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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
09/26/2008	00	Approved new document to establish the technical basis for the development of a radiation exposure matrix for General Atomics. Incorporates formal internal and NIOSH review comments. Modified document to include final SEC Evaluation Report language. Section 5.0, Estimating Exposures From Residual Radioactivity, expanded to address dose reconstruction methodologies for exposures received during the residual period. Section 6.0, Estimating Exposure From DOE Remediation, expanded to address dose reconstruction methodologies for exposures received during the remediation period. Includes Attributions and Annotations. Training required: As determined by the Task Manager. Initiated by Bernard M. Olsen.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ANL-E	Argonne National Laboratory-East
AWE	Atomic Weapons Employer
cm	centimeter
Ci	curie
CIRGA	Critical Isotope Reactor, General Atomic
cph	counts per hour
cpm	counts per minute
D&D	decontamination and decommissioning
DAC	derived air concentration
DHHS	U. S Department of Health and Human Services
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
EBOR	Experimental Beryllium Oxide Reactor
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ESTES	Engineering Scale Tritium Extraction System
FFF	Fuel Fabrication Facility
fpm	feet per minute
ft	foot
g	gram
GA	General Atomics
gal	gallon
GM	Geiger-Müller
HCF	Hot Cell Facility
HEPA	high-efficiency particulate air
HEU	highly enriched uranium
HLC	high-level cell
hr	hour
HTGR	high-temperature gas-cooled reactor
ICA	Isotope Committee Approval
ICRP	International Commission on Radiological Protection
IFM	irradiated fuel material
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
IPC	internal proportional counter
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1,000 electron-volts
kg	kilogram
kVp	applied kilovoltage; peak kilovoltage
kW	kilowatt

L	liter
LINAC	linear accelerator
LLC	low-level cell
LLD	lower limit of detection
lpm	liters per minute
m	meter
mA	milliampere
MC	metallurgy cell
mCi	millicurie
MCi	megacurie
MDL	minimum detection level
MeV	megaelectron-volts, 1 million electron-volts
MFP	mixed fission product
MGCR	Marine Gas-Cooled Reactor
mi	mile
min	minute
ml	milliliter
MPC	maximum permissible concentration
mR	milliroentgen
mrem	millirem
NCRP	National Committee on Radiological Protection and Measurements
NERVA	Nuclear Engines for Rocket Vehicle Applications
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
NTA	nuclear track emulsion, type A
OCAS	Office of Compensation Analysis and Support
ORAU	Oak Ridge Associated Universities
PAEC	potential alpha energy concentration
PAEE	potential alpha energy exposure
PIC	personal ionization chamber; pocket ionization chamber
pCi	picocurie
POC	probability of causation
ppm	parts per million
psi	pounds per square inch
QF	quality factor
R	roentgen
RAM	Radiation Area Monitor
RERTR	Reduced-Enrichment Research and Test Reactor
RWP	Radiation Work Permit
SEC	Special Exposure Cohort
SNAP	Space Nuclear Auxiliary Power
SNM	special nuclear material
SRDB Ref ID	Site Research Database Reference Identification (number)
STSF	Sub-Critical Time of Flight Spectrum Facility
SV	Sorrento Valley

TCF	Thermionic Critical Facility
TLD	thermoluminescent dosimeter
TRIGA	Training, Research, Isotopes, General Atomics
U.S.C.	United States Code
USASI	United States of America Standards Institute
WA	Work Authorization
W	watt
WL	working level
WLM	working level month
yr	year
α	alpha
β	beta
γ	gamma
µCi	microcurie
µg	microgram
§	section or sections

1.0 PURPOSE AND SCOPE

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historic background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

In this document the word “facility” is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an “atomic weapons employer facility” or a “Department of Energy [DOE] facility” as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility as “any building, structure, or premise, including the grounds upon which such building, structure, or premise is located … in which operations are, or have been, conducted by, or on behalf of, the Department of Energy (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program)” [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled “Exposure in the Performance of Duty.” That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer “shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation[1]] guidelines established under subsection (c) …” [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation, 42 C.F.R. Pt. 82) define “performance of duty” for DOE employees with a covered cancer or restrict the “duty” to nuclear weapons work (NIOSH 2007a).

The statute also includes a definition of a DOE facility that excludes “buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program” [42 U.S.C. § 7384l(12)]. While this definition excludes Naval Nuclear Propulsion Facilities from being covered under the Act, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled “Exposure in the Performance of Duty”] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally-derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external occupational radiation exposures are considered valid for inclusion in a dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived (NIOSH 2007a):

- Background radiation, including radiation from naturally occurring radon present in conventional structures

- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

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[1] The U.S. Department of Labor (DOL) is ultimately responsible under the EEOICPA for determining the POC.

Under EEOICPA, employment at an AWE facility is categorized as either (1) during the DOE contract period (i.e., when the AWE was processing or producing material that emitted radiation and was used in the production of an atomic weapon), or (2) during the residual contamination period (i.e., periods that NIOSH has determined there is the potential for significant residual contamination after the period in which weapons-related production occurred). For contract period employment, all occupationally derived radiation exposures at covered facilities must be included in dose reconstructions. This includes radiation exposure related to the Naval Nuclear Propulsion Program and any radiation exposure received from the production of commercial radioactive products that were concurrently manufactured by the AWE facility during the covered period. NIOSH does not consider the following exposures to be occupationally derived (NIOSH 2007a):

- Background radiation, including radiation from naturally occurring radon present in conventional structures
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

For employment during the residual contamination period, only the radiation exposures defined in 42 U.S.C. § 7384n(c)(4) [i.e., radiation doses received from DOE-related work] must be included in dose reconstructions. Doses from medical X-rays are not reconstructed during the residual contamination period (NIOSH 2007a). It should be noted that under subparagraph A of 42 U.S.C. § 7384n(c)(4), radiation associated with the Naval Nuclear Propulsion Program is specifically excluded from the employee’s radiation dose. This exclusion only applies to those AWE employees who worked during the residual contamination period. Also, under subparagraph B of 42 U.S.C. § 7384n(c)(4), radiation from a source not covered by subparagraph A that is not distinguishable through reliable documentation from radiation that is covered by subparagraph A is considered part of the employee’s radiation dose. This site profile covers only exposures resulting from nuclear weapons-related work. Exposures resulting from non-weapons-related work, if applicable, will be covered elsewhere.

NIOSH has determined, and the Secretary of the U.S. Department of Health and Human Services (DHHS) has concurred, that it is not feasible to reconstruct internal radiation doses from exposures to thorium for General Atomics (GA) employees during the period from January 1, 1960, through December 31, 1969. As stated in the findings of the Secretary of DHHS, “Although NIOSH is unable to adequately estimate total internal exposures for members of the [SEC] class, internal dose due to intake of uranium can be reconstructed for exposures starting in October 1963, and tritium internal doses can be estimated after September 1965. NIOSH considers the reconstruction of occupational external radiation doses, including medical dose to be feasible for members of the class.” (Leavitt 2007). The Secretary of DHHS has designated a single class of employees from GA for inclusion in the Special Exposure Cohort (SEC) (Leavitt 2007).

The approved SEC class includes all Atomic Weapons Employer (AWE) employees who were monitored or should have been monitored for exposure to ionizing radiation while working at the following GA locations: Science Laboratories A, B, and C (Building 2); Experimental Building (Building 9); Maintenance (Building 10); Service Building (Building 11); Buildings 21 and 22; Hot Cell Facility (HCF) (Building 23); Waste Yard (Buildings 25 and 26); Experimental Area (Buildings 27 and 27-1); LINAC Complex (Building 30); HTGR-TCF or Experimental Critical Facility (Building 31); Fusion Building (Building 33); Fusion Doublet III (Building 34); Sorrento Valley (SV)-A (Building 37); SV-B (Building 39); and SV-D (no building number) for a number of workdays aggregating at least 250 workdays from January 1, 1960 through December 31, 1969, or in combination with workdays within the parameters established for one or more other classes of employees in the SEC. Although the *SEC Petition Evaluation, Report SEC-00064* (NIOSH 2006a) focused on the inability to estimate internal dose from thorium during the period from January 1, 1960, through December 31, 1969, it is not currently feasible to estimate internal doses for other inadequately monitored GA source terms (uranium before October 1963 and tritium before October 1965). Monitored exposures during the residual radiation and remediation periods and bounding doses for the residual and remediation period source terms can be used to determine the dose for GA personnel.

This site profile provides information to reconstruct those doses for which applicable monitoring and source term information is available.

Section 2.0 of this document describes the GA site and its history including information about the radiological processes and source terms as well as the radiological controls and monitoring practices. Sections 3.0 and 4.0 discuss internal and external dose assessment, respectively. Section 5.0 provides information for assigning dose during the residual radiation period beginning in 1970, after the cessation of operations under the U.S. Atomic Energy Commission (AEC), and extending through 1995. Section 6.0 provides information for assigning dose during the DOE remediation period from 1996 through 1999.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 7.0.

2.0 GENERAL SITE DESCRIPTION AND OPERATIONAL HISTORY

General Dynamics Corporation acquired property in La Jolla, California, in 1956 where the John J. Hopkins Laboratory for Pure & Applied Science – later named the General Atomic Division of General Dynamics facility – was built. The information that follows applies to a period of AEC radiological operations at GA in La Jolla, California, from January 1, 1960, to December 31, 1969. This document assumes that the residual contamination period was from January 1, 1970, through December 31, 1995. DOE remediation activities are listed by the DOE Office of Worker Advocacy and are assumed to run from January 1, 1996, through December 31, 1999. Although beryllium exposure is not considered in this part of the program, it is noted that the covered period for beryllium began a year before the period for radiation exposure.

The GA source term includes depleted, normal, and highly enriched uranium (typically greater than 20% and up to 93.5% ^{235}U by mass) in the form of scrap and reactor fuel both in a fresh form and in an irradiated form that includes uranium as well as activation and fission products. Thorium was used in fuel fabrication work. Plutonium work also occurred on the site. Four linear accelerators (LINACs) and several nuclear reactors were also available. Sealed sources were used for instrument calibrations and other work. Byproduct materials (atomic numbers 3 to 83) and tritium were also used.

According to a company pamphlet, General Dynamics Corporation established General Atomic Division on July 18, 1955, in San Diego, California, to develop commercial applications for nuclear technology. GA was also known as John Jay Hopkins Laboratory for Pure and Applied Science; Gulf General Atomic; Gulf General Atomics Company; Gulf Energy & Environmental Systems; and GA Technologies. At one point they were held in partnership between Gulf Oil Corporation and Scallop Nuclear, and the partnership was known as Valley Pin Associates.

2.1 SITE DESCRIPTION

The current GA main site, a 60-acre complex on Torrey Pines Mesa in La Jolla, California (GA 1998), is the facility location that is covered by EEOICPA. The main site is approximately 300 ft above sea level, 1 mi from the Pacific Ocean, and 13 mi northwest of downtown San Diego. The GA site is in the center of Torrey Mesa Science Center, a 304-acre industrial park. The majority of GA buildings are on top of the mesa, but some work [e.g., at the Fuel Fabrication Facility (Building 37)] and the Fusion Energy Facility (also known as DIII-D National Fusion Facility) occurred (and some still occurs) at the base of the mesa to the east of the central set of site buildings. Radiological work at facilities on the mesa included the HCF (Building 23) activities; Training, Research, Isotopes, General Atomics (TRIGA) reactor facilities activities; LINAC work; and neutron studies. In addition, research and development, including metallurgical and chemistry work, was occurring in the Science Laboratory and Experimental Buildings. The Fuel Fabrication Facility and the HCF were both in operation by the end of the 1950s. The HCF included support for high-temperature gas-cooled reactor (HTGR) and TRIGA reactor fuel research and development. Beginning in October 1966, a Plutonium Laboratory that was equipped with gloveboxes and high-efficiency particulate air (HEPA) filtration exhaust systems was activated for the first time in the HCF (Gurren 1966a). General Dynamics/Convair work at the Kearny Mesa Site in San Diego, California, which is not listed as a covered facility by the DOE Office of Worker Advocacy, is not specifically considered in this site profile, but is mentioned because of the possibility that dosimetry records for some workers might include exposures from both the San Diego site and the La Jolla site. Work on the Experimental Beryllium Oxide Reactor (EBOR) project might have occurred both in California and at the Idaho National Laboratory, and dosimetry records from both sites might be needed to reconstruct dose for these workers.

GA possessed and used radioactive materials under several licenses and its radiological programs were regulated by the Federal Government and the State of California.

2.1.1 Facilities

Due to facility changes at GA over approximately 50 years of site operations, Figure 2-1 was considered to be the best figure available to indicate locations of buildings on the site that would have been related to EEOICPA work.

The following is a list of facilities that have been identified from available documents. This list is not considered to be a comprehensive listing of all the GA facilities and their associated radiological operations, but rather provides an overview of facilities and identifies most radiological operations onsite.

Main Campus (10955 John Jay Hopkins Drive)

- Building 1 Administration Building
- Building 2 Science Laboratories A, B and C
- Building 9 Experimental Building (E Building)
- Building 10 Maintenance Building (current Health Physics offices)
- Building 11 Service Building (early Shipping and Receiving)
- Building 13 Technical Office Building (T.O. Bldg)
- Building 14 Technical Office East – Building #1
- Building 15 Technical Office East – Building #2
- Building 21 TRIGA Building (previously contained three TRIGA reactors)
- Building 22 TRIGA Fuel Laboratory Building
- Building 23 HCF (also contained BeO, EBOR, tritium production, and a room where plutonium was handled, which was sometimes referred to as the Plutonium Laboratory)

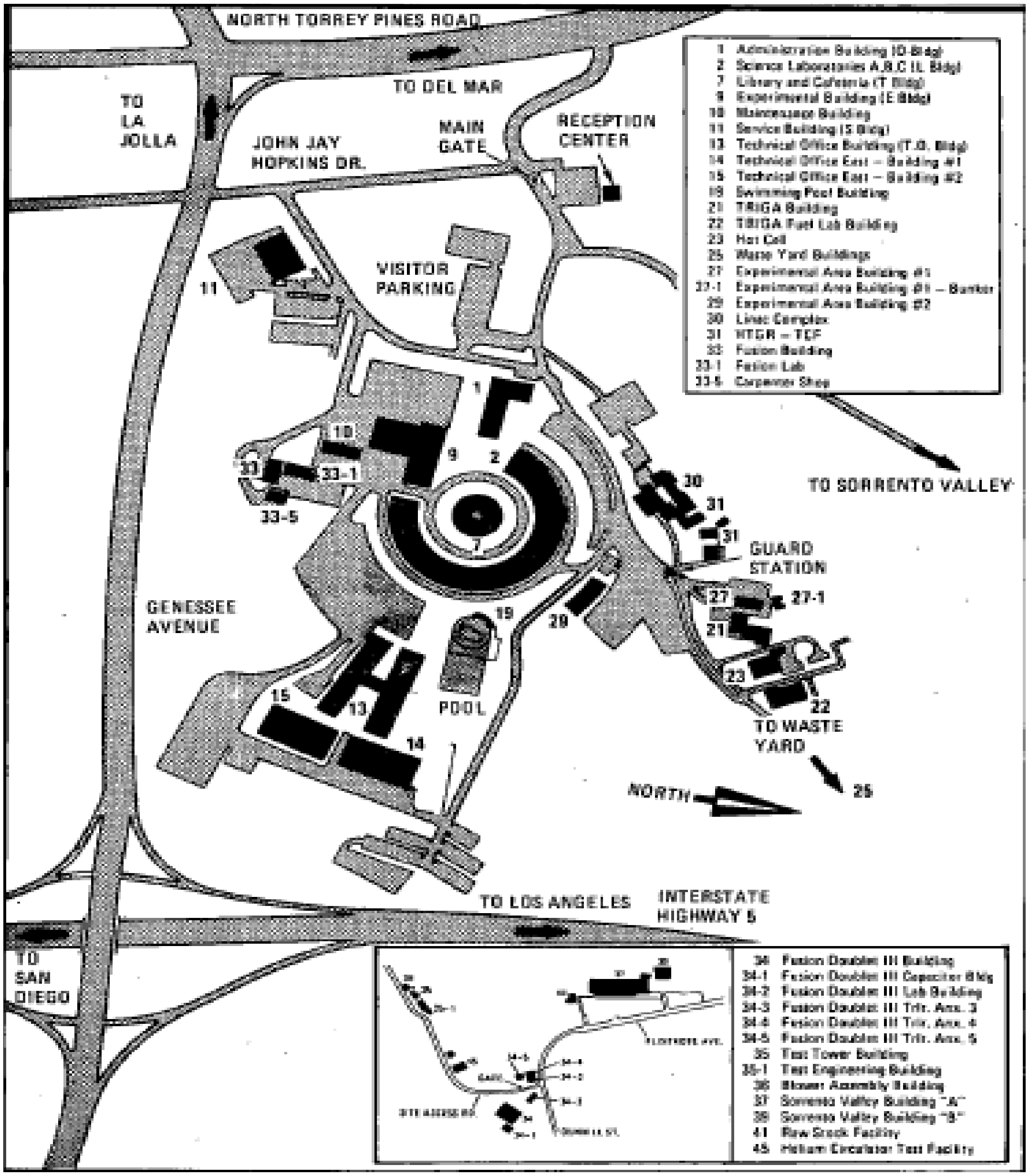

Figure 2-1. Site map, 1977 (GA 1978, p. 61).

- Building 25 Waste Yard/Incinerator
- Building 27 Experimental Area Building #1
- Building 27-1 Experimental Area Building #1 – Bunker
- Building 29 Experimental Area Building #2
- Building 30 LINAC Complex
- Building 31 HTGR-TCF
- Building 33 Fusion Building

- Building 34 Fusion Doublet III Building

Facilities Off the Main Campus

- Building 37 Sorrento Valley Building "A" (SV-A, 11220 Flintkote Avenue)
- Building 39 Sorrento Valley Building "B" (SV-B, 11220 Flintkote Avenue)
- Building (number unknown) Sorrento Valley Building "D" (SV-D, 11388 Sorrento Valley Road)

Administration Building 1

This is an office building. No records of radiation, radioactivity, or radioactive contamination in this building were found.

Building 2, Science Laboratories A, B, and C (SNM-69 1963, p. 17)

These two-level buildings (in addition to Laboratory B) are attached end-to-end and make up the outer annular ring that surrounds the central library and cafeteria. (From an aerial view, these buildings distinctly appear as the outer ring of a bull's-eye.) The Laboratory buildings were approximately the same size and approximately equal to one-third of the entire structure. They were arranged in a counter-clockwise direction from A to C to B beginning with the southernmost section. These buildings contained offices, scientific and electronic laboratories, shops, and low-level hot cells. The majority of the activities of the GA research in metallurgy, chemistry, and experimental physics have been conducted in these buildings.

Building 9, Experimental Building (E Building) (SNM-69 1963, pp. 18 and 194)

The Experimental Building was involved in much of the initial fuel development work that was conducted on the site. The building housed a drafting room, engineering and metallurgical development facilities, machine shops, a special materials machining room, and a hydriding room. The major early operation in the facility was the fabrication of TRIGA reactor fuel elements (UZrH, a uranium-zirconium hydride alloy). There were two radiologically controlled areas in the structure that were designated as Zones A (South Control Zone: A-1, A-2, and A-3) and B (Hot Suite: B-1, B-2, and B-3).

All operations in which generation of particulate matter was possible were conducted in hoods or gloveboxes, or were serviced by individual elephant trunks from an exhaust system. Because GA had contracts with commercial and government clients, a major aspect of working in these areas was material control to minimize the chance of commingling different enrichments of special nuclear material (SNM). Procedural and administrative controls were in place to minimize this occurrence and these controls typically aided the health physics group in relation to personnel protection. There were two SNM storage vaults adjacent to the east wall of the facility and a Security storage area that was designed to store materials in Zone A for short periods.

At a later date, a plutonium storage vault was located in the Experimental Building (SNM-69, Amendments 1-41, p. 156). It appears that the material was designated as being "C.4, encapsulated," on the license. It is not clear when this SNM vault came into existence or what the material was ultimately used for, although it appears that work with thermionic cells was conducted at some time in the building. It appears the vault was used to store encapsulated plutonium used in the Space Nuclear Auxiliary Power (SNAP) 15A program, because the SNAP Plutonium Handling Room had been relocated from the Sorrento Valley B (SV-B) Building in October 1965 (Bold 1965a).

