in the nuclear industry.

2.2.3 BNI Responsibilities and Organization

BNI and GA have developed this Plan for decommissioning of SVA. BNI shall (1) develop a detailed implementation schedule for the decommissioning project, (2) design, prepare, and furnish work instructions and procedures for the project, and (3) perform decommissioning activities. BNI shall perform all obligations and duties necessary for the successful completion of the project, including furnishing design, engineering, equipment, material, tools, supplies, labor, services, and project supervision in accordance with the Plan. The work will be in accordance with the facility license and all applicable laws, rules, and regulations. To accomplish the foregoing, BNI will coordinate activities with GA.

2.2.3.1 Decommissioning Project Organization

The BNI project organization is displayed in Figure 2.2-4. The identified positions are staffed with senior employees qualified by virtue of their past and continuing work experience on similar or related projects. Their responsibilities include project management and control, administrative and office support, technical and engineering, procurement, materials control, and quality assurance.

Responsibilities and minimum qualifications for each position identified in Figure 2.2-4 are as follows:

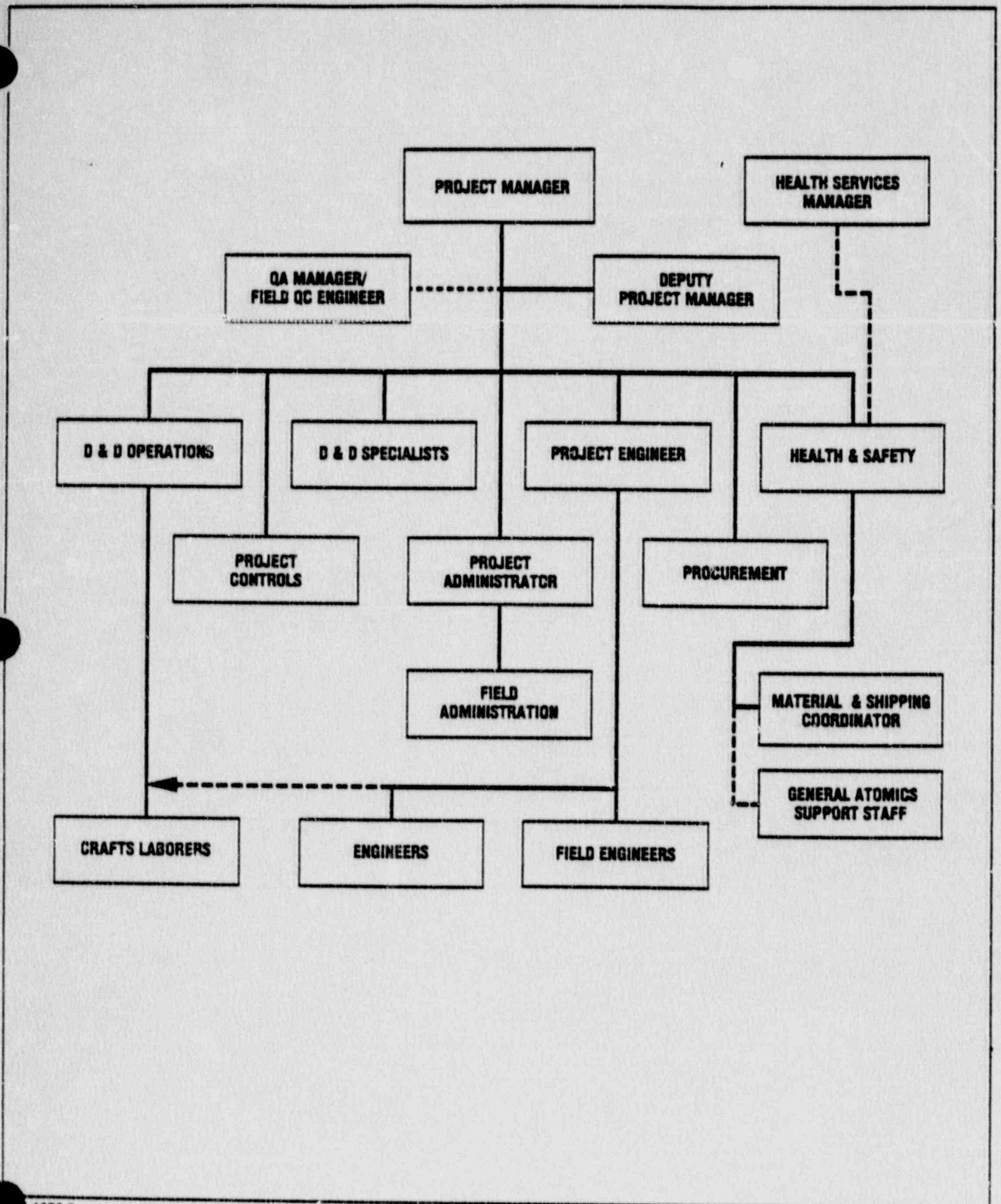

Figure 2.2-4
BNI ORGANIZATION FOR SVA DECOMMISSIONING PROJECT

Project Manager

The Project Manager is BNI's management representative for the project and has the responsibility for its successful completion. He has the authority to fulfill the responsibilities of the decommissioning contract and is responsible for the overall management and execution of the project. Responsibilities include:

- o Approving procedures and controls required for implementation of the Project Decommissioning Plan. These include lines of communication, working relationships, interfaces, controls, and reporting requirements both within the SVA Decommissioning Project and with VPA

- o Approving the Project scope and objectives defined in the Decommissioning Plan, Project plans, schedules and budgets and ensuring that manpower requirements are established

- o Assembling a project team qualified to complete the decommissioning effort

- o Approving subcontracts

- o Assuring that required records of project activities are maintained

Minimum experience and eduction requirements for the Project Manager include:

- o Previous project management experience

- At least two years experience in a project management/ engineering position on previous decommissioning or decontamination projects

- Prior involvement in the detailed preparation of a decommissioning program

- An accredited four-year college degree

Deputy Project Manager

The Deputy Project Manager reports to and assists the Project Manager. In addition, responsibilities include support of the following:

- Health and Safety, Environmental and Radiological Control,and Licensing

- Other activities assigned by the Project Manager

- Assistance to the Project Manager, as required

Deputy Project Manager qualifications include:

- Previous experience in project management

- Experience in a project management position on at least one previous decontamination or decommissioning project

- Past assignments involving planning, scheduling, subcontract, and cost control responsibilities

- Nuclear license compliance experience

Procurement Manager

The Procurement Manager is responsible for the purchasing of materials and administration of subcontracts. He/she receives project direction from the Project Manager. Responsibilities include:

- Preparation of procurement and subcontract procedures
- Approval of subcontracts and changes
- Issue and control of project purchase orders
- Conduct of procurement activities per appropriate procedures

Minimum experience and education requirements for the Project Procurement Manager are:

- Experience in purchasing and subcontract control relative to a decontamination or decommissioning project
- Previous responsibility for procurement activities in support of a remote jobsite
- Knowledge of radiological and decontamination work and the equipment/materials/services used for such work
- Specialized training in business administration, law, accounting, and/or other related subjects

Project Controls Manager

The Project Controls Manager is responsible for operating the Project Management Control System, including; cost estimating, cost tracking, scheduling, and management reporting. He/she receives project direction from the Project Manager.

Minimum experience and education requirements for the Project Controls Manager include:

- o Experience in project cost estimating, scheduling, tracking, and management reporting
- o Prior experience with project field activity tracking for remote jobsites
- o Knowledge of engineering, construction and business management techniques customarily acquired by prolonged course of specialized instruction. Knowledge of current industry and technical developments.

Project Engineer - (Manager of Decommissioning Engineering)

The Project Engineer is responsible for engineering activities on the project. The Project Engineer is responsible for complying with BNI Engineering Department Project Instructions (EDPIs) applicable to the scope of work and for executing the responsibilities of the "Project Engineer" as defined in the EDPIs. He/she receives project direction from the Project Manager. Responsibilities include the following:

- o Reviewing and approving reports and studies for technical quality

- Reviewing and approving project specifications and material requisitions

- Approving disposition of nonconformance reports

- Approving design changes (Field Change Requests/Field Change Notices, FCR/FCN) required during the decommissioning project

- Exercises operational supervision over the engineers of all disciplines assigned to the project or in support of the project

- Directing and coordinating engineering activities for the project

- Establishing or assistance in establishing the detailed scope project work plans and procedures

Minimum experience and education requirements for the Project Engineer include:

- Prior experience as a senior engineering staff member

- Two years experience as a senior engineer or project engineer on a decommissioning project

- Previous involvement in the detailed preparation of a decommissioning program

- A technical knowledge of related engineering systems, engineering calculations, and applications of engineering and construction methods and materials

- A knowledge of related construction practices and economics and current knowledge of new methods of design and construction

- A knowledge of industry and regulatory standards and design criteria pertinent to the project

- A recognized degree in architecture or an engineering or scientific discipline from an accredited college or university

  or

  A professional license in an appropriate discipline from a recognized licensing board

  or

  specialized courses in general engineering, architectural design or an engineering discipline and progressive work experience, including supervision, preferably in project and staff design engineering or architectural design, but may include other relevant types of engineering including startup, field engineering or plant operation in large industrial plants, utilities, process plants or engineering/construction firms

BNI Health and Safety Manager

The BNI Health and Safety Manager (HSM) has the responsibility for direct interface with the GA Licensing, Safety, and Nuclear Compliance organization in matters of environmental protection, fire protection, occupational Health and Safety, industrial hygiene, radiation protection, protection from hazardous chemicals exposure, and licensing activities. The Health and Safety Manager reports to the BNI Health Services Manager and has the organizational freedom and authority to require changes in work practices, identify problems and propose solutions, and, if necessary, stop work activities that could pose a danger to personnel or the environment. He/she receives project direction from Project Management. Responsibilities include:

- o Ensuring regulatory and operational compliance with Occupational Health and Safety Administration (OSHA), the Environmental Protection Agency, the project Health and Safety Plan, and BNI requirements

- o Preparing and/or approving Site Health and Safety Plans and procedures

- o Establishing and maintaining logs, records, reports and files in accordance with project requirements related to the Health and Safety aspects of projects

- o Review and concurrence with Unit Work Instructions (UWI), Work Authorizations (WA), and Radiation Work Permits (RWP)

- o Reviewing and concurring with reports and studies which present project Health and Safety information and/or conclusions

- o Health Physics requirements, including interfacing with General Atomics Health Physics Staff

- o Compliance with environmental control measures

- o Licensing activities and interfacing with General Atomics licensing staff

Minimum experience and education requirements for the Health and Safety Manager include:

- Health Physics experience in operating nuclear facilities, decontamination projects or decommissioning proje
- Previous assignment as a project/site health and safety officer or as a facility radiation protection officer
- Prior work in a management or supervisory position on a decontamination or decommissioning project
- Previous involvement in the detailed preparation of a decontamination or decommissioning program
- Nuclear license compliance experience
- A four year degree plus two years of directly applicable experience or a two year degree plus four years of directly applicable experience

Project Quality Assurance Manager

The Project Quality Assurance Manager is responsible for developing and maintaining the Project Quality Assurance Program Plan (PQAPP) and for coordinating its implementation with the Project Manager. He/she reports directly to the Oak Ridge Quality Assurance Manager and coordinates project activities with the Project Manager. Responsibilities include the following:

- Reviewing and concurring with project plans and procedures for quality concerns
- Performing audits and surveillance of project activities for compliance with the Project Quality Assurance Program Plan (PQAPP), and implementing procedures

- o Providing quality assurance indoctrination and training to project personnel and assisting in procedure training

- o Reporting regularly to management on project quality assurance activities and items of concern

xxx o Identifying the need for corrective action and initiating, recommending, coordinating, and providing solutions for project quality problems. Initiating "stop work orders" when deemed necessary to correct quality problems

Minimum experience and education requirements for the Project Quality Assurance Manager include:

- o Two years experience in QC/QA activities for licensed nuclear facilities

- o Experience as lead QA representative for a small project, preferably related to decontamination or decommissioning of a nuclear facility

- o A recognized degree in an engineering or scientific discipline from an accredited college or university

  or

  specialized courses in engineering, quality, personnel administration or a specialized discipline and practical work experience in fields of engineering, procurement, industrial construction, business administration and/or work experience as a QA Engineer or equivalent

<u>Decommissioning Engineers</u>

Decommissioning Engineers are responsible for providing engineering and technical support for the decommissioning project. They report to the Project Engineer and coordinate with operations at the job-site. Their responsibilities include:

- o Preparing procedures, plans, and unit work instructions
- o Providing for Hold and Witness points in unit work instructions
- o Initiating FCR/FCNs as required for necessary design changes
- o Maintaining field design document control
- o Dispositioning of reported discrepancies
- o Ensuring compliance with all plans, instructions, and procedures
- o Preparing calculations, specifications, material requisitions, studies, drawings, and reports
- o Providing technical review/approval of plans, specifications, and procedures. Other responsibilities as assigned by the Project Engineer

Minimum qualifications for Decommissioning Engineer include:

- o Engineering experience either at an operating nuclear facility or on a decontamination/decommissioning project
- o Familiarity with the detailed preparation of procedures, plans, and specifications for similar projects

Operations Superintendent

The Operations Superintendent is responsible for the direction of the work force during decommissioning activities. He/she reports to and receives project direction from the Project Manager. Responsibilities include:

- o Directing and controlling the crafts and laborers
- o Scheduling and coordinating site activities
- o Ensuring that craft personnel are trained for their assigned activities
- o Assuring that the Health and Safety and quality procedures are followed

Minimum experience and education requirements for the Operations Superintendent include:

- o Two years prior experience as an operations supervisor/superintendent at a nuclear facility involving decommissioning or decontamination
- o Experience as a group supervisor on at least one decontamination or decommissioning project
- o Involvement in the detailed preparation of a decommissioning program
- o Experience in worker training and safety instruction
- o Specialized courses in construction technology and management or work experience leading to a thorough understanding of the responsibilities described above

Field Quality Control Engineer

The Field Quality Control Engineer reports directly to the Project Quality Assurance Manager and receives Project direction from the Project Manager. Responsibilities include:

- o Performing audits and surveillances of site activities
- o Performing inspections of selected site activities
- o Assistance in training site personnel
- o Concurring with disposition of nonconformance reports
- o Approving UWI's and changes thereto

The qualifications for the Field Quality Control Engineer include experience in a QC/QA organization involved with a nuclear facility or a decontamination/decommissioning project.

2.2.3.2 Decommissioning Health and Safety Responsibility

Since GA is fully responsible for radiological health and safety on the site, BNI and BNI subcontractors will comply with GA radiological health and safety policies and procedures. Further assurance of adequate health and safety coverage by BNI throughout the project will be provided by the following:

- o A full-time BNI Health and Safety Manager will be located on-site for the duration of required field operations. This employee will have a health physics background and significant experience in other safety disciplines. He/she will routinely interact with the GA counterparts to assure compliance with the Health Physics and Safety Programs, as well as their continued upgrades.

- Completion of a comprehensive training program covering all aspects of health and safety will be required for GA employees and for employees of BNI and BNI subcontractors on-site (see Section 2.3).

- Utilization of a complete set of task/activity-specific work procedures that places significant emphasis on health and safety.

- Other health and safety expertise available in the home office.

The program identified above is designed to protect workers, the public, and the environment by applying comprehensive training to all site personnel. The program includes both radiological and industrial safety training. This training will include both operational classroom and on-the-job training, with periodic retraining available as needed.

2.2.3.3 Decommissioning Operations Staff Responsibility

The responsibilities, minimum qualifications, and reporting functions of the decommissioning operations staff, as well as those of home office and field operations technical and administrative support personnel, are defined in Section 2.2.3.1.

2.2.4 Coordination of Decommissioning

Decommissioning activities will be coordinated and controlled by the project staff. Coordination between BNI and GA will ensure compliance with the facility license and minimize interference with other licensee activities.

GA will review and approve project procedures prior to their use in the facility. GA will be advised of project daily work, pre-shift briefing times and locations and may attend the meetings. BNI will provide GA with copies of detailed schedules and Unit Work Instructions to ensure that GA is aware of activities scheduled within the facility.

2.2.5 Qualifications

2.2.5.1 BNI Qualifications

BNI is the prime contractor for the decommissioning effort. The BNI team has a broad base of experience in nuclear facility decommissioning planning, implementation and decontamination experience. BNI's experience ranges from remedial action studies and development of standards to facility and systems design and construction. It includes the characterization of contaminated facilities, decommissioning planning, decontamination, decommissioning, and management of radioactive wastes.

The following brief descriptions of previous relevant projects provide examples of BNI's experience in the performance of nuclear clean-up work and related activities.

1) HUMBOLDT BAY POWER PLANT - DECOMMISSIONING PLAN AND ENVIRONMENTAL REPORT

Client: Pacific Gas and Electric Company

BNI prepared the decommissioning plan and the environmental report for Humboldt Bay Power Plant Unit 3. The decommissioning plan addressed both the 30-year SAFSTOR option and the delayed dismantlement of the unit. The environmental report was prepared to satisfy NEPA documentation requirements for the license application amendment to change the unit's status from operating nuclear power plant to possession-only of licensed material. In addition to evaluating alternatives for plant decommissioning, the following tasks were completed:

- o Site characterization
- o Pathways analysis of nuclide migration from the plant
- o Estimation of dose received by workers, the public, and environmental components
- o Assessment of radwaste treatment alternatives and implications to health and environmental quality

2) EDWIN I. HATCH NUCLEAR POWER PLANT - DECOMMISSIONING PLAN AND COST ESTIMATE

Client: Georgia Power Company/Southern Company Services, Inc.

BNI prepared the decommissioning plan and cost estimate with sufficient detail to permit use of the information in establishing a basis for rate case hearings as they pertain to decommissioning the Hatch Plant, Units 1 and 2. The study addressed the prompt removal/dismantlement (DECON) method for decommissioning.

3) JOSEPH M. FARLEY NUCLEAR POWER PLANT - DECOMMISSIONING PLAN AND COST ESTIMATE

Client: Alabama Power Company

BNI prepared a decommissioning plan and cost estimate for the prompt removal/dismantlement (DECON) method of decommissioning. Development of this plan took into account recent information regarding the decommissioning of nuclear power plants and new regulations promulgated by the Nuclear Regulatory Commission.

4) KOREAN ADVANCED ENERGY RESEARCH INSTITUTE - TRIGA REACTOR DECOMMISSIONING PLAN AND COST ESTIMATE

Client: Korean Advanced Energy Research Institute

BNI prepared a decommissioning plan and cost estimate to decommission a TRIGA reactor. The two options considered were prompt removal (DECON) and decontamination followed by refurbishing as a national museum.

5) THREE MILE ISLAND UNIT 2 RECOVERY PROJECT

Client: GPU Nuclear

As program manager, BNI was responsible for investigations, design, and cleanup activities to stabilize TMI-2. BNI provided a staff for evaluating health risk assessments to workers, community residents, and the environment. In addition, BNI provided technical support for scoping the remediation program. The program included preparation and implementation of work plans for decontamination and fuel removal, including disposal of radioactive waste.

6) MAYAGUEZ TRIGA REACTOR DECOMMISSIONING

Client: U. S. Department of Energy

BNI managed the operations activities and developed plans and procedures to decommission the TRIGA Reactor at the University of Puerto Rico, Mayaguez site.

7) SURPLUS FACILITIES MANAGEMENT PROGRAM (SFMP)

Client: U. S. Department of Energy

BNI is, or was, respectively, the program manager for the DOE in decontamination and restoration of two surplus facilities: the Niagara Falls Storage Site (NFSS) and the Weldon Spring Site (WSS). These are storage facilities for pitchblende residues resulting from the processing of uranium ore. Primary responsibilities include:

- Site and facility characterization
- Design, installation, and operation of radwaste treatment facilities
- Alternatives assessment for waste disposition
- Decontamination and remedial action implementation
- Personnel monitoring and dose assessment
- Environmental monitoring
- Regulatory compliance
- Industrial hygiene planning and compliance

8) FORMERLY UTILIZED SITES REMEDIAL ACTION . ROGRAM (FUSRAP)

Client: U. S. Department of Energy

BNI is the program manager for the Department of Energy in decontamination and restoration of 29 sites contaminated by hazardous substances, particularly low-level radioactive materials

2.2.5.2 GA Qualifications

Since starting operations in 1962, GA's nuclear fuel operations have centered on:

- o design of coated particle fuel
- o development of fuel processes
- o design-construction-and-qualification of manufacturing equipment
- o fabrication of fuel for several reactors and for other programs

Over 50,000kg of coated particle fuel have been produced using 93% enriched uranium. GA is one of four companies in the United States to be licensed by the NRC for highly enriched uranium (HEU).

HEU fuel production commenced in the 1960's with several thousand kilograms of NERVA and ROVER rocket fuel, two cores for the Peach Bottom HTGR plant, and fuel particles for classified military applications. This was followed by the initial core for the Fort St. Vrain (FSV) power plant, four FSV reload segments, and a core for the Compact Nuclear Power Supply (North Warning System) at

Los Alamos. Recently, test particles were made for the Savannah River Plant (SRP) extrusion experiments.