There were many operations that took place in this facility. Uranium stock material had to be sized to less than 0.125-in. cubes. The uranium was cold-rolled into a sheet, annealed, sheared to size in a ventilated hood, and then sent to a storage vault. The shaped product fuel (including uranium, zirconium, and other additives) was either formed from the uranium-zirconium mixture being directly melted and cast in an induction furnace or the mixture was compacted (pressed), arc-melted, cast into 4-in.-diameter cylinders again using arc melting, machined, canned in copper, extruded, pickled to remove the outer copper layer, and then cut to length. Each of these operations took place in the Experimental Building.

Building 10, Maintenance Building
There was not much information found during data capture activities that described the operations that took place in this building. It is presumed that due to its location in proximity to Building 9 (the Experimental Building) that it primarily supported activities in that structure. It currently houses the site's Health Physics professional staff. A radioanalytical laboratory toward the center of the building analyzes operational, environmental, and remediation samples.

Building 11, Service Building (early Shipping and Receiving) (SNM-69 1963, p. 20)
The building no longer exists, but it originally housed early administrative functions such as purchasing, employment, and shipping and receiving for the entire site. Shipments were received, surveyed, and transported from the facility to other buildings on the site for storage or use. In August 1961, storage of radioactive items among surplus items began (Bethard 1961a). The New Delhi TRIGA reactor was stored here for a period. A control area was set up to segregate this material from other nonradioactive material.

Building 13, Technical Office Building (T.O. Building)
This is an office building. Documentation was found that indicated that a portion of the New Delhi TRIGA reactor was stored in the basement of this facility and later relocated to Building 11. No other records of radiation, radioactivity, or radioactive contamination in this building were found.

Building 14, Technical Office East, Building #1
This is an office building. No records of radiation, radioactivity, or radioactive contamination in this building were found.

Building 15, Technical Office East, Building #2
This is an office building. No records of radiation, radioactivity, or radioactive contamination in this building were found.

Building 21, TRIGA Building (previously contained three TRIGA reactors)
This building is north of the Science Laboratories and contained three TRIGA reactors. Three reactor licenses were found for the site (R-38, -67, and -100). Licenses R-38 and -67 mention two TRIGA reactors (Mark I and F), which were shown on figures as located at the site. These reactors came on line in 1957 and 1960, respectively. The Mark F reactor was sometimes designated as the Flash Irradiation Reactor (FLAIR). An additional reactor, which came online in the mid-1960s and was designated as Mark III, was noted in several documents, but no figure (other than a very rough sketch, SNM-69 1963, p. 114) has been found to identify its location. These pool-type reactors were designed to be inherently safe in that the fuel design eliminated the potential for a reactor to go supercritical. The reactor facility was used to test the various designs of the GA-designed reactors.

Building 22, TRIGA Fuel Fabrication Facility (SNM-69 1963, p. 227)
Amendment 61 (January 9, 1975) of SNM-696 identified the TRIGA Fuel Laboratory Building as being in the TRIGA Fuel Fabrication Facility. This building is east of the HCF (Building 23). It is constructed primarily of 7-in.-thick prestressed or reinforced concrete. (In addition to the Fuel Laboratory, the building also contains several nuclear material storage areas or vaults.) The Fuel Laboratory was divided into six separate zones: metal preparation, furnace room, hydride room, machine shop, assembly and inspection room, office, and locker room. All building operations that generated airborne radioactivity were conducted in closed equipment, hoods, or gloveboxes, or the radioactivity was exhausted from the area via elephant trunks.

Building 23, HCF (also contained BeO, EBOR, tritium production, and a plutonium handling room) (GA 1998, p. 18)

The HCF was constructed in 1959 and was operated for over 30 years. It was utilized primarily to facilitate remote inspection of irradiated, highly radioactive HTGR and TRIGA fuel as well as thermionic fuel elements for space power applications (Aycock 1995). Some of the very earliest examinations involved UO_2-BeO fuel for the Marine Gas-Cooled Reactor (MGCR), which later became known as EBOR. Figure 2-2 shows the layout. Approximately 12% of the building footprint was taken up by the hot cell (SNM-69 1963, p. 31). The hot cell was made up of three sections: the high-level cell (HLC), low-level cell (LLC), and metallurgy cell (MC). The HLC was used as a remotely operated hot machine shop to cut open and section irradiated fuel samples for inspection. The MC was used for analytical inspection and research activities; including the preparation of samples, which entailed grinding, polishing, and mounting. The LLC was used as an interface between the other hot cells and other areas of the building, and housed all shipping cask off-loading and packaging operations. Thick (up to 42-in.) high-density concrete walls were in place to minimize personnel exposure to penetrating radiation.

In addition to activities that took place in the hot cells, there were four additional, notable operations that took place in the building: (1) development, fabrication, and inspection of UO_2-BeO fuel materials in support of the EBOR fuel project (Rooms 108/108A) from 1960 to 1972; (2) tritium extraction laboratory for preparation and testing of irradiated fuel samples in Room 109 from 1969 to 1987; (3) an Engineering Scale Tritium Extraction System (ESTES) that was used to demonstrate the ability to extract tritium from New Production Reactor target compacts (Rooms 109, 119, and 120) from 1987 to 1991; and (4) plutonium transport experiments using PuO_2 (Room 116). The two rooms added to support EBOR were included late in 1959 during the final phase of construction to accommodate the Fuel Fabrication Facility (Aycock 1995).

The airflow throughout the facility was directed from clean areas towards areas with high potential for contamination. A continuous air monitor was located in the operation gallery (workspace immediately adjacent to the hot cells) to signal elevated air concentrations, and continuous health physics coverage was provided for the facility.

Building(s) 25, Waste Yard/Incinerator

The waste processing facility includes several buildings that are associated with waste handling, minimization, processing, and storage; SNM and byproduct material storage; and a gamma counting facility. Liquid wastes that were generated on the GA site were treated by placing them in evaporation pools. A low-level radioactive waste incinerator was also used to burn unirradiated industrial trash containing ^{235}U. Typical materials included paper, plastic shoe covers, gloves, mops, plastic bags, etc. from GA laboratories and fuel fabrication facilities. Only authorized personnel were allowed into the fenced-off incinerator area, and they were required to wear a film badge and self-reading dosimeter and to submit bioassay samples for analysis (SNM-69 1963, p. 325).

Several locations were used in the waste processing facility area to store radioactive materials including TRIGA hot storage, a temporary storage yard, the Butler Building, and the By-Products Vault.

A gamma counting facility is also located in the controlled area for the purposes of confirming the amount of ^{235}U present in containers before receipt by the waste processing facility.

Building 27, Experimental Area Building #1 (EA 1 Laboratories)

EA-1 was constructed between September 15, 1964, and April 7, 1965, on a site adjacent to and immediately west of the TRIGA Reactor Facility. (This facility was constructed to be an Annex to the TRIGA Reactor building, Building 21 [1]). The building was constructed of concrete masonry material with exterior dimensions of 41.3 ft by 112 ft with interior floor space of approximately 4,400 ft^2. There

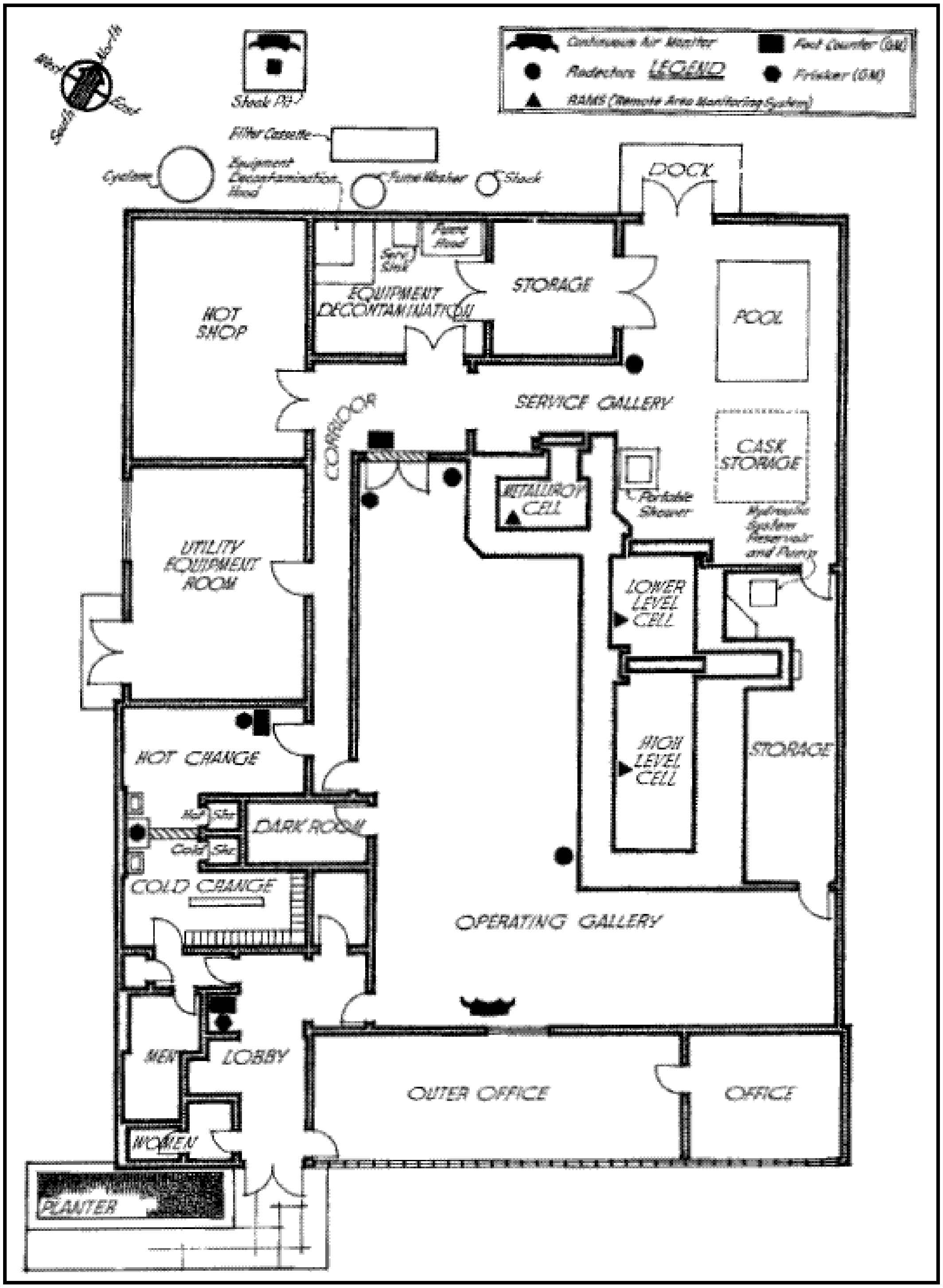

Figure 2-2. HCF floor plan about 1960 (GDC and H&N 1960, p. 89).

were no interior load-bearing walls, so the facility could quickly and easily be modified (Hill 1997). The GA Neutron Activation organization occupied the building immediately after construction. This group established offices in the south end and northeast corner, and set up radiochemistry laboratories in the rest of the building. The laboratories were used primarily to prepare target samples for neutron irradiation and to perform postirradiation radiochemical separations. The GA Analytical Chemistry and Radiochemistry groups were combined into the GA Chemistry group, which then occupied the whole building except the laboratory in the northwest corner. Operation of the GA

Chemistry organization ended in 1996, and administrative control of the facility was assigned to the GA Decontamination and Decommissioning (D&D) organization (Hill 1997).

Building 27-1, Experimental Area Building #1 – Bunker
Construction details were not obtained from document capture, but rather from a discussion with a site radiochemist who was and is still employed at the GA site [1]. The facility was constructed in the same period as Building 21 and is immediately north of that building. Much of the structure was built into the mesa. During facility remediation, the southernmost walls of the structure were left in place to support the face of the mesa [2]. The Neutron Activation organization used the facility to conduct activities that did not generate much long-lived contamination. Some postirradiation examinations of Peach Bottom fuel elements were conducted there [1]. Later a three-unit hot cell facility was set up in the building to “milk” ^{90}Y from large 50-Ci quantities of ^{90}Sr. The facility was called the Strontium Facility.

Reports mention the ^{90}Y production facility, molybdenum facility, and chemistry laboratories. Monthly reports in the mid-1960s discuss the use of titanium tritide associated with the TRIGA reactor building. It appears that targets were irradiated in the TRIGA reactor and then the Activation Analysis group worked with the targets in the Experimental Area Building #1 – Bunker before they moved to their new location. Thermionic generators were also in these areas where the Analytical Analysis was conducted.

Building 29, Experimental Area Building #2
Phone conversation with the Lead GA D&D technician indicated that this facility was and is an office building and that radioactive material was never handled or processed in the facility [2]. Another conversation with an individual in the Site Radiochemistry group indicated that the building originally housed the medical offices and that Frank Bold, one of the early Health Physics Department managers, had an office there [1]. At present the facility is occupied by a company that conducts classified work.

Building 30, LINAC Complex (SNM-69 1963, p. 18)
The LINAC complex includes both a 17-MeV and a 45-MeV electron accelerator that can be coupled to provide even greater acceleration. This facility has been in continuous operation since September 1958. For the majority of accelerator experiments, the LINAC was not used in conjunction with the critical assemblies. The console and administrative office areas were separated from the experimental area. There was a large concrete enclosure called “the cave” in the center of the experimental area that was a highly shielded area for neutron-thermalization studies.

Building 31, HTGR-TCF (SNM-69 1963, p. 332)
This building complex appears to have been a critical test facility for several programs including the HTGR and Thermionic Critical Facility (TCF), as well as being called the Experimental Critical Facility. Information about the activities that took place in the Experimental Critical Assembly Cell indicates that many different critical assembly tests were conducted in this facility.

Building 33, Fusion Building
Divided into a machine building and a setup laboratory annex, the Fusion Buildings were designed for large-scale experimental apparatus for the study of controlled thermonuclear reactions. Experiments were conducted here with high-temperature plasmas. Activities in this facility preceded the activities in Building 34.

Building 34, Fusion Doublet III Building
This facility is in Sorrento Valley at the termination of Tower Road, approximately 1,000 ft south of Building SV-A (below). It conducts research in magnetic fusion technologies. The operations

generate some intense but short-lived radiation sources. Administrative and procedural controls are in place to minimize personnel exposure.

Building 37 or SV-A, Component and Fuel Manufacturing Building (SNM-696 multiple dates, p. 11)
The Component and Fuel Manufacturing Building, SV-A, was approximately 1 mi north of the GA Main Campus facilities, off the Torrey Pines Mesa in the Sorrento Valley at an elevation of 40 ft above the floor of the valley. The building was in an area of approximately 60 acres. The area was physically separate from the Main Campus, and transportation of radioactive materials between the two GA sites had to take place over U.S. Department of Transportation-regulated roads. The facility was approximately 400 ft long by 120 ft wide. The front (east) portion of the building had two floors that included offices, change rooms, laboratories, storerooms, and a cafeteria. A metal machine shop, sheet metal shop, and assembly area for mechanical parts also were located in the building. The northern portion of the building (about 160 ft between columns 25 and 41) was devoted to fuel fabrication activities. This operational area was physically separated from all other areas in the building by either a sheet metal wall (to the south) or a masonry wall (to the east). At some time after original building construction (and original license submittal), there was an addition to the north end of the facility that was “created primarily for storage.”

The use of SNM in the building was primarily for the fabrication of HTGR fuel materials and elements and took place in the 160-ft northern section of the facility. A letter and number designation was used to identify each station in the fuel fabrication area of the building, and each station is further defined in SNM-696 (multiple dates). The processes were identified in the following manner:

- Fabrication of pyrolytic carbon-coated uranium oxide or carbide and uranium-thorium oxide or carbide particles at the process stations that were designated by the letter A.
- Fabrication of compacts that consisted of coated particles in combination with other materials. These process stations were designated by the letter B.
- Assembly of compacts and graphite components into finished fuel elements took place at process stations that were designated by the letter C.
- Fabrication of compacts composed of uranium oxide (up to fully enriched ^{233}U, or up to fully enriched ^{235}U with maximum enrichment in ^{236}U) and graphite and further into finished fuel elements. [No letter designation was given for these operations.]
- Fabrication of fuel containing a minimum ratio of 3.60 thorium-to-uranium at process stations that were designated by the letter E.
- Fabrication of fuel elements that contained fuel rods that were composed of coated uranium-thorium oxide or carbide particles in combination with other materials took place at process stations that were designated by the letter F.

Radiation protection surveys included 35 air sample locations that were in place in the facility that were supported by a house vacuum system. Each air sampling location had a dry gas meter that measured total sampled air and used 2-in.-diameter filter paper (not defined). Air samples were collected for 8 hr on each shift in all areas where SNM was handled in a form that could become airborne. The samples were counted initially after a 4-hr decay and again after 3 days in the Health Physics Counting Room and analyzed for both gross alpha and beta activity. Site personnel who regularly worked in the fuel fabrication areas were provided with dosimetry that measured beta, gamma, thermal, and fast neutrons. In addition, area dosimetry was in place throughout both the operating and office areas in the building. All operating personnel were on a routine bioassay monitoring program. Both direct and indirect radiation surveys were normally conducted on a daily basis. Protective clothing was company-supplied and required to be worn.

Building 39 or SV-B, Pilot Plant Building or HTGR Fuel Production, Research, and Development Building (SNM-696, p. 83)
The HTGR Fuel Production, Research, and Development Building, SV-B, was immediately northwest of SV-A. Its main purpose was process development and pilot-scale operations in relation to fuel production for HTGRs. Process development operations involved bench and pilot-scale operations, which typically involved fuel particle kernel manufacture, application of coating onto kernels to form coated fuel particles, forming carbonaceous matrix fuel bodies that contained the coated fuel particles, and fuel element assembly. The kernel formation included work with both uranium- and thorium-bearing solutions. The typical kernel forming process took place by flowing nitrate solutions through an orifice to make spherical droplets. These droplets were gelled by contact with an ammonia solution and subsequently dried. Calcining and sintering furnaces were used to form high-density oxide microspheres and these particles were then typically carbon-coated in fluidized bed reactors. Various metering and blending techniques were developed and tested for optimal fuel body performance in an HTGR. The facility conducted process development and refinement of fuel element assembly operations in which the fuel bodies were loaded into graphite fuel elements.

Each handling location in the facility was limited to a maximum of 350 g of ^{235}U. These locations consisted of fume hoods, gloveboxes, workbenches, and other types of processing equipment. Ventilation (using either fixed or flexible ducting) was provided wherever airborne contamination might have occurred. Work with powdered SNM was conducted in fully contained glovebox enclosures. Radiation protection surveys in the facility were conducted at least weekly. The action level for wipe survey results was compared to a limit of 6 dpm/100 cm^2 for controlled areas and, in the event that loose contamination was detected exceeding this level contamination control, boundaries were set up. Uncontrolled areas in the building were maintained below this level.

SV-D, Special Nuclear Systems (SNM-69 1963, p. 130)
This concrete-reinforced structure (floor, walls, and roof are concrete) was approximately 3 mi northeast of the main GA campus and assembled thermionic and thermoelectric fuel elements. A concrete block firewall was in place to physically separate the two operations. The thermionic operations in the facility had a site possession limit of 2.5 kg of ^{235}U that was received and handled in the form of dense high-fired pellets of UO_2. It is not clear from information provided in the site license (SNM-696), but encapsulated plutonium (PuN, PuO_2, etc.) in a nuclear battery assembly was also handled in this section of the building. Radiation protection surveys were conducted throughout the facility, including wipe surveys (at least monthly in all areas) and air sampling (of several areas including welding operations that plutonium and before continuing any other operations in which positive contamination was detected). Thermoelectric generators were assembled in another area of the building.

The SV-D facility possession limit, as of September 15, 1972, was 300 g of plutonium. Laboratories [e.g., Mound Laboratory or Harwell of the United Kingdom Atomic Energy Agency] supplied the double-walled, encapsulated-plutonium-containing fuel capsules in shipping containers. Capsules contained 0.18 g of ^{238}Pu and 20% ^{239}Pu (no amounts were specified for $^{236/240/241/242}$Pu, but lesser amounts of these isotopes would be expected). Receipt surveys of the entire outer surface were conducted and compared to an action level of 6 pCi (~10 dpm). Once the package was received, it was moved to the Plutonium Handling Room, placed in a glovebox where it was unpacked, and additional surveys with portable alpha scintillators and wipes for alpha activity were conducted. Fuel capsules were handled and generators were assembled in a sealed glovebox. When not being handled, the fuel capsules were stored in a safe in the Plutonium Handling Room. Assembled generators and shipping containers were checked by a Health Physics surveyor before their removal from the glovebox. The radioactive material was encapsulated in the generator and might have been transported to other locations in SV-D or shipped to other facilities at the GA site for quality assurance examinations.

General Dynamics/Convair (Kearny Mesa Site, San Diego, California) (SNM-69 1963, p. 170)
This site is not part of the covered facility. It conducted early electron-beam welding services for ^{235}U-fueled thermionic generators that appear to have been manufactured in the Experimental Building on the main campus site (SNM-69 1963, p. 173). The generator was required to have an insulator that was welded into place, and apparently GA did not have the facilities at one time to perform this work. The units were inspected (e.g., spectrometric leak testing, radiographs, as well as other inspections that are not specifically described) on the GA main campus before shipment to the facility. If any defects were noted, the affected assembly was rejected and not shipped.

The uranium fuel, in solid form, was encapsulated in tungsten containment. A GA health physicist was with the generator at all times during the welding activities. Air sampling and a portable alpha survey meter were used at all locations where the generator was handled or tested, and the health physicist routinely performed alpha wipe surveys (SNM-69 1963, p. 189). The welding at the Kearny Mesa site did not take place on the tungsten itself, but rather on a tantalum transition flange brazed to the tungsten. This ensured the integrity of the tungsten container would not be breached. The welded assembly was immediately returned by a GA representative back to the Main Campus.

Green Farm Experimental Test Site
This site is not part of the covered facility. The Green Farm Experimental Test Site was a 500-acre, U.S. Air Force explosive-firing facility that GA operated approximately 10 mi east of the main campus site near the Air Force Sycamore Canyon Missile test site (GA undated). The facility was originally set up for Project ORION. The facility had offices, shops, laboratories, explosive magazines, and firing bunkers. There was also a Flash X-Ray Facility at the Green Farm site to test both high-explosive and radiation effects experiments. The instrument operated at a nominal operating voltage of 2 MeV, and delivered 6,000-R X-ray doses in 30 nanoseconds at the tube face. A smaller flash X-ray machine was installed “in a special shield room” at the Torrey Pines site. This machine was used for performing experiments at low dose rates and for laboratory checkout of the instrumentation and experiments before irradiation at the LINAC

2.2 PROCESS DESCRIPTIONS

GA conducted irradiation and time-of-flight studies with four LINACs, developed and fabricated reactor fuels, operated three TRIGA reactors on the site, performed fusion research since the mid-1960s, ran experimental criticality test facilities, used an experimental facility that is understood to have conducted a diverse set of tests with radioactive materials, and worked with SNM and radioactive tracers in its laboratory facilities. Hot cells were used for handling radioactive materials that were associated with high radiation fields or very large quantities of radioactive materials that might be subject to processes that would disperse the material.

AEC Contract Number AT(40-1)-3367 (AEC 1965), was for reclamation of highly enriched uranium (HEU) from unirradiated Nuclear Engines for Rocket Vehicle Applications (NERVA) scrap fuel. The scrap was identified in the contract as being in scrap groups III (uranium compounds such as oxide) and IV (combustibles such as paper, tissue wipes, filters, etc.). According to the terms of the contract, the scrap materials were to be shipped from the Westinghouse Electric Corporation’s Astrofuel Facility in Cheswick, Pennsylvania, to GA for processing into uranium oxides (Type III recovered uranium; UO_2, UO_3, and U_3O_8). This reclamation was to take place “no later than 150 days after delivery of the scrap” to GA. According to the “Special Conditions of the Contract,” the recovered HEU oxides were to be shipped to Portsmouth Gaseous Diffusion Plant. The individual who managed the GA Fuel Operations Division during the 1960s and through the 1980s (and currently works part time at GA) was interviewed and indicated that this HEU processing was one of several similar contracts to process NERVA/Rover process scraps that were conducted through approximately 1972. The scrap recovery operations took place in the GA Fuel Fabrication Facility, Building 37, sometimes called SV-A (Figure 2-3). No other AEC contracts have been located.

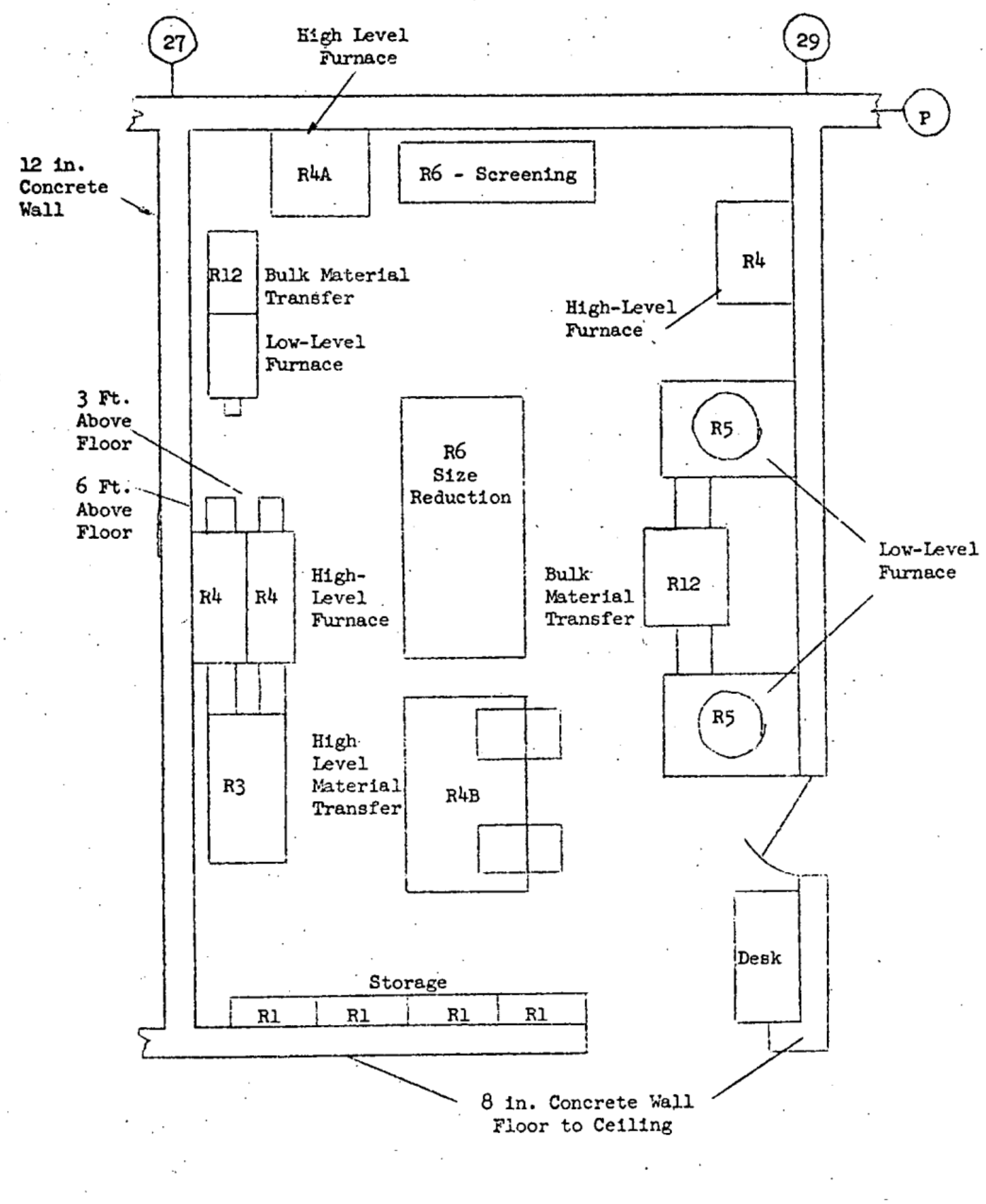

Figure I-1 - Recovery Operations Area

Figure 2-3. Recovery Operations Area (SNM-696, p. 176).

The HCF, Building 23, consisted of centrally located hot cells (HLC, LLC, and MC), an adjacent operating gallery, support laboratories, change rooms, and administrative office areas. The HCF was designed and built in 1958 and 1959, and it was used from 1960 through 1969 to conduct the postirradiation examinations of DOE fuels, structural materials, reactor dosimetry materials, and instrumentation inspection of irradiated fuel. In late 1959, during the final phase of construction, GA decided to expand the original building design to include an additional 1,200 ft^2 of laboratory space to accommodate the EBOR Fuel Fabrication Facility. Portions of the building have been used to conduct other radiological activities, including the development and manufacture of EBOR fuel (in

Rooms 108/108A), the engineering-scale tritium extraction system to support the New Production Reactor (Rooms 109, 119, and 120), and a plutonium transport experiment that used plutonium oxide (Room 116). The building was maintained in a surveillance and maintenance mode from 1991 until it was remediated. This building and adjacent area were remediated from 1992 to 1994 and were released by the State of California from further radiological control in 1994.

GA maintains Isotope Committee Approval (ICA) and Work Authorization (WA) records that provide information that identified major radionuclides and quantities involved in the activity [many of these are currently available in Oak Ridge Associated Universities (ORAU) Team records]. Radiological controls to be used during the activity were specified, and the records were signed by the responsible investigator (sometimes names of other workers who were involved in the work were included).

2.2.1 Fuel Fabrication

Before this work moved to the Fuel Fabrication Facility (SV-A) in 1962, many of the activities were carried out in the Experimental Building. Related activities occurred in the Science Laboratory. Uranium (up to 93.5% enriched in ^{235}U by mass) and thorium were used in this facility. Radioactive materials were coated, blended, compounded, heated, rolled, extruded, machined, compacted, and eventually made into fuel elements in the Fuel Fabrication Facility. The station mass limits were based on fully enriched uranium and no credit was taken for the neutron absorption effect of thorium. In those operations in which either hoods or gloveboxes were used, a minimum airflow of 150 fpm was supposed to be maintained to the facilities hood ventilation system. Figure 2-4 shows the layout with station designations in the Fuel Fabrication Facility and Attachment A provides a step-by-step description of the fuel fabrication process that is described in SNM License 696. Figure 2-5 shows the mezzanine of the Fuel Fabrication Facility.

2.2.2 Uranium Purification

SNM in the form of impure uranium-bearing scrap materials such as oxides or carbides were processed to recover and purify the contained uranium. The material was accumulated from several sources (both on and off the site), including out-of-specification SNM product, process equipment, scrap, waste, and solid residue from liquid evaporation. Some of the materials were products from the facility recovery operations. Figures 2-6 and 2-7 show the layout of the purification operations at the Fuel Fabrication Facility. The uranium was recovered from unirradiated scrap, but these scraps probably included some uranium recovered from recycle operations. Records indicate that this area might have been used for thorium recovery as well. A GA letter (Bissonnette 1971) to the AEC on airborne radioactivity in excess of permissible limits in the scrap recovery (uranium-thorium) area of the Fuel Fabrication Facility on February 3 and 4, 1971, was the earliest documentation found that related to thorium scrap recovery. It is unknown whether thorium scrap recovery was occurring before 1970.

SNM License 696 descriptions of the uranium material extraction and purification steps, and activities at specified workstations follow. The general steps are summarized (SNM-696, pp. 191–192):

1. The impure uranium-bearing material was dissolved in an acid solution.
2. The uranium was extracted from the acid solution by means of an organic extracting agent.
3. Impurities that were extracted along with the uranium into the organic phase were removed by scrubbing the organic phase with an aqueous salting solution.
4. The uranium compound was separated from the organic phase by an aqueous extraction.

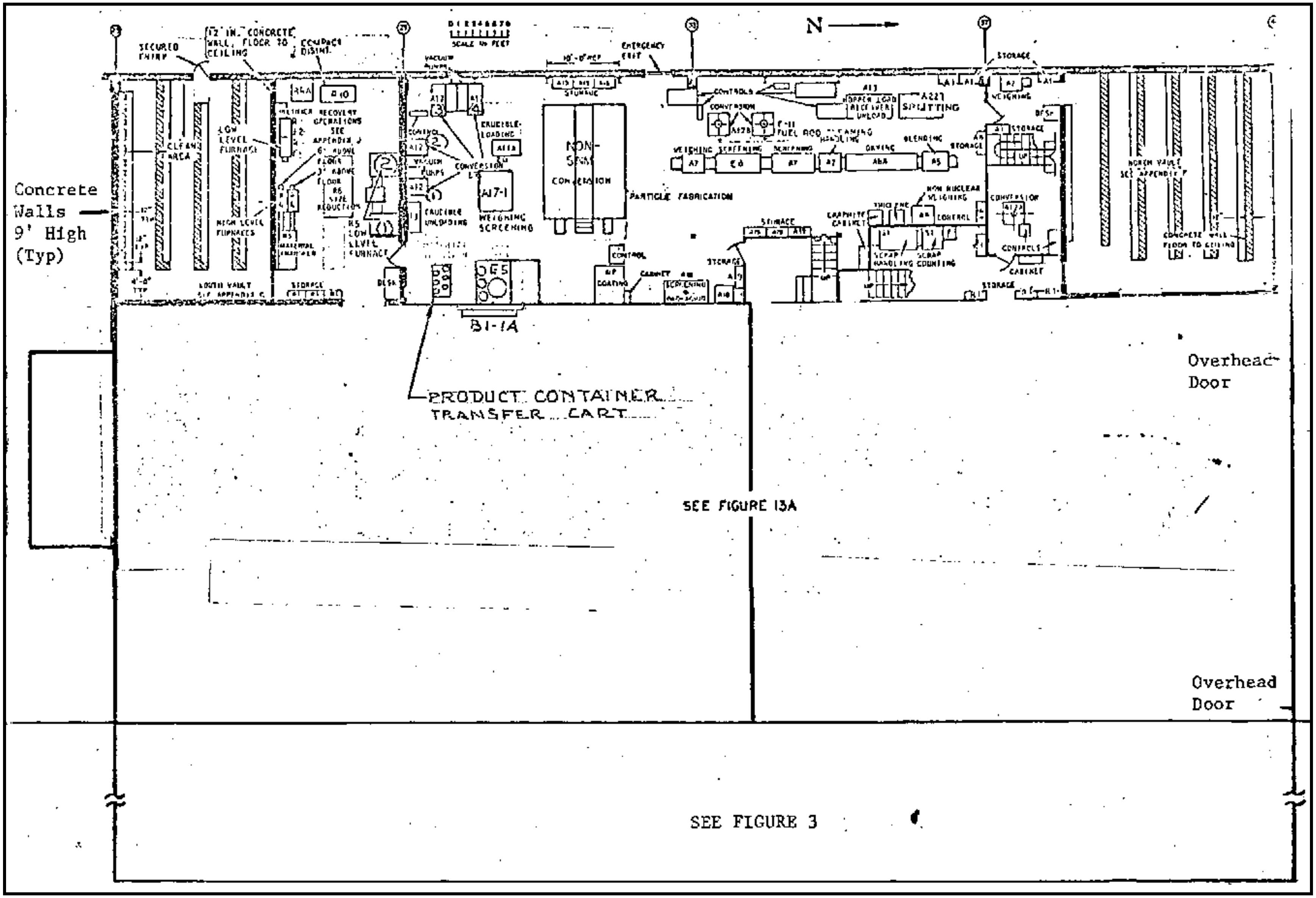

Figure 2-4. Fuel Fabrication Facility layout, 1974 (SNM-696, p. 241).

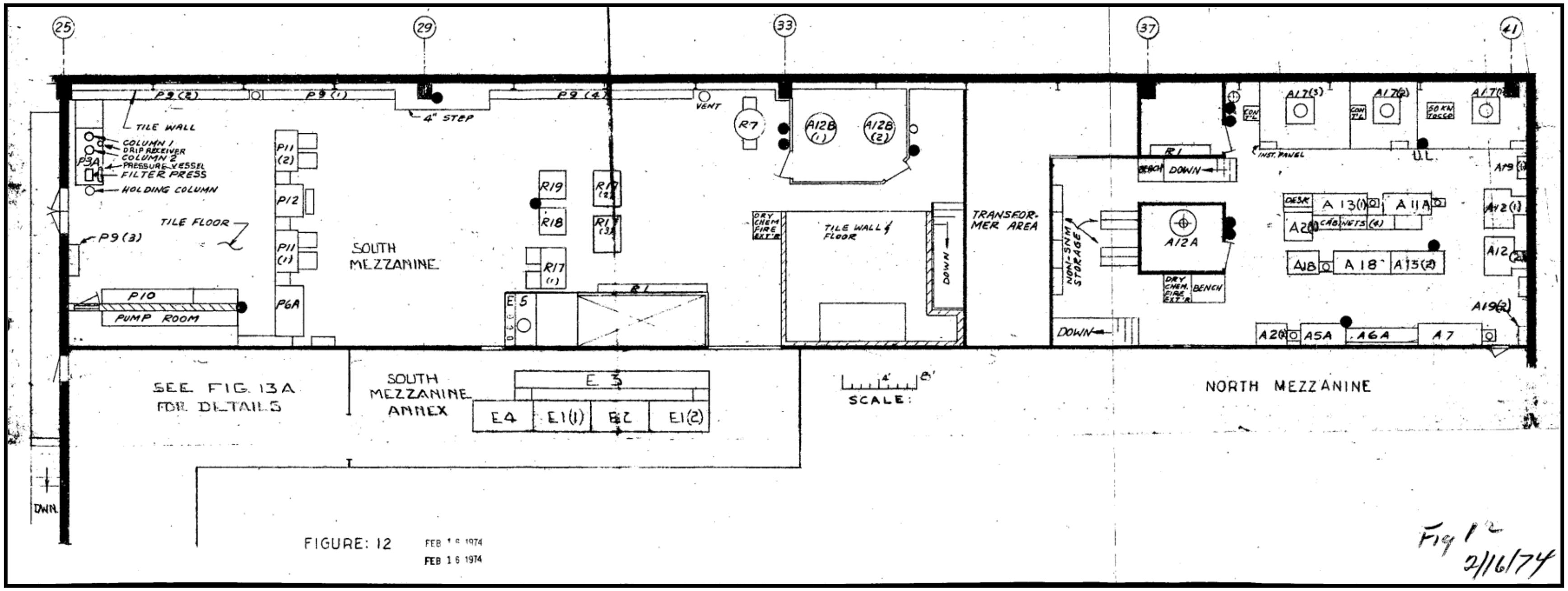

Figure 2-5. Fuel Fabrication Facility Mezzanine, 1974 (SNM-696, p. 240).

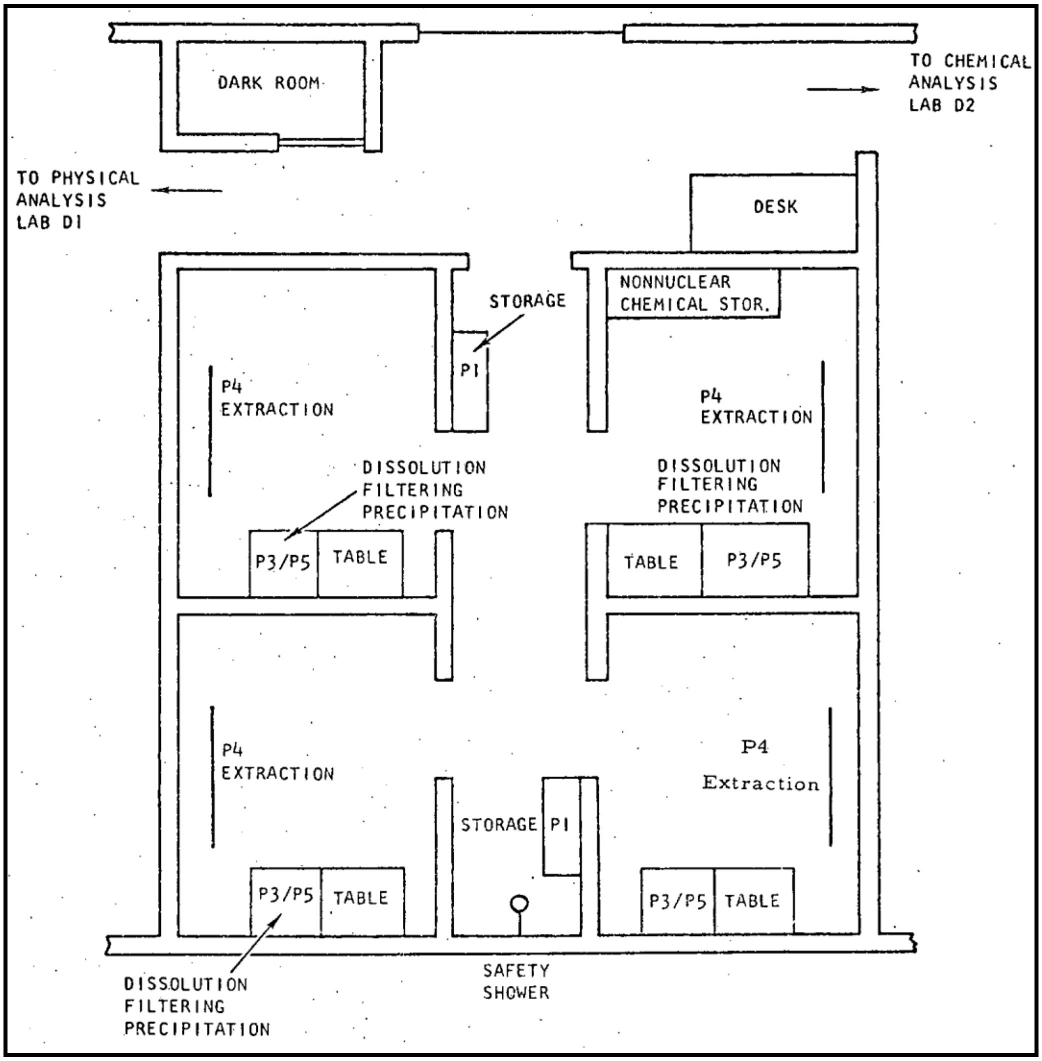

Figure 2-6. Uranium Purification Area (SNM-696, p. 205).