GA's fuel manufacturing plant is a stand-alone facility providing all the services, quality control, uranium storage vaults, uranium recovery, waste processing, uranium accountability systems, and security systems required to produce HEU HTGR fuel. It is fully approved by the NRC, through a "broad scope license" which allows movement and introduction of new HTGR processes without requiring substantial NRC license changes.

The history of HTGR fuel development and fuel manufacturing has been an evolution of new processes introduced within an environment of elaborate Quality Assurance and extensive NRC regulation. GA has the experience and management system that allowed these changes to be implemented expeditiously while continuing to supply high quality fuel on schedule within predicted costs.

In recent years, GA has decommissioned several of its own facilities formerly used for nuclear operations, including the following:

- A former Nuclear Waste Processing Facility (80,000 ft^3 of low-level waste)

- The Callan Pond, a shutdown sanitary sewage plant that had become contaminated (15,000 ft^3 of uranium-contaminated low-level waste)

- The E-Building, a 20,000 ft^2 facility which used thorium, and natural and depleted uranium to demonstrate HTGR fuel recycling systems

- Numerous small laboratories

These facilities were decontaminated by GA and were released for unrestricted use by the NRC and the State of California.

2.3 TRAINING

All personnel will receive instruction through orientation/training concerning the project radiological protection and health and safety programs. Each decommissioning project worker will receive orientation/training which will consist of instruction in jobsite radiological protection, health and safety, hazard recognition and control, fire extinguisher training, and emergency and safety awareness. This training program shall also include applicable decontamination procedures for work in areas of real or potential radiological significance, radiation effects, radiation safety, radiation dosimetry, contamination control, and the role and obligation of workers with respect to regulatory requirements. Training shall consist of classroom training and appropriate practical drills in which individuals demonstrate the ability to perform various aspects of their assigned functions. During the practical drills, on-the-spot correction of improper actions will be made and a demonstration of the proper performance offered by the instructor. Specialized training applicable to specific conditions will be given as the progress of decommissioning activities dictates.

Supervisory safety training will be an important part of the safety training program. Each foreman will receive a foreman's safety orientation detailing the safety responsibilities of his position and how to develop good safety practices among the workers. Records will be kept of personnel attendances, level of accomplishment, follow-up sessions, etc., as necessary to ensure that the appropriate awareness and competency have been demonstrated.

For the purpose of defining training requirements for the decommissioning project, personnel will be assigned to three categories: visitors, intermittent personnel, and decommissioning workers.

- **Visitors**

  Individuals who do not frequent controlled areas (i.e., visitors such as tour groups or visiting managers) shall be escorted by GA or BNI employees while in the controlled areas and will not require training. These individuals may not enter areas where they are likely to receive > 2 mrem in any hour or > 100 mrem in any year, whichever is the more restrictive.

- **Intermittent Personnel**

  Individuals who frequent the controlled area (more than 40 hours per quarter) but who do not work in the area will be required to take a four-hour Contractor Radiation Safety Course and pass a written exam with a grade of at least 70 percent.

  Other individuals included in this category are workers with special skills who are brought in for a specific task of less than 40-hour duration. These individuals may work in the radioactively controlled areas under direct supervision of a qualified decommissioning worker. They will be required to take and pass the four-hour Radiation Safety Course.

- **Decommissioning Workers**

  Individuals who will routinely handle radioactive material or contaminated equipment associated with the decommissioning project must have satisfactorily completed a two-day Radiological Safety Course prior to working in the controlled area.

2.3.1 Radiological Safety Course

The Radiological Safety Course required for decommissioning workers includes the following topics: ALARA practices, an introduction to 10CFR19 and 10CFR20, radiological instrumentation and controls, decontamination procedures, fire protection, and emergency procedures. General subjects such as the nature and sources of radiation, methods of controlling contamination, interactions of radiation and matter, biological effects of radiation, use of monitoring equipment, principles of nuclear criticality safety, and risks from occupational radiation exposure also are covered. The course is comprised of lectures and demonstrations, augmented with selected audiovisual aids. It also includes industrial and general safety training. The content of the Radiological Safety Course may be revised as the need becomes apparent during the project, provided that such changes do not decrease the effectiveness of the training program.

The comprehension of each worker and the overall effectiveness of the training course will be evaluated by a written examination given at the end of the course. Workers will be required to pass the examination with a grade of at least 70 percent. The GA Manager of Health Physics or a qualified designee will be responsible for the radiological safety training course content and for conducting the training.

2.3.2 Retraining

Additional radiological and general safety training that is specifically directed toward planned work activities will be conducted by representatives of Health Physics and Safety prior to the start of the activity. Unit Work Instructions will be issued for each task and will include reference to the applicable procedure(s) and specific concerns/precautions related to health and safety.

In addition to the above-mentioned training, retraining on selected subjects will be conducted on an as-needed basis, but at least annually. Individual comprehension and the program's overall effectiveness will be evaluated by a written examination, which requires a score of at least 70 percent.

2.3.3 Health Physics Technician Training

Health physics technicians shall successfully complete the Radiological Safety Course described above. In addition, they must receive three (3) weeks on-the-job training. Additional training, including characteristics of criticality, principles of nuclear safety, and details of postulated criticality accidents, will be given to enable technicians to recognize potential problem areas.

2.3.4 Equipment Operator Training

A number of specialized powered tools and equipment will be utilized through the course of the decommissioning project.
Operator qualification for this equipment will be required to ensure personnel safety and operational efficiency requirements are met. Qualification will be attained by completion of the following:

- o Study of vendor-supplied literature and of the applicable Operating Procedure(s)
- o Functional demonstration of the use of the equipment
- o Hands-on training under qualified supervision

Whether operator retraining/requalification is necessary at any time will be determined on a continuing basis. The BNI Operations Superintendent will have the responsibility to require operator requalification, depending on operating complexity, operator continuity of service, or other reasons as deemed appropriate.

2.3.5 Nonradiological/Industrial Safety Training

Work shall be conducted in accordance with the California Division of Industrial Safety Construction Safety Orders Title 8 Subchapter 4 of the California Administrative Code.

Each individual shall be given instruction regarding the hazards and safety precautions applicable to the type of work in question and shall be directed to read the Code of Safe Practices. Only qualified persons shall operate equipment and machinery.

Individuals shall be instructed concerning workplace hazards, such as flammable liquids and gases, chemicals, and confined spaces and shall be instructed in the procedures for protecting themselves from injury.

Examples of special training requirements:

- o Crane/hoist operation (including forklifts)
- o Powered platforms and scaffolds
- o Use of personal protective equipment in elevated work areas
- o Disposal of chemicals and hazardous materials
- o Use of cutting and welding equipment

Workers will be instructed in the use of ladders, in protecting floor and wall openings, in maintaining emergency egress capability, and in the proper use of scaffolding.

Individuals shall be given instructions in accident prevention during the daily readiness reviews.

2.3.6 Training Records

Records will include but are not limited to:

- o The name of the supervisor and attending employees
- o The subject of the meeting and a brief description
- o The date, time, and duration of the training or meeting
- o Written examinations

Such records will be kept in accordance with 10CFR Part 19 or 20.

2.4 SUBCONTRACTOR ASSISTANCE

During the preliminary planning stages of the decommissioning project, the need for utilizing the services of specialty contractors was evaluated, to identify areas of expertise where subcontracts may be desirable. Use of these contractors will increase the overall cost-effectiveness of the project by improving the efficiency of specialty operations and reducing the need for specialized staff training.

2.4.1 Scope of Work

For each subcontracted task, a comprehensive bid specification will be prepared for subcontractor quotes. The selection of the subcontractors will be made well in advance of the scheduled work, to incorporate their schedules into the overall project schedule.

The services to be provided by qualified subcontractors may include:

- Hauling of radioactive and non-radioactive materials
- Laboratory and testing services
- Medical and emergency services
- Concrete cutting services
- Dismantlement services

It is not anticipated that the subcontractors will possess a nuclear license.

2.4.2 Qualifications

Potential subcontractors for each task identified will supply their qualifications as part of their bids. The qualifications will emphasize the following:

- o Ability to accommodate the overall project schedule
- o Experience with similar work
- o Adequacy of qualified workers

2.4.3 Administrative Controls for Health and Safety

Administrative controls for Health and Safety will be based on experience and prudent practices for protecting contractor and subcontractor personnel from known and/or suspected potential hazards. Contractor and subcontractor personnel will adhere to health and safety rules/standards set forth by GA in addition to applicable instructions, procedures, and directions for the performance of the work and will undergo training and qualification per Section 2.3.

When subcontracts are awarded, the work will be performed in compliance with GA's license, utilizing the aforementioned procedure approval format (as described in Section 2.1.11.2).

2.4.4 Quality Program

BNI subcontractors will perform their activities in accordance with BNI quality assurance requirements. BNI will perform surveillances and audits of the subcontractor's activities to verify compliance with all quality and technical requirements (see Section 7.0).

2.5 RESUMES

The following resumes highlight the experience of key personnel who are assigned to the SVA Decommissioning Project.

PERSONNEL	FUNCTION
A. General Atomics	
R. C. Noren	Project Manager
K. E. Asmussen	Manager, Licensing, Safety & Nuclear Compliance
J. J. Saurwein	Manager, Quality Control
R. N. Rademacher	Manager, Security
M. G. Dunlap	Manager, Project Quality Assurance
R. J. Bott	Manager, Industrial Safety
R. A. Rucker	Manager, Nuclear Safety
C. L. Wisham	Manager, Nuclear Material Accountability
J. M. Bolin	Manager, Statistics & Measurement Control
L. Quintana	Manager, Health Physics
R. Vanek	Manager, Nuclear Waste Processing Facility
S. P. Massey	Project Engineer, Quality Assurance
B. Bechtel National, Inc.	
J. A. Chapin	Project Manager
R. J. Cohose	Procurement Manager
G. D. Cremeans	Project Engineer
J. V. Hargreaves	Project Controls Manager
P. K. Jackson	Health and Safety Manager
J. F. Mattson	Operations Superintendent
B. Peacock	Quality Assurance Manager

ROBERT C. NOREN

PROGRAM DIRECTOR

PROFESSIONAL SPECIALTY

Management of high enriched fuel manufacturing and uranium recovery production plant. Design, procurement, installation, and shakedown of fuel manufacturing equipment.

EDUCATION

B.S., Mechanical Engineering, Purdue University, 1959

EXPERIENCE

Twenty-eight years experience in an NRC-licensed high-enriched uranium fuel manufacturing facility with overall management responsibility for production of HTGR fuel, TRIGA fuel, and other experimental fuel; development of HTGR commercial fuel fabrication processes and equipment; development and implementation of fuel manufacturing control systems; management of large groups of people (200 to 300).

Managed Process Development Department responsible for development of HTGR commercial fuel processes and construction of full-size production equipment to qualify these processes.

Developed new packaging and transportation system for disposal of low-level radioactive waste at DOE sites.

Over a period of three years, Manager for decommissioning of two GA sites involving disposal of 100,000 ft^3 of radioactive waste.

Production Manager for the world's first prismatic-block HTGR core [Fort St. Vrain, 330 MW(e)]. Directed production, material control, scrap recovery, waste handling, and equipment engineering. This first-of-a-kind effort presented many technical and managerial problems requiring innovative resolution in concert with quality control, quality assurance, core physics, core material design, and the customer.

Managed Equipment Engineering Branch responsible for design, procurement, installation, shakedown, and maintenance of HTGR fuel manufaccuring equipment and facilities.

Project Engineer for manufacture of reactor components (helium circulators, steam generators, etc.). Responsible for liaison between Engineering and Manufacturing.

Commissioned Officer in the U.S. Navy; Shipboard Damage Control Officer in the Engineering Department

PROFESSIONAL ASSOCIATIONS

Registered Professional Engineer (Nuclear), California
Registered Professional Engineer (Manufacturing), California

PATENTS

Holder of three patents; two for nuclear fuel production equipment, one for nuclear fuel design.

OTHER DATA

DOE Q and L, NRC U, and DoD SECRET clearances

DR. KEITH E. ASMUSSEN
MANAGER, LICENSING, SAFETY AND NUCLEAR COMPLIANCE

EDUCATION

Ph.D., Nuclear Engineering, Iowa State University of Science and Technology, 1969

M.S., Nuclear Engineering, Iowa State University, 1966

B.S., Engineering Operations (Industrial Engineering), Iowa State University, 1965

EXPERIENCE

Dr. Asmussen joined General Atomics' (GA) Nuclear Analysis and Reactor Physics Department as a Se[illegible] Reactor Physicist in 1969. His initial responsibilities involved n[illegible]ear fuel management analyses and reactor physics calculations. In 1972 he was temporarily assigned to the Fuel Performance Branch where he was responsible for developing the reactor core thermal safety limit and other fuel related technical specifications for a large High Temperature Gas-cooled Reactor (HTGR).

In 1973, and again in 1976, he served as a site physicist at the Fort St. Vrain (FSV) HTGR. His responsibilities involved planning, coordinating and participating in the initial fuel loading, subcritical testing and monitoring, zero power physics testing and rise-to-power testing. Beginning in 1974, he spent 18 months working in the HTGR physics group of Hochtemperatur Reaktor Bau (HRB) located in Mannheim, West Germany. At HRB he acted as GA liaison and consultant regarding HTGR core and fuel design. In 1976, he returned to GA's San Diego offices and became a section leader engaged in Lead Plant HTGR core physics design and nuclear analysis. Late in 1977, he was given the special assignment of coordinator of all testing (in-pile and out-of-pile) related to resolving the FSV core temperature fluctuation problem.

In 1979, he became Manager, Fort St. Vrain Fuel Engineering where he was given the additional responsibility for directing all the technical analyses required to design, manufacture and license FSV reload segment fuel. Other responsibilities included fuel accountability, core reactivity monitoring and monitoring the performance of the core and fuel. He played a key role in developing revised Technical Specifications for the FSV reactor and obtaining NRC release for unrestricted full power operation. He worked intimately with Public Service Company of Colorado licensing personnel on a variety of issues involving personnel interaction with NRC staff. In 1983, he became Coordinator, Fort St. Vrain Core Activities. In this capacity his technical responsibilities remained unchanged but he assumed responsibility as project manager of these and related tasks.

From 1979 to 1985, Dr. Asmussen served on GA's Fuel Material Review Board which reviews and dispositions nonconformance reports, waivers, etc., related to the FSV Fuel Specifications.

In 1985, he became Manager of Licensing and Nuclear Material Control. His areas of responsibility were broadened in 1986 when he became Manager, Licensing, Safety and Nuclear Compliance. In this capacity, he is responsible for administering GA's licenses, liaison with regulatory agencies and reviewing and approving all work involving radioactive material for compliance with applicable regulations and license conditions. In addition, he is responsible for the overall planning and coordination of GA's special nuclear material control, nuclear safety, health physics, industrial safety and emergency services programs.

PROFESSIONAL ASSOCIATIONS

Registered Professional Engineer, Nuclear Engineering, California

JOHN J. SAURWEIN
MANAGER, FUEL QUALITY CONTROL

PROFESSIONAL SPECIALTY

Managing HTGR fuel quality control activities. Specialist in developing quality control methods and sampling/inspection plans for HTGR fuel and MHTGR-NPR targets.

EDUCATION

B.S., Mathematics and Chemistry,

EXPERIENCE

Ten years experience in nuclear fuel performance analysis. Performed computer analyses to predict the irradiation conditions and fuel performance in LHTGR cores, developed computer models to predict the temperature history of test elements irradiated in the Peach Bottom HTGR, and served as lead engineer in developing methods for remote examination of irradiated fuel elements and in performing nondestructive and destructive examinations of fuel elements irradiated in the Fort St. Vrain HTGR. Has developed a thorough understanding of fuel performance requirements and the base for fuel product specifications.

Three years experience as Supervisor of the Fuel Quality Control Laboratory. Was responsible for the day-to-day operation of the laboratory, including test method development and qualification, equipment maintenance, personnel deployment, inspection activities (schedule and accuracy), identification of manufacturing problems from quality control results, and timely reporting of quality control results to Manufacturing.

Manager of Fuel Quality Control for past four years. Responsible for all quality control activities, including procedure development, inspection, quality engineering, and customer interface in support of GA's CNPS fuel manufacturing campaign for Los Alamos National Laboratory (LANL). Developed the quality control procedures and quality control sampling plans, and directed product inspection activities in support of GA's ATR-1 target manufacturing campaign for EG&G, Idaho.

PROFESSIONAL ASSOCIATIONS

American Society of Quality Control (ASQC) Certified Quality Engineer

PUBLICATIONS

McCord, F., and J. J. Saurwein, "Remote Examination of Fort St. Vrain HTGR Fuel and Reflector Elements," *Trans. Am. Nucl. Soc.* (USA), Vol. 46, pp. 776-8 (1984).

Wallroth, C. F., N. I. Marsh, C. M. Miller, J. J. Saurwein, and T. L. Smith, "Development of Surveillance Robot for Dimensional and Visual Inspection of Fuel and Reflector Elements of the Fort St. Vrain HTGR," *Trans. Am. Nucl. Soc.* (USA), Vol. 33, pp. 890-1 (1979).

Wallroth, C. F., and J. J. Saurwein, "HTGR Nuclear and Thermal Design Verification in Peach Bottom," *Trans. Am. Nucl. Soc.* (USA), Vol. 28, pp. 686-8 (1978).

RODNEY N. RADEMACHER
VICE PRESIDENT, HUMAN RESOURCES
MANAGER, SECURITY

B.A. Industrial Psychology, San Diego State University, 1962.
Graduate Studies, San Diego State University.
Graduate Studies, University of Colorado, Colorado Springs.

Mr. Rademacher has been employed by General Atomics since early 1974 in various management capacities prior to assignment to his present position in March 1988. Presently, he has responsiblity for all human resource functions at the corporate level. In particular, Mr. Rademacher is responsible for designing, developing and directing company Human Resource programs, policies and procedures so as to effectively support the company's overall business objective. He functions as chief advisor on the personnel implications of company problems, business procedures and other management actions. The Security Force Department was added to his responsibilities in August 1985 and the Licensing, Safety and Nuclear Compliance Department was added in March of 1986. As such, he has a very broad understanding of the company's people, programs and business needs and requirements. He is intimately familiar with his organization's operations, requirements and applicable NRC and other government requirements. Because of his strong Human Resources and Safety orientation, he has developed an influential, positive working relationship with most GA managers and employees.

Before coming to GA, Mr. Rademacher was Director of Employee Relations for Getz Brothers & Company, Inc., 1973-74; Manager of Corporate Employment and EEO for Colorado Interstate Corporation, 1968-1973; Personnel Generalist for SDG&E, 1957-1968; and the U.S. Marine Corps Reserves, 1955-1956.

MICHAEL G. DUNLAP
MANAGER, QUALITY SYSTEMS & DEFENSE QA

PROFESSIONAL SPECIALTY

QA System Development and Assessment

EDUCATION

Master of Arts in Industrial Arts, San Diego State University, 1979.
Bachelor of Arts, San Diego State University, 1974 - Major Industrial Arts.
Associate of Arts, San Diego Mesa College, 1973, Major Biology.

PROFESSIONAL EXPERIENCE

Currently manage GA's QA audit program including internal division/program/project audits, and audits of vendors which are on GA's Approved Suppliers List (ASL). For independent third-party reviews, develop review objectives, criteria, schedules and budgets, and assign appropriate personnel. Primarily responsible for content and format of review reporting to customers. Responsible for records and procedures control, including development, implementation, and revision of the GA QA Manual, Quality Division Instructions, and the GA QA Topical Report, and the corrective action program.

Recent experience includes participation in the independent third-party design, construction, and Quality Assurance reviews, of six Nuclear Power Plants. Responsibilities included assuring that commitments in the PSAR and FSAR were implemented into released design, construction, and QA procedures. In addition, verified that the actual hardware in the plants was procured, installed, inspected, and tested per the applicable drawings and procedures. Technical and administrative responsibilities included the direction of professional personnel, and the accomplishment of project milestones, budget requirements, and report preparation. Also participated in three independent third-party assessments of operating nuclear power plants which addressed, among other areas, Project Management, Operations and Maintenance Support, Nuclear Licensing, Nuclear Safety, and Site Quality Assurance/Quality Control. These assessments evaluated the adequacy of procedural compliance and the overall effectiveness of the plant activities through a review of records, personnel interviews, and observations.

PROFESSIONAL ASSOCIATIONS

Member American Society for Quality Control
Certified Quality Engineer, American Society for Quality Control
Certified Lead Auditor to ANSI N45.2.23
Member Epsilon Pi Tau, Beta Alpha Chapter, San Diego State University
Community College Teaching Credential (California 1979)

RONALD J. BOTT
INDUSTRIAL SAFETY ENGINEER

EDUCATION

B.S., Mechanical Engineering, San Diego State University, 1970
Numerous special courses in industrial safety and materials engineering.

LICENSES

Registered Safety Engineer, California #1593

QUALIFICATIONS

Twelve years experience in developing, implementing, and reviewing company safety, fire, and health programs. Broad working knowledge of federal and state occupational safety and health codes, hazard communication programs, and environment regulations.

Experienced in accident/incident investigations involving worker's compensation insurance and loss prevention programs.

Strong technical background in mechanical engineering and manufacturing processes, including experience with plastic processing and fabrication.