5. The uranium compound was separated from the aqueous solution by precipitation as a uranium salt.

6. Purified U_3O_8 was obtained from the precipitate by heating in a controlled temperature furnace.

A more detailed description of the process is provided in Attachment B.

2.2.3 LINAC Facility

The LINAC complex in Figure 2-8 consists of two LINACs, one 17 MeV and one 45 MeV, which could be coupled to obtain greater acceleration. Plans were made to increase the coupled output to 94 MeV when the 45-MeV klystrons were upgraded (GA undated). The accelerator beam was capable of being targeted to any of the three targeting cells (A, B, and C-D) as shown in the figure. The targeting could be changed during operations that required strict personnel entry controls and interlocks.

The LINAC facility has been used for a variety of research activities over the years. The major program at the LINAC in 1962 (GA 1962) was Integral Neutron Thermalization; this program, which was sponsored by AEC, had as its goal the measurement of integral phenomena of neutron thermalization and moderated systems. The electron beam was aimed at a suitable target, which

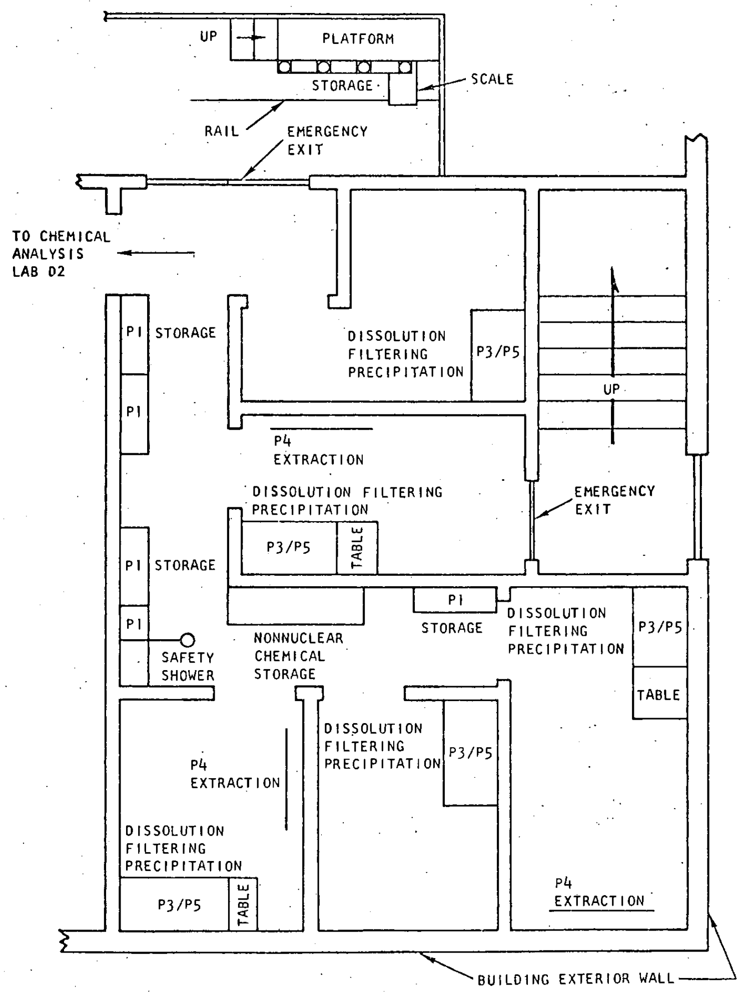

Figure J-2 - Uranium Purification Area

Figure 2-7. Uranium Purification Area (SNM-696, p. 206).

produced a burst of neutrons that passed through the test moderator. Moderated neutron energy spectra were then measured using time-of-flight techniques.

Differential Neutron Thermalization. In this program, which was also sponsored by AEC, investigators attempted to measure such neutron parameters as transport cross-sections and a variety of materials. The data included energy loss per collision for number of moderators as well as inelastic scattering cross-sections as a function of several variables for different moderated materials.

Multiplying Assembly. This program used a semicritical, semihomogenous uranium water system to produce an enhanced burst of neutrons when exposed by a neutron beam. The goal was to approximate the environment of an operating reactor.

Neutron Capture. This program was to determine the neutron capture cross-sections for a large number of materials.

In 1964, the above studies were still being performed with the addition of the following (GA 1964):

- **Fission.** This was a study to investigate the intensity of the delayed radiations from the fission of ^{238}U, ^{235}U, ^{239}Pu, and ^{232}Th.

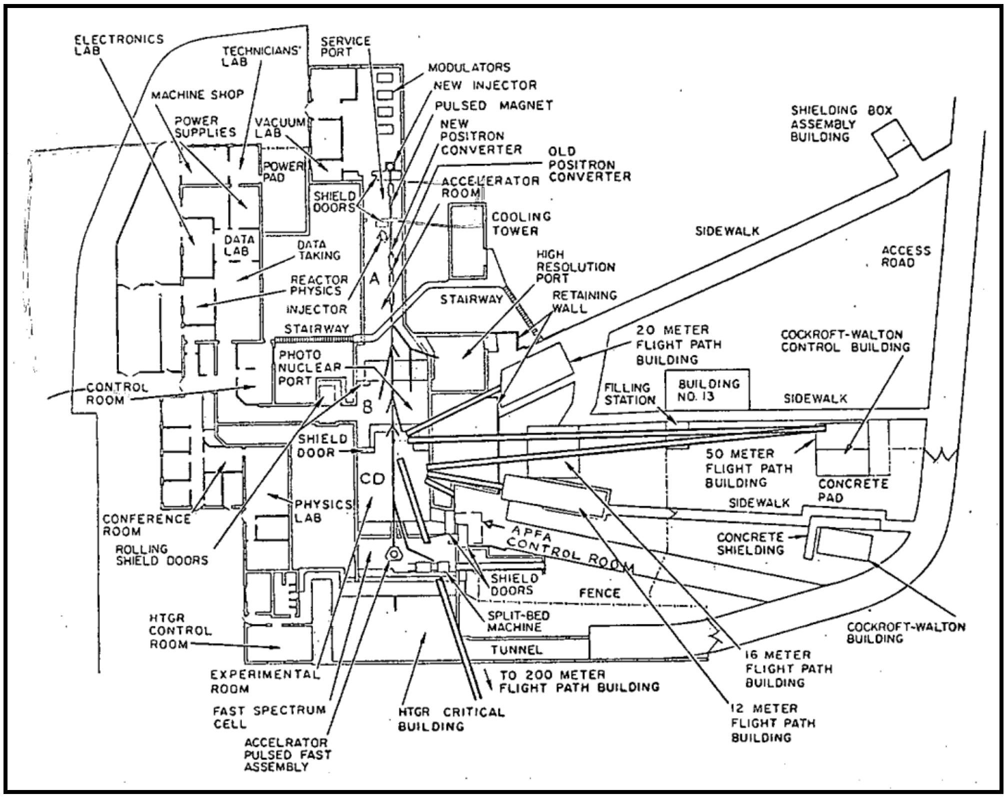

Figure 2-8. LINAC Complex

- **Positron and Photonuclear Transport.** This created the world's most intense positron beam by bombarding a tungsten target that was located after the first section of the accelerator. The positron beam was used for such studies as the comparison of electron- and positron-induced nuclear reactions.

- Using positron annihilation in flight, it was possible to generate monoenergetic gamma rays that were continuously variable between 3 and 30 MeV.

- **High-Energy Neutron Transport.** This program studied the deep penetration characteristics of high-energy neutrons. This was an important consideration in the nuclear heating of liquid hydrogen rocket propellants and the shielding design for space nuclear systems.

To ensure personnel safety, the LINAC used a combination of administrative controls with physical interlocks. Every electron beam radiation cell (a cell was any room into which the electron beam could be deflected) was equipped with a minimum of five safety devices and safety measures that were designed to protect personnel when in a cell. One of the safety devices was considered a primary safety device and controlled the high voltage on the klystrons. Four safety devices were considered secondary because they were in the personnel safety interlock chain. The five safety devices were as follows (Trimble et al. 1967, pp. 12–13):

1. **Cell Door Interlock.** This interlock prevented the operation of the LINAC while the cell door was open. The cell door was required to be open during occupancy.

2. **Cell Safety Plugs.** These were required to be in the receptacle by the door for the system to be energized. This interlock prevented targeting of the LINAC beam into the cell while a safety plug was removed. The safety plugs were at the entrance to each cell and had to be worn on the external clothing during cell occupancy.

3. **Cell Safety Switch.** This was a primary safety device that prevented operation of the LINAC whenever the switch handles were in the down position.

4. **Radiation Area Monitor (RAM).** This was a gamma detection system that would shut down the LINAC when a preset level was reached in the cell. Each cell contained a local readout of the radiation in the cell, so the radiation level in the cell could be determined immediately. A remote reading was provided in the LINAC console room. When the preset level was reached the LINAC was shut down and an alarm sounded. The cell was immediately evacuated when the alarm sounded.

5. **Mode Switch.** The mode switch prevented the entry of an electron beam into an active cell by placing certain personnel safety locks (e.g., a beam stopper) into the personnel safety interlock chain. The mode switch was in the LINAC console room and was controlled by the duty LINAC operator.

The two safety measures were as follows:

1. A Radiation Warning Light was activated when the LINAC was switched on in preparation of the electron beam to enter the active cell, and it remained on during LINAC operation.

2. All the shielding doors had switches that allowed them to be opened from the inside.

There was an additional interlock for any target that required cooling such that the LINAC would shut down should there be a loss of cooling (Trimble et al. 1967, p. 14).

In the early days of operation this led to some potentially unsafe conditions. In one case an individual entered one of the cells without removing the necessary plug; this led to his almost being exposed to potentially fatal radiation (Bold and Trimble 1966). This incident was one of the prime reasons a set of radiological safeguard regulations was issued for the use of the LINAC in 1967 (Trimble et al. 1967) with a recommendation of penalties for noncompliance (Bold and Trimble 1966). These regulations contained rules for non-LINAC personnel (experimenter), the Duty LINAC Operator, the experimenter, and the health physics surveyor. The following is a summary of these rules:

- **Machining of Radioactive Material.** Machining on any radioactive material in the LINAC machine shop (or in any other area of the LINAC facility) was subject to prior approval of the chairman of the LINAC safeguard committee. Any material that registered over twice background on an alpha, beta, or gamma detector was considered radioactive.

- **Electron Beam Irradiation Cell Entry Procedure.** A cell was considered to be in one of two conditions during an 8-hour work shift:

  - Inactive – The accelerator was either not attempting to operate during the shift or the cell was opened for the rules that follow.

  - Active – The cell was prepared to receive the electron beam during the shift and the accelerator was only temporarily shut down.

Entry to an inactive cell was permitted if the LINAC was not scheduled to operate during the shift, provided the health physics department had surveyed it since the last accelerator operation. The door was required to be open throughout occupancy. If the LINAC was operating, the operator had to declare the cell open with entry allowed subject to the following (Trimble et al. 1967, pp. 8–9):

1. The request had to be made to the Duty LINAC Operator.

2. If entry was deemed feasible, the Duty Health Physics Surveyor and Experimenter in charge had to be notified.

3. Confirmation of the securing of all electron beam stoppers, removable shielding pieces, and deflecting magnets had to be made by the Duty LINAC Operator.

4. The LINAC mode switch setting had to be confirmed to show that it was not set to the cell being opened by the Duty LINAC Operator.

5. The LINAC operator opened the door and a health physics employee performed a survey. If the beam was not steadily on target in an unoccupied cell, and the subject cell needed to be occupied, the HP surveyor had to remain on duty in the occupied cell until the desired targeting was achieved. When 100% power was achieved, the Duty LINAC Operator notified the surveyor, who did a final survey of the cell, and the Duty LINAC Operator would then determine if the cell could remain open.

6. The cell door was to be left open during occupancy. The RAM had to be checked with a source, and the LINAC scram had to be proven to be operable with a check of all interlocks by the surveyor.

7. Cell safety plugs were required to be worn on the external clothing of all cell occupants and returned to the plug board on leaving the cell.

All cells were closed before the accelerator was moved to a new target.

An active cell was never declared open. Entry into such a cell was made only by the Experimenter in charge or the Duty LINAC Operator and those under their immediate control. The following procedure had to be followed:

1. The accelerator was turned off by removing high voltage from the klystrons, and the operating key was removed from the console.

2. At least one member of the party that was entering the cell had to carry a gamma survey instrument and act as a surveyor for the party.

3. All persons who were entering the cell had to wear a safety plug on the external clothing and return it on leaving. An exception was made for the Duty LINAC Operator who carried the LINAC console key.

4. If the constant air monitor for the cell exceeded its trip level or if fissionable targets were in use, protective clothing and shoe covers had to be worn.

5. A health physics surveyor had to be contacted immediately if significant smearable contamination was suspected due to fire, leakage of coolant, etc.

6. The cell door had to be left open at all times during occupancy.

7. Persons who were leaving an active cell had to check their hands, feet, and clothing for contamination as well as any irradiated items that were removed from the cell.

2.2.4 Critical Assembly Facilities

A fenced area between the LINAC Complex (to the southwest) and TRIGA/Experimental Area Building #1 (to the northeast) was used to conduct many critical and subcritical assembly experiments at the GA site. A January 24, 1961, document indicated three buildings within the fenced area that appear to have originally been set up to support both the MGCR and HTGR programs (GDC and H&N 1961, p. 95). In addition to these two buildings, a fuel assembly building was also located in the area. (The facility was later known as the Experimental Critical Facility.) The LINAC facility was also used to conduct these criticality experiments. (At some point, the northernmost portion of the LINAC facility housed the HTGR critical assembly.) Over the years, several additional reactor fuel development programs have occupied the critical experiments buildings:

- Nuclear Rocket or Thermal Tungsten Rocket
- Modified HTGR
- Advanced HTGR
- Thermionic

A paper attached to Isotope Committee Amendment #1101 indicated that the purpose of critical assembly experiments was to test the criticality safety of fuel designs to establish safe handling procedures (Richey, Lloyd, and Clayton 1965). A similar study (Russell 1967, p. 3) stated:

> *A portion of this program [thermionic] consists of constructing subcritical assemblies of the critical facility fuel elements and measuring the neutron spectrum from these assemblies using the General Atomic Electron Linear Accelerator and time-of-flight techniques. These spectra will be used to aid in the understanding of the critical experiment performed.*

These fuel assemblies were either irradiated with accelerator-produced neutrons or with neutron triggers (e.g., Po-Be or Pu-Be sources).

The design of the fuel and assembly was such that if an undesired criticality were to occur the fuel would either rapidly expand (Russell 1967) or the subcritical fuel beds would separate (Pound 1974, Attachment), which induced the neutron excursion to terminate. In the first case, a phenolic resin that fails at 140°C was used to bind fuel compacts, and conical springs would separate the fuel to subcritical conditions. In the latter, beds of fuel were brought together and attached electromagnetically. If a criticality occurred, the magnet would fail and one of the beds would slide away by gravity.

A March 29, 1974, memorandum requested that the reactor building and associated fuel assembly buildings that made up the Experimental Critical Facility (also known as the TCF) be assigned for storage of Fort St. Vrain reload fuel elements (Pound 1974).

2.2.5 Reactor Facilities

GA operated several reactors in the TRIGA Reactor Building (Building 21). The first reactor that was identified through data capture documents was the Critical Isotope Reactor, General Atomic (CIRGA). Although it is clear that fuel assemblies were constructed for the reactor in January 1958 (Ray 1958a), very little information was found to describe the reactor itself, and it is not clear if this reactor ever operated. The critical assembly operated under License CX-3 (Ray 1958a). Reactor Pit 1 was "used for subcritical experiments and other tests" (GA 1969, p. 18), and it is possible that this area was used with CIRGA.

Three separate TRIGA reactors – Mark I, Mark F, and Mark III – were operated in Building 21, each under its own license – R-38, -67, and -100 respectively. The original building housed Mark I and was constructed in 1957. The building was later expanded east (1960) and then north (1966) to house the Mark F and III reactors, respectively. These reactors were all below-ground pool reactors (GA 1969, p. 13) and were designed to provide university training, provide a source of neutrons and/or gamma radiation for research, aid in the investigation of the effects of radiation on materials, and produce radioisotopes.

The Mark I was graphite reflected and operated routinely at steady-state thermal levels up to 250 kW and could also be operated in pulsed mode. The Mark F reactor (also known as the Advanced TRIGA Prototype and the FLAIR) also operated routinely at steady-state thermal power levels up to 1,500 kW and could be operated in pulsed mode, but it was water reflected. The Mark III reactor operated routinely at steady-state thermal levels up to 1,500 kW and was water reflected.

The TRIGA core contained reactor fuel, dummy fuel assemblies, and one or more neutron source holders. As a thermal reactor, the TRIGA reactor depended on slow or thermal neutrons to perpetuate chain reactions in its fuel. Because the neutrons born in the fissioning of ^{235}U atoms were too energetic to produce significant numbers of further fissions, a moderator was employed to slow down thermalized fast neutrons. In the TRIGA reactor, neutron thermalization was achieved by using zirconium hydride, water, and graphite as moderating materials. To reduce leakage of neutrons from the core, reflectors were installed. Figure 2-9 shows the TRIGA Reactor Facility.

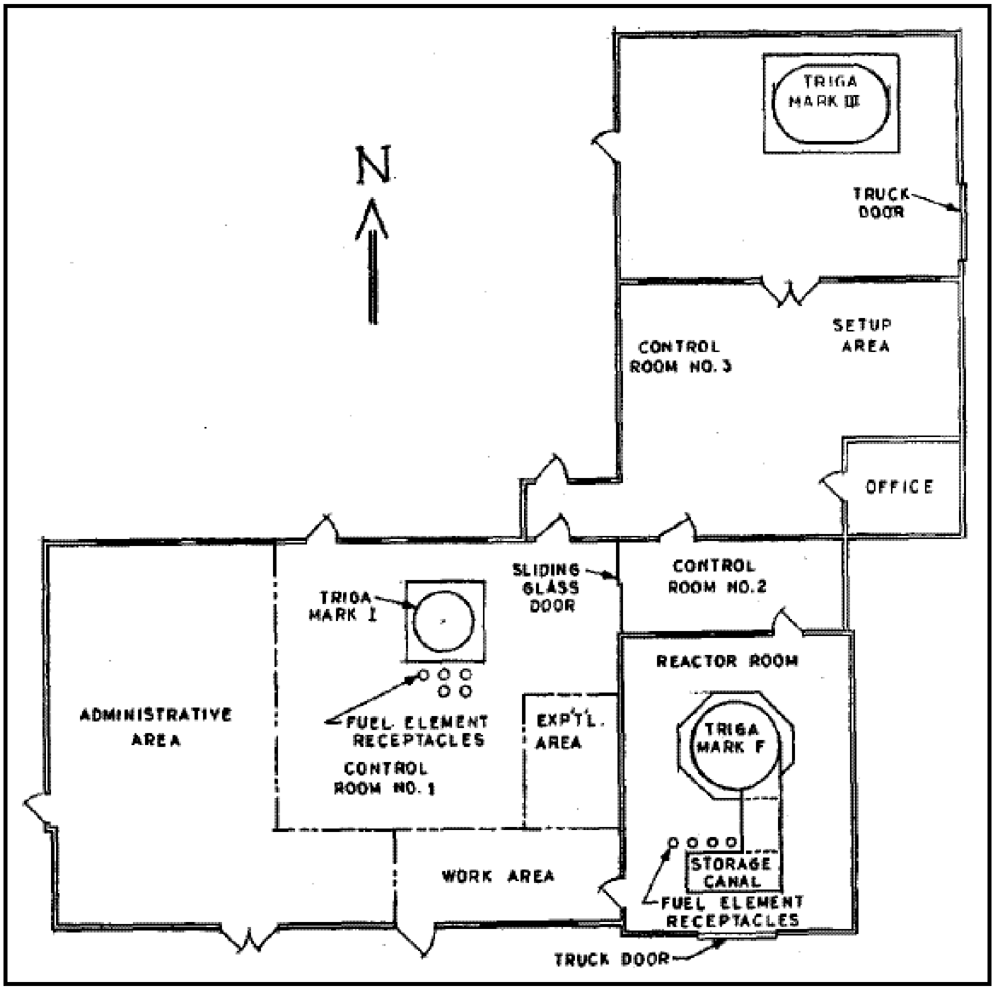

Figure 2-9. Plan view of TRIGA Reactor Facility (GA 1969).