EXPERIENCE

Five years' experience in environmental compliance with Federal, State, and local regulations, including Instructor of Record for Sixth Annual Environmental Compliance Conference in San Diego for the University of California Extension.

GENERAL ATOMICS - 1973 to present

Manager of Industrial Safety and Health (1988-1990) - Administer Industrial Safety and Health Program including incorporation of plant safety committee and the supporting divisional safety committees. Maintain program for review of hazardous work, use of extremely hazardous materials, and the proper storage and disposal of hazardous material and hazardous waste. Responsible for interface with Federal, State, and local regulatory agencies, as well as coordination with loss control personnel and insurance audits. Provide assistance and direction to company fitness center and the wellness program.

Industrial Safety Engineer (1984-1988) - Administer industrial safety at General Atomics, including accident statistics, hazardous work requests, safety committee development, worker's compensation reporting, year-end reporting, accident investigation of serious incidents, liaison with nuclear waste management, and coordination of industrial safety programs with Industrial Hygiene, Health Physics, Emergency Services, and Medical. Responsibilities include non-nuclear waste transportation projects and hazardous material management.

Senior Engineer (1982-1984) - As Task Engineer, responsible for development, fabrication, and installation of large electrical coils (18 feet in diameter) and patch board systems. Basic design of electrical coils including drawings and specifications (stress, electrical, cooling), design procurement and installation of a complete coil winding facility (water heating systems, vacuum systems, winding tables, sandblasting, insulation wrapping, special power tools, copper handling, solvent cleaning). Complete fabrication of coils and vacuum potting in epoxy matrix. Electrical testing (high voltage, high current).

Safety Engineer (1979-1982) - Reviewed hazardous work requests and developed appropriate safety measures, including hazardous chemical waste, specific processes with potential safety considerations (i.e., cryogenic, explosive atmospheric, flammable liquids, industrial hygiene, hoisting/lifting.) Also conducted system safety analysis, and safety inspections. Oversaw company safety procedures. Reviewed state codes and federal regulations. Responded to emergency response fires, industrial accidents, vehicle emergencies, and occupational illnesses and accidents.

Safety Chief (1975-1979) - Developed, implemented, and maintained comprehensive accident prevention program involving line management. Implemented controls to eliminate or minimize potential hazards (laser, high voltage microwave radiation, industrial work practices). Responsible for training and indoctrination of 70-150 employees.

Mechanical Engineer (1973-1975) - Supervised manufacturing of equipment for fusion research and development. Work included construction of large electrical coils, stainless steel cooling systems, ceramic insulators, high vacuum systems, and cryogenic supply systems Assisted physics personnel in development of diagnostic systems and vendor liaison. Supervisor of machine shop, welding laboratory and sheet metal fabrication area.

PROFESSIONAL ASSOCIATIONS

Member, American Society of Safety Engineers, Research and Development Section

Member, Pacific Coast Electrical Association, Safety and Health Committee

Director & Chairman, Water Committee, Industrial Environmental Association, San Diego, CA

Director and Chairman, Emergency Response Committee, Riverview Water District, Lakeside, CA

ROBERT A. RUCKER
NUCLEAR ENGINEER

EDUCATION

B. S., Nuclear Engineering, University of Michigan, 1970
M. S., Nuclear Engineering, University of Michigan, 1971

PROFESSIONAL SPECIALTY

- o 18 years nuclear experience with General Atomics specializing in core nuclear design, safety analysis, and criticality analysis, including 11 years in HTGR design and fabrication.

- o Experienced with diffusion, transport, nodal, point kernel, Monte Carlo, and cross section generation techniques used for nuclear core, shielding, cask, and fuel fabrication facility design and analysis.

- o Manager of Statistics & Measurement Control from June 1984 to April 1990. Deputy Nuclear Safety engineer for four years, and Manager of Nuclear Safety from January 1990 to present. Acting branch manager for MHTGR core physics from January 1990 to July 1990.

EXPERIENCE

Manager of Nuclear Safety, January 1990 to Present, and as deputy nuclear safety engineer since 1986. Have been involved in nuclear safety inspections, and given nuclear safety lectures at Radiological Safety training courses. Was also responsible for criticality calculations for use of boronation in the fuel fabrication process at GA, and for the facility to store spent fuel at the Fort St. Vrain Nuclear Generating Station.

As Manager of Statistics and Measurement Control, was responsible for producing and maintaining procedures, measurement equipment, and statistics data to assure proper control of special nuclear material in compliance with NRC regulations; calculating measurement error components for bimonthly inventories, and annual contractor audits related to measurement control. Performed analysis of scale and balance errors, computer code write-ups, and inventory control system review in response to NRC concerns.

As acting branch manager for MHTGR Core Physics, was involved in nuclear analysis and cost/performance optimization of the modular HTGR core, including fuel management, control rod patterns, and detailed power distributions. Also supplied data for safety analysis for Fort St. Vrain reload fuel, and served as on-site core physicist at Fort St. Vrain. Performed core design and analysis on several TRIGA reactors. Was also responsible for the nuclear analysis and design of the core and blankets of a demonstration size fast breeder reactor. Designed and analyzed fuel irradiation and critical assembly experiments for verification of computer codes and modeling methods.

Performed shielding analysis for design of the OCRWM spent fuel shipping cask using Monte Carlo (MCNP), transport, and point-kernel techniques for the calculation of neutron and gamma dose rates. Also performed nuclear core design for military and space applications.

Was on off-site assignment at Bechtel Power Company, Norwalk, CA, for 18 months. Was responsible for radiation analysis and shielding design for two Westinghouse PWRs. Involved determination of radiation dose rates and shielding design adequacy near primary and secondary shielding, seismic gaps, demineralizers and other balance of plant equipment.

COURSES & CONFERENCES

Attended various ANS conferences in San Diego, San Francisco, and Scottsdale, Arizona. One semester courses taken in Managerial Accounting and in Computer Fundamentals at local college

Fundamentals of Nondestructive Assay of Nuclear Material, held at Los Alamos, New Mexico, October 1-5, 1984

Institute of Nuclear Materials Management (INMM) Conference on use of process monitoring data for MC&A, Nashville, February 13-15, 1985

Nuclear Criticality Safety Short Course, sponsored by the University of New Mexico, June 1-5, 1987

Monte Carlo Analysis course, University of Tennessee at Knoxville, August 13-17, 1990

SECURITY CLEARANCES

DOD	SECRET	
DOE	L	(DOE Q clearance is currently being processed)
NRC	L	
NRC	R	

Registered Professional Engineer in Nuclear Engineering - 1977 (CA)
Registered Professional Engineer in Mechanical Engineering - 1983 (CA)
Member of American Nuclear Society, National and Local Section

CHESTER L. WISHAM
MANAGER, NUCLEAR MATERIAL ACCOUNTABILITY

EDUCATION

A.A. (Accounting), Mesa College, 1983

EXPERIENCE

General Atomics, San Diego, CA (1960 to date)

Manager, Nuclear Material Accountability (NMA) (1985 to date). Responsibilities include developing, revising, implementing, and enforcing nuclear materials control, safeguards, and accounting procedures. Manage the overall system of nuclear material control, including shipping, receiving, storage, and audits. Knowledgeable in the preparation for shipment of low-level waste, DOT regulations and disposal site requirements. Maintain liaison regarding nuclear material safeguards and control with other licensees, regulatory agencies, and all departments within the company.

Instrumental in the development of a computerized accountability system used by GA to implement the Nuclear Regulatory Commission's 10 CFR 70.57 and 70.58 and the related requirements of 10 CFR 70.51. Maintain a highly reliable and accurate system for the accountability of Special Nuclear Material (SNM).

Nuclear Materials Accountant (1963). Responsible for the maintenance and management of the SNM accountability system. Responsibilities included training nuclear material custodians, scheduling inventories, collecting and reviewing data for entry into the GA SNM material accounting system and accomplishing or supervising nuclear material control and safeguards reporting activities to assure compliance with the various regulatory agencies.

Nuclear Materials Assistant (1961). Maintained the company's accountability records and coordinated SNM control activities within the material balance areas.

Nuclear Material Processor (1960). Involved in the company's early activities involving low-level radioactive waste disposal.

JOHN M. BOLIN
SENIOR ENGINEER

PROFESSIONAL SPECIALITY

Nuclear material measurement and measurement control, heat transfer, probabilistic risk assessment, severe accident assessment, and radiological dose assessment.

EDUCATION

B.S., Nuclear Engineering, University of Cincinnati, 1982.

EXPERIENCE

Completed MHTGR-NPR Source Term Report and Review of Generic Plant Description and Source Term Report. Provided safety analysis for NPR Containment Trade-off Study. Responsible for NPR deterministic safety assessment.

Audited and reviewed special nuclear material control and accounting system and measurement control program as member of the Safeguards Management Review Committee from 1985 to 1987.

Performed consequence and uncertainty analysis of MHTGR for preliminary safety information document and probabilistic risk assessment. Analysis included plant transient response to licensing basis and beyond licensing basis events, fuel safety performance, fission product release and transport, containment response, and Monte Carlo assessment of dose uncertainties.

Developed two-dimensional finite-difference heat transfer model of MHTGR used in analyzing depressurization transients with loss of all forced convection cooling.

Developed MHTGR model used in analyzing fuel failure and fission product release from core during thermal transients.

Collected and analyzed measurement control data throughout the HEU fuel fabrication facility. Calibrated, maintained, and designed engineering tests for the various non-destructive assay systems. Developed software for statistical measurement control. Completed the development and installation of a high-resolution gamma ray assay system. Participated in uranium holdup measurement project in which radioactive tracers were used. Responsibilities included interaction with NRC during inspections.

Completed LOFT instrumentation error analysis and coauthor of instrument error analysis code while employed at EG&G Idaho.

PROFESSIONAL ASSOCIATIONS

Member, American Nuclear Society
Associate Member, American Society of Mechanical Engineers
Member, Institute of Nuclear Materials Management

PUBLICATIONS

T.E. Northup and J.M. Bolin, "Safety Yardstick for Evaluating Nuclear Power Reactors," Transactions American Nuclear Society Annual Meeting, June 1989.

J. Bolin, K. Verfondern, T. Dunn, and M. Kania, "Modeling of Fission Product Release from HTR Fuel for Risk Analyses," Proceedings Post SMiRT Conference Seminar on Small and Medium-Sized Reactors, August 1989.

SECURITY CLEARANCE: NRC "L"
DOE "Q" being processed

LAURA R. QUINTANA
MANAGER, HEALTH PHYSICS

EDUCATION

B.S., Biology, Chemistry, New Mexico Highlands University, 1976
M.S., Applied Nuclear Science (Health Physics), Georgia Institute of Technology, 1979

EXPERIENCE

General Atomics, San Diego, CA

Manager, Health Physics (8/82 - present). Assures compliance with 10 CFR Parts 19 and 20 as well as state and .S. Nuclear Regulatory Commission license-imposed radiological safety requirements. Provides review and approval of radiological safety of activities involving special nuclear materials or other radioactive materials, monitors activities involving special nuclear or radioactive materials, personnel monitoring, dose rate measurement, radioactive material detection and assay, air and water sampling and environmental monitoring.

Provides radiological safety support in decontamination/decommissioning of facilities, including low-level radioactive waste disposal. This involves the identification of radionuclides, quantities and classifications as well as radiation and contamination measurements.

The Salk Institute, La Jolla, CA (2/80 - 5/82)

Assistant Radiation Safety Officer and subsequently Radiation Safety Officer. Responsible for the radiation safety program and the radioactive material licensing of two affiliated companies, La Jolla Biological Associates and the Salk Institute Biotechnology Industrial Associates, Inc..

Oak Ridge National Laboratory, Oak Ridge, TN (6/76 - 9/78)

Initially assigned a research project for the Environmental Sciences Division. Subsequently, joined the Health Physics Division as a health physics technician

RICHARD P. VANEK
MANAGER, HTGR FUEL FABRICATION

PROFESSIONAL SPECIALITY

Management of HTGR fuel manufacturing activities and of low-level radioactive waste processing, packaging, shipment and disposal operations. Nuclear fuel process design and development and industrial and nuclear safety analysis.

EDUCATION

Broome Technical Community College, Chemical Technology, 1948-1950
B.Ch.E., Syracuse University, 1958
M.Ch.E., Syracuse University, 1959

EXPERIENCE

Manager of HTGR Fuel Fabrication Department supervising production, operations, and equipment engineering branches manufacturing 93% enriched fuel for FSV power station. Engaged in production scheduling, process testing, procedure initiation and revision, personnel administration. Interaction with health physics, nuclear and industrial safety, facilities personnel.

Manager of HTGR fuel production branch supervising all manufacturing and scrap recovery operations performed by 30 exempt and nonexempt personnel. Production operations involved mixing, drying and sizing equipment, high temperature heat treating, pyrolytic carbon and silicon carbide fluidized bed microsphere coating and injection molding. Scrap recovery of enriched uranium involved crushing, grinding, acid dissolution and liquid-liquid solvent recovery systems.

Production superintendent directing activities of 40 production operators fabricating HTGR fuel with processes described above on day shift. Troubleshooting mechanical, hydraulic, and high temperature equipment in order to maintain set production schedules.

Process Engineer, Fuel Operations Department. Fabrication of FSV type test fuel rods. Development of a new process for the fabrication of nuclear fuel microspheres for HTGR's involving equipment and process development, formulation and implementation of development programs, construction supervision of production scale units, and process consultation during production. A patent was obtained for this process. Process and equipment development involving the application of pyrolytic carbon coatings on nuclear fuel particles.

PROFESSIONAL ASSOCIATIONS

Member, ANS Local Section, San Diego, California
Registered Professional Engineer (Nuclear), California

PATENTS

Held patent (now expired) for nuclear fuel kernel spheroidization furnace.

OTHER DATA

DOE Q clearance

STEPHEN P. MASSEY

QUALITY ENGINEER

PROFESSIONAL SPECIALTY

Project Quality Engineering

EDUCATION

B.S., Geological Science, San Diego State University, 1984

Successfully completed several QA-related training programs, including the DOE QA Workshop for Safety Analysis Reports for Packaging, and General Atomics (GA) Industry Courses in QA Codes & Standards and Lead Auditor Training.

EXPERIENCE

Five years of quality engineering experience.

<u>Professional Experience at General Atomics (GA). (Since 6/87)</u>

Project Quality Engineer for the following projects:

- **Callan Pond Site Waste Removal and Burial Services.**

  QA Program focused on the management of low level radioactive soil that was removed from our Callan Pond Site and delivered to the Nevada Test Site (NTS) at Mercury, Nevada, for burial. QA Program was tailored to comply with all applicable DOE, DOT, and NRC requirements.

- **Radioisotope Separation Services for Industry.**

  QA Program applies to the production, packaging, and delivery of radioactive isotopes for medical applications.

- **Decontamination and Decommissioning (D&D) of GA Nuclear Fuel Fabrication Facility.** (Present assignment)

Support to other projects include:

- Writing conceptual design quality procedures for GA's **Modular High Temperature Gas-cooled Reactor - New Production Reactor Program**. These "tailored" quality procedures satisfy the QA requirements of ANSI/ASME NQA-1-1989; DOE/NP-0001, Appendix R; and 10CFR50, Appendix B.

- Performing independent "Red Team" technical evaluations of GA Defense-related proposals. Evaluations include **R&D of an Automated Radiographic Inspection System** and **R&D of Nonaqueous Equipment Decontamination System.**

- Developing a quality cost accounting system for government **TRIGA Reactor Project**. The program was designed to easily identify and track costs related to quality (i.e., prevention, appraisal, and failure costs).

PUBLICATIONS

ANSI/ASME NQA-1 Quality System Evaluation Matrix. This 121-page handbook identifies the NQA-1 QA requirements endorsed by the NRC, DOE, and ASME, including the specific additions and exceptions made by each agency. The handbook is the property of GA, and has been used as a course text in the GA Codes and Standards Course for Industry.

PROFESSIONAL ASSOCIATIONS

Member, American Society for Quality Control (ASQC)
ASQC Certified Quality Engineer (#19906)
ASQC Certified Quality Auditor (#1089)
Certified Auditor per ANSI N45.2.23

JOHN A CHAPIN
PROJECT MANAGER

SUMMARY

Twenty-eight years' experience in engineering and management with a broad background in planning and directing decommissioning activities for nuclear facilities.

In the broader decommissioning context, Mr. Chapin has conducted numerous studies for End-of-Life planning for nuclear facilities to identify available options and measures that can be taken during the plant's operation life to minimize EOL problems and costs.

RELEVANT EXPERIENCE

- As the Study Manager on the General Atomics Nuclear Fuel Fabrication Facility, Mr. Chapin directed the efforts to develop a decommissioning study and cost estimate for Valley Pines Associates.

- As the Study Manager for the KAERI TRIGA Mark II Decommissioning Cost Study and as a senior member of the Georgia Power Company Hatch decommissioning study team and the Alabama Power Company Farley decommissioning team, Mr. Chapin had lead responsibility for the development of the overall decommissioning methodology, task work scope definition, detailed task manpower requirements, and the evaluation of the decommissioning cost estimate.

- As Program Manager for the Humbolt Bay Decommissioning Project, directed the on-site staff in the development of decommissioning plans for both the SAFSTOR and DECON decommissioning alternatives, the Environmental Report, an environmental sampling program, and the revision of the plant operating tech [illegible]l specifications.

- At the Three Mile Island, Unit 2 Recovery Project, Mr. Chapin developed the operations scenario at TMI-2 for conducting the reactor containment building gross decontamination experiment. He also developed a conceptual plan for stabilization and decontamination of the Auxiliary Building and Fuel Handling and Systems at TMI-2.

- As project engineer at the Idaho National Engineering Laboratory, Mr. Chapin developed an R&D program for decontamination and decommissioning, and researched techniques for decontamination of radioactive soils. He was the primary developer of the Decontamination and Decommissioning Long Range Plan at the major DOE facility.

EDUCATION

B. S., Electrical Engineering, New Mexico State University

ROBERT J. COHOSE
PROJECT PROCUREMENT MANAGER

SUMMARY

Sixteen years' experience in purchasing, cost, and contracts on nuclear, fossil, and government facility projects.

RELEVANT EXPERIENCE

- Oak Ridge Office Procurement Manager - provides supervision and oversight of Oak Ridge Office procurement activities.
- Project Procurement Manager on FUSRAP Project, a decontamination and restoration project. Also, previously held the position of Senior Contracts/Purchasing Supervisor on this project.
- Provided field procurement activities on several nuclear power plant sites.

EDUCATION

B.S. Construction Engineering, Arizona State University

SPECIAL TRAINING

- Construction Contracts, Pepperdine University, Certificate, July 1983
- Government Contract Law, American Graduate University, Certificate, February 1985
- Financial Management & Pricing, American Graduate University, Certificate, February 1985
- Masters Institute in Government Contracting, Federal Publications, Inc., Certificate, December 1988
- Small and Disadvantaged Business Utilization Program, U. S. Department of Energy, Certificate, June 1989

GEORGE D. CREMEANS
PROJECT ENGINEER

SUMMARY

Twenty years' project engineering and construction management experience on various projects throughout the United States, with primary responsibility for the management of operational superintendents and field engineers. This experience includes: construction and maintenance of fossil and nuclear power plants, contract administration, and initial fuel loading and refueling of nuclear propulsion reactors. Most recently completed the Three Mile Island Unit 2 Recovery Project as the Project Engineer of Systems Engineering.

RELEVANT EXPERIENCE

- Project Engineer for the Three Mile Island Unit 2 recovery engineering group. Responsible for the design and installation of systems to support the decontamination and decommissioning of the plant, including software preparation, technical overview, personnel administration, resolution of technical problems, and coordination of information and related activities with other recovery groups. Two major design efforts accomplished during this assignment were; 1) the development of a radwaste removal system used to collect, sample, and package accident related sludges and resins, and; 2) the development of a program to dispose of approximately 2.3 million gallons of accident generated water by forced evaporation, and; 3) the collection and disposal of approximately 161 tons of radioactive waste from the process.

- Task Supervisor for TMI-2 Reactor Disassembly and Decommissioning. Responsible for directing projects pertaining to the reactor disassembly and defueling activities including; tooling for core topography using ultrasonic techniques, debris removal form the reactor core for characterization, and flushing of the reactor plenum with vessel closure head in place.

- Assigned to various projects with previous employer, the majority of which were nuclear facilities. This included an assignment at the DOE Idaho National Engineering Laboratory (INEL) as Manager of the Constructability/Feasibility Group responsible for the acceptance review of all designs installed at the INEL facility.

EDUCATION

B.S. Mechanical Engineering, Clayton University, Clayton, MO.

J. VICTOR HARGREAVES
PROJECT CONTROLS MANAGER

SUMMARY

Eighteen years' experience in all aspects of Project Controls including: conceptual and definitive cost estimate development, critical path method scheduling, cost controls, variance analysis and incentive bonus systems development, value engineering, and subcontract administration. Performed supervisory functions and developed specific controls on projects ranging up to multibillion dollar revenue projects. Worked on a variety of nuclear and non-nuclear facilities.

RELEVANT EXPERIENCE

- Cost/Schedule Engineering on hazardous waste projects including EPA "ARCS" superfund programs, DOE "FUSRAP" programs, remedial investigations and flexibility studies of various radioactive sites.