2.2.6 Science Laboratory

The Science Laboratory consists of three arc-shaped buildings that surround the laboratory center; they contain 150 individual laboratories and associated offices (Figure 2-10) where work in chemistry, experimental physics, electronics, metallurgical research, reactor physics, and thermoelectricity was performed. These are two-story buildings in a ring around the central library and cafeteria buildings. Laboratory Buildings A, B, and C have 46,000, 49,000, and 41,200 ft^2, respectively.

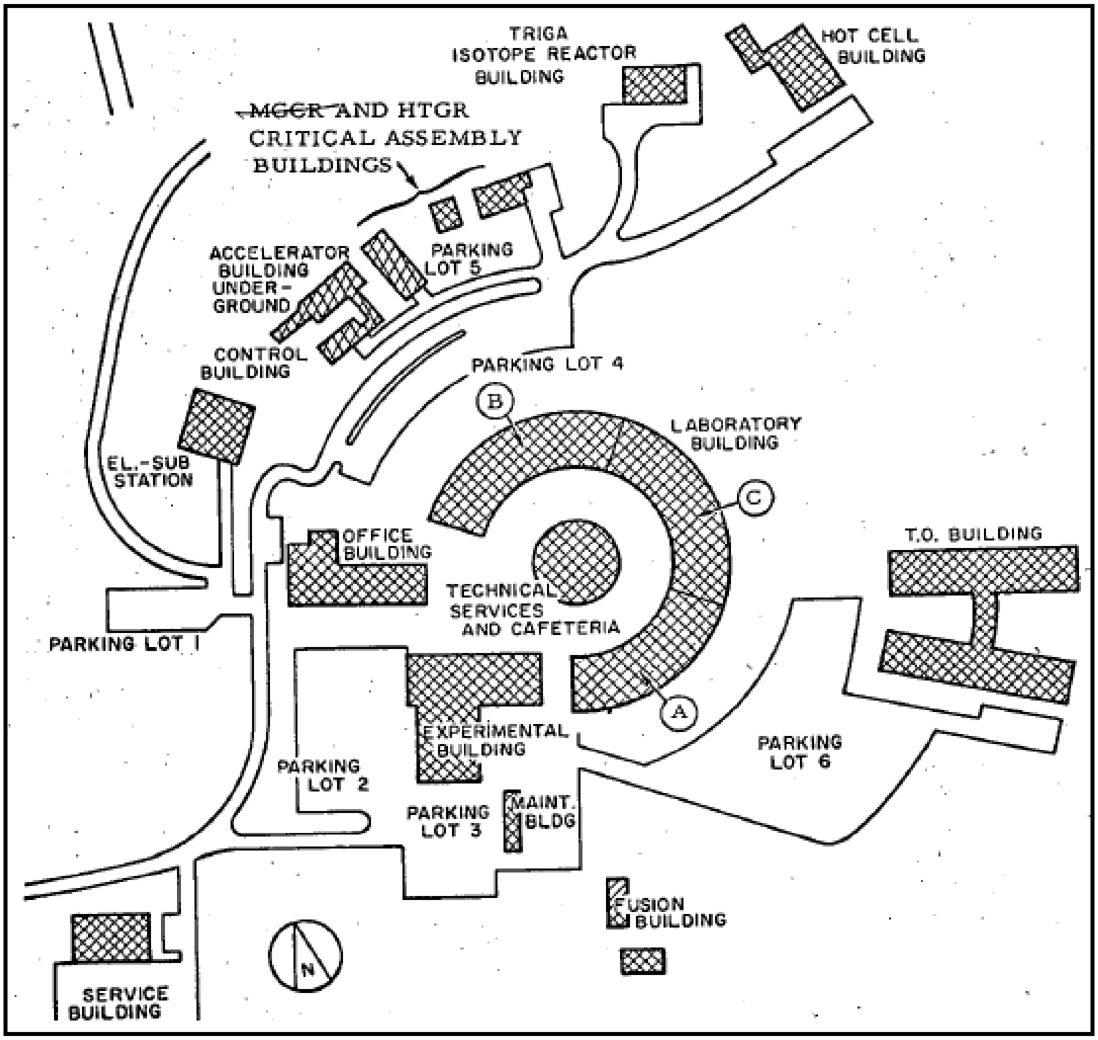

Figure 2-10. Site map showing location of Science Laboratory (SNM 1963).

These buildings are arranged and named in a counter-clockwise direction with Building A, then Building C, then Building B, beginning from the southernmost segment. The specific size of these buildings and the floor plans for Buildings A and C are indicated in Figures 2-11 and 2-12, respectively. Some room numbers might have changed over time.

2.3 SOURCE TERM

There are three main sources that describe the amounts and types of radioactive material that were handled at GA: federal and State of California licenses for the possession and use of radioactive materials; descriptions and reviews of proposed experiments or jobs that involved handling of radioactive material in the form of ICAs, WAs, and Radiological Work Permits (RWPs), which contain information on radionuclides, quantities, and recommended safety precautions for the described activity and inventory and material handling (accountability) records.

The use of SNM was governed by AEC regulations and licenses, SNM-69 (Docket No. 70-72), and SNM-696 (Docket No. 70-734). Byproduct materials were governed under AEC License

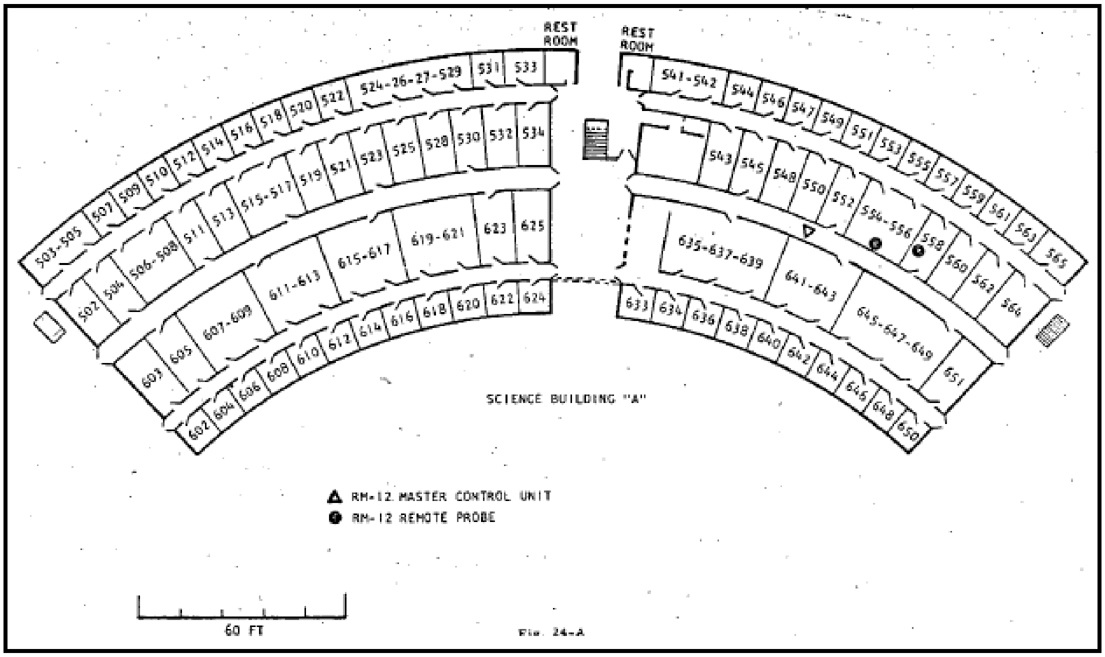

Figure 2-11. Floor plan of Science Building A (SNM 1963).

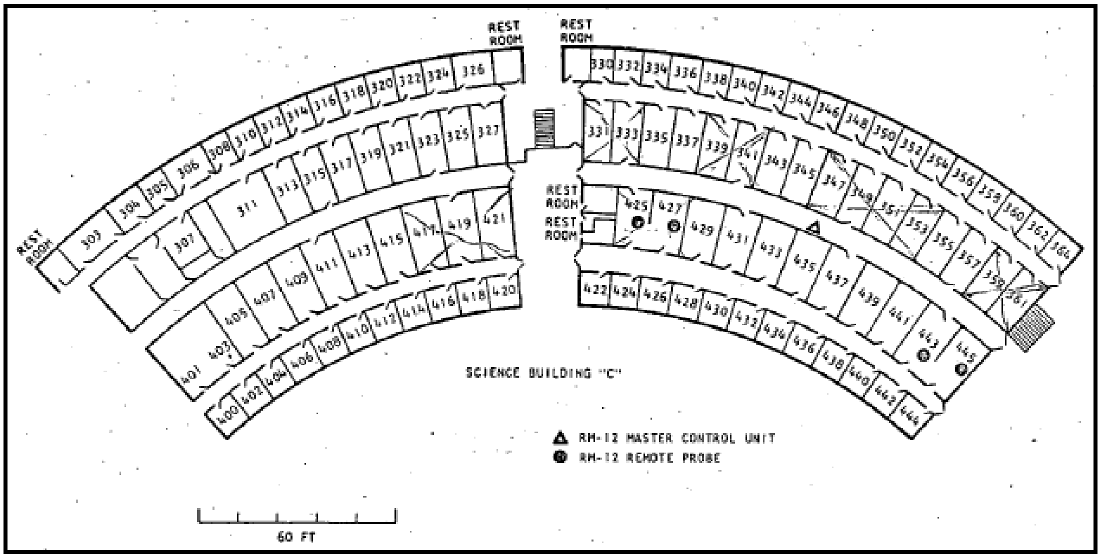

Figure 2-12. Floor plan of Science Building C (SNM 1963).

No. 4-1611-11 (J-60), which appeared to have been transferred to State of California jurisdiction in later years. A request for amendment to the byproduct material license in December 1960 or later requested additional authorization to possess and use 1 MCi of fission products and to increase the amount of byproduct materials 3 to 83 to 100,000 Ci each and up to 500,000 Ci total (GDC and H&N 1960). Another license, SNM-55, was identified in License Amendment #59 (SNM-69 Amendments 41-62), but has not been reviewed. Attachment C provides a summary of licensing limits that are described in GA records. It appears that in the earliest years of operations, the site was licensed by the AEC to possess byproduct materials and SNM, and at some point the request to increase in the byproduct material limit and add fission products appears to have been accepted based on the California license summary from about 1973. The AEC did not license accelerator-produced material or naturally occurring material (except for source material, i.e., uranium and thorium). Information on early radioactivity limits for accelerator-produced or naturally occurring radioactive material and source material has not been found, but these materials are included in the California license summary information.

The GA radiological source term included uranium, thorium, plutonium, fission and activation products, strontium, and tritium, as well as sealed photon- and neutron-producing sources and radiation-producing equipment. Radioactive materials were received from off the site or created on the site in the reactor or with the LINAC.

Uranium. Uranium in the form of metals, oxides, and carbides was used for GA fuel fabrication, reactor fuel, research studies in the hot cells, laboratories, and reactor areas. The typical amounts of uranium in use in any one area ranged from milligrams to 100s of kilograms. Occasional work with other uranium chemical forms, such as UF_6 or uranyl nitrate, was also conducted. Uranium isotopic forms included depleted, normal (i.e., natural enrichment), and enriched (up to 93.5%), as well as ^{232}U, ^{233}U and ^{236}U. Uranium from recycling operations would have included relatively small activities of nonuranium isotopes, such as ^{99}Tc, ^{237}Np, ^{230}Th, and ^{239}Pu.

As discussed in Section 2.1.1 (Building 37 or SV-A) ^{233}U was used for fabrication of some of the HTGR fuel compacts that were incorporated into finished HTGR fuel elements. Uranium-233 was also used to conduct neutron yield studies at the LINAC (Bold 1967a, p. 114) and used for various research and development activities at various GA locations (see Attachment D). Site documentation (Bold 1965b, p. 242) indicates that a special glovebox designed for handling up to 100 g of ^{233}U for a special experiment in one of the Science Building laboratories was inspected in January 1965. Uranium-233 is produced by the neutron irradiation of ^{232}Th, which captures a neutron and becomes ^{233}Th, which decays to ^{233}Pa, which decays to ^{233}U. The ^{233}U at GA was most likely from the Hanford Site (Parker 1965, pp. 3–7).

In addition, ^{233}U always contains the impurity ^{232}U due to the neutron irradiation of ^{232}Th. Correspondence related to the ^{233}U program suggests that ^{232}U impurity levels in the ^{233}U at GA would have been 8 ppm (Parker 1965, pp. 3–7). The ^{232}U impurity in the ^{233}U results in elevated gamma and beta dose rates due to the ingrowth of ^{232}U progeny (^{228}Th and its progeny). In addition to the presence of ^{232}U at GA as an impurity in ^{233}U fuel material, site documentation indicates that ^{232}U was being used for diffusion studies at the site.

In October 1966, ^{232}U was the first radionuclide that was handled in the HCF Plutonium Laboratory glovebox (Gurren 1966a). Documentation from about March or April 1967 indicates that ^{232}U in solution was deposited onto disks and then the liquid was evaporated in gloveboxes at the HCF Plutonium Laboratory (Gurren 1967a and 1967b). According to Gurren (1967b), the discs were approximately 1×10^5 cpm alpha and were used for diffusion studies. The reports also indicate that there were no contamination problems during the ^{232}U work; however, representative sampling of debris from the HCF HLC in 1994 indicated that the concentration of ^{232}U was higher than any other uranium isotope (Sills 1994). It appears likely that the 1994 debris sampling included materials that had been used in making the ^{232}U discs from previous years. On the other hand, ^{232}U would have also been present in the HCF HLC from examination of irradiated HTGR thorium-bearing fuel.

(**NOTE:** The Heath report (Heath 1971) cover page says it might contain patentable material).

Thorium. Thorium use and chemical forms were similar to uranium. The total mass of thorium that was used on the site was probably less than that of uranium, but the thorium activity in use in an area at a given time could have been greater or less than uranium activity.

Plutonium. GA conducted the SNAP-15A program for AEC, which utilized a small thermoelectric generator that was powered by ^{238}Pu. According to site documentation, over 20 SNAP 15A generators had been successfully tested by the end of September 1964 (GA 1964). GA also documented the presence of plutonium on the site in October 1964 when the SNAP-15A Plutonium Handling Room was completed at the SV-B Building (Bold 1964a), but plutonium was probably handled elsewhere on the site earlier. The SNAP Plutonium Handling Room was later relocated to the Experimental Building concrete vault in October 1965 (Bold 1965a).

Site documentation indicates that encapsulated PuN and or PuO_2 were used in nuclear (thermoelectric) battery assemblies that were assembled in the SV-D Building (SNM-69 1963). Laboratories (e.g., Mound Laboratory or Harwell in the United Kingdom) supplied the double-walled, encapsulated-plutonium-containing fuel capsules in shipping containers. Receipt surveys of the entire outer surface were conducted and compared to an action level of 6 pCi (~10 dpm). Site documentation indicates that fuel capsules contained a nominal 0.18 g of ^{238}Pu as nitride with 20% ^{239}Pu (no amounts were specified for $^{236/240/241/242}$Pu, but lesser amounts of these isotopes would be expected) (SNM-69 1963; GA 1971a).

Plutonium use was also noted in the LINAC and laboratory areas, the reactor areas, and the Plutonium Handing Room at the HCF. On October 6, 1964, 82 ^{239}Pu foils that were sealed with a plastic coating were inspected on receipt in an absolute filtered glovebox at the LINAC (Bold 1964a). The foils were presumably going to be used in an experiment that involved the measurement of delayed gamma-ray intensities from the photofission of ^{239}Pu and ^{235}U foils that was temporarily discontinued the following month (Bold 1964b). After a receipt inspection in October 1968, plutonium-aluminum alloy fuel elements (jacketed rods) for a Euratom plutonium-fueled experiment were received at the LINAC for placement into stainless-steel pressure jackets and pressure testing (Bold 1968a). The Euratom plutonium fueled experiment was completed on October 31, 1968, and the rods were placed into storage. Wipes that were taken on the fuel rods, pressure tank, and glovebox where the experiment was carried out “indicated no significant contamination” (Bold 1968b). In addition, authorization to use an 8.5-L plutonium nitrate solution that contained at least 4.6% ^{240}Pu in a shielded container in a glovebox at the LINAC facility was granted in 1968 (SNM-69 1963). A December 1968 report indicated that dry runs with a simulated plutonium nitrate solution at the LINAC were being performed (Bold 1969).

Receipt surveys of plutonium fuel elements for use in the HTGR Critical Facility were documented in several reports beginning in September 1966 (Bedsaul 1966). Site documentation indicates that receipt survey precautions were undertaken to ensure that plutonium fuel elements (Bold 1966a) and thermoelectric sources (Bold 1964a) were not leaking.

Table E-1 in Attachment E provides information on the types of work and radiological monitoring that were performed at the Plutonium Laboratory in the HCF. The work appears to have involved mostly unencapsulated fuel-grade plutonium in gloveboxes. The mention of beta as well as alpha activity measurements in the HCF Plutonium Laboratory (Table E-1, November 1968) seems to imply that the beta activity fraction was as high as or higher than the alpha activity fraction, which might have been due to the presence of ^{241}Pu, which might have been significant (about 85%) in terms of activity in a $^{239/240}$Pu source term, but its maximum energy beta emissions of 21 keV (similar to ^{3}H beta energy of 18.6 keV) would be hard to detect unless a windowless counter was used. Perhaps the photon emissions from the plutonium isotopes and ^{241}Am, as well as the bremsstrahlung from the ^{241}Pu beta emissions, were detected and attributed to beta emissions. There is no mention of ^{241}Am in association with the plutonium source term, and for fresh source terms the ^{241}Am activity would be insignificant, but by 5 years the ^{241}Am ingrowth would account for approximately 8% of the total radioactivity in an unirradiated reactor fuel-grade source term. On the other hand, the beta contamination that was noted in the November 1968 survey in Table E-1 might have been due to contamination that was tracked into the Plutonium Laboratory from elsewhere within the HCF.

The isotopic composition of the various types of plutonium that would have been encountered at GA is presented in Section 3.1.3.

Tritium. Little information has been found on the quantities and chemical forms of tritium that were used at GA, but site documentation indicates that a Texas Nuclear Corporation Neutron Generator console at the LINAC was being used with a “new 3 to 5 curie tritium target” in January 1963 (Bold 1963a). In February 1965, it was noted that the Activation Analysis group would move into the new laboratory building (Experimental Area Building No. 1) and that the bioassay schedule was updated to include all personnel who worked in the area (Bold 1965c). The October 1965 monthly report for TRIGA and Activation Analysis Facilities (Cockle 1965a) noted that 25 used tritium targets that totaled 111 Ci were transferred from GA for storage and ultimate disposal. The fusion group might have been using tritium targets on the order of 1 Ci. Tritides (particulate), and tritiated water vapor and tritiated gas (the latter compounds would be released during off-gassing of the tritium targets) were probably the main chemical forms on the site.