- Senior Planning Specialist for the multibillion-dollar DOE project to characterize a repository site for the underground storage of high level commercial and defense generated nuclear waste.

- Project Controls/Contract Administration Engineer on loan to the U.S. Department of Energy, Carlsbad, New Mexico. Assisted in the construction management of a $700 million pilot facility for the underground storage of defense-generated nuclear waste.

- Provided validation assistance and overview of the DOE C/SCSC; contracts administration, and value engineering.

EDUCATION

B.S.I.E. - Evaluated U.S. equivalent, Educational International, New York, NY, 1981

PROFESSIONAL AFFILIATIONS

Senior Member, Institute of Industrial Engineers
Member, American Associate of Cost Engineers
Member, Society of American Value Engineers

P. KENNETH JACKSON
HEALTH AND SAFETY MANAGER

SUMMARY

Sixteen years' experience in nuclear decontamination, radwaste processing, radioactive material transportation, and health physics in nuclear facilities. Technical experience includes hands-on work as a nuclear plant operator, health physics technician and supervisor, plant chemistry specialist, decontamination specialist and supervisor, and radioactive material handling and shipping supervisor.

RELEVANT EXPERIENCE

- Operations Manager at the BNI Service Center at Oak Ridge, a licensed nuclear facility. Responsible for the daily operation of the facility, including scheduling and manloading, staff direction, technical guidance, radioactive shipment preparation and movement both domestic and internationally, and radioactive work controls and license compliance. Was the alternate Radiation Protection Officer for the facility (as approved by the State of Tennessee Division of Radiological Health).

- Radwaste Shipment Manager on the CEER, Mayaguez, Puerto Rico decontamination project, assuming Site Manager responsibilities for the latter stages of the project. The project resulted in the release of a former research and training reactor for unrestricted use by the University of Puerto Rico. Activities included removal of nonradioactive and radioactively contaminated materials, equipment, structural components, large irradiation sources, and shipment of these materials.

- Senior Bechtel Site-Representative at the Koeberg Nuclear Power Station in South Africa. Responsible for administering contracts for decon and health physics support services. Also served as a consultant in the areas of radwaste transport and disposal, licensing amendment, radworker training, and ALARA evaluations.

- Decontamination Technical Planning Manager. Responsible for the development of long-range plans relative to the decontamination approach during the recovery program at Three-Mile Island Unit 2.

EDUCATION

A.S., Physical Science, University of New York, Albany NY

J. F. MATTSON
OPERATIONS SUPERINTENDENT

SUMMARY

Seventeen years in nuclear waste management and radwaste systems operation, nuclear plant decontamination, and nuclear facilities decommissioning, the majority in various supervisory capacities. Eight additional years in conventional industrial plant maintenance and supervision of craft personnel.

RELEVANT EXPERIENCE

- Operations Superintendent and Radwaste Operations Site Manager for Peach Bottom Nuclear Power Plant. Responsible to the Utility for plant area cleaning, housekeeping, and decontamination; processing and management of liquid and dry radiological waste, including solidification and volume reduction; supervision of approximately 200 radwaste support employees, including administrative, engineering, labor, and operators/technicians.

- Limerick Nuclear Power Plant Startup Supervisor for radwaste handling and decontamination.

- Field Operations Supervisor, TMI-2. Responsible for reactor building decontamination operations, including employee training in decontamination methods.

- Special Projects Supervisor, EG&G, Idaho. Budget tracking, work schedule development and maintenance.

- Field Operations Superintendent, EG&G, Idaho. Decommissioning of OMRE and SPERT II, III, and IV reactors and associated facilities.

- Field Operations Superintendent, EG&G, Idaho. Decommissioning of EBR-1 to National Monument status.

- Maintenance Supervisor, EG&G, Idaho. Maintenance and refueling support for numerous Idaho reactors.

EDUCATION AND QUALIFICATION

2-1/2 years of general engineering college credit

Licensed Radioactive Materials Shipper, 1975-78

BRET PEACOCK
PROJECT QUALITY ASSURANCE MANAGER

SUMMARY

Nineteen years' experience in Quality Assurance, Quality Control, and startup on nuclear, fossil and space projects. Five years' experience as a reactor plant operator and technician on naval nuclear power plants.

RELEVANT EXPERIENCE

- Project Quality Assurance Supervisor for NASA's Component Test Facility and BNI's Service Center which performs work in radiological and hazardous material decontamination, repair, and maintenance.

- Project Quality Assurance Engineer on the Skagit Nuclear Power Project responsible for the development and implementation of the quality assurance program.

- Project Field Quality Control Engineer on the Kori 5 & 6 Nuclear Power Project in Korea and for two years on the Intermountain Power Project coal fired plant. Responsible for the original development and implementation of Quality Control Programs including QC procedures and instructions, inspection plans, inspection of work, surveillance of contractors, supervision of civil testing and NDE laboratories and contractors, and training and certification of QC inspectors.

- Mr. Peacock was also a quality control supervisor in areas of quality control, spare parts procurement, configuration management, and equipment environmental qualification during TVA's restart efforts at Browns Ferry and Sequoyah Nuclear Plants.

EDUCATION

B.S., Electrical Engineering, Sacramento State College

PROFESSIONAL CERTIFICATION

Professional Engineer, California No. QU4668

QUALIFICATIONS

Mr. Peacock has been certified as a Lead Auditor per ANSI N45.2.23 and a Level III Inspector in the electrical and control system areas per ANSI N45.2.6. Mr. Peacock is familiar with and has utilized and recognized industry standards for implementation of specific quality programs. These include ANSI/ASME NQA-1, IOCER50 App. B, ANSI N45.2 and daughter standards and 10CFR50.49.

3.0 METHODS FOR PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

3.1 RADIOLOGICAL HISTORY

3.1.1 Radioactive Material Work Locations

Building 37 North and its ancillary structures house the fuel fabrication areas, offices, clothing changerooms, laboratory, storerooms, eating area, and shops. The process materials used in the above facility include oxides and carbides of uranium and thorium; the uranium material was approximately 93% enriched in the U-235 isotope. This section addresses SVA, comprised of:

- o Building 37 North
- o West Shed
- o Service Building
- o North Annex
- o Tunnel
- o West Vault
- o Soot Filter
- o Fume Scrubbers

Plans of the facility are shown in Figures 1.3-1 and 1.3-2. Specific locations where work with radioactive material occurred, which are considered to be radioactively contaminated, are as follows:

Building 37 North - Main Floor

Final Heat Treat Furnace (Lower Furnace)
South High Bay
Thorium Processing Facility
Chemical Preparation Room
VSM 10 and 11 (Lower)

HCL Furnace (Lower)
North High Bay
Vacuum Cleaner Service and Repair Station
Dry Coater Control Room
Graphite Matrix Mix Room
Dry Scrap Recovery Area
Particle Production Area (Old Mix Line)
Decon Rooms (2)
VSM 6
Vault Office and Assay Room
Thorium Storage Vault (South Vault)
DESMA Rod Press Area
Thorium Spheres Production Area

Building 37 North - Second Floor

HCL Furnace (Upper)
VSM 10 and 11 (Upper)
Carbonization Room Extension
Thorium Room Roof
South Mezzanine
Transformer Area
North Mezzanine
Old Coater Mezzanine (Coater Platform)
Quality Control Laboratory
Fuel Rod Carbonization Furnace Area (Upper Furnace)
Upper Furnace Load Prep Room
Thorium Spheres Mezzanine
HEPA Filter Area
VSM 6 (Upper)

West Shed

HEPA Ventilation Equipment Area
Electrical Equipment Area

Service Building

Shop
Stock Room

North Annex

Mop Water Area
Transit Storage Area
Liquid Rad Waste Dumping Station
Laundry Storage Area
Waste and Barrel Handling Area
Transformer Enclosure Area
Scrap Disposal Area
Dry Coater
Coater Cooling Room
Fuel Block Loading Room
Fuel Block Packaging Area

Tunnels

North Tunnel Annex
North Tunnel Fuel Storage Area
Non-SNM Tunnel Annex South Storage Area
South Tunnel Annex

West Vault
Soot Filter
Fume Scrubbers

A radiological survey of the facility has characterized several contaminated areas (refer to Appendix A for details). In general, contamination was found where it was expected. Only three locations were discovered where contamination had penetrated more than 1/4 in. into structural concrete. These locations were on the first floor. However, contamination was found under paint, tile, and other surface coatings. No contamination was detected in soil samples taken from below Building 37 North. Even those samples taken near floor joints had no detectable radioactivity (above background), inferring that contamination was contained by the floor. The North Annex areas have been maintained as non-shoe-cover areas since 1985, and contamination on floors and equipment is expected to be minimal except for waxed or painted areas where contamination has been fixed.

Of the 35 sample locations on the first floor, 16 were noncontaminated, including two background samples. Of the 19 contaminated locations, three locations near VSM-10 and -11 and locations in the particle production area, and in the mop water treatment area (North Annex) were contaminated beyond the first 1/4 in. of concrete. The first floor of SVA contains two highly contaminated rooms, VSM-10 and -11, and the thorium processing facility used for thorium oxide and carbide processing. The interior walls, floors, and equipment in these rooms are covered with thorium oxide dust. Personnel working in these rooms may require extra protective clothing, and breathing air. The balance of this area has been a "non-shoe-cover" area, while equipment within may have low-level external contamination and may be highly contaminated internally. Of twenty-five sample locations on the second floor, eight were noncontaminated. Thirteen of the seventeen contaminated locations were in the overheads or on walls. In the tunnel, one location was uncontaminated, while the other three were contaminated. On the roof, there was one uncontaminated location and one contaminated location in the vicinity of the HEPA filter area ventilation discharge stacks.

The particle production area is currently maintained as a shoe-cover area and contains some highly contaminated equipment such as the grinder, fissile and fertile beds, and VSM [illegible]. The HEPA room, west shed, control room, vault office, and vault floors are non-shoe-cover areas. The HEPAs and vault storage racks have low contamination levels.

The HEPA filters and their associated floor areas on the second level of Building 37 North are contaminated. The thoria sphere enclosure is highly contaminated. The balance of this area is either not contaminated or slightly contaminated.

The thorium room roof is a shoe-cover area. The equipment on the roof is internally contaminated internally. The Quality Control lab equipment, benches, and hoods have been removed, exposing a slightly contaminated floor. The offices and hallway in the area are uncontaminated.

The south mezzanine area has some highly contaminated equipment, such as the VSM-10 and -11 upper room and the fissile mix line. The scrap recovery area also contains contaminated hoods, furnaces and the upper floor of the grinder tower. The scrap recovery area has been used for dry and wet scrap processing. Liquid spills and leaks have occurred in this area and shoe covers are required. The north mezzanine contains the upper level of the VSM-6 room, which is highly contaminated. The balance of this mezzanine requires shoe covers but has low-level contamination. Shoe covers are not required for entry to the coater platform and old coater mezzanine, but contaminated equipment is stored there.

As indicated in the Characterization Report (Appendix A), contaminants found were U-238, U-235, and thorium and its daughters. Concentration of the contamination averaged around 2-6 pCi/g, with maxima of approximately 25pCi/g U-238, 400 pCi/g U-235, and 600 pCi/g of thorium and its daughters.

Loose contamination ranged from < 1 dpm/100 cm^2 to 13,500 dpm/100 cm^2 for alpha and < 5 dpm/100 cm^2 to 7000 dpm/100 cm^2 for beta. As expected, the higher levels of contamination were found in and around the main process equipment.

3.1.2 Radiation Levels, History of Spills/Releases

Radiation levels are less than 1 mr/hr with the exception of a few areas. The thorium bed converter vacuum pumps used in thorium processing tended to concentrate thorium daughter products internally, which resulted in somewhat higher radiation levels - up to 50 mr/hr. Minor spills of contaminated liquids and soot occurred periodically in this area but were routinely remediated by a decontamination crew which was an integral part of the work force.

Large amounts of uranium and thorium oxide powder were utilized in the fuel manufacturing process. Heat treatment processes were used to convert these oxides to carbides, which tended to back-oxidize, forming fine powders. Since these operations were dusty, HEPA-filtered glove boxes were used extensively to control airborne contamination. The generated dusts were carried into the ducts leading to the filters. In the case of systems containing U-235, the ducts were periodically removed and cleaned to minimize the U-235 hold-up. The current U-235 hold-up distributed throughout thousands of feet of ducting is estimated to be around 700 grams, as measured with an Eberline two-channel stabilized assay meter.

The large amounts of thorium oxide that were processed may have led to appreciable hold-up in selected ducting for which only cursory accountings have been made.

In certain areas of SVA construction, asbestos-bearing materials were used (see Section 2.1.3.1). Floor coverings containing asbestos may have been used both in original construction and facility modifications. All such materials, with the exception of floor coverings, will be removed prior to the start of decommissioning work to minimize worker exposure to asbestos. Since some of the floor coverings may have trapped existing radioactive contamination, they will be removed during the decommissioning project, with necessary precautions to protect workers. In addition, workers will be trained to recognize other potential sources of asbestos and to report such findings.

Because SVA was originally constructed in the early 1960's, lead-based paint may have been used. Systematic sampling of painted surfaces will be conducted prior to paint removal to check for lead or other heavy metal contamination.

Hazardous materials, including trichloroethylene, methylchloroform, methyltrichlorosilane, trichlorotrifluoroethane, ethyl alcohol, kerosene, nitric and hydrofluoric acids, various spray paints, and hydraulic and lubricating oils, have been utilized in the facility throughout its operating history. All such materials will be removed prior to the start of decommissioning activities; however, workers will be trained to be alert for the presence of any such material that may have been overlooked.

3.2 ASSURANCE THAT EXPOSURES ARE ALARA

3.2.1 ALARA PROGRAM

As a licensee engaged in activities under licenses issued by the Nuclear Regulatory Commission and the State of California, GA is required to make every reasonable effort to maintain radiation exposures (and releases of radioactive materials in effluents to unrestricted areas) as low as is reasonably achievable (ALARA), taking into account the state of technology, the economics of improvements in relation to benefits to the public health and safety, the utilization of atomic energy in the public interest, and other societal and socioeconomic considerations. In addition, GA must meet requirements set forth in 10CFR20, and Title 17, California Administrative Code, Chapter 5, Subchapter 4, Group 3.

Even though current occupational exposure limits provide a very low risk, GA is obligated to avoid unnecessary exposure to radiation. The ALARA objective is therefore to reduce occupational exposures to levels as far below the acceptable limits as is reasonably achievable. This is accomplished by means of good radiation protection planning and practice and by a commitment to policies that foster vigilance against departures from good practice.

In addition to maintaining individual radiation doses ALARA, the collective dose received by all exposed individuals should also be maintained at the lowest practicable level. It would not be desirable, for example, to hold the largest doses to individuals to some fraction of the applicable limit if this involves exposing additional people and significantly increasing the collective exposure.

All GA operating groups and contractors are required to conduct their activities in accordance with federal, state, and local rules and regulations; license regulations; and company policy and established practices. Such policy requires activities to be conducted in an ALARA manner.

Any individual or organization desiring to perform decommissioning work must obtain advance authorization through submittal and approval of a Work Authorization (WA) and/or a Radiological Work Permit (RWP). Work authorizations related to the decommissioning project will be reviewed annually.

Decommissioning workers must complete a training course which will adequately prepare them to perform assigned tasks in a radiologically safe manner.

Facilities, equipment, supplies, a staff of radiological health specialists, and training will be provided to maintain personnel exposures and releases ALARA. It is incumbent upon each employee to use these resources to assure that they perform their work in an ALARA manner.

3.2.2 Management Positions Responsible for Radiation Protection and Maintaining Exposures ALARA during SVA Decommissioning

The management positions responsible for radiation protection and maintaining exposures ALARA during SVA decommissioning, as well as the specific responsibilities of these positions, are identified below. The locations and relationships of these positions within the GA organizational structure are shown in Figure 2.2-3.

As described in Section 2.2.2.2, the GA Decommissioning Project Manager shall be responsible for the safe operation and control of decommissioning activities for the protection of the environment from potential sources of radiation/contamination exposures from decommissioning activities. The GA Project Manager, with the assistance of the GA Licensing, Safety and Nuclear Compliance organization and the BNI management shall ensure that the conduct of decommissioning activities is in compliance with applicable criteria and GA's rules and practices.

The Health and Safety Manager (HSM) shall be responsible for ensuring that all work performed by BNI personnel and BNI's subcontractor personnel is conducted in compliance with the Decommissioning Project Health Physics program as defined in this Plan and implemented by approved procedures. The BNI HSM will report to the BNI Project Manager and will interface, as required, with the GA Health Physics Manager on matters related to radiological health and safety.

The GA Licensing, Safety and Nuclear Compliance organization will be responsible for assuring compliance of decommissioning project activities with applicable license and regulatory requirements controlling radiological and nuclear safety and for safeguarding licensed material.

The compliance functions responsible for radiation protection and maintaining exposures ALARA are described below and are headed by managers or supervisors whose titles are synonymous with the functions for which they are responsible. Responsibilities and minimum qualifications* for these positions are described below.

* A two year degree plus five years of directly applicable experience or a high school diploma (or equivalent) plus 10 years of directly applicable experience may be considered equivalent to an accredited four-year college degree and appropriate work experience, subject to a determination by the Vice President of Human Resources that proficiency has been demonstrated.

o **Licensing, Safety, and Nuclear Compliance (L&NC)**

This function administers licenses and reviews and approves all Work Authorizations (WA) for compliance with applicable regulations and license conditions. It also provides interpretation of licenses and regulations, determines the need for licensing actions, coordinates the preparation and processing of applications, disseminates license requirements to operating organization, and maintains or oversees maintenance of master license records for NRC or GA audit or review. This function is responsible for the overall planning, coordination, and administration of the special nuclear material (SNM) measurement control and accounting, nuclear safety, health physics, and industrial safety functions.

The Manager, Licensing, Safety and Nuclear Compliance shall have demonstrated his/her proficiency in activities relevant to the functions assigned. Demonstration of this proficiency shall be based on the manager's qualifications which shall include an accredited four-year college degree in science, engineering, or other related field, a minimum of two years applicable work experience and/or training in nuclear industry management, and a knowledge of nuclear safety and health physics.

o **Health Physics**

This function ensures compliance with Title 10 Code of Federal Regulations, Parts 19 and 20 and with license-imposed radiological safety requirements. It is responsible for review-and-approval of proposed activities involving radioactive material, monitoring/ auditing of such activities, and providing services such as personnel and environmental monitoring, dose rate measurement, radioactive material detection, and assay. The Manager, Health Physics, or designee, will be responsible for ensuring that decommissioning activities are conducted in compliance with GA's license and the Health Physics program described in this Plan.

The GA Health Physics Manager will have demonstrated his/her proficiency in nuclear health physics activities and the evaluation of potential radiological hazards. Demonstration of this proficiency shall be based on the manager's qualifications, which shall include an accredited four-year college degree in the physical sciences, biological sciences, or other related field and a minimum of two years of experience in phases of nuclear health physics relating to GA licensed activities and the evaluation of potential radiological hazards therefrom, or equivalent work experience or training.

The GA Health Physics Manager and the BNI HSM shall each have the authority to stop any decommissioning activity which threatens the safety and health of workers, visitors, or the public or the quality of the environment.

Criticality and Radiation Safety Committee (CRSC)

In addition to the above functions, GA maintains a Criticality and Radiation Safety Committee (CRSC). The CRSC reports to the office of the President via a designated vice president. Committee members who have applicable expertise and qualifications are appointed by the company's Executive Committee. A written policy statement establishes committee and member responsibilities. At a minimum, the committee (1) acts in an advisory capacity for radiological safety, (2) reviews policies and criteria on safety of SNM operations, (3) provides second echelon review for nuclear safety analysis, and (4) audits work involving radioactive materials for conformance with and effectiveness of applicable procedures and practices to ensure radiological and nuclear safety, including conformance with ALARA policy.

3.2.3 Methods for Occupational Exposure Protection

This section provides a discussion of the work activity control program to be implemented for the SVA Decommissioning Project. Further information on the equipment, techniques, and practices to be used in meeting the standards for protection against radiation of 10 CFR Part 20 including the program for controlling radiation exposure and contamination, the respiratory protection program, and the program for controlling the handling and storage of radioactive materials, is provided in Section 3.3 of this Plan.

3.2.3.1 Work Activity Control Program

SVA decommissioning activities will be controlled through Work Authorizations (WA's) and Radiological Work Permits (RWPs). All activities which may involve exposure of workers to radiation and which are of significant scope and/or duration must be performed under a WA which has been approved in accordance with the approval procedure detailed in Section 3.2.3.1.1 prior to their start. RWP use and control is discussed in Section 3.2.3.1.2.

3.2.3.1.1 Work Authorization (WA) Approval

The procedure for obtaining approval of a WA is shown schematically in Figure 3.2-1 and is described below.

1) The individual responsible for the decommissioning work to be performed (i.e., the "Principal Investigator") shall prepare a WA request identifying the proposed work, the quantity and form of radioactive materials involved, the equipment to be used, individuals authorized to do the work, and the applicable work procedures. The WA shall be reviewed and approved in writing by the GA Decommissioning Project Manager or his/her designee(s), who must assert that all license and company criteria will be met and that required procedures for radiological safety, criticality, material accountability and control, and physical protection will be followed. Also, he/she must assert that other applicable safety related features of the work, such as potential for fire or explosion, etc., are adequately considered and that suitable provisions have been incorporated. The approved WA and backup information shall then be forwarded to Health Physics.

2) The WA and appropriate safety-related information shall be approved in writing by Health Physics, Nuclear Safety, Nuclear Material Accountability, Industrial Safety, and Licensing, Safety, and Nuclear Compliance. In addition, the WA shall be sent (1) to Security for approval if physical protection requirements for the area are to be modified, and (2) to the CRSC for approval if the WA is for work involving a new or revised nuclear safety analysis and involves a quantity of SNM greater than 350 grams of U-235. Any reviewer may seek outside expertise as deemed necessary.

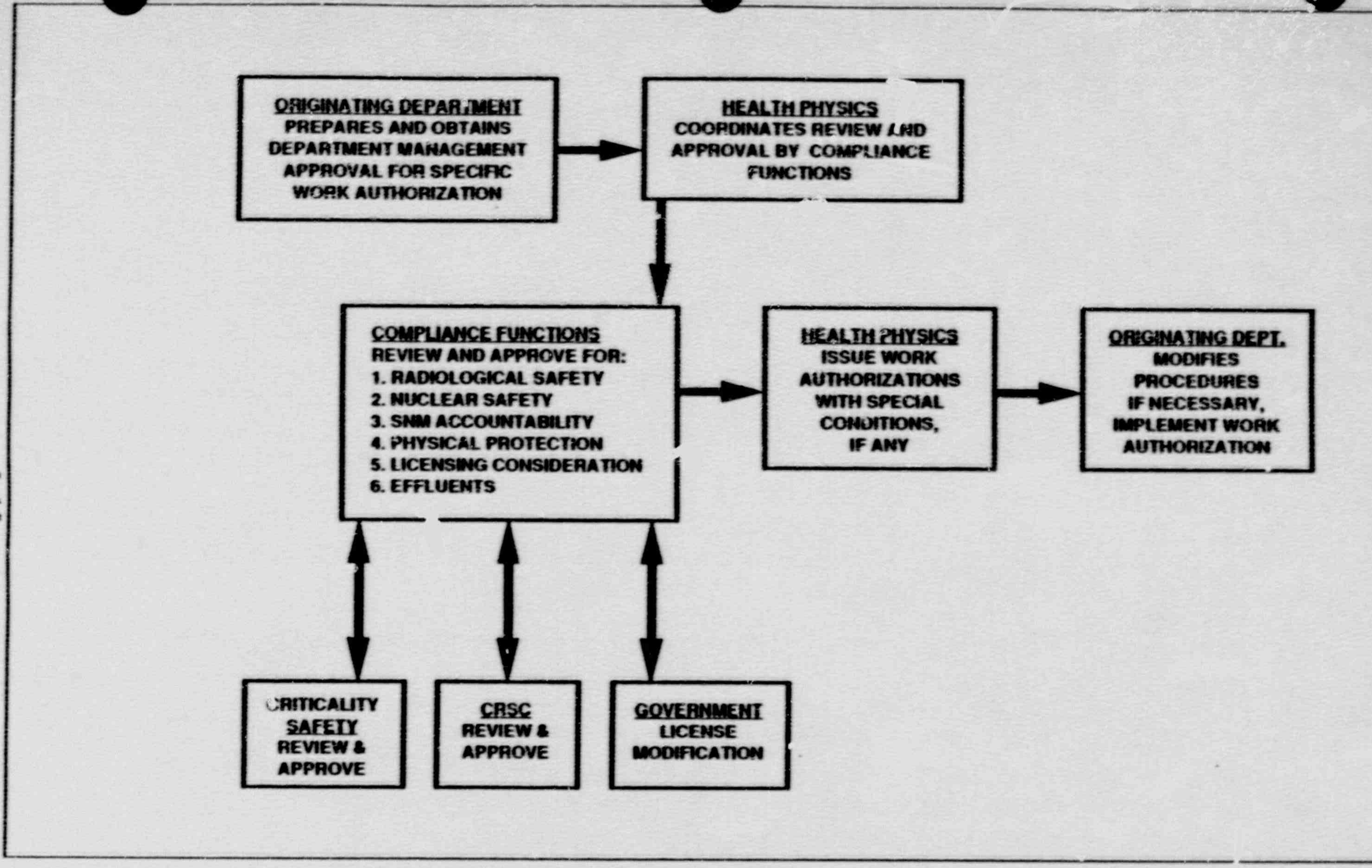

Figure 3.2-1
WORK AUTHORIZATION APPROVAL PROCESS

1 24 1698.1

The evaluation of each request for WA approval shall include but not be limited to the following determinations. The manager of the responsible compliance function, or his/her designee, shall determine:

- That the proposed work is within currently authorized license limits

- That radiological safety is properly assured. This includes a review of operational methods to maintain radiation exposures ALARA

- That criticality safety has been reviewed and adequate provisions are incorporated

- That a new or revised nuclear safety analysis, if applicable, has been separately verified by two independent parties. (Analysis of solid angles or interaction requires verification by a single independent party.)

- That proper material protection for SNM is provided and that applicable limits and regulations are observed

- That an assessment of the effluents has been performed and that proper provisions for their control exist

- That industrial safety practices and procedures have been adequately considered

- That proper SNM accountability is provided

3) After the WA has been approved in writing, authorization shall be deemed granted. Authorization may be contingent on

special conditions as required by the license or as deemed necessary by the WA reviewers.

4) The Principal Investigator shall revise the operating procedures as necessary to incorporate any special conditions imposed by the WA reviewers before beginning the authorized activity. The current version of the WA and the associated approved procedures shall be maintained in or near the wor' are

5) All Decommiss ing Project WAs will be reviewed and the approvals updated annually.

3.2.3.1.2 Radiological Work Permit (RWP)

Personnel on Work Authorizations performing work in radiologically controlled areas will perform the work under the authorization of a Radiological Work Permit (RWP).

> NOTE: The only exception to this requirement is that in accident or emergency conditions, personnel access to Controlled Areas shall be controlled as specified in the Emergency Procedures.

The RWP is also intended for authorization of employees or outside contractor or subcontractor personnel not on a WA who will perform limited work in a controlled area, but whose exposure to possible radiation or contamination is incidental to their function (e.g., maintenance, facility modification, janitorial, etc.)

An RWP for the task-specific work will be generated for the Unit Work Instruction (UWI) that involves work with radioactive materials or in contaminated areas. The function of the RWP is to define radiological conditions and personnel protection measures required to perform the tasks specified in the UWI. The RWP will be approved by the GA Health Physics Manager or his designee, who will also make the determination if other GA approvals such as Nuclear Safety are required.

All personnel working under an RWP will sign an RWP sign-in sheet, indicating that they understand the conditions and requirements for the work.

The RWP shall be initiated by the individual responsible for the work to be performed. The request for an RWP shall include the specific work location, the time period during which the proposed work will be performed, and a description of the work. Health Physics will indicate the following on the RWP:

- Protective clothing to be worn
- Requirements for checking personnel and personnel clothing for contamination
- Personnel monitoring requirements
- Conditions under which Health Physics is to be contacted for specific actions
- Any special Health Physics instructions

3.2.4 Emergency Planning and Procedures

The administrative and operating procedures in effect at GA and the specific procedures to be used for SVA decommissioning are designed to prevent the occurrence of accidents. However, the possibility always exists that accidents or natural disasters may occur, despite the most effective safeguards. Because of this possibility and the presence of radioactive material, GA maintains an NRC-approved Radiological Contingency Plan. The plan establishes an organization capable of coping with emergencies. This organization will classify emergencies according to their severity, define and assign responsibilities, and clearly outline the most effective measures to mitigate the consequences of an emergency to the employees and the public. GA's emergency procedures will be placed in effect whenever an emergency exists.

3.3 HEALTH PHYSICS PROGRAM

The Health Physics program to be implemented during the SVA decommissioning is described in this section.

3.3.1 Work Area Monitoring

Radiation and contamination surveys of areas shall be performed as required by 10CFR20.201.

3.3.1.1 Radiation Surveys

Radiation surveys shall be conducted as appropriate, using portable instruments.

3.3.1.2 Contamination Surveys

During equipment disassembly, surveys will be performed concurrently and documented daily. Surveys of other facility areas will be performed weekly.

The action levels for fixed and removable contamination are described in Table 3.3-1. The levels are based upon the requirements given in Table 2 of Regulatory Guide 8.24. When a level is exceeded, an investigation shall be made and appropriate corrective action taken.

Table 3.3-1

CONTAMINATION ACTION LEVELS

	Contamination (dpm/100 cm^2)			
	Fixed		Removable	
	alpha	beta	alpha	beta
Unrestricted	1000	2500	200	200
Restricted Areas*			*2,000**	2,000**
Skin			0	
Personal Clothing Worn Outside Restricted Areas			200	

* Determined by amount of external radiation likely to be received by personnel

** Excluding enclosed areas where respiratory protection use is required and has been approved and the interior of glove boxes, hoods and other equivalent enclosures

3.3.1.3 Air Sampling

Air sampling results assist in confirming the adequacy of the ventilation system. Air sampling equipment shall be maintained and used where there is a potential for particulate airborne radioactivity. At a minimum, this equipment shall consist of a sampling head, filter, air volume flow measuring device, and pump. The standard method of evaluating the air samples shall be based on removal of the sample filter to suitably calibrated laboratory counting equipment. Various types of air sampling equipment will be used during the decommissioning project. This equipment includes fixed air samplers, at least ten (10) portable air samplers, two (2) high volume air samplers, and four (4) lapel air samplers. In addition, four (4) continuous air monitors (CAM's) will be used in the facility for continuous monitoring of airborne concentrations. Air sampler placement shall conservatively represent the worker's breathing zone.

Fixed air samplers will be taken in "breathing zones" whenever possible. The collector will be oriented so that the collection orifice is vertical at a height of about six (6) feet above the ground. Positioning of a sample head in this manner allows the collection of respirable dust particles suspended in the air rather than non-respirable particles that might fall own on the collector. Portable air samplers and high volume air samplers will be used in locations where fixed air samplers are not available or in locations where the fixed air samplers are not in the worker's "breathing zone". Lapel air samplers will be used in areas where high airborne concentrations are likely. The Continuous Air Monitors (CAM's) will be used for general area monitoring and to provide an immediate alarm if airborne concentrations exceed a pre-established level. This level will be determined based on the activity, the radionuclide involved, the length of time expected in the area, and other factors. Each of the air samplers with suitable collection filters shall be coupled to a measuring device to determine the volume of air that flows through the filter. The collection filters will be evaluated for alpha, beta, and gamma activity as appropriate to the operation. Samples shall be collected for each work shift, usually eight hours. Typically, the samples will be analyzed within 24 hours. Samples collected on Fridays and holidays are analyzed on the next regularly scheduled work day.

Average airborne concentrations are anticipated to be maintained at a level well below the Maximum Permissible Concentrations (MPC's). If these <u>average</u> levels for an area or for the facility (whichever is applicable based on occupancy) are maintained below 15% of the applicable MPC when averaged over a calendar quarter, MPC-hours will not be maintained for personnel.

If the average levels increase above this level (based on a quarterly average), MPC-Hours will be maintained beginning the next calendar quarter. MPC-hour action levels shall be set at 20

MPC-Hours/40 hour workweek. If this level is exceeded for any individual, an investigation and corrective action (as needed) shall be completed.

Any one sample measuring $\geq$ 1 MPC shall be reported to the HP Manager, who will investigate and take appropriate corrective action.

Where air sample data indicates airborne radioactivity exceeding 50% of the MPC in a specific area or location averaged over one week, the results shall be investigated and appropriate corrective action taken.

3.3.2 Personnel Monitoring

3.3.2.1 Dose Limits

The regulatory dose limits for occupational exposure are provided in 10CFR20 and in State of California Radiation Control Regulation (Title 17).

3.3.2.2 External Dosimetry

Film dosimeters or thermoluminescent dosimeters (TLD's) shall be used to measure personnel external occupational radiation dose. Dosimeters shall not be issued for a period longer than one calendar quarter. Table 3.3-2 presents a summary of the typical characteristics of such devices. Self-reading pocket ionization chambers (dosimeters) may be used in addition to film dosimeters or TLD's.

Decommissioning workers will be observed by a health physics technician to ensure that they properly frisk when leaving a controlled area. Visitors and intermittent workers will be monitored by a health physics technician. Monitoring will include checking the hands, shoes, and clothing for possible contamination.

TABLE 3.3-2

FILM BADGE AND TLD DOSE RANGES

RADIATION TYPE	DOSE RANGE FROM	TO	ENERGY
Film Dosimeter			
X-ray and gamma	10 mRems	40 Rems	4 KeV to 20 KeV
	10 mRems	40 Rems	20 KeV-100 KeV
	10 mRems	750 Rems	100 KeV-3 MeV
Beta	50 mRems	750 Rems	Above 1 MeV
TLD Dosimeter			
X-ray and gamma	10 mRems	10^5 Rems	10 KeV and up
Beta	50 mRems	10^5 Rems	1 MeV and up

3.3.2.3 Internal Dosimetry

A bioassay program shall be conducted for decommissioning workers. This program will be used where air sampling is necessary for personnel protection. Urinalyses, as well as in-vivo counting techniques will be used to determine the extent of inhalation or ingestion of radioactive materials, if any. Thorium lung counts shall be performed using a Helgeson "Do-It-Yourself" whole body counter, located on the GA site. U-235 lung counts shall be performed using a mobile counter from Helgeson Scientific Services, Inc.

Uranium Bioassay Type & Frequency Criteria

An individual shall submit a urinalysis specimen prior to working in controlled areas. The individual will undergo a U-235 lung count on the first scheduled visit of the in-vivo measurement equipment.

The minimum bioassay program for uranium (U-235) includes semiannual lung counts and quarterly urinalyses.

Other analysis types and frequencies shall be in accordance with Tables 2 and 3 of NRC Regulatory Guide 8.11, dated June 1974 and entitled "Applications of Bioassay for Uranium."

Interpretation of Uranium Bioassay Results (action levels and actions)

Appropriate action based on bioassay results depends first upon the underlying purpose of the measurement.

Preparatory Evaluation

An investigation will be conducted if urinalysis reveals the presence of uranium in personnel prior to job assignment.

Exposure Control

When work is in progress and bioassay measurements are made routinely, the results will be reviewed and action taken as follows:

- o Urinalysis results which exceed 20% of the Maximum Permissible Limit (MPL) will be reviewed and investigated. For U-235 in a urine sample, the Maximum Permissible Limit assumes 93% enriched uranium. A result of 0.625 dpm/liter is equal to 100% MPL for 93% enriched uranium. Appropriate corrective action will be taken as required. Additional measurements will be obtained as required.

  If the results of the urinalysis exceed the MPL and are unexpected, work restrictions will be established for affected individuals. Individual case studies will be performed.

- In-vivo "U-235 lung count" measurements exceeding 40% of the Maximum Permissible Lung Burden (MPLB) will be reviewed and investigated. Appropriate corrective action will be taken. **For 93% enriched uranium, the MPLB is 253 micrograms.**

  If the result is >100% of the MPLB, work restrictions will be established for affected individuals. Individual case studies will be performed. Appropriate corrective action will be taken.

Thorium Lung Counting

Individuals shall receive a thorium lung count prior to working in controlled areas, semiannually thereafter, and at the end of employment.

In-vivo "thorium lung count" measurements which exceed 40% of the Maximum Permissible Lung Burden (MPLB) will be reviewed and investigated. Appropriate action will be taken. **For natural thorium the MPLB is 30 milligrams.**

If the result is >100% of the MPLB, work restrictions will be established for affected individuals and individual case studies will be performed. Contamination control and/or air sampling capabilities will be reviewed and corrected, if necessary.

Intermittent workers will be added to a bioassay program appropriate to their exposure potential. The bioassay requirements will be specified on the Radiological Work Permit. Internal monitoring will not be required for visitors.

3.3.3 Environmental Monitoring

An environmental surveillance program has been established and will be maintained to evaluate the effectiveness of the radiological safety program and to provide information facilitating timely corrective action, as needed.

The program includes sampling of (1) air, (2) water (including sewage), and (3) external gamma radiation measurements.

3.3.3.1 Air and Water Sampling and Radiation Measurements

Air - Stack Sampling

Atmospheric radioactive releases are controlled to maintain the annual average activity at the property boundary as far below the Maximum Permissible Concentration (MPC) for unrestricted areas (as defined in 10CFR Part 20) as practicable. Continuous evaluation of air sampling results shall be used to assess the cumulative amount discharged.

Effluents shall be continuously sampled if calculations indicate that concentration levels of SNM at the site boundary (averaged over a calendar quarter) are equal to or greater than 10% of the appropriate concentration listed in 10CFR20, Appendix B, Table II.

An investigation will be conducted and appropriate action will be instituted if stack sample results indicate levels which on a weekly basis exceed 25% of the MPC listed in 10CFR20, Appendix B, Table I.

Air - Environmental

Air will be sampled at no less than 15 locations on, adjacent to, and near the GA site. The sampler filters will be changed and analyzed for long-lived alpha and beta radioactivity on a weekly basis. The samples will also be composited and gamma-scanned monthly. The minimum detectable concentration for 7-day samples shall be no more than 0.025 pCi/m^3.

If the air sampling data indicates levels which on a quarterly basis exceed 25% of the MPC listed in 10CFR20, Appendix B, Table II, an investigation will be conducted and corrective action will be taken as required.

Water Sampling

Influent tap waters will be sampled daily during the normal work week, composited, and analyzed for gross-alpha and gross-beta concentrations on a weekly basis.

Filtered liquids from effluent waste streams from SVA will be collected in holdup tanks/drums and will be sampled and analyzed for radioactivity content. If radioisotopic concentration levels are in accordance with 10CFR20.303 criteria, the hold-up tank/drum contents will be released to the metropolitan sewer system. Contents of the hold-up tank/drum will be sparged for sampling purposes and to facilitate its complete discharge. If the measured levels are above discharge limits, the liquid will receive additional filtration and ion exchange treatment, as needed, until its discharge to the sewer system is permissible. The treatment residue will be solidified (see Section 2.1.10.2) for disposal as radioactive waste. All discharges to the sewer system will be recorded.

Sewage Sampling

Sewage will be continuously sampled, and an aliquot collected on a daily basis during the normal work week will be evaluated for gross-alpha and gross-beta concentrations. Any aliquot indicating levels that exceed 25% of the applicable MPC (10CFR20 Appendix B, Table 1, Column 2) shall be investigated and appropriate corrective action will be taken.

> **Investigation** - When the sample results indicate that the level averaged over one calendar quarter exceeds 50% of the MPC, an investigation shall be made to determine the required corrective action.

External Gamma Measurements

Film or thermoluminescent dosimeters (TLD's) will be used to measure the integrated dose at selected air sampling locations. The dosimeters are changed on a quarterly basis.

3.3.3.2 Location Criteria for Environmental Monitoring

- The current sampling stations were established to accommodate HTGR fuel fabrication activities and will continue to provide adequate coverage during decommissioning operations. No additional stations are anticipated. Each sampling station was established with consideration for:

  a. Type and quantity of material which may be discharged to the environment

  b. Postulated mode of release

c. Characteristics of the local environment, including local population distribution and land use

d. Ability of the station to detect the release of radioactive material from the facility

- Sampling stations which present statistically significant results above the average for the reference site (which is two miles away from the GA site) will remain in service. The deletion of any sampling station will be documented.

- Where construction or other activities require removal of a sampling station, a new station with a new identification will be established as close to the original location as practicable.

- Additional temporary sampling stations may be added where suspect conditions indicate that they may be needed. These new locations may be deleted if the sampling results indicate no statistically significant variation from the original sample stations.

3.3.4 Exposure and Contamination Control

3.3.4.1 Controlled Access

Access to restricted areas shall be controlled for purposes of radiation safety. Posting of areas within a restricted area will comply with 10CFR20.

3.3.4.2 Protective Clothing

Personal protective clothing and equipment will be used to provide protection against airborne contamination, skin absorption, body and/or clothing contamination, and sanitation and impact hazards. This clothing and equipment shall be donned at the access control location. Health Physics shall specify the appropriate protective clothing and equipment for existing circumstances. Selection will be based on the degree of hazard presented by the contaminant(s) and the working conditions under which exposure may occur. Table 3.3-3 lists the protective clothing to be utilized for decommissioning operations.

In situations where the routine use of protective clothing may be questionable, a hazards assessment will be performed to determine the proper selection.

TABLE 3.3-3

PROTECTIVE CLOTHING

	Non-Manual[1] in Shoe Cover Area	Manual in[2] Shoe Cover Area	Manual in Air Hood	Manual in Lunch Room
Lab Coat - Clean (S)				X
Lab Coat (S)	X			
Street Shoes (P)	X			O
Street Clothing (P)	X			O
Shoe Covers (S)	X	X	X	
Safety Glasses (S)	X	X	X	
Safety Shoes (S)		X		
Underwear (S)		X	X	X
Socks (S)		X	X	X
Coveralls (S)		X		
Hot Safety Shoes (S)			X	
2 pr. Coveralls (S)			X	
Hat/Snood (S)				
Hard Hat (S)	X	X		

P - Employee Owned
S - Supplied by Employer
O - Optional

[1] Non-Manual - engineers, supervisors, personnel not directly performing hands-on operations.