Strontium. GA conducted research and development on thermoelectric generators that were powered by radioisotopes for space power applications for AEC. By the end of September 1964, a 250-W ^{90}Sr-powered thermoelectric generator had been investigated and test generators in the 4- to 10-W range were being fabricated and tested (GA 1964). Strontium bioassays also indicate its presence on the site. Radioactive strontium could have been associated with the irradiated fuels or the ^{90}Y production facility (about which little is known) as well as other programs. A 1971 HCF report indicates that a ^{90}Sr source had been loaded into a generator housing and transferred to the SV-D Building (Bedsaul 1971). Specific information on chemical form or forms of ^{90}Sr at GA is unavailable, but strontium that is associated with the irradiated fuels is likely to be in the form of an oxide and not titanate; however, if strontium was being used for the production of nuclear batteries (thermoelectric generators), it might have been in the form of $SrTiO_3$ (CBWI 2006; OTA 1994; Grissom 1994).

Fission Products. Irradiated fuel would have a variety of fission and activation products. GA documentation mentions ^{134}Cs, ^{137}Cs, and ^{182}Ta as well as ^{60}Co and ^{154}Eu. It was noted that a molybdenum production facility was located on the site, although it is not clear if this referred specifically to radioactive molybdenum. Additional fission and activation products that are associated with intakes at the TRIGA reactors on the site can be obtained from the current revision of ORAUT-OTIB-0054, *Technical Information Bulletin: Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2007a).

Activation Products. The TRIGA reactors and the LINAC were used to activate a variety of materials including soils, steel, cladding materials, uranium, and plutonium. A more detailed list of possible activation products at the LINAC is provided in Attachment F.

Other Radioactive Materials. Sealed sources, including ^{226}Ra, ^{137}Cs, Pu-Be, and Am-Be, were available on the site for irradiation or calibration work. There is mention of releases of noble gases and iodines from the reactors and use of ^{252}Cf in a glovebox. GA was licensed to possess 50,000 Ci of $^{170/171}$Tm, but no documentation was found to describe its use or form. A possible use of $^{170/171}$Tm is in isotope-powered thermoelectric converters (Kay 1994). No other information has been found on this potentially very large activity source with low-energy beta emissions. There was also a mention of using unsealed sources of ^{226}Ra.

Radiation-Producing Devices. Reactors and a variety of particle accelerators and X-ray producing machines were used on the site. Some descriptions are in the Reactor, Critical Assembly, LINAC, and Laboratory sections of this document. External radiation from neutrons as well as photons and electrons could be generated on the site from the reactors and the LINAC, and from sources such as Pu-Be or Am-Be that were listed in GA documentation. Table 2-1 lists radiation-producing devices in the Science Laboratory.

Table 2-1. Radiation-producing equipment in the Science Laboratory as of April 1971.

X-ray machines and/or location	Manufacturer	Voltage (kVp)
L-556	Phillips Electronics	5
L-556	Picker X ray Corporation	50
L-556	Picker X ray Corporation	40
Pulsed X-ray machines location Science B	Field Emission Corporation Model 265	700
X-ray diffraction location		
L-147	Phillips Electronics	60
L-147	Phillips Electronics	60
L-409	Phillips Electronics	60
Electron microscope location L-543	Hitachi	100
Electron microprobe location L-154	Applied Research Laboratory	50

Source Term in the Laboratory

The determination of the amount and variety of radionuclides that were the source term for exposures in the Laboratory is a complex process due to changes in radionuclides over time as well as fluctuations in the amount of the SNM (^{235}U, ^{233}U, and plutonium) that appears to have been omnipresent in the Laboratory. The number of laboratories with the variety of activities that are associated with work in chemistry, experimental physics, electronics, metallurgical research, reactor physics, and thermoelectricity imply the potential for a large variety of radionuclides as well as variant quantities of these radionuclides in the Laboratory at any given time. Beyond these considerations, there is the potential to view individual laboratories as separate entities in the Laboratory, each with an individual source term. The alternative – viewing all the individual laboratories as one entity – has validity based on inventory changes alone but is most strongly supported by the conclusions of a1970 evaluation of the Laboratory (Bold 1970a). There are also sources of information on specific radionuclides in the Laboratory, such as the example of annual throughput data (Attachment D), which is only a partial list of radionuclides that would have been in the Laboratory and other GA facilities (GA 1972).

Because there is no compilation of all the radionuclides and their activities in the Laboratory at any point in time, several information sources were examined for data that contributed to defining the Laboratory source term. Because these information sources were not created to define source terms, but rather for other purposes, only portions are relevant for defining source term. The primary sources of this information are:

- SNM throughput data for the Laboratory (Attachment D),
- License information (Attachment C),
- ICA data, and
- Health physics reports.

Isotope Committee Approval forms

There was a requirement at GA that a request to use radionuclides be formally reviewed by a standing group that was referred to as the Isotope Committee. Approval from this committee was necessary to proceed with work. The approval forms generally included a reference to where the work was to be performed, the radionuclides to be involved, and a brief description of the physical form of the radionuclide. Some ICA summaries are included in Table 2-2 to indicate this level of information. The data were taken from about 6 months of 1962 and as such represent some of the activities that took place in these laboratories during this time. The variety and quantities of radionuclides in the Science Laboratories at other times might have been similar or not.

Table 2-2. Summary information for selected ICA forms.

ICA	Laboratory room	Radionuclide(s)	Date
326	L-343	Natural uranium 3E-2 Ci solution-nitrate Cesium-137 3E-2 Ci solution-nitrate Strontium-90 3E-4 Ci solution-nitrate Natural thorium, 3E-3 Ci, solution-nitrate	4/10/1962
328	L-647 Metallurgy X-ray Laboratory L-657	Graphite sample	3/22/1962
327	L-307 L-147	Zn-65 5E-3 Ci metal	4/12/1962
352	L-635 L-637,39	U-235, <1 mCi, pyrolytic coated Tl-232, <1 mCi, (U-Th) C_2 particles in graphite compacts	4/12/1962
333	L-540 L-637, -639	Unknown isotope, <100 mCi, irradiated graphite	4/9/1962
334	L-141	Any nuclides resulting from neutron irradiation of steel including Fe-55, Fe-59, Cr-51, Ni-63, Ni-65, Mn-56, Co-58, Co-60, Mo-93, Mo-99 The nuclide of initial interest is: Fe-59' 0.001 Ci, solid steel plate I-131, 0.015 Ci, solid iodine crystals	4/23/1962
349	L-137	Fission products, 0.1 Ci, as produced in uranium carbide	5/22/1962
400	L-218 High temperature lab to 3500ºC	BeO irradiated and graphite, solid Ci to be determined in hot cell	9/10/1962
402	L-425 L-427 L-429	U-238, wire	9/17/1962
406	L-554 L-556 L-558 L-439 L-441	U-238, UO_2, UC_2 up to 1,500 g U-235, UO_2, UC_2 up to 300 g Th-232, ThO_2, ThC_2 up to 1,500 g	9/25/1962
407	Unstated laboratory	U-238, U-235, Th-232, 10 g	9/21/1962
412	L-343	*Z* = 3 to 93, 1 mCi	9/21/1962
413	L-151	Ir-192, 10 µCi	9/25/1962
411	L-554 L-556 L-558 L-439 L-441	U-235, U-233, Th-232	3/22/1962
320	L-236 L-143	Z = 32 to 83, solid or liquid, 0.1 Ci *Z* = 90 to 92, solid or liquid, 0.1 Ci Pneumatic system between TRIGA and L-236 Micarta (a proprietary polycarbonate reinforced cotton material), polyethylene encapsulated)	3/26/1962

Health Physics Reports

The level of information provided in the health physics reports varies considerably based on when and by whom they were written. Some reported items included discussion, radionuclides, and quantities; others were quite brief. Some of this information is summarized in Table 2-3.

Health physics reports also included occasional discussions of monitoring or exposure estimating methods.

Table 2-3. Radionuclides in specific locations in the Laboratory.

Location	Source term information
L-540	150 g of normal U
L-441	Spill, 50,000 cpm α
L-403	MFPs
L-307	Cs-134, 10 mCi
L-441	Uranium fire, 127 g of U-235
L-236	Sr-89 plus Sr-90, 2.5 R/hr β+γ at 16 in., contact 160 R/hr β+γ
L-218	Ta-182
L-407	Ce-144/Pr-144, 0.1 μCi
Science Building B service core	K-85 release, 5 Ci

A memorandum from May 22, 1963, described a fire in L-540 that involved uranium chips and turnings, which produced an estimated maximum air concentration of 4.4×10^{-7} μCi/ml (Bold 1963b). The memorandum went on to state that "although there were no air samplers in operation at the time," the volume of the room was 114 m^3 and therefore based on the estimated amount of uranium that was consumed in the fire the maximum air concentration estimated was 4.4×10^{-7} μCi/ml. The report went on to state that as the maximum permissible concentration (MPC) for natural uranium was 6×10^{-11} μCi/ml and the time-adjusted MPC based on the exposure time was 9.6×10^{-9} μCi/ml.

This memorandum raises several questions, such as why there were no operating air samplers in adjacent areas of the Laboratory, because if there was a sampler in L-540 at the time it was either not functioning or not operating (L-540 is characterized in Bold 1970a as one of two laboratories that were designed to be most appropriate for handling the most significant radionuclides in the Science Laboratory).

2.4 SAFETY

The GA program broke ground in 1956. For December 15, 1960, through December 31, 1964, Table 2-4 lists the number of film badges that were distributed (biweekly before January 6, 1962; once a month thereafter), the number of pocket dosimeters that were distributed (two per employee), the number of dosimeters that were provided to visitors, the number of bioassays that were reported for a given period, the number of wipes that were counted (alpha and beta counts might have been counted separately), the number of air samples that were counted (sometimes first and second counts or alpha and beta counts are tabulated separately), and the number of people who were considered trained to handle radioactive material.

Safety precautions were included in the ICAs and WAs. A review of the first 100 ICAs (there are at least several hundred more of these documents that have not yet been obtained) shows that the authorized user's name is on the applications, but the names of associates were not always included. A review of the first 100 ICAs and claim file information resulted in identification of a single name that appeared in both information sets, although it should be noted that the current ICA capture is not complete.

Formal health physics procedures were found for the early to mid-1970s (after the operational period); before then, policy documents appear to have governed site radiological activities. A discussion with Paul Maschka, a radiological control technician, who began work at GA in the mid-1960s and who still works there part time, indicated that early site radiological protection personnel were required to cycle through training at all areas where radiological activities occurred, before working unsupervised. It was estimated that this training took 6 months to complete.

Table 2-4. Radiological monitoring from December 15, 1960, through December 31, 1964.

Period	Film badges[a]	Dosimeters reset and read	Visitor dosimeters	Bioassays processed and evaluated	Qualified to handle radioactive material	Wipe and air samples counted	Wipes[b]	Air samples[c]
12/1–31/1964	791	1,770	878	188	-	9,193	-	-
11/1–30/1964	755	1,795	730	232	-	8,620	-	-
10/1–31/1964	785	2,273	861	185	-	-	3,088	6,057
9/1–30/1964	692	1,613	674	285	-	9,699	-	-
8/1–31/1964	686	1,916	622	374	-	9,704	-	-
No July report	-[d]	-	-	-	-	-	-	-
6/1–6/30/1964	880	2,320	868	199	-	-	4,500	6,536
4/1–31/1964	810	1,705	750	254	-	-	5,695	3,380
3/1–31/1964	692	1,831	484	222	-	6,134	-	-
2/1–29/1964	644	1,990	617	262	-	6,341	-	-
1/1–31/1964	718	2,259	504	274	-	7,574	-	-
12/1–31–1963	873	1,825	636	210	-	6,358	-	-
11/1–30/1963	867	2,049	731	257	-	6,736	-	-
10/1–31/1963	879	1,901	824	165	-	8,258	-	-
9/1–30/1963	819	1,992	593	(e)	-	-	2,222	4,463
8/1–31/1963	808	2,251	722	(e)	-	-	3,146	4,709
7/1–31/1963	746	2,035	600	(e)	-	-	2,692	3,400
6/1–30/1963	719	2,096	502	(e)	-	-	2,274	2,708
5/1–31/1963	783	1,796	636	(e)	-	6,318	-	-
4/1–30/1963	658	1,890	480	(e)	-	-	2,112	2,582
3/1–31/1963	629	2,063	563	(e)	-	-	2,084	2,322
2/1–28/1963	611	1,798	364	(f)	215	-	2,426	2,106
1/1–31/1963	563	2,187	519	(f)	-	-	2,760	2,040
12/1–31/1962	528	1,669	453	(e)	-	-	1,636	1,066
11/1–30/1962	527	1,619	562	(e)	203	-	3,084	888
10/1–31/1962	528	1,659	492	(e)	180	-	3,436	830
9/1–30/1962	472	1,322	477	(e)	-	-	3,474	662
8/1–31–1962	374	2,461	687	(e)	-	-	2,250	858
7/1–31/1962	336	1,684	501	(e)	-	-	2,720	5,080
6/1–30/1962	331	1,631	542	(e)	-	-	2,454	618
5/1–31/1962	330	1,322	484	(e)	-	-	3,000	700
3/16–4/30/1962	598	2,242	710	(e)	116	-	3,151	570
2/1–3/15/1962	308	1,340	372	(e)	-	-	2,335	370
1/15–2/15/1962	380	2,109	621	(e)	-	2,536	-	-
12/16/1961–1/15/1962	474[g]	1,757	680	(e)	-	1,268	-	-
8/16–9/15/1961	864[g]	1,836	948	(e)	-	-	-	-
7/16–8/15/1961	558[g]	2,334	984	(e)	-	-	-	-
6/16–7/15/1961	519[g]	2,880	1,232	(e)	-	-	-	17
5/16–6/15/1961	934[g]	2,422	707	(e)	-	-	-	-
4/16–5/15/1961	528[g]	1,247	684	(f)	-	-	-	36
3/16–4/15/1961	541[g]	1,280	640	(e)	-	-	-	-
2/16–3/15/1961	465[g]	2,742	622	(e)	-	-	-	-
1/16–2/15/1961	556[g]	1,657	-	(e)	-	-	-	-
12/15/1960–1/15/1961	546[g]	1,300	-	(e)	-	-	-	-

a. Film badges were biweekly before January 6, 1962 and monthly thereafter.
b. Wipes sometimes include alpha and beta counts separately.
c. Air samples are sometimes added in twice (or possibly more) as an initial count and a delayed count, or as an alpha count and as a beta count.
d. No data found.
e. Although the Health Physics Reports for these time frames do not indicate that bioassays were performed or the number of bioassay analyses performed, there is documentation to indicate that some bioassays were performed during some of these times (GA 1956–1963, GA 1963–1967, GA 1963–1970).
f. The Health Physics Reports for these periods made references to the analysis of bioassay samples, although specific numbers were not called out.
g. Monitoring was biweekly during this period, and monthly during later periods.

Engineering controls to reduce radiation exposure appear to have been present throughout the site. The HCF was designed with thick concrete and leaded glass shielding to minimize external dose to individuals who worked in the area. An evaluation of the laboratory in June 1970 compared the radiotoxicity of radionuclide uses and limits to the laboratories that contained them. The basis for this evaluation was an article published entitled "United States of America Standards Institute (USASI) Design Guide for Radioisotope Laboratory" (Bold 1970a). This article set guidelines for radiochemistry laboratories based on the four classifications of relative radiotoxicity – very high, high, moderate, and slight. Recommended maximum activities for each laboratory were given. The laboratories were classified as A, high-level laboratory; B, radioisotope laboratory; and C, good chemical laboratory. A description of what the four classes of isotopes entails is included in Attachment G. At GA, most were classified as Class 2 and Laboratory types B and C. More importantly, these references are necessary to be able to understand the discussion of the original description of the laboratories in the Science Building that follows (Bold 1970a):

> *When the facilities of the Gulf General Atomic were originally constructed, two laboratories located in the Science Buildings, L-307 and L-540 were designed as radioisotope laboratories. They were marginal in design and probably fell somewhere between the USASI classification of a good chemical laboratory and a radioisotope laboratory. As the need for additional facilities to do radiochemistry work arose various chemical laboratories were modified by the addition of high efficiency filters on the chemistry hoods, shielding, and controlled access to allow for this work. Thus, some of these originally designed chemical laboratories were developed into what could be classed as relatively good radioisotope laboratories*.

Ventilation equipment with particulate filtering appears to have been used in many areas throughout the site to minimize airborne contaminants. However, reports of work activities that did not use appropriate ventilation are common, especially in the early years. Open doors would occasionally cause contamination to blow into a lobby or similarly undesirable place. Air sampler placement was not always ideal in relation to measuring what exposures (Bold 1963b). On the other hand, it was not unusual to use a portable air sampler or localized ventilation to measure or control some of the potentially higher hazard exposures.

The use of personal protective equipment (gloves, coveralls, respirators, air hoods, etc.) and personal survey requirements were required based on the perceived hazard, and hot and cold change rooms were available in some areas. Much work occurred in the early years to keep the cold side cold.

Available site documentation (Quimby 1966) indicates that supplied air hoods were required in the Fuel Fabrication Facility thorium process area by January 1966. The use of supplied air hoods was also noted at the HCF in 1968 (Cockle 1968a), but the use of supplied air hoods probably dates back before 1966. In June of 1971, GA received approval from AEC and State of California to use air-line respirators (hoods) for personnel in controlled areas where airborne concentrations averaged over 40 hours in any 1 week exceeded the limits that were specified in Federal and State regulations (Bold 1971a).

Little information is available on directly measured contamination levels at GA areas, but numerous results of wipe tests are available for the 1960s. Area contamination appeared to be extensive in some areas, such as the Fuel Fabrication Facility, and more localized in other areas. Most contamination measurements were reported in terms of removable activity and adjusted to picocuries or picocuries per unit area. Site documentation indicates that the removable contamination limits for clean areas were 6 pCi alpha and 30 pCi beta based on wiping an area of approximately 413 cm^2 using a 2-in. filter paper (SNM-69 1963, p. 36). These levels were twice the sensitivity of the counting equipment. For controlled areas, contamination limits were established that depended on the specific operations, but alpha contamination limits were not to exceed 500 cpm as measured with a portable alpha counter that was equipped with a 60-cm^2 detector on a large-area masslinn wipe of approximately 100 ft^2. Site documentation indicates that contamination levels were expressed in terms of 100-cm^2 areas by the 1972–1973 time frame (GEES 1972, p.65).

GA utilized a dedicated decontamination crew throughout the operational period (Bold 1965d,e) and during much of the residual period (Bold 1977) to minimize contamination levels in controlled areas. It appears that contamination levels were high enough at the Fuel Fabrication Facility during the operational period to require decontamination of the floors twice every 24 hours during times of increased workload (Bold 1965d) and continuously during the residual period (Bold 1977).

Workers were to be monitored or check themselves for contamination when leaving contaminated areas, but in the early years there were a number of reports of missed opportunities and instrumentation that was not working properly. GA designed and fabricated a number of its own instruments, and although its innovations were admirable, and probably very acceptable, there is a sense that calibrations of radiation instrumentation in the early years sometimes lagged. Alpha hand and foot counters were available in the earliest days, but they were frequently partially out of service (some detectors did not work) and detection thresholds are unavailable.

Low-energy beta measurements did not appear to be a concern in the earliest years, but these radiations gained interest by the mid-1960s. An internal proportional counter (IPC) reportedly measured tritium at levels approximately 20 times higher than the low beta counter. During October 1969, the highest tritium value from the IPC was reported as 32,000 pCi and a minimum value of <100 pCi. Because of tritium's very low beta energy (0.0186 MeV maximum), quantitative measurement of tritium on wipes is very dubious, especially on samples in which significant self-absorption or self-shielding can take place or contamination from other radionuclides might be present.

Unreliable performance of the IPC reduced the tritium wipe and air sample counting throughput in November 1965 (Cockle 1965b), and it was noted that the tritium monitor had not detected significant activity to date. In January 1966, a cleanup of the bunker area took place and several areas were to be individually cleaned by Activation Analysis Facility personnel until clean. A preliminary procedure was written for interlock operations, target changes (to be performed in a "glovebox type" enclosure), and maintenance at the bunker. Tritium tritide air concentrations were measured as $<1 \times 10^{-10}$ $\mu Ci/cm^3$ during the January 1966 time frame. The June 1966 health physics report (Gurren 1966b) suggested a complete cleanup of the neutron generator bunker room before the next target change, investigation of a practical method to calibrate the tritium monitor, and adherence to the bioassay schedule. In January 1967, Eberline tritium film badges were distributed to all personnel in the facility, and additional badges were placed at various locations within the facility (Kesting 1967a) in an effort to determine whether it was feasible to use the film badges in lieu of bioassay samples. Results reported in February were all less than the 1,000 pCi detectable limit of the film (Kesting 1967b, p. 52). How these results might be used for dose reconstruction is unclear, but the next month it was noted that quarterly samples for tritium had been provided by all personnel on the March 1967 schedule (Kesting 1967c). In June 1967, it was noted that the semiannual bioassay had been completed.