[2] Manual - crafts, technicians, laborers working directly with equipment and hands-on operations.

The following will be considered:

- o Care should be taken to ensure that the specified clothing and equipment are compatible with other specified protective equipment (e.g., respirators, etc.)

- o No single combination of protective clothing and equipment is capable of protecting against all hazards. Protective clothing and equipment should be used in conjunction with other protective methods, if possible. The use of protective clothing and equipment can itself create significant physical stress which must be considered in order to select the right combination.

- o Clothing and equipment should be selected to provide an adequate level of protection for specific work assignments. Over-protection, as well as under-protection, can be hazardous and should be avoided. Selection will be based on the following criteria:

  - Permeation, degradation, penetration
  - Heat transfer characteristics/temperature effects
  - Durability
  - Flexibility
  - Ease of decontamination
  - Compatibility with other equipment
  - Duration of use
  - Special conditions (fire, explosion, radiation)
  - clo Value (thermal insulation value)

- o Gloves may be an integral, attached, or a separate item from other protective clothing.

- Safety boots protect feet from contact with chemicals, compression, crushing, or puncture. Shoe covers protect footwear from contamination and chemical damage.

- Safety helmets, hoods, and protective head coverings protect the head from impact, chemical splashes, and entanglement of hair in machinery or equipment.

Workers are responsible for properly wearing the clothing and equipment assigned by Health Physics personnel for specific work operations. The use of protective clothing and equipment will be enforced by management.

Health physics and safety procedures, WAs, and RWPs will list the protective clothing and equipment needed in a restricted work area.

Health Physics is responsible for maintaining a protective clothing and equipment program that is in compliance with 1) U.S. Environmental Protection Agency (EPA), Standard 40CFR191.15, "Individual Protection Requirement," and 2) U.S. Department of Labor, Occupational Health and Safety Administration (OSHA) Standard, 29CFR1910, "Occupational Health and Safety Standards."

The requirements for training are described in Section 2.3. Training shall be given in conjunction with site orientation and the respiratory training program.

3.3.4.3 Ventilation Systems

The SVA ventilation system will be maintained to confine hazardous materials. When work is in progress, room ventilation shall provide at least four air changes per hour.

Monthly checks shall be performed and pressure differentials shall be maintained to ensure that air flow is from zones of lesser contamination potential to zones of greater contamination potential.

Operations involving materials in a form which can become airborne shall be carried out in enclosures or exhaust ventilation systems having a face velocity of 150 linear feet per minute over 90% of the opening. The enclosure or systems shall be equipped with HEPA filters as appropriate to the operation being performed. Work shall be halted if the average face velocity falls below 100 ft/min.

Air flow measurements for hoods, glove boxes, and inert gas boxes shall be conducted monthly.

Room exhausts shall be filtered through HEPA filters. HEPA filter systems shall have the following characteristics:

- o Rated operational efficiency of 99.95% for particles of 0.3 micron size

- o Fire resistance

  - Fire resistant - These filters will be made of fire resistant material capable of withstanding the UL spot test and capable of continuous operation at 150°F, or

  - Fire proof - These filters shall be made of fireproof materials and capable of operation at temperatures above 150°F.

- o Differential pressure indicators or monitors as appropriate

For ventilation systems, GA will perform the following tests:

1) Visual inspection to verify housing integrity shall be performed during changing of the filters.

2) Weekly checks of the Magnehelic gauges to determine if the filters require changing. HEPA filters using pressure indicators or monitors shall be replaced when the pressure drop across the filter reaches 4 inches of water. Prefilters shall be changed when differential pressure indicators indicate a reading greater than 0.9 inches of water.

Every operating stack used in the facility will be monitored. The samples are to be collected and analyzed weekly. Each result above an alert level (>10% of the Maximum Permissible Concentration) will be investigated and corrective action taken.

3.3.4.4 Respiratory Protection

Respiratory protective equipment may be used when the airborne concentration limits of 10CFR20.203(d)(1)(ii) cannot be practically achieved. Use of respiratory protective equipment is conditional upon the following:

- The selected respiratory protective equipment will provide a protection factor greater than the multiple by which peak concentrations of radioactive materials are expected to exceed the values specified in 10CFR20, Appendix B, Table I, Column 1. The equipment is used to ensure that the average concentration of radioactive material inhaled during any period of uninterrupted use in an airborne radioactivity area will not exceed the values specified in 10CFR20, Appendix B, Table I, Column 1.

- o Each respirator user is advised that he/she may leave the area for relief from respirator use in the event of equipment malfunction, physical or psychological discomfort, procedural or communication failure, significant deterioration of operating conditions, or any other condition that might cause reduction in the protection afforded the wearer.

- o A respiratory protection program will be maintained by GA, and will include:

  - Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposure, and to permit proper selection of the respiratory protective equipment

  - Procedures to ensure proper selection, supervision, and adequate training of personnel using such protective equipment

  - Procedures to ensure the proper fitting of respirators and the testing of equipment for operability

  - Procedures for maintenance to ensure full effectiveness of respiratory protective equipment, including issuance, cleaning/decontamination, inspection, repair, and storage

  - Implementation of written procedures for proper control, issuance, use, and return of respiratory protective equipment, including provisions for planned limitations on duration of respirator use.

- Bioassays and other surveys, as appropriate, to evaluate individual exposures and to assess protection actually provided

- Sufficient records to permit periodic evaluation of the adequacy of the respiratory protection program

- Determination that an individual is physically able to perform the work and to use the respiratory protective equipment, made prior to work assignment. A physician will determine what health and physical conditions are pertinent. The medical status of each user is to be reviewed at least annually.

o The SVA Decommissioning Project will use equipment which is listed under appropriate Approval Schedules in 30CFR11 of the U.S. Bureau of Mines-National Institute for Occupational Safety and Health and as set forth in 10CFR20 Appendix A.

o Where equipment of a particular type has not been approved under the schedules in 30CFR11, or where no existing schedule for approval of certain equipment exists, such equipment is not to be used except as authorized by the Commission.

o Unless otherwise authorized by the Commission, GA will not assign protection factors in excess of those specified in 10CFR20 Appendix A when selecting and using respiratory protective equipment.

Employees required to use respiratory protection will receive special instructions and training in the proper use and care of the respiratory protective equipment. Health Physics will ensure the compliance of the Respiratory Protection Program with regulations identified above.

Storage

Nondisposable apparel and equipment will be properly stored to prevent damage or malfunction due to exposure to dust, moisture, sunlight, damaging chemicals, extreme temperature, etc.

Quality Control

Defective protective clothing and equipment will be properly disposed.

3.3.4.5 Radioactive Material Control

An inventory of all radioactive material in SVA shall be maintained by Health Physics and/or Nuclear Material Accountability (NMA). Radioactive material will be stored in designated areas which are designed and posted as radioactive material areas.

3.3.5 Equipment for Radiation and Contamination Surveys and Personnel Monitoring

Instrumentation for detection and measurement of radiation to be provided is listed below:

1. Geiger-Muller (GM) Counters

   - **Six (6)** -- Ludlum Model 3 ratemeters used with a variety of detectors, i.e., HP-210L (a shielded GM pancake counter), Ludlum 44-40 (a tungsten shielded probe), and HP-260 (a pancake GM probe) manufactured by Ludlum Measurements Inc., or equivalent, will be used to survey equipment and other surfaces for contamination. Typical ranges of these instruments are 0-500,000 counts per minute (four ranges).

- o Nine (9) --Model TBM-28 and/or Model TBM-15 geiger counters manufactured by Technical Associates, (or equivalent) will be used for surveying. The instruments contain a pancake Geiger-Mueller (GM) detector which has a window thickness of less than 7 mg/cm^2. The instrument has three ranges covering from 0-50,000 counts per minute (cpm).

2. Alpha Counters

- o Eight (8) -- Model 177 alarm rate meters used with Model 43-5 alpha scintillation [ZnS(Ag)] probe manufactured by Ludlum Measurements Inc., (or equivalent) will be used for personnel monitoring. The instrument has four ranges covering from 0-500,000 counts per minute (cpm).

- o Eleven (11) -- Ludlum Model 12 ratemeter used with a Ludlum Model 43-4 or Model 43-1 probe (alpha scintillation ZnS (Ag) probes). The Model 12 has four counting scales covering from 0-500,000 cpm.

- o Two (2) -- Ludlum Model 21 ratemeter used with Model 43-5 alpha scintillation probe with counting ranges similar to the Model 12 ratemeter.

3. Two (2) Hand Monitors -- Ludlum Model 48 hand monitors will be used for personnel monitoring. The hand monitors have gas flow proportional detectors. They feature a display panel, seven electronic control modules, movable detectors for back of hand contact, fixed detectors for palm of hand contact, a background detector, and input-output flow meters. The detector face are protected by 70% open

stainless steel screen. The detector window is made up of two layers of 0.5 mg/cm^2 aluminized mylar. The alpha meter scale is 0-500 cpm and displays the sum of the back and palm alpha response for each hand. The beta/gamma meter scale is 0-5,000 cpm and displays the sum of the back and palm beta/gamma response for each hand. A hand monitor will be available in both the men's and women's change room.

4. Two (2) Floor Monitors -- Two (2) Ludlum Model 239-1F floor monitors will be used primarily for surveying floors and walls of the facility. The Model 239-1F is a gas proportional floor monitor detector mounted on a roll-around cart. The instrument features a static-flow system, quick-connects, a portable gas bottle and a means to adjust the height of the detector from the floor for optimum performance. The detector is a Model 43-47 large area gas proportional detector used for the detection of alpha, beta or gamma radiation. The probe was designed to be used on the Ludlum Model 239-1F floor monitor. The detector has an active area of 434 cm^2. It is about 30% efficient for alpha, and 10-70% efficient for beta and 1% efficient for gamma radiation. The background count rate is 2.6 counts per minute at 1200 volts and 800-1000 counts per minute at 1700 volts.

5. Gamma Detectors

- **Two (2)** -- Model 19 microR meters with NaI (Tl) scintillation detectors manufactured by Ludlum Measurements Inc. will be used as needed. The instrument has three ranges from 0-5 microR/hr, the lowest range being from 0-25 microR/hr.

- **Two (2)** -- Ludlum Model 16 ratemeter with Model 4-2 High Energy gamma scintillator (1" x 1" NaI (Tl) scintillator for surveying for U-235. The instrument has four scales from 0-500,000 counts per minute.

- Two (2) -- Model RO-2 dose rate ionization chambers type meters manufactured by Eberline Instruments (or equivalent) shall be used to measure dose rates. The instrument detects beta and gamma radiation. The instrumants have several ranges covering from 0-5000 mR/hr.

6. <u>Four (4) Wipe Counters</u>
   - Two (2) Beta wipe counters (Model BC-4) manufactured by Eberline (or equivalent) shall normally be available for use. The instrument uses a pancake GM tube with a 1.75-inch diameter window which is 1.4-2.0 mg/cm^2 thick. (May decrease this to 7 mg/cm^2 by using a 0.002 inch mylar window.) The background is typically less than 50 counts per minute. The gamma sensitivity is typically less than 1200 cpm per mR/hr of Co-60.

7. <u>Fixed and Portable Air Sampling Equipment</u> -- Fixed samplers are located throughout the facility in at least 30 different locations. In addition, at least 10 portable Air Samplers will be available for other locations.

8. <u>Two (2) High Volume Air Samplers</u> -- Two high volume air samplers manufactured by SAIC/RADECO will be used as needed.

9. <u>Four (4) Continuous Air Monitors (CAM's)</u> -- Four CAM's manufactured by Technical Associates (Model FM-5ABN) will be used during the project to monitor airborne concentrations in a general area.

10. Four (4) Lapel Air Samplers -- Four lapel air samplers manufactured by Sensidyne may also be used during the project (as needed).

11. Gamma Spectrometry MCA system using Ge(Li) or a high purity germanium detector manufactured by Canberra Industries, Inc. The lower sensitivity depends on the counting time and type of sample.

12. Low level Alpha/Beta counting systems manufactured by Canberra were used to analyze wipe samples. The lower sensitivity depends on the counting time and type of sample.

13. Alpha spectrometry MCA system -- The lower sensitivity depends on the counting time, type of sample and other factors.

14. Helgeson "do-it-yourself" whole body counter -- measures the total body burden of gamma emitters and also approximates where the emitters are deposited within the body. Both qualitative and quantitative analysis of major radioactive depositions, determination of external contamination and true whole body counting. The whole body counter will be used to determine the thorium lung burden. The minimum sensitivities for thorium in the lungs is 6-8 milligrams depending on the subject's physical parameters.

15. Helgeson mobile in-vivo whole body counter (for U-235 lung counting) -- The minimum sensitivities for U-235 in the lungs is 30-50 micrograms, depending on the subject's physical parameters.

16. Meteorology system -- The meteorology system would be used to determine wind speed, wind direction, etc. during a

radioactive airborne release. The lower sensitivity for wind speed is 0 miles per hour.

Portable survey meters will be used to measure dose rate. Portable instrumentation shall be calibrated after repair and at least semiannually. The calibration will use sources with calibration data traceable to the National Institute of Standards and Technology (NIST). An exception to the above is microR meters, which will be calibrated annually. Effective meter ranges are shown in Table 3.3-4.

A daily calibration check will be conducted with the low-level alpha beta counting systems and the gamma spectroscopy system.

TABLE 3.3-4

PORTABLE METER DOSE RATE RANGES

	Dose Rate Range		
Radiation Type	From	To	Energy
X-ray and gamma	0	5000 r/hr	6 keV to 3 MeV
Beta	0.5 mRad/hr	50 Rads/hr	N/A

3.3.6 Health Physics Records

Health Physics will maintain a "Records and Reports" program which complies with federal, state, and local requirements. These records include external and internal monitoring files, training records, radiological histories of individuals, investigations of radiation incidents, radiological work permits, and work authorizations.

3.3.7 Health Physics Program Inspections and Audits

3.3.7.1 Health Physics and Nuclear Safety Inspections and Management Reviews

Health Physics and Nuclear Safety employees will perform inspections and reviews of decommissioning activities. A Health Physics inspection of the activities shall be made at least quarterly by the Manager, Health Physics or qualified designee. The Manager, Nuclear Safety or designee will perform criticality safety inspections quarterly or as often as deemed necessary by the Manager, Nuclear Safety. These inspections may include the scope of identified discrepancies, unusual events, corrective actions, and review of operations involving any substantially new activity. Inspection results will be reported in writing to project management; to the Manager, Licensing, Safety and Nuclear Compliance; to the Vice President of Human Resources; and to the Chairman of the Criticality and Radiation Safeguards Committee (CRSC). The Manager, Licensing, Safety and Nuclear Compliance (or designee) will assure that necessary corrective actions are taken.

3.3.7.2 CRSC Audits

CRSC will conduct audits as necessary to ensure that work is being done in compliance with approved plans and procedures. Such audits will be reported in writing to the Project Manager, to the Manager of Licensing, Safety and Nuclear Compliance, and to the Vice President of Human Resources. The Manager of Licensing, Safety and Nuclear Compliance or designee will investigate problem areas and implement needed corrective action.

3.3.7.3 BNI Health and Safety Services Reviews

BNI Health and Safety Services (H&S) maintains management oversight of all BNI operations from a health and safety standpoint. H&S will conduct periodic inspections of health and safety program implementation on the SVA Decommissioning Project as discussed below:

- Prior to the start of each new activity, the Bechtel Safety Representative will review with the responsible Field Superintendent, the requirements placed on the workers by the safety program. An example is that prior to erecting a scaffold, the Safety Representative and Field Superintendent will review the manufacturer's instructions, CAL-OSHA Title 8 and the Bechtel Safe Practices rules for scaffold erection.

- The Bechtel Safety Representative and the Bechtel Health and Safety Manager will conduct individual walk-thru inspections daily. Safety problems will be immediately identified to the Field Superintendent for action. Recurring safety problems will be reported to the Site Operations Superintendent for corrective action.

- The Bechtel Safety Representative and Field Superintendent will appoint a Weekly Safety Marshall who will inspect for safety program implementation at least daily. Recurring safety problems will be reported to the Site Operations Superintendent for corrective action.

- A Bechtel Safety Committee, made up of non-manual employees, will hold weekly meetings and discuss safety program implementation and specific actions necessary for the next work week.

- Monthly meetings will also be held with BNI and GA project management and safety representatives to discuss the safety program performance and corrective action taken during the previous month's activities.

3.4 CONTRACTOR PERSONNEL - RADIATION PROTECTION POLICIES

SVA Decommissioning Project personnel and subcontractor personnel will operate within the policies of the GA ALARA and health physics programs. This requirement is designed to ensure that project and subcontract personnel receive the full benefit of the programs described in Sections 3.2 and 3.3. An additional advantage of this approach is that project radiation protection activities and subsequent documentation are administered by a single organization.

GA will provide the BNI Oak Ridge Exposure Records Management Unit (ORER-MU) copies of all worker medical examination records, physician statements, bioassay results, quarterly exposure data and-end-of job exposure data. Reports regarding unusual radiological occurrences (e.g., personnel contamination, airborne contamination uptake) will be forwarded to both the GA and BNI managers of health and safety. Descriptions of actions taken will be included to preclude recurrence of the problem. These actions will ensure that project and subcontract personnel receive adequate radiation protection.

3.5 RADIOACTIVE WASTE MANAGEMENT

3.5.1 Estimate of Radioactive Waste Volume

A major consideration in planning the decommissioning waste handling program is the volume of radioactive waste requiring processing and disposal. Consequently, an accurate radioactive waste volume estimate by waste type and by physical and radiological characteristics is important to the planning activity. This section provides that estimate. A summary of the radioactive waste volume associated with the decommissioning project is provided by the following table.

TABLE 3.5-1

RADIOACTIVE WASTE VOLUME ESTIMATE

Description	Volume (cu.ft.)
Radioactive Material (Total)	86,000
Radioactive Waste (After Processing)	23,000
Packaged Radioactive Waste (Burial)	32,000

These quantities take into account the volume reduction and solidification activities described in Section 2.1.10 of this plan. Waste types and the anticipated volumes of each, radionuclide activity, and package details are discussed in subsequent sections of the plan.

3.5.2 Waste Type

3.5.2.1 Projection of Waste Types - Solids

Planning a waste control program requires knowledge of the types of wastes that will be present, the mass/volume inventory of the waste that exists (and that which will be generated by the decommissioning activities), suitable methods for processing and packaging of both solid and liquid waste materials, and the appropriate disposal methods for each waste type. Waste material will be either of two types: radioactive or nonradioactive. The emphasis in this section is placed on the radioactive materials since they present significant handling, packaging, and shipping problems.

Radioactive waste material results from the deposition of radioactive contaminants in or on the surfaces of facility components, materials, or structures. Examples of facility components and materials that are or may become contaminated include:

- Filter media, such as HEPA filter cartridges
- Ventilation/electrical duct work
- Surfaces of building and structures
- Surfaces of concrete walls or slabs (potentially up to several inches in depth)
- Soils
- Miscellaneous items such as rags, plastic sheeting, and clothing

Operationally accumulated waste and waste generated during decommissioning must be disposed of as part of the decommissioning effort. Operationally accumulated wastes are those that have accumulated during operation or are otherwise present at the time of final shutdown. Wastes generated during decommissioning include those generated during decontamination of equipment and structures, as well as those resulting from the dismantling and cutting of equipment and the demolition of structures.

Decontamination of the floors, walls, and other surfaces within the facility will be accomplished using standard techniques such as washing with solutions and/or physical removal of surface layers. Structural surface decontamination can generate a significant amount of waste that must be processed for disposal. The equipment in the various systems has been contaminated during the operation of the facility. This equipment either will be decontaminated and disposed of as nonradioactive waste or volume-reduced, packaged, and disposed of as radioactive waste. In general, this equipment will include a heat exchanger; filter vessels; tanks; radwaste system components; and various pumps, valves, piping, electrical conduits, and structural materials.

An estimate of the various waste volumes associated with the facility and equipment within the facility is:

- Concrete, Concrete Block (shredded, fluffed) 6000 ft^3
- Stud and Plaster (shredded) 4000 ft^3
- Misc. (tile, floor, wall, glass, etc.) 1500 ft^3
- Ceiling 500 ft^3
- Shredded equipment (50% of equip. vol) 5000 ft^3
- Non-shredded equipment (50% of equip. vol) 5000 ft^3

3.5.2.2 Projection of Waste Types - Liquids

Radioactive liquids will be disposed of as solidified or absorbed wastes. The volume of this waste will be small and is included in Table 3.5-1 under Radioactive Waste (after processing).

3.5.3 Estimate of Nuclide Activity Concentrations

Earlier sections of this plan have identified the radionuclides to be disposed of as U-238 and U-235, and thorium and its daughters. Concentrations of the radionuclides range from background levels of around 1-2 pCi/g for thorium and its daughters, and U-238 and 0.5-1 pCi/g for U-235 to maximum levels of 600 pCi/g for thorium and its daughters, 25 pCi/g for U-238, and 15 pCi/g for U-235. Radionuclide concentrations in the waste are expected to remain within this range. Radioactive waste packages will be surveyed and/or sampled to determine their radionuclide content and total package activities as they are prepared for shipment and disposal. Appendix A provides radionuclide concentration data in more detail.

3.5.4 Waste Packaging

3.5.4.1 Package Types

The selected package types for project radioactive waste shipment and disposal are identiried below.

- o Strong tight metal drums

- o Strong tight metal or wooden boxes

- o Self-contained package, described as a radioactive item whose geometry and materials permit it to be sealed and its exterior surfaces to be decontaminated, so that it forms its own "strong, tight container"

3.5.4.2 Waste Loading Procedures (Packaging)

Radioactive waste generated during the decommissioning process will be packaged for disposal in accordance with packaging criteria and procedures established by GA and the disposal facility. Applicable administrative, engineering, safety and health physics, and operations procedures will be utilized to ensure proper waste packaging.

Nonradioactive waste will be prepared for transport and disposal as required by local and state regulations, as well as any requirements from the receiving landfill.

3.5.4.3 Observation of Waste Loading

To prevent the inclusion of hazardous materials in radioactive waste packages, waste package loading shall be witnessed by a GA employee who is knowledgeable about the past operation of the SVA facility. This observer will be trained to recognize hazardous materials which have been used in the facility (see Section 3.1.2) and which may have been overlooked during SVA decommissioning. This precaution is necessary to prevent the generation of mixed waste.

Observers will document their acceptance of each container to certify that the waste packages do not contain mixed waste (see Section 3.5.6.2).

3.5.5 On-site Waste Storage

Packaged radioactive waste will be staged in designated controlled areas. Early in the decommissioning schedule, these areas will be cleared and modified as required to accommodate such storage. Measures will be implemented in these areas to control the spread of contamination, limit radiation levels, and prevent unauthorized access, unauthorized material removal, tampering, and weather damage.

3.5.6 Waste Shipping

3.5.6.1 Transportation

Radioactive waste will be transported to an authorized radioactive waste disposal site in accordance with DOT (49CFR), NRC (10CFR), and EPA (40CFR) regulations and the requirements of the waste disposal site and its host state. It is anticipated that radioactive waste will be transported by truck in exclusive-use vehicles.

3.5.6.2 Waste Certification

GA is required to certify that waste shipped to a radioactive waste disposal site meets the waste acceptance criteria. The GA Certification Official (or designee) shall affix his/her signature to a certification label on each waste package, thereby certifying that the waste package is in compliance with disposal site requirements. The Certification Official and an alternate will be specified by name to the disposal site.

The Certification Official or his/her alternate shall also sign a statement, to be included with the shipping papers for each waste shipment, certifying that the shipment meets disposal site requirements. This statement also certifies that the nuclide content of each waste container is as listed on the shipping manifest.

The Certification Official will use the documentation generated by the waste package loading observers (see Section 3.5.4.3) and the waste shipment checklists (see Section 3.5.7.2) as the bases for certifying that waste packages and waste shipments meet disposal site requirements.

3.5.6.3 Waste Shipment Documentation

GA will complete the necessary waste shipment documentation for each container on each shipment and will have the responsibility for their completeness and accuracy of the information provided.

The documentation will include:

- GA Shipping Order
- Bill of Lading
- DOE and NRC Nuclear Material Transaction Report
- GA Radioactive Material/Removal Record
- Shipper Instructions to the Carrier
- Other documentation as required by the radioactive disposal site

This documentation will include the nuclide content of each package in the shipment and the total weight and volume of the shipment. The instructions to the carrier will include emergency instructions and instructions for maintenance of the exclusive use status of the vehicles.

3.5.6.4 Waste Shipment Notification

GA will provide advance notification of radioactive waste shipments in accordance with the requirements of the radioactive disposal site to which the shipment will be made.

3.5.6.5 Notification to State Governors

Per 10CFR71.47, advance notification to state governors or their designees is required for a radioactive waste shipment only if a waste package contains more than a Type A quantity of radioactive material or if the fissile material exemption in 10CFR71.53 is exceeded.

3.5.6.6 Health Physics Survey (Waste Shipment)

Health Physics will perform surveys to establish that exposure rates and/or contamination levels for waste packages and transport vehicles are within the limits of 49CFR Sections 173.441 and 173.443, respectively. Health Physics also will complete the Health Physics section of GA's Form-558 "Radioactive Material Removal Record" and will verify that the radiation readings on the packages are consistent with the listed contents.

The Health Physics surveys required for waste shipments will be performed in accordance with existing written and approved GA Health Physics procedures, described briefly below.

3.5.6.6.1 Packages

A Health Physics technician will conduct radiation and contamination surveys of each package scheduled for shipment and will record the results on the package. The maximum radiation and contamination levels shall not exceed the following:

- o 1000 millirem per hour at any point on the external surface of the package

- o

Contaminant	dpm/cm^2	$dpm/100\ cm^2$
Beta/gamma-emitting radionuclides; all radionuclides with half-lives less than 10 days; natural uranium, natural thorium; uranium-235; uranium-238; thorium-232; thorium-228 and thorium-230 when contained in ores or physical concentrates	22	2200

All other alpha-emitting radionuclides	2.2	220

Health Physics will also verify that the indicated radiation level of the package is consistent with the listed radionuclides and quantities. Inconsistencies will be investigated and resolved prior to release of the package for shipment.

3.5.6.6.2 Trailer/Trucks

Before the transport vehicle is loaded, a Health Physics technician will survey the vehicle to ensure that no contamination is present and will record the results in accordance with approved procedures.

After loading the vehicle and completing and verifying the shipping documents, a Health Physics technician will conduct a vehicle radiation survey and complete GA's "Transport Vehicle Survey Form." All vehicles transporting radioactive waste from SVA will be consigned as exclusive-use (sole use) vehicles. Radiation measurements will be entered on GA's Form GA-558 and shall not exceed the following:

- 200 millirem per hour at any point on the external surface of the vehicle

- 10 millirem per hour at any point 2 meters (6.6 feet) from any surface of the vehicle

- 2 millirem per hour in any normally occupied position in the vehicle ("sleeper" portion of the truck cab shall be considered normally occupied).

The transport driver will get final clearance from Health Physics. Checkout includes monitoring the driver, as required, and a wipe survey of the loaded vehicle. The wipe survey will include two wipes on each of the side surfaces, the back door, and the floor inside doors. Results will be documented on the "Transport Vehicle Survey Form." The contamination levels must be at normal background levels. If contamination levels exceed background, the vehicle must be decontaminated.

A checklist will be completed to document that required Health Physics tasks have been performed and that radiation and contamination surveys for each package and for the vehicle meet regulatory requirements.

3.5.7 Quality Control (Radioactive Waste)

In accordance with GA's procedures for waste packaging and transportation, GA Quality Control shall perform and/or witness inspections and measurements on radioactive waste packages and radioactive waste shipments to ensure compliance with DOT (Title 49CFR, Parts 171 through 177), and disposal site requirements.

3.5.7.1 Inspections

GA Quality Control shall perform and document the following inspections, measurements, or verifications in support of radioactive waste disposal activities:

- o Inspect shipping containers prior to loading. The inspection shall verify that the containers are strong and tight (i.e., have no holes or cracks through which material can escape). If a container has been stored outside, the inspection shall verify that the container contains no water.

- o Witness loading of waste containers to preclude inclusion of hazardous materials (to be performed by GA QC or other

GA personnel having plant knowledge in accordance with Section 3.5.4.3).

- o Survey ducts for U-235 content.

- o Inspect all waste packages to verify that the shipping containers have not been damaged during loading and that they have been properly sealed to prevent the escape of any material.

- o Verify that waste packages are marked/labeled in accordance with disposal site requirements.

- o Verify that radiation and contamination surveys of waste packages and transport vehicles are performed in accordance with procedural requirements.

- o Verify that the measured external dose rates and contamination levels are within the applicable limits of 49 CFR Sections 173.441 and 173.443, respectively (see Section 3.5.6.6).

- o Witness loading of waste packages onto the transport vehicle to verify the identity of each waste package against the shipping manifest and to verify that the packages are adequately braced and that the gross weight and U-235 limitations for the shipment are not exceeded.

- o Witness tamper sealing of the transport vehicle and verify vehicle placarding.

3.5.7.2 QA Checklists

A checklist will be used to ensure that packaging and shipment of radioactive waste is accomplished in accordance with approved procedures. The checklist will identify required activities and the organization(s) responsible for each activity. It will include a location for sign-off for each activity by the responsible organization(s). Before a radioactive waste shipment can leave the site, the checklist for that shipment must be reviewed for completeness and signed off by both GA Quality Assurance and the authorized Waste Certification Official or his/her designee (see Section 3.5.6.2).

Figure 3.5-1 shows a typical checklist used by GA for radioactive waste shipments to a commercial radioactive waste disposal site. A similar checklist, tailored to the requirements of the particular disposal site, will be used for SVA Decommissioning Project waste shipments.

3.5.7.3 Records

Records documenting conformance of radioactive waste shipments with DOT and disposal site waste certification requirements shall be specified, prepared, and maintained. These records will include but not be limited to:

- Radiation and contamination survey records
- U-235 measurement records
- Waste Shipment Checklists
- Documentation identified in Section 3.5.6.3

The organization responsible for retention of records and for selecting the method (e.g., hard copy, microfilm, etc.) and duration of record retention will be specified in the GA Quality Assurance Program Document (QAPD) for the SVA Decommissioning Project (see Section 7.2).

SHIPMENT NO__________ SHIPMENT DATE___________

ACTION	HWPF	HP	QC	NMA	DATE COMPLETED	INITIALS
1. DETERMINE CONTAINERS TO BE SHIPPED	•					
2. INSPECTION OF OPEN CONTAINERS			•			
3. CAULK AND SEAL CONTAINERS	•					
4. INSPECTION OF CAULKED AND SEALED CONTAINERS			•			
5. CONTAINERS MARKED AND LABELED	•					
6. INSPECTION OF CONTAINER LABELS			•			
7. COMPLETE MANIFEST	•					
8. CHECK CONTAINER AGAINST MANIFEST	•					
9. SURVEY AND WIPE CONTAINERS		•				
10. MONITOR SURVEY &WIPING OF CONTAINERS			•			
11. RECORD SURVEY RESULTS ON EACH CONTAINER		•				
12. MONITOR RECORDING OF RESULTS			•			
13. VERIFY RESULTS (CONSISTENT WITH LISTED RESULTS)		•				
14. MONITOR VERIFICATION OF RESULTS			•			
15. RECORD RESULTS OF SURVEY FROM CONTAINER TO MANIFEST	•					
16. SURVEYING DOCUMENTS COMPLETE	•					
17. REVIEW OF APPLICABLE DOCUMENTS				•		
18. DOE/NRC FORM 741 & OTHER DOCUMENTS COMPLETE				•		
19. GA-211 SHIPPING CONTROL FORWARDED TO SHIPPING	•					
20. SURVEY EMPTY TRANSPORT VEHICLE		•				
21. MONITOR SURVEY OF EMPTY TRANSPORT VEHICLE			•			
22. LOAD TRANSPORT VEHICLE	•					
23. MONITOR LOADING OF TRANSPORT VEHICLE			•			
24. CHECK CONTAINERS AGAINST THE MANIFEST			•			
25. SURVEY LOADED TRANSPORT VEHICLE		•				
26. MONITOR SURVEY OF LOADED TRANSPORT VEHICLE			•			
27. TAMPER-SAFE SEAL TRANSPORT VEHICLE (IF APPLICABLE)	•					
28. CHECK TAMPER-SAFE SEAL			•			
29. PLACARD TRANSFER VEHICLE	•					
30. CHECK PLACARDING			•			
31. OBTAIN DRIVER SIGNATURE ON APPLICABLE DOCUMENTS	•					
32. APPLICABLE SHIPPING DOCUMENTS TO DRIVER	•					
33. LAND BURIAL SITE NOTIFIED BY TELEPHONE	•					
34. SHIPPING DOCUMENTS MAILED TO LAND BURIAL SITE	•					
35. APPLICABLE SHIPPING DOCUMENTS FORWARDED TO NMA FOR RETENTION	•					

CERTIFICATION OFFICIAL

DATE

QUALITY ASSURANCE

DATE

Figure 3.5-1
RADIOACTIVE WASTE SHIPMENT CHECKLIST

1 24 1996.7

3.5.8 Nonradioactive Waste Disposal

Decommissioning will also involve the handling and disposition of solid and liquid waste materials that are nonradioactive. Nonradioactive solid waste materials are expected to include:

- o System piping and components (pumps, valves, tanks, and heat exchanger)
- o Ductwork and associated equipment
- o Cable trays and associated wiring
- o Electrical equipment (motors, generators, switchboards, and panels)
- o Building and structure materials (concrete, beams, columns, and siding)

The wastes described above include (1) wastes which were never contaminated and, (2) contaminated wastes that have been decontaminated to meet release criteria.

3.5.8.1 Release Criteria

The criteria for release of equipment (e.g., system piping and components, ductwork and associated equipment, motors, generators, etc.) and structural debris (e.g., concrete, beams, columns, siding) to landfill disposal as nonradioactive or industrial waste shall be the same as the criteria specified in Section 4.2.1 for release of facilities and equipment to unrestricted use (see Tables 4.2-1 and 4.2-2).

3.5.8.2 Release Procedure

All equipment and structural debris will be subjected to at least two radiation and contamination surveys. The initial surveys will be performed at the work station if the area is sufficiently clean and free from background radiation, or within a designated area having relatively low background. These surveys will permit designation of items as either radioactive or nonradioactive waste, depending on whether or not they meet the criteria for release as nonradioactive waste. All survey results will be documented on the appropriate Health Physics survey form(s).

Items designated as nonradioactive waste will be moved to an on-site storage area restricted to nonradioactive waste (see Section 3.5.8.3), in accordance with procedures to preclude the possibility of contamination.

Final contamination and radiation surveys will be performed on waste items at the on-site storage area to verify that they meet the criteria for release as nonradioactive waste. The results will be recorded on the appropriate Health Physics survey form(s). These surveys will be witnessed by a GA QA/QC representative who will be responsible for verifying that the survey is performed in accordance with the applicable Health Physics procedure(s) and that the survey results demonstrate compliance with the release criteria. Upon verification of the surveys and survey results, the QA/QC representative will sign the survey form(s).

Upon completion and QA/QC verification of the final surveys, nonradioactive nonstructural wastes may be immediately transported for landfill disposal.

Nonradioactive wastes whose source of origin is the structure of the SVA facility (e.g., concrete floors, walls, etc.) will be disposed of by a different procedure. Following QA/QC verification of the final survey, GA will notify NRC Region V a minimum of 30 days in advance of the scheduled disposal date. The NRC will (1) check/survey the waste within the 30 day period and authorize disposal, (2) waive its verification survey and authorize disposal, or (3) take no action during the period (considered to be disposal authorization). Upon receipt of NRC disposal authorization, the nonradioactive waste will be transported for landfill disposal.

3.5.8.3 On-Site Storage

On-site storage of industrial waste awaiting disposal will be controlled and documented. Section 3.5.8.4 describes the necessary disposal records. Storage locations will be identified, with controlled access to prevent items not inspected and/or not surveyed from being placed in the area. Each waste storage location may be surrounded by a berm or curb to control runoff of rain or spilled liquids. This control ensures environmental protection while the materials await final survey and shipment. Other protective measures may become necessary and will be implemented as appropriate. Because items placed in on-site storage will have been surveyed and documented as nonradioactive per the criteria defined below, no potential for radioactive contamination of the storage area is expected.

3.5.8.4 Records

Records describing items disposed of as nonradioactive waste will be kept. The records will contain item descriptions, survey data sheets, comparisons to release criteria, and copies of release certification as required. This information may consist of one or more forms designed for collecting the required data. The nonradioactive waste disposal record will also contain forms relevant to the final disposition of the waste, (e.g., shipping notice, waybill, and disposal site receipt form).

3.5.8.5 Transportation

Nonradioactive waste will be transported to the landfill disposal site by a commercial waste contractor. The waste transporter will be required to indicate in writing that all nonradioactive wastes shipped from the site have been properly delivered to an authorized disposal facility.

3.6 INDUSTRIAL SAFETY

BNI will be responsible for industrial safety and will implement its industrial safety program. Hazardous Work Permits for decommissioning activities will be issued by BNI as necessary. GA's industrial safety group will oversee the BNI program to verify that it is effective and meets all regulatory requirements. GA's oversite of the BNI industrial safety program will include verification that an evaluation of new equipment and new activities for potential safety hazards has been performed. Concerns to be addressed in the evaluation will include fire protection, industrial safety, as well as provisions for containment and loss of utility services. Typically, this evaluation will be part of the review and approval process for procedures and work authorizations which is conducted when preparing for the use of new equipment or new activities. This control process is detailed in Sections 2.1.11 and 3.2.3 of this plan.

3.6.1 Hazardous Work Permits (HWP)

BNI and subcontractor personnel will review hazards involved in the planned work operation, with specific attention to recent monitoring results, respiratory protection, and specialized safety equipment. Any potential for hazards will be discussed with the work crew before they enter the work area. At a minimum, this will occur during the daily readiness review.

BNI and/or subcontractor personnel will obtain hazardous work permits (HWP) prior to work activities which could result in serious injury, illness, or death. Hazardous work could be a result of the following:

- Chemical exposure
- Oxygen deficiency
- Extreme temperatures
- Excessive noise
- Biological hazards
- Electrical hazards

The HWP will be requested by the Field Superintendent, who will submit the request to the BNI Site Operations Superintendent for approval. HWP approval by the Site Operations Superintendent and the BNI Health and Safety Manager or his designee(s) is required. Approval of an HWP will be contingent on inspection of the work site, proper preparations for the work, results of area hazardous gases sampling and checks, and worker awareness of hazards associated with the work.

3.6.2 Hazardous Materials Exposure

GA, BNI, and subcontractor personnel will adhere to the BNI Hazard Communication Program. The program will ensure that appropriate measures are taken to inform and protect those employees who work with hazardous materials. The Hazard Communication Program will include the following:

- Employee Information - GA, BNI, and subcontractor personnel will be provided with information regarding potential exposure in the workplace. Material safety data sheets (MSDS) will be provided for detailed safety and health information.

- Employee Training - GA, BNI, and subcontractor personnel will be trained to recognize and understand the potential hazards to safety and health associated with the work to be performed.

- Hazardous Material Inventory - An accurate, up-to-date inventory will be maintained for hazardous materials utilized in the performance of the work.

- Warning Labels - Warning labels will meet commercial standards. If warning labels are missing or illegible, personnel will not utilize the material until adequate information is obtained.

3.6.3 Nonradiological Monitoring - Ambient Air, Liquid Waste

Personnel and area monitoring programs will be devised to ensure the identification of areas and work activities for which engineered controls and/or respiratory protection are required. Monitoring will be conducted to confirm that levels of protection, provided by the respiratory protection program and by engineered controls, are adequate to protect the worker, the environment, and the public.

Respiratory protection shall be mandatory for tasks involving potential for airborne exposure(s) (e.g., working in areas containing asbestos, working where dust contamination cannot be controlled, etc.). Direct-reading measurements will be supplemented by analysis of breathing zone samples by a method approved by the National Institute of Occupational Safety and Health (NIOSH).

Monitoring may include assessments of airborne contaminants in work areas and at area boundaries. Swipe and grab samples may be collected to identify contamination on surfaces and equipment. Properly calibrated, and controlled equipment adequate to meet monitoring needs shall be available. Depending on the operation, surveys will be performed as required to determine the following:

- Airborne concentrations of chemical or biological materials
- Combustible/explosive gas/vapor levels
- Toxic gas levels
- Oxygen levels
- Noise levels
- Personnel contamination potential
- Surface contamination in work areas
- Contamination of personnel protective apparel and equipment
- Suitability of equipment and material for release to unrestricted areas

3.7 FIRE SAFETY

Some activities may present potential fire hazards. Flame cutting, welding, and grinding are examples of such activities which are likely to occur on the project. To reduce this potential hazard, a project fire safety program will be implemented.

The project fire safety program will consist of employee awareness, fire prevention, and fire protection. The goal of the program is to reduce the likelihood of fire and to minimize the effect of a fire should one occur. Employees will receive training and instruction in fire prevention and fire protection. The GA General Emergency Plan provides for effective control of emergency situations throughout the GA complex, including SVA. It describes the actions to be taken in the event of fire. Decommissioning project personnel will respond to fire (and to other emergencies) as required in the emergency plan. Program implementation will be the responsibility of both BNI and GA, with specific responsibilities divided between the organizations.

BNI will have primary responsibility for project fire prevention activities, including ensuring employee awareness of fire safety. Fire safety training will be conducted by GA with additional instruction provided by either GA or BNI, as appropriate.

GA will continue to be responsible for the implementation of the General Emergency Plan, control of emergency responses to fire at SVA, operation and maintenance of fire alarm and major fire suppression systems, and coordination of outside support.

3.7.1 Fire Prevention

Fire prevention on the project will include utilization of good housekeeping practices, proper control and storage of flammables, use of cutting and burning permits, the presence of fire watches for hot work, implementation of a weekly project safety marshall/fire warden program, including fire safety instruction as part of the initial training, and weekly safety meting programs. Fire safety inspections will be conducted regularly by both BNI and GA.