Individual monitoring programs were part of the GA program, although bioassay before October 1963 occurred in response to nose wipes and incidents and was not routine. Routine uranium urinalysis and some mixed fission product (MFP) and beta urinalyses were also added on then. Bioassay for tritium and strontium isotopes came a bit later. A review of the NIOSH-Office of Compensation Analysis and Support (OCAS) Claims Tracking System data indicates that external dosimetry was provided to workers who were directly involved in radiation work. Workers who were less likely to have elevated radiation exposures, such as a telephone operator, showed only sporadic individual external dosimetry. Radiation workers were provided with a film badge and two pocket dosimeters.

The film badge could measure fast neutrons, and the dosimeter measured slow neutrons. It is not clear if the dosimeter results are included in the individual records. Some extremity badges were available, but it was common to substitute a less desirable wrist dosimeter for the unavailable ring dosimeters. According to the explanatory notes in the early reports of external dosimetry (e.g., p. 2 of both Landauer 1959 and Landauer 1961) a beta dose was reported only for betas of energy greater than 1.5 MeV. One report mentioned that the ratio of this lower energy beta to gamma dose might be on the order of 50. One surveyor's anecdote on surveying while a worker adjusted an X-ray beam indicated that the worker had directed the beam to his forehead, and that perhaps head shields should be made available, which points out the difficulty of assessing nonuniform radiation doses, which were not unusual in some GA areas.

Air sampling devices (including giraffe, portable, and continuous air monitors) were used on the site. Early on, complaints of a lack of these samplers and the equipment to count them were common. The number seemed to increase over time. Many air monitoring results (for individual, area, and environmental samples) were observed at GA as handwritten entries that began appearing in technician logbooks back in the 1960s. Some limited environmental monitoring data were collected in the early years, but were not consolidated. For tritium monitoring at GA, Cockle (1965a) noted the following:

> *Air Sampling: Air samples have been taken on a daily basis since October 18, 1965. Sampling has been specifically for suspected air-borne titanium tritide in the generator room and the console room. The present sampling is not for gaseous tritium or tritiated water vapor. Plans have been discussed to use the tritium monitor when opening the drift tube to sample for tritium gas.*

Area contamination measurements of floors and work areas are available in technician logbooks (which have not all been captured) and some of the more notable results appear in health physics reports. A table of instruments that were on hand and the desired instruments and how they would be used is in Attachment H.

2.5 INCIDENTS

The table in Attachment I of information on radiological incidents and accidents for 1960 to 1969 is based on information from GA documents; it is not a complete list of radiological incidents and accidents, but rather provides a general sense of these events at GA. It should not be construed from this summary that documentation of more significant incidents will not be found.

2.6 PHYSICAL EXAMINATIONS – X-RAYS

According to 1961 reports, "Special tests for employees working around BeO. Special radiation tests, X-rays and Spirograms," ranged in number from 93 to 169 per month (Bethard 1961a,b,c,d). No other documentation on occupationally required medical X-ray examinations has been found. Current health physics personnel were unaware of a requirement for occupationally required X-ray examinations of BeO workers in the early years of operations; however, it was thought that workers in the respiratory protection program had chest X-ray examinations. Two claimants indicated that chest X-ray examinations were given either every 3 months or annually. A former GA nurse recalled that chest X-rays were required for beryllium workers and stated the X-ray examinations were conducted at La Jolla Radiology. To date no information on equipment and techniques has been found. Medical records that indicate if an X-ray examination was made might be available at La Jolla Radiology, but copies or originals of the X-ray films are believed to no longer exist. Many of the inactive medical records at GA have been destroyed in the past several years, but it is possible that medical files exist, either on or off the site, for some of the still-active workers.

In the early years at GA, complete blood counts were used to assess potential biological detriment from external and internal radiation exposure. A checklist was used (Form 1-1) to document medical history, previous radiological exposure, assignment of a film badge or pocket dosimeter, attendance at radiological safety training, receipt of a controlled copy of the Radiological Safety Guide, etc.

2.7 SUMMARY OPERATIONAL PERIOD ASSUMPTIONS, WORKDAYS, WORK HOURS, WORK CATEGORIES

When bioassay data, film badge results, and other monitoring results are used to estimate internal and external exposures during the AEC operations, an estimate of the number of workdays or work hours per year is generally not important. However, whenever airborne concentrations are used for dose reconstruction the number of hours in a standard workweek needs to be known. The use of a 56 hour MPC in GA site documentation (Bold 1963e, p. 62) and (Bold 1963f, p. 71) indicates that some employees worked 56 hours per week. The inhalation intake rate estimates for unmonitored workers presented in Tables 5-1 through 5-7, and in Tables 6-2 and 6-3, are based on a 10 hour workday for 250 workdays per year. If claimant files contain exposure time information, the intakes should be adjusted accordingly.

No attempt has been made to sort workers into exposure categories. Individual dosimetry appears to be available for most workers. One unusual job category at GA was the custodian, who had the responsibility for surveying clean trash and decontaminating areas. In addition, it was noted that a number of different workers might have changed filters on hoods and exhaust systems, including the metallurgist and Administrative Services staff.

2.8 RESIDUAL CONTAMINATION AND REMEDIATION PERIODS

Documentation that was reviewed by NIOSH (2006c) indicates a potential for significant residual contamination outside the period of weapons-related production at GA. The residual contamination period was from January 1, 1970, through December 31, 1995. A DOE remediation period is listed by the DOE Office of Worker Advocacy as 1996 through 1999, and this document assumes it ran from January 1, 1996, through December 31, 1999. The residual and remediation periods are discussed further in Sections 5.0 and 6.0 respectively.

3.0 ESTIMATING INTERNAL EXPOSURE

This section addresses internal exposure during the operational period from 1960 through 1969.

Air monitoring was achieved with both fixed-head and portable air samplers, but a summary report from December 1960 stated that insufficient uranium air sampling was occurring in uranium process areas (Ray 1960a). Iodine-specific sampling in the facilities did not appear to be in place at startup, as indicated by a December 1960 health physics report notation that 11 mCi of ^{131}I had been released through the stack of the HCF, but “the calibration of our iodine measurement is subject to question” (Ray 1960b), and by a statement by Ray (1960c) that iodine was not suitably detected by the GA April 1960 methods of evaluating radioactive particulate concentrations (collection of sample on a filter and counting). The February 15 to March 16, 1961, progress report stated, “A new filter paper holder that incorporates a charcoal filter as well as a paper filter has been designed for use on the ’giraffe' air samplers. The charcoal filter is used to sample radioactive iodine. With a minor modification of the filter paper holders on the Continuous Air Monitors, the charcoal filters can be used in them” (Bold 1961a). Fixed-head air samplers were often located near hoods and gloveboxes. Air samples included considerable activity from thoron (^{220}Rn) progeny, and so required a couple of days’ wait before long-lived air concentration measurements were available. No explicit concentration measurements of thoron or its progeny have been found for 1960 through 1969. A later set of measurements from 1975 is not considered representative of 1960 through 1969.

The initial focus of the bioassay program appeared to be uranium. Results were commonly reported in units of micro-microcuries, which is equivalent to picocuries. As noted in section 2.4, bioassay before October 1963 occurred only in response to nose wipes and incidents and was not routine. Records of urinalysis for fission and activation products are available as early as 1963. The early (1963) urinalysis procedures for uranium have in general been described. In December 1963, the HP Laboratory Report (Andrews 1964a) stated:

> *The urinalysis procedure for uranium is being changed. The old procedure required that the entire volume of urine be boiled dry with an excess of HNO_3. This resulted in excessive odor and noxious fumes. The new procedure follows the MFP scheme and precipitates the uranium. Most of the urine is then poured down the drain and the precipitate is redissolved and ashed with a minimum of nitric acid. The procedure is essentially unchanged from here on. Extraction of the uranium with ether, back extraction into H_2O and drying in a planchet complete the sample preparation. The use of ether remains a safety hazard.*

The December 1963 HP Laboratory Report (Andrews 1964a) also reported that the Eberline MAC with a ZnS(Ag) phosphor would be used in lieu of an internal proportional counter to analyze urine samples. It was additionally noted in that report that the MAC ZnS (Ag) phosphor in contact with the sample would have essentially the same background and efficiency as the internal proportional counter, and so the MAC was available for counting the urine samples, which would reportedly speed up counting by about 500%.

No records of thorium or plutonium bioassays have been found for 1960 through 1969.

In vivo bioassay measurements were first performed on site in 1966 by Helgeson Nuclear Services (HNS) in its mobile counting van. Based on a review of available Helgeson reports, the van came to the site every 4 months (HNS 1967, HNS 1969a, HNS 1969b, HNS 1969c, GA 1970) . According to the site documentation (GA 1970) individuals were selected for counting based on their potential for exposure to beta-gamma emitters and ^{235}U as well as on the results of their urinalyses results. Both total body counts and chest counts were performed. A review of some of the 1969 Helgeson reports indicates that ^{60}Co, ^{58}Co, ^{137}Cs, ^{182}Ta, ^{95}Zr/Nb, ^{131}I and ^{85}Sr were occasionally detected (HNS 1969a, HNS 1969b, HNS 1969c). The focus of the early chest count was ^{235}U; a program for detection of thorium was added in the 1970s (HNS 1975), which is after the operational period. An in-house counting system, based on a ***Health Physics Journal*** article design, including a 4- by 4-in. sodium iodide crystal and an office chair lined with lead was used when counting was desirable between Helgeson's scheduled visits (GA 1966, GDC 1966-70, Tammany 1973).

Both *in vitro* and *in vivo* bioassay data are available for GA employees. For uranium bioassay results, it is favorable to the claimant to assume 93.5% enrichment unless there is documentation that indicates a different enrichment. For bioassay results in the operational period that were reported as gross alpha, it is favorable to the claimant to assume that the gross alpha result is attributable to the alpha-emitting radionuclide (^{234}U or ^{233}U) that results in the greatest organ dose, unless there is documentation that indicates that the gross alpha result is representative of only one radionuclide (or radioelement). For bioassay results that were reported as gross beta, it is favorable to the claimant to assign radionuclide-specific intakes of mixed fission and activation products using the methods in ORAUT-OTIB-0054 (ORAUT 2007a), using ^{90}Sr as the indicator radionuclide that document describes, as discussed in Section 3.3.6.2 of this document.

In vitro data, specifically urinalyses for the 1960s and later exist at the site in three formats: laboratory records, transcribed results on individual cards, and an electronic database that was developed later to better assimilate individual results. It appears that the bioassay analyses were performed on the site. Some nasal swab results are available, apparently in response to incidents. Wound monitoring to identify photon emitters was observed in at least one incident report. Some fecal samples were analyzed.

The general codes for bioassay data are listed in Table 3-1.

Table 3-1. Bioassay codes [3].

PU, PA, and PX	Uranium urinalysis (results written as U-235 in urine can be assumed to be total uranium. These samples could either have been analyzed by fluorimetry or gross alpha counting. Any alpha emitter that made it through the chemical processing of the urine would be detected by the gross alpha counting methods, but the actual chemical recovery for nonuranium alpha emitters is unknown, so only uranium is assumed).[a]
UX	Whatever radionuclide is stated in the heading of the database column, these are samples that were sent to U.S. Testing.
PT	Tritium.
PF	Mixed fission and activation products.
PC	Cesium (assumed to Cs-137 unless Cs-134 or another isotope is specifically stated).
PS	Sr-90.
PI	I-131.

a. The individual GA Internal Radiation Monitoring reports (*in vitro* section) typically include the summary heading: "Analysis Type: Gross Uranium Urinalysis (U-235)."

3.1 *IN VITRO* DATA

Most of the information in this section applies to urinalysis, but the documentation indicates that some fecal samples might have been analyzed at GA or by an offsite laboratory. No fecal sample results have been located in the files.

Based on a review of urine sample calculation sheets (GDC 1965a; GDC 1965b), most individual urinalyses appear to be based on the collection of spot samples with results recorded as picocuries per 24 hours. GA normalized these results to a 24-hour sample on the basis of collection time; that is, they multiplied the results of these spot samples by a "yield factor" that was the ratio of 24 hours to the urine sample collection time to determine the 24-hour sample result.

There are a number of bioassay history cards; these are available in two electronic image files. One set, Bioassay History Cards 00000 to 49999 (GA 1963–1967), was reviewed to better understand the format. The file numbers appear to be assigned to specific individuals (i.e., everyone was assigned a unique file number). This data set is organized in batches based on individual personnel (file number) at GA. The whole dataset contains results for about 1,000 people from November 25, 1963, to June 18, 1970. There are several card formats as shown in examples below:

Example 1

Date	*µµc/24 hrs*	*Analysis*	*Reference*
11-25-63	*<1.1*	*α*	*PUXXX*
		Gross α	
		Gross α	*PUXXX*
		U-235	*PUXXX*
		MFP	*PFXXX*
	<13.0	*β*	*PFXXX*
		T^3 *or* H^3	*PTXXX*

Example 2

Date	*pCi/24 hr*	*% MPL*	*Analysis*	*Sample No.*
	<0.93	*<2.3*	*α*	*PUXXX*
	0.80	*<1.1*	*α*	*PUXXX*

Example 3

Date	*pCi/24 hr*	*20*	*% MPL*	*Reference*
	0.89	*0.82*	*2.1*	*PUXXX*
	0.1	*0.8*	*<2.1*	*PUXXX*
	0.4	*0.6*	*<1.4*	*PUXXX*

Name, sometimes Social Security Number (SS# or SSN), Department, Room Number, File # (which corresponds to each individual person), Date of sample collection, etc., were included on reports. The less-than symbol (<) was used starting in 1963 with the earliest samples for alpha results, but not many beta results have a < symbol. The heading that appears to be 20 means 2-sigma and refers to the error that was associated with the result. It is likely that the error was based primarily on the counting statistics.

The code IH in the reference column could mean that either the sample was taken for industrial hygiene purposes rather than health physics purposes or, before about 1963, the sample was being analyzed by the GA Industrial Hygiene group.

The bioassay data card that contains the data entry from the calculation above (GDC 1965a, p. 22, sample PU321) was identified in claim files and is shown below:

Date	*μμc/24 hrs*	*Analysis*	*Reference*
2-26-65	*0.6*	*α*	*PU321*

Most GA dosimetry files also contain a database printout with the heading “INTERNAL RADIATION MONITORING REPORT” that contains the worker’s name, Social Security Number, birth date and sex. The format of the results is shown below with the results from the bioassay card and claim file above.

```
Analysis Type:          GROSS URANIUM URINALYSIS (U-235)
DATE          RESULT                2 STD DEV         UNITS
02/26/65 PU321 = 0.600              0.00              DPM/24HRS   DPM/L = 0.425
```

Note that the result entry begins with “PU321;” the “PU” entry, as noted in Table 3-1, signifies a total uranium result, so this result is assumed to be a gross alpha result. The favorable to claimant assessment for uranium, as described in 3.3.1, would be used for dose reconstruction.

Based on a review of claimant files, there are a number of instances where the bioassay data card result was expressed in pCi/24 hr but was entered into the database with units of DPM/24HRS as in the example above. In addition, less-than values entered on bioassay data cards appear to have been entered as a 0.000 result in the database. The dose reconstructor should carefully review all of the internal dosimetry data for a claimant to ensure that the data supports the reported units (i.e., results that are reported in “DPM/24HRS should be 2.22 times greater than other results reported pCi/24HRS). If the claimant dosimetry data review indicates the same numerical values with differing units, the most primary data source should be used. For the purposes of this direction, the order of primacy of internal dosimetry data records is (1) urine sample calculation sheets and offsite laboratory reports, (2) bioassay data history cards, and (3) computer-generated summary reports.

3.2 *IN VIVO* DATA

The Helgeson Nuclear Services mobile counter was brought to the site about every 4 months beginning sometime in 1966. The perceived potential for internal exposure was the early basis for including a worker in the *in vivo* bioassay program. Many individual measurement spectra are included with results that identify the measured radionuclide and quantity. Background and calibration measurements were also observed in the records from HNS (1966a, 1969a, 1969b, and 1976).

In addition to the *in vivo* counting by Helgeson, site documentation indicates that GA conducted its own in-house internal dosimetry measurements beginning in 1966. These occurred in Room L-122, where the worker sat in an office chair lined with 0.25-in.-thick lead and a 4- by 4-in. NaI (Tl) detector was in a lap geometry, and appeared primarily to be in response to incidents (GA 1966, GDC 1966-70, Tammany 1973).

Site documentation indicates that workers with suspected intakes were sometimes sent off the site for evaluation. At least two workers in the 1960s with ^{137}Cs intakes were sent to the University of California at Los Angeles. Later documentation from 1975 (Bold 1975) and (Jordan and Kesting 1975) indicates that sometimes personnel who were involved in contamination incidents were sent to the Sacramento Municipal Utility District facility for whole-body counting by Helgeson.

Whole body counts (sometimes noted in GA dosimetry records as “total body count”) are counts performed of a broader area of the body than that for the lung (chest) counts. The whole body counts are performed to detect higher energy gamma emitters (typically fission products), while the chest counts are performed to detect the lower energy x-rays (as is done to detect ^{235}U) in the lung.

The GA dosimetry record entries for in vivo count results typically appear as separate pages on the Internal Radiation Monitoring Report are headed either as “In-Vivo Total Body Count Gross Gamma Scan,” or “In-Vivo Lung Count (^{235}U, natural Thorium, etc.). Both sets of results include the analytical result and the two sigma result for the count. Both sets of summary results typically include the individual count data in addition to the summary results (as appears on the Internal Radiation Monitoring Report). The individual count data sheets additionally typically contain the name, height, weight, date of count, time of count, count identifier, value of % MLPB for positive results, and individual results for the analyzed energy regions-of-interest for the particular count. The Helgeson raw gamma spectra are also typically included in the dosimetry record. The lung count results are presented in mass units (micrograms for ^{235}U, milligrams for natural thorium), and the total body results are presented in activity units (nanocuries) [4].

By August 1970, whole-body counting was performed three times a year by Helgeson (GA 1970). Whole-body counting was performed on all individuals who might have become “internally contaminated with beta-gamma emitters.” The frequency of whole-body counting was determined by the job or exposure hazard index of particular personnel with the maximum frequency of three times per year.

3.3 RADIOELEMENT-SPECIFIC INFORMATION

3.3.1 Uranium

3.3.1.1 General

As noted in Section 1.0, it has been determined that it is not feasible to reconstruct internal radiation doses for GA employees from exposures to uranium before October 1963 (Leavitt 2007a). The uranium at GA included depleted, normal, and enriched uranium. In addition, ^{232}U, ^{233}U, and ^{236}U were used on the site. As discussed in Section 2.3, uranium as depleted, normal, or enriched uranium, or as ^{233}U were used in various locations and at various times on the site; accordingly, if multiple work locations must be inferred for dose reconstruction, then the dose is reconstructed assuming the uranium is either highly enriched, or ^{233}U/^{232}U, whichever is favorable to the claimant. Assuming that the ^{233}U fuel material at GA contained ^{232}U impurity at 8 ppm as discussed in Section 2.3, the ^{233}U to ^{232}U activity ratio would have been 55.7 to 1 [5]. The forms might have included foils, capsules, solutions, metal, or graphite compacts. Recycled uranium would also have been encountered. In general, uranium was in the form of metals or oxides at GA, although scrap recovery operations could have resulted in other chemical forms. Table 3-2 lists half-lives and specific activities for uranium isotopes.

Table 3-2. Uranium isotopic half-lives and specific activities [6].

Radionuclide	Half-life (yr)	Specific activity (pCi/µg)
U-232	72	22,000,000
U-233	160,000	9,800
U-234	240,000	6,300
U-235	700,000,000	2.2
U-236	23,000,000	65
U-238	45,000,000,000	0.34

Table 3-3 provides some factors that might be useful for interpreting uranium results. In general, when no information is provided on the enrichment or isotopic composition of a uranium result, it is favorable to claimants to assume that the material is either highly enriched uranium plus the recycled uranium contaminants, or ^{233}U/^{232}U with an activity ratio of ^{233}U to ^{232}U of 55.7, whichever is more favorable to the claimant. Where uranium urinalysis results are based on gross alpha counting, the uranium enrichment is not needed to account for the total uranium activity in a urine sample. Uranium mass measurements in urine should be interpreted as natural uranium. HEU mass measurements (e.g., pCi per µg) reported for in vivo chest counts are valid.