Daily fire safety inspections will be conducted by the BNI Site Safety Representative and the BNI Safety Marshall, who is appointed weekly. GA Emergency Services Technicians will inspect fire suppression equipment monthly and conduct general fire safety walkdowns of the facility monthly. They will also inspect areas requiring cutting or burning permits daily before the start of work.

When using an ignition source, e.g., cutting, welding or grinding, a cutting or burning permit is required. The permit will be issued and controlled by GA Emergency Services, in accordance with the National Fire Protection Association (NFPA) Code NFPA-1, Appendix H. Any hot work will require the presence of a dedicated fire watch with ready access to a fire extinguisher. Fire watchers will be trained in fire causes and types, fire extinguisher selection and use, and specific fire watch duties and responsibilities.

3.7.2 Fire Protection

Project personnel will sound the fire alarm by (1) word of mouth to co-workers in the area, (2) activating one of the fire alarm pullboxes located throughout the facility, and/or (3) telephoning Ext. 2000, the GA Emergency Telephone Number. The BNI project Emergency Response Team will provide an initial response to any local fire alarm. That response is limited to an attempt to confine or control the fire until GA Emergency Services personnel and/or off site fire departments arrive. Emergency Response Team Members will be first aid/CPR trained, trained in the use of fire suppression equipment, and at least some member will be trained in SCBA use.

Fire suppression systems available for use on the project include portable equipment, a fire main capable of supplying water to fire hoses, and heat activated sprinklers and a nitrogen blanket system for HEPA filter protection. The sprinklers and the nitrogen system are automatic and were part of the facility fire protection system during operation. The fire extinguishers and fire hoses will be used as appropriate in controlling fire emergency situations. In the event of a fire alarm, GA Emergency Services Technicians will respond with the company fire trucks.

GA Emergency Services will take control of the scene upon their arrival and will determine if outside support is needed. If outside support is required, GA Emergency Services will notify the appropriate outside organization and provide assistance to them as necessary.

3.7.3 Fire Suppression System Changes

Progress on the project may necessitate reconfiguration and even removal of some of the fire suppression equipment. In that event, temporary fire suppression will be made available. These activities will be performed at the latest possible point in the project. Proposed system reconfiguration and/o removal will be reviewed by GA Emergency Services and appropriate outside organizations will be advised as necessary prior to performing the work. Whenever a portion of a fixed fire suppression system must be deactivated, replacement capability will comply with the requirements of the OSHA General Industry Standard for Portable Fire Extinguishers (29 CFR 1910.158), the OSHA Construction Industry Standard, Subpart E on Fire Protection (1926.150) and NFPA Standard No. 10, Standard for Portable Fire Extinguishers.

Isolation of any portion of the SVA North fire suppression system will not impact the fire protection for SVA-South because the systems are designed and installed to be isolable. The fire alarm system can also be secured on a zone-by-zone basis, thereby, not adversely affecting the alarm system capability in the remainder of the facility.

4.0 PLANNED FINAL RADIATION SURVEY AND SITE RELEASE

The final radiation survey (release survey) is planned to provide assurance that the site meets the criteria for release to unrestricted use. The release survey will be conducted by or under the guidance of the GA Health Physics organization. The survey data and results will be used in preparing a final report and a request to the NRC for confirmation surveys in support of an application to the NRC for release of the site to unrestricted use and for amendment of GA's license. Similarly, a copy of the final report will be provided to the State of California with an invitation to participate in the confirmatory survey in support of obtaining State release to unrestricted use. This section of the Decommissioning Plan provides the information necessary to plan and perform the facility release survey. It also describes the content of the proposed final report, which will be submitted when applying for release of the site to unrestricted use and for amendment of GA's NRC and State of California licenses.

4.1 Final Report Content

The final report will contain the following sections:

- Introduction - Scope of the decommissioning effort, involved parties, and a brief description of the decommissioning project

- Site Description - Brief physical description of the site and its contained facilities

- Historical Use of the Facility - Brief description of function of the facility and processes conducted within it

- Project Description - Descriptions of project activities, including methods and technology used, waste disposal sites, waste volumes, waste radioactivity data, personnel exposure data, and project duration

- Criteria for Release to Unrestricted Use

- Measurements for Release to Unrestricted Use - Description of the measurements taken and the measurement equipment utilized, including sensitivities, accessories, and calibration information

- Final Radiation Survey - Description of the final radiological survey methodology and reports of data collected. Resulting analysis and discussion will be included in this section. Surveys for radiation dose rate and contamination will be described

- Summary - Will summarize the findings of final radiation surveys, project costs, and project lessons learned

- Conclusion - Will provide conclusions based on final survey data and analyses

4.2 Criteria for Release for Unrestricted Use

The criteria for release of the SVA equipment, facility, and soil for unrestricted use are based on the following:

- o Meeting the criteria incorporated into GA's license SNM-696 (Annex C) for release of equipment to unrestricted use (Tables 4.2-1 and 4.2-2) and

- o Release of the facility and/or soil for unrestricted use will be determined using criteria derived from a pathway analysis. The analysis will either be based on the mix of radionuclides found at the facility or will be based on a more conservative approach, defined by the most restrictive radionuclide present. The levels determined by this approach, as criteria for facility release, will be such as to ensure that the annual total effective dose equivalent (TEDE) does not exceed 10 mrem/yr, taking into account the technical basis for new criteria presented in NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning, Technical Basis for Translating Contamination Levels to Annual Dose."

4.2.1 Equipment Criteria

Table 4.2-1 is taken from USNRC's "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material," dated July 1984. It has been incorporated into GA's SNM-696 license by License Condition 25. Table 4.2-2 is taken from the guidelines incorporated into GA's State of California Radioactive Material License No. 0145-80, Item 13 (a). These guidelines, "DHS Criteria for Release of Facilities and Equipment to Unrestricted Use," are also known as "DECON-1."

The criteria defined in these tables will be met prior to releasing process and support equipment to unrestricted use.

4.2.2 Facility and Soil Criteria

- Release of the facility and/or soil for unrestricted use will be determined using criteria derived from a pathway analysis. The analysis will either be based on the mix of radionuclides found at the facility or will be based on a more conservative approach, defined by the most restrictive radionuclide present. The levels determined by this approach, as criteria for facility release, will be such as to ensure that the annual total effective dose equivalent does not exceed 10 mrem/yr, taking into account the technical basis for new criteria presented in NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning, Technical Basis for Translating Contamination Levels to Annual Dose."

The site characterization study conducted during 1988 (see Appendix A) included the collection of concrete/soil samples from various areas of the building, namely: the thorium spheres production area (location 1), the area near the VSM-10 and -11 (location 2), the mop water area (locations 6 and 22), and the particle production area (location 8).

Gamma spectroscopy results indicated that soil radionuclide concentrations were at normal background levels. However, in the unlikely event that the soil underneath the building or around underground drain pipes or floor drains is found to be contaminated, the following soil criteria shall be used.

Table 4.2-1
USNRC ACCEPTABLE SURFACE CONTAMINATION LEVELS

Nuclide[a]	(dpm/100 cm^2)[b, f]		
	Average[c]	Maximum[d]	Removable[e]
U-nat, U-235, U-238, and associated decay products	5,000	15,000	1,000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000	15,000	1,000

a) Where surface contamination by both alpha and beta-gamma emitting nuclides exists, the limits established for alpha and beta-gamma emitting nuclides should apply independently.

b) As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

c) Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

d) The maximum contamination level applies to an area of not more than 100 cm^2.

e) The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with the appropriate instrument of known efficiency. When removable contamination of objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

f) The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

Table 4.2-2
STATE OF CALIFORNIA ACCEPTABLE SURFACE CONTAMINATION LEVELS[1, 2]

Nuclide[a]	(dpm/100 cm^2)[b]		
	Average[c]	Maximum[d]	Removable[e]
U-nat, U-235, U-238, and associated decay products	5,000	15,000	1,000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000	15,000	1,000
H-3, C-14 except as DNA.[f]	20,000	60,000	4,000

a) Where surface contamination by both alpha and beta-gamma emitting nuclides exists, the limits established for alpha and beta-gamma emitting nuclides should apply independently.

b) As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

c) Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

d) The maximum contamination level applies to an area of not more than 100 cm^2.

e) The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with the appropriate instrument of known efficiency. When removable contamination of objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

f) DNA precursors mean molecules or compounds that are directly incorporated into the DNA molecule during DNA biosynthesis, e.g., purine and pyrimidine bases and their analogs, nucleotides and nucleosides. The acceptable surface contamination levels for H-3 and C-14 in DNA precursors are as tabulated in paragraph (d) for beta-gamma-emitters.

Footnotes:

1) ANSI, Control of Radioactive Surface Contamination on Materials, Equipment and Facilities to be Released for Uncontrolled Use, final draft, proposed American National Standard N-328, Atomic Industrial Forum, Inc., N.Y. (June1974)

2) NRC Regulatory Guide 1.86 Termination of Operating Licenses for Nuclear Reactors, Washington, D.C. (June 1974)

4.2.3 Demonstration of Compliance with Release Criteria

4.2.3.1 Direct Radiation

Upon completion of any required decontamination, a final survey will be made to verify that all significant contamination has been removed and the area meets the release criteria.

The external radiation levels will be used, as appropriate, in calculating TEDE to ensure that 10 mrem/yr is not exceeded.

If any large (> 30' x 30') diffuse areas are found to exceed the exposure rate criteria, they will be divided into grids about 30' x 30' for surveying purposes.

4.2.3.2 Inhalation Pathway

The inhalation pathway will be included as appropriate, in calculating the TEDE to ensure that 10 mrem/yr is not exceed.

If a mixture of radionuclides exists, the following formula will be applied to show compliance:

the sum of $C_i/L_i \leq 1$, where
C_i = the average concentration of radionuclide i, and
L_i = the derived maximum limit for radionuclide i

To demonstrate compliance with the established dose limit, the following criterion must be met:

Representative soil samples will be collected from large (> 30' x 30') diffuse affected areas, if any, at the intersections of a 30' x 30' grid. Samples will be taken from the first 1 inch (1") of soil and analyzed for the various isotopes. If the concentration of isotopes in a sample exceeds the release criteria, the results of the four adjacent samples from the 30' x 30'grid can be averaged with the initial sample's result. If this average value is below the release criteria, compliance with the release criteria will have been demonstrated.

Subsurface soil samples will also be collected. "Subsurface soil sample" refers to a soil sample taken starting at a depth of 1" and extending to a depth of 6" or 1', depending on the area and estimated depth of contamination. After an affected area has been decontaminated to meet the release criteria for surface soil, compliance with the subsurface requirement will be demonstrated by analyzing a random 10% of the subsurface soil samples from the area meeting the release criteria for surface soil.

If a subsurface random sample exceeds the limit for release, the four adjacent subsurface samples from the 30' x 30' grid will be analyzed and the results averaged. If the average is below the limit, the inhalation dose release criteria will have been met. If the averaged result is above the limit, further decontamination will be conducted.

If discrete areas (less than 30' x 30') are found to be contaminated (e.g., underneath drain pipes), radiation measurements will be taken using a microR meter. Individual soil samples will also be collected in the areas where the highest soil contamination is expected, as well as in random locations. Composite samples will be collected as required.

4.2.3.3 Ingestion Pathway

SVA is located within a semi-arid region zoned for light industry and research and development. The area is not, nor is it likely to be, zoned for residential or agricultural land usage. There is no potable water originating on the site or in its environs. A brackish water table is approximately 275-300 feet deep, at about the same level as the nearby salt water backwater and marshes. Therefore, the pathway for ingestion of any residual contamination via any food or water is highly improbable.

4.3 Measurements for Demonstrating Compliance with Release Criteria

4.3.1 Instrumentation

- Beta/gamma survey meters - Model TBM-28 and Model TBM-15 Geiger counters manufactured by Technical Associates (or equivalent) will be used to survey surfaces for beta/gamma contamination. The instruments contain a pancake Geiger-Mueller (GM) detector with a window thickness of less than 7 mg/cm^2. The instrument has three ranges covering 0-50,000 counts per minute (cpm). The instruments are calibrated semiannually using a Cs-137 standard or a pulser.

- Alpha Survey Meters - Ludlum Model 12 rate meters with Model 43-44 air proportional alpha probes manufactured by Ludlum Measurements, Inc. (or equivalent) will be used to survey for alpha contamination. The instrument has four ranges from 0-500,000 counts per minute. Alpha counters are calibrated semiannually using an alpha source traceable to NIST.

- Gamma Survey Meters - Model 19 microR meters with NaI scintillation detectors manufactured by Ludlum Measurements, Inc. will be used in all locations. The instrument has three ranges covering 0-5 mR/hr. The microR meters are normally returned to the manufacturer for annual recalibration. Gamma spectral analysis on selected samples will be performed using a high purity germanium detector manufactured by Canberra Industries, Inc.

- Low-level alpha/beta counting systems manufactured by Canberra will be used to analyze wipe samples.
- Gamma Spectroscopy - Gamma Spectroscopy will be accomplished using a high purity germanium detector system or a Ge(Li) detector system. The system will be calibrated with sources traceable to NIST for each type of sample analyzed.

The alpha/beta systems and the germanium detector system are calibrated for various types of samples. This calibration is checked on a daily or weekly basis depending on usage.

4.3.2 Measurement Methods for Conducting Surveys

Final survey results will be converted from cpm (counts per minute) for a particular detector to dpm/100 cm^2. Detector "background" cpm are subtracted from measured readings, and resultant values are corrected for efficiency and geometric factors associated with the instrumentation to complete the conversion.

- **Background**

  A "standard" natural background will be determined for each instrument and subtracted from the survey readings. The surface beta/gamma readings will be taken with the TBM Model 28 and Model 15 GM counters, or equivalent survey meter, which have a background concrete reading of about 80 ± 20 cpm.

- **Efficiency**

  National Institute of Standards and Technology (NIST) traceable standards will be used to calibrate instruments for various efficiencies. The standards consist of concrete blocks (one for each radionuclide) having an area of approximately 25 cm x 25 cm spiked with an NIST-traceable standard liquid solution of the specific radionuclide, (Cs-137, 93% enriched uranium, depleted uranium or thorium). Surface measurements of the blocks will be taken with the instruments and efficiency percentages (correction factors) obtained. The readings will be corrected for efficiency by multiplying by the appropriate correction factor.

- **Geometry Correction**

  The acceptable values listed in Tables 4.2-1 and 4.2-2 are in dpm/100 cm^2; therefore, a correction factor will be applied to the measured values to correct for the smaller area of the detector. For example, the Model TBM-28 and Model TBM-15 beta/gamma counters have a detector with a 19.6 cm^2 area. The reading is therefore corrected by multiplying by 5.1 (the ratio of 100 to 19.6). The alpha detectors used for the surveys "see" an area of about 78.7 cm^2, which requires a correction factor of 1.27.

If other instruments are used, the methods outlined above will be applied to obtain dpm/100 cm^2.

4.3.3 Site Survey Grid

Averaging of surface contamination levels over one square meter areas is permitted in determining compliance with release criteria. With that in mind a grid overlay of the site will be developed. The grid will be based on an existing site reference grid, primarily defined by column locations within the SVA Building structure. The existing grid is on ten-foot (3.05 meters) coordinates, designated numerically from south to north and alphabetically from east to west as shown in Figure 4.3-1.

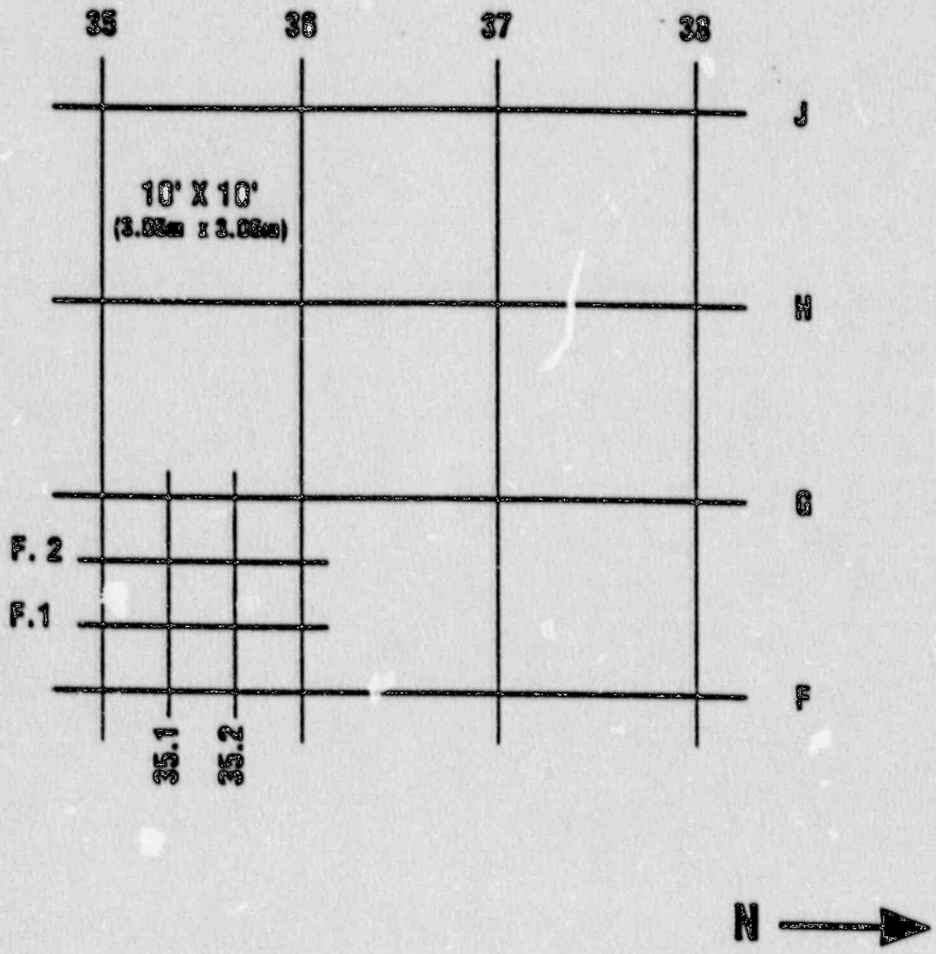

Figure 4.3-1
REPRESENTATIVE SURVEY GRID COORDINATES

This grid will be further subdivided into an approximate one meter by one meter grid layout.

Numbering of the release survey grid will identify the one square meter grid locations within the larger grid. This will be accomplished by numbering of the larger grid number. That is, within grid square 35F the one meter squares will be numbered 35,F; 35.1,F; 35.2,F; 35,F1; and so on. Figure 4.3-1 shows this grid breakdown. As with the site reference grid, all numbers will start in the southeast corner of the grid elements. Significant features within a grid area will be indicated on the grid survey maps used for identifying survey radiation reading and sample locations.

Marking of the grid on surfaces will be made with permanent markers in such a manner that major grid points (i.e., site reference lines) are readily visible to radiological survey crews.

A similar grid arrangement will be used for vertical surfaces. Numbering for these surfaces will start at the lower left corner and be made left to right and bottom to top, with a V designator on the number to indicate vertical surface. For example, the surfaces would be numbered 3V,A; 3.1V,A; 3.2V,A; 3V,A.1; 3V,A.2; etc. Each vertical area will be grid-numbered independent of adjacent surfaces not on the same plane.

Data sheets will identify grid location by elevation for horizontal surfaces and by map direction (i.e., N, S, E or W) for vertical surfaces. A description of grid location will also be supplied, according to the sample identification given in Appendix A, Characterization Report.

4.3.4 Fixed Contamination Survey Protocol

The surfaces of each grid area with a high potential of having radiological contamination will be surveyed for alpha and beta/gamma contaminants over 100% of the area. Grid locations with a low potential of being contaminated may be randomly surveyed. The average and maximum readings (in cpm) will be recorded.

These data will be used for calculating average and maximum contamination levels in dpm/100 cm^2 for comparison to the release criteria.

4.3.5 Removable Contamination Survey Protocol

Removable contamination smear samples will be collected at approximately 10-20% of all grid locations including areas with elevated surface contamination.

The smear samples will be counted in a low-level alpha/beta counting system. The results will be reported in dpm/100 cm^2.

4.3.6 Soil Sampling and Analysis

In addition to the routine soil sampling required by GA's NRC license, any area identified as being contaminated will also be sampled. Soil samples will be taken to haracterize the site as follows:

- **Collection**

    1. Surface samples (0-1" depth) will be collected until approximately 2 kilograms of soil sample are obtained

2. A composite soil sample (1"-6" or 1"-12" depths) will be collected, depending on the area and estimated depth of contamination

3. If subsurface soil contamination is found, additional samples (core samples) will be taken at sufficient depths to provide a reliable estimate of the extent of contamination.

4. The locations where soil samples are taken will be marked and identified on a map.

- **Preparation**

1. The samples will be free of vegetation and rocks.

2. Samples will be dried to remove moisture.

3. The individual samples will be crushed, mixed and homogenized to eliminate clumping.

- **Analysis**

The samples will be analyzed utilizing gamma ray spectroscopy, gross alpha, gross beta, delayed neutron activation analysis, or chemical analyses performed in accordance with written procedures. If results indicate contamination levels above the release criteria, additional decontamination will be conducted until the release criteria are met, or the material will be disposed of at an authorized disposal site.