Table 3-3. Uranium mixtures specific activity and isotopic fractions [7].

Highly enriched (93.5%)	**Activity fraction**	**Mass fraction**	**Activity ratio to U-235**	**Mass ratio to U-235**
U-234	0.96811	0.010606	32.64	0.0113
U-235	0.02966	0.934633	1	1
U-236	0.00197	0.002075	0.0664	0.00222
U-238	0.00026	0.052687	0.00877	0.05637
Useful factors				
U (93.5%) pCi/µg	68.109			
Uranium activity ratio to U-235 mass U (93.5%) (pCi): U-235 (µg)	72.9			
Natural uranium	**Activity fraction**	**Mass fraction**	**Activity ratio to U-235**	**Mass ratio to U-235**
U-234	0.4886	5.37E-05	21.4	0.00745
U-235	0.0228	7.20E-03	1	1
U-238	0.4886	9.93E-01	21.4	138
Useful factors				
U-natural pCi/µg	0.683			
Uranium activity ratio to U-235 mass U (pCi): U-235 (µg)	94.8			
Depleted	**Activity fraction**	**Mass fraction**	**Activity ratio to U-235**	**Mass ratio to U-235**
U-234	0.1546	1.00E-05	14.45	0.00502
U-235	0.0107	1.99E-03	1	1
U-236	0.0005	3.11E-06	0.0467	0.00156
U-238	0.8342	9.98E-01	78.0	501
Useful factors				
DU pCi/µg	0.4021			
Uranium activity ratio to U-235 mass DU (pCi): U-235 (µg)	202			

It does not appear that large quantities of pure ^{232}U were used routinely and continuously by GA employees. As discussed in Section 2.3, during the operational period ^{232}U in solution was being deposited onto disks and then evaporated in gloveboxes at the HCF Plutonium Laboratory (Gurren 1967a,b). According to Gurren (1967b), the discs were approximately 1 x 10^5 cpm alpha and were used for diffusion studies. The latter reports also indicated that there were no contamination problems during the ^{232}U glovebox work so, except where a ^{232}U incident is identified for a claim in the operational period, it is reasonable to make the assumptions in the previous paragraph for dose reconstruction.

Human and animal studies have indicated that oxides of uranium can be very insoluble in the lung (ICRP 1995), which indicates absorption type S (0.1% and 99.9% with clearance half-times on the order of 10 minutes and 7,000 days, respectively). Other *in vitro* dissolution studies of compounds that are found at uranium facilities have shown that oxides of uranium exhibit moderate solubility (Eidson 1994; Heffernan et al. 2001), which suggests absorption type M (10% and 90% with clearance half-times on the order of 10 minutes and 140 days, respectively). *In vitro* dissolution tests on oxides produced from uranium metal during depleted uranium armor penetrator tests have indicated multicomponent dissolution rates, with 25% of uranium dissolving with a half-time of less than or equal to 0.14 day and 75% dissolving with a half-time of 180 days. Because there was no specific information on the solubility of aerosols produced, this analysis assumed that both types M and S were available. The selection of absorption type should depend on the organ of interest. Dose reconstructions should assume International Commission on Radiological Protection (ICRP) Publication 66 default parameters for particle deposition (ICRP 1994).

The scrap processing might have resulted in the formation of more soluble uranium compounds. GA documents include occasional mentions of UF_6 and uranyl nitrate, which are absorption type F, but the quantities of these chemicals were relatively small, and so, unless specific information indicates that type F exposure occurred, type M or S should be assumed.

All of the uranium that was that was handled at GA should be assumed to have been composed of recycled uranium. An estimate of contaminants that might contribute the most to internal doses, based on a review of recycled uranium contaminants at the Hanford and Fernald sites, is listed in Table 3-4. The relative internal dose contributions from ^{99}Tc, ^{228}Th, and ^{232}Th were low enough to be ignored. In addition, the activity fractions assume that the uranium specific activity is based on depleted uranium, which increases the proportion of the contaminants by activity. The contaminant levels for depleted uranium overestimate the contaminants in uranium of normal enrichment by about 40%. Plutonium is assumed to behave as absorption type M or type S material and neptunium as absorption type M.

Table 3-4. Estimated recycled uranium activity fractions for internal dose reconstruction.

Uranium	Pu-239	Np-237
1	0.00246	0.00182

3.3.1.2 Uranium *In-Vitro* Analyses and Data

Some uranium urinalysis results for 1958 were found, but a 1960 memorandum (Andrews 1960) reported the development of a uranium urinalysis procedure to count samples from 18 people who were involved in a December 3, 1959 incident. A reported detection threshold of 1 cph for the 0.5-cph background alpha counter and a 30% uranium recovery reportedly corresponded to a detection threshold of 30 μg/L for the samples. Two of the 18 samples exceeded the counter background, but none exceed 1 cph. The "weights" of the samples (50 to 100 mg/cm^2) exceeded the values on their graphs of sample self-absorption correction factors.

The Industrial Hygiene group activated the bioassay laboratory in early October 1960 and assumed responsibility for beryllium analyses from the Health Physics group. Analytical procedures for enriched uranium, normal uranium, and thorium were reportedly outlined that year. The 1960 Medical Department (Bethard 1960) also noted, "A laboratory technician ... was hired November 7, 1960, and trained in tentative uranium procedures," and 13 urine samples were analyzed for uranium (as of December 19, 1960).

Fires in the Hot Suite of the Experimental Building on March 18 and October 31, 1960, resulted from spontaneous combustion of uranium zirconium hydride (Skeehan 1960); 24-hour urine samples were collected from involved personnel. The March 1960 health physics report states that after the March 18, 1960, fire in the filter plenum of the Enriched Uranium Suite, 48-hour urine samples were collected and sent to U.S. Nuclear Corporation for analysis, and each was found to contain <1 µg/L of uranium (Ray 1960d), which would have been considered a very low sensitivity for that time by fluorimetric methods. However, if the uranium was highly enriched, the sensitivity in terms of activity would have been 68 pCi/L. In the October 1960 incident, 24-hour samples were collected, and might have been counted on the site. The sensitivity of the October urinalysis was given as 10 dpm (4.5 pCi) per 24-hour sample and all results were less than this amount (Bold 1960a).

The June 22, 1961, health physics report stated, “Receipt of a Jarrel-Ash fluorimeter has enabled us to establish calibration curves for fluorimetric determination of uranium” (Bethard 1961b). The September 18, 1961, report (Bethard 1961d) stated, “One urine sample analyzed for uranium contained less than our detectable amount by fluorimetric method (10 µg/L). Radioassay of the sample is in process” (Bethard 1961c).

There might have been a few samples analyzed thereafter, but in September 1963 it was reported that the bioassay program for counting alpha and beta activity in urine had been set up, equipment had been ordered and “one set of samples had been run with apparent success” (Andrews 1963). The October 1963 health physics report included the first reporting of the number of bioassay samples analyzed for the period.

In December 1963, the Health Physics Laboratory Report (Andrews 1964a) stated:

> *The urinalysis procedure for uranium is being changed. The old procedure required that the entire volume of urine be boiled dry with an excess of HNO_3. This resulted in excessive odor and noxious fumes. The new procedure follows the MFP scheme and precipitates the uranium. Most of the urine is then poured down the drain and the precipitate is redissolved and ashed with a minimum of nitric acid. The procedure is essentially unchanged from here on. Extraction of the uranium with ether, back extraction into H_2O and drying in a planchet complete the sample preparation. The use of ether remains a safety hazard.*

Andrews (1964a) also reported that the Eberline MAC with a ZnS(Ag) phosphor in contact with the sample would have essentially the same background and efficiency as the internal proportional counter, and so the MAC was available for counting the urine samples, which would reportedly speed up counting by about 500%. It was noted that the timers on the MAC should be replaced so they could be set with more precision.

The nominal minimum detectable activity for uranium urinalyses after September 1963 is assumed to be 1 dpm/24 hr for gross alpha counting, and 10 µg/L for fluorimetry analyses [8], (Bethard 1961d).

According to an August 1970 GA internal radiation monitoring program description, urine sample specimens from individuals working alpha emitters, tritium and strontium were routinely analyzed while personnel working with other radionuclides were monitored using whole body counting techniques (GA 1970):

> *Bioassay for alpha emitters in urine is done using a gross alpha technique involving co-precipitation of alkaline earth phosphates followed by ether extraction. This procedure is simple and effective for use where enriched uranium is the principle isotope of interest. Plutonium and other alpha emitters will be carried along in this procedure.*

Gross alpha urine sampling frequency varied from weekly to annually depending on an individual's job or "exposure hazard index" and samples were to be the first voiding after normal bed rest.

3.3.1.3 Uranium *In-Vivo* Analyses and Data

Early ^{235}U lung (chest) counting results (1966 to about 1967) were considered "experimental" by HNS (1966a) or "not thoroughly refined" (HNS 1966b) and either were associated with some large uncertainty or assumed to be invalid. Workers were chosen for lung counts based on their potential for ^{235}U exposure and the results of their uranium urinalyses. The early GA ^{235}U measurements are indicators that a worker had the potential for ^{235}U exposure. Helgeson (HNS 1976, p. 8) reported ^{235}U chest count minimum sensitivities of 30 to 60 µg based on 20-minute counts. For some results, 2-sigma counting errors (HNS 1969b) might be in the claimant data, and these could be used to estimate positive results (frequently defined as greater than 3-sigma). .As an alternative, 60 µg is a reasonable choice for the nominal lung (chest) detection sensitivity, unless there are reported specific data to use instead of the nominal value [9].

Uranium-235 chest counts were available for some workers beginning in 1966, but are not considered valid for dose reconstruction purposes until 1968 based on an entry on a whole-body monitoring data card (GDC 1966–1970, p. 69), which states that results are valid using a "new curve fitting technique." In addition, a review of approximately 1,000 whole-body monitoring data cards (GDC 1966–1970) indicates that before January 1968, ^{235}U chest counts were "not valid" (e.g. GDC 1966–1970, pp. 564–567). The reason for the "not valid" determination in the pre-1968 counts is explained by HNS as constant bias in the ^{235}U data on p. 45 of HNS 1967. By mid January 1968, HNS had solved the bias issue in counting for ^{235}U (HNS 1968). If the phrase "not valid" or "not corrected for chest wall thickness" does not appear on the record of a positive in-vivo count performed after 1967, the reported result can be used in the dose reconstruction [10]. By August 1970, ^{235}U at 25% of the lung burden could be detected using *in-vivo* counting (GA 1970).

3.3.2 Thorium

Thorium in the form of oxides, carbides, and nitrates was documented at GA. One of the prime thorium uses was fabrication of thorium-uranium fuel. This document assumes that absorption types M and S were both on the site. Thorium appears to have been on the site before and throughout the operational period. No *in vitro* bioassay results specifically for thorium have been found in the GA records; former employees indicate that thorium *in vitro* bioassays were not performed. *In vivo* bioassays on 12 workers for thorium were performed in March 1966, but results were not included in the report because the measurements were experimental and were undergoing careful review before final results were provided (HNS 1966a). Thorium *in vivo* counting was not mentioned again until May 1969, when HNS (1969a) observed no activity in one count. In October 1975, the thorium chest counts appeared to become more routine.

There does not appear to be a defensible way to ratio thorium intakes to uranium intakes, because some tasks included pure thorium handling. Thorium source terms that were associated with internal exposures would have most commonly been associated with the Fuel Fabrication Facility, the scrap handling facility, the laboratory buildings, the Experimental Building High Bay and HTGR critical assembly, and the waste handling areas.

Air samples were measured in alpha counters with no differentiation between radionuclides. Gross alpha air concentrations in the Fuel Fabrication Facility thorium work areas ranged from 1.8 × 10^{-12} to 9.6 × 10^{-9} µCi/ml alpha plus beta during parts of July, August, and October 1966.

Air concentration controls were based on enriched uranium concentration limits, because GA concluded that in most areas, although the thorium air concentration limits were lower, thorium was only a small component of the radioactive source term. This probably was true in most areas; however, there appear to be some exceptions.

No bioassays specific for thorium exposures have been found for the GA covered operational period. In later years, chest counts were used to identify thorium intakes, but care would have to be taken to address the actual period of exposure. These later chest counts, when available, can be used to estimate intakes of thorium based on a worker’s individual history.

Consideration was given to using later chest counts to estimate coworker exposures, but availability of records and their representativeness especially for earlier exposures was deemed questionable.

As noted in Section 1.0, it has been determined that it is not feasible to reconstruct internal radiation doses for GA employees from exposures to thorium during the period of January 1, 1960, through December 31, 1969.

3.3.3 Plutonium

There is insufficient information to characterize Pu in the breathing zone air for workers during the operational period. This section is provided to characterize the operational period Pu source term, since that information is needed to assist in developing intake estimates during the Residual and Remediation periods (Sections 5 and 6).

Much of the plutonium that was used at GA was in the form of sealed sources, but unencapsulated plutonium was also used. Records indicate that reactor-grade plutonium and heat sources were handled as well as ^{239}Pu foils (assumed to have the composition of weapons-grade plutonium), and plutonium that was associated with irradiated fuel examination at the HCF. In addition, ^{241}Am would have been associated with some of these source terms.

Attachment E contains a summary of the only available workplace monitoring data at GA during plutonium-specific work during the operational period.

Chemical forms of plutonium at GA included metals, oxides, and nitrides. Table 3-5 contains the isotopic composition of plutonium by activity from the HCF based on the analysis of representative debris collected from the HLC in 1994 (Sills 1994). Table 3-6 contains the theoretical isotopic composition of unirradiated plutonium grades that were handled at GA.

3.3.4 Strontium

It appears that strontium analyses involved chemically removing the strontium from the urine sample, counting the new sample, and counting again when the ^{90}Y has started to grow back in to determine the activity of ^{90}Sr. Strontium could have been associated with the irradiated fuels, or the ^{90}Y production facility (about which little is known), as well as other programs. A 1971 HCF report indicates that a ^{90}Sr source had been loaded into a generator housing and transferred to the SV-D Building (Bedsaul 1971). Specific information on chemical form or forms of ^{90}Sr at GA is unavailable,

Table 3-5. HLC alpha sampling results.[a]

Radionuclide	Sample alpha concentration (µCi/g)	Relative alpha fraction
U-232	4.50E-01	4.94E-02
U-233	3.80E-01	4.17E-02
U-234	7.90E-02	8.66E-03
U-235	8.00E-04	8.77E-05
U-236	1.20E-03	1.32E-04
U-238	5.00E-04	5.48E-05
Th-228	3.10E-01	3.40E-02
Th-229	2.40E-02	2.63E-03
Th-232	2.00E-03	2.19E-04
Pu-238	5.70E+00	6.25E-01
Pu-239/240	5.00E-01	5.48E-02
Pu-241	5.50E-04[b]	6.03E-05
Pu-242	ND[c]	
Am-241	7.10E-01	7.79E-02
Am-243	1.60E-02	1.75E-03
Cm-242	2.00E-03	2.19E-04
Cm-243	4.00E-03	4.39E-04
Cm-244	9.20E-01	1.01E-01
Cm-245	ND[c]	
Cm-246	ND[c]	
Cm-247	1.00E-03	1.10E-04
Cm-248	5.00E-04	5.48E-05
Cm-250	1.40E-02	1.54E-03
Cf-248	8.00E-04	8.77E-05
Cf-250	3.00E-04	3.29E-05
Cf-252	1.00E-03	1.10E-04
Total	9.12E+00	1.00E+00

a. Adapted from Sills (1994).

b. Based on 0.0022% alpha decay (which would have been detected by gross alpha air sample analysis) from 25 µCi/g of Pu-241 determined by beta analysis (Sills 1994).

c. ND – not detected

Table 3-6. Isotopic composition by weight and activity percent of unirradiated plutonium grades.[a]

	Heat source/thermoelectric generator		Unirradiated reactor grade fuel		Weapons grade (Pu-239 foils)	
Isotope	Weight %	Activity %	Weight %	Activity %	Weight %	Activity %
Pu-238	90.0	99.8	1.5	2.12	0.05	1.73
Pu-239	9.1	0.0365	58.1	0.297	93.6	11.8
Pu-240	0.6	0.0088	24.1	0.451	6	2.76
Pu-241	0.03	0.2	11.4	97.1	0.4	83.7
Pu-242	0.01	0.000003	4.9	0.016	0.05	0.0004

a. Adapted from DOE (2006).

although strontium that was associated with the irradiated fuels is likely to be in the form of an oxide and not titanate; but, if strontium was being used for the production of nuclear batteries (thermoelectric generators), it might have been in the form of $SrTiO_3$ (CBWI 2006; OTA 1994; Grissom 1994). If a thermoelectric generator cannot be ruled out as the source of strontium being assessed, the larger organ dose between the following options should be assigned: (1) all ^{90}Sr (as material type S), or (2) strontium as part of an ORAUT-OTIB-0054 (ORAUT 2007a) evaluation, which is reactor-based and in which strontium is assessed as material type F.

3.3.5 Tritium

The possibility of counting urine samples for tritium and maybe fission products “by using a plastic phosphor wet with urine” was mentioned in 1964 (Andrews 1964a). The March 1965 health physics report (Bold 1965c) noted that the bioassay schedule was updated to include all personnel working in the Activation Analysis area (it is not clear if bioassay was for tritium or perhaps other radionuclides)(Cockle 1965a). In October 1965, personnel using the Bunker Facility were sampled routinely for tritium, and the samples were analyzed by an outside service (Cockle 1965a). In November 1965, all personnel who were associated with the bunker were on bioassay for tritium (Cockle 1965b).

As noted in Section 1.0, it has been determined that it is not feasible to reconstruct internal radiation doses from exposure to tritium before October 1965.

By August 1970 urinalysis for tritium was performed using liquid scintillation (GA 1970). Routine urine sampling at 1-month intervals for tritium was performed for individuals who regularly worked with 0.1 Ci or more of tritium. Individuals working with 10 Ci or more were to be sampled the following day.

3.3.6 Mixed Fission and Activation Products

3.3.6.1 General

Urine samples that contained mixed fission and activation products were likely placed in a container and gamma-counted, or evaporated to dryness and beta-counted. The radionuclides were typically not identified, and the results were reported in terms of gross activity or gross beta and adjusted to a 24-hour time, if a 24-hour sample was not provided or analyzed.

At some point, maybe in 1963, the urine samples were chemically processed and then beta-counted, but it is not known if the gamma-counted results were chemically processed or if results that were designated as MFPs or beta represent the same or different analysis types.

Because of the variety of activities ongoing at GA, there is much uncertainty about which radionuclides to attribute these gross results. Long-lived beta/gamma emitters were typically not identified in the early GA source term, probably because the short-lived activity tended to dominate.

Following a ^{134}Cs hand contamination incident on April 23, 1964, in Sorrento Valley, urine was collected immediately and from every voiding for 48 hours after the incident. Graphs of the excretion rate as a function of time showed that most of the ^{134}Cs showed up in the urine the first day. Subsequently, single specimens were collected once a week. Additional samples were collected 1 day after the incident and showed little activity due to ^{134}Cs, even though they were counted for 1 hour on a 256-channel analyzer. “The rapid excretion of Cs-134 is probably due to the versene soak which complexed the cesium and got it quickly into the bloodstream” (Andrews 1964b). The intake was modeled as an ingestion intake.

For this incident, urine specimens were wet-ashed with nitric acid and hydrogen peroxide; then the salts were transferred to a planchet and counted in contact with the 2- by 2-in. NaI(Tl) crystal for 50 minutes. Background corrections were made, but no correction was made for ^{40}K in the sample. Count rates for all but the first few samples were on the order of 10 cpm. All counts were compared to a ^{134}Cs standard that was prepared by the Analytical Chemistry group. Decay corrections were made to correct the sample count to the time at excretion. The lowest urine activity for April 26, 1964, at 2100 hours to April 27 at 0800 hours was 54 pCi/24 hr and results ranged from 54 to 1,000 pCi/24 hr.

From reviews of bioassay cards and urine calculation sheets during the operational period, a nominal range for the minimum sensitivity of a gross beta result (without normalization to time or urine volume) would be 5 to 10 pCi, and the nominal minimum sensitivity for a mixed fission product result would be on the order of 70 pCi [11] (GDC 1965a, GDC 1965b, GA 1963-1967, GA 1963-1970).

3.3.6.2 Applicability of ORAUT-OTIB-0054 to Experimental High-Temperature Gas-Cooled and Reduced-Enrichment Research and Test Reactor Fuels

Five basic categories of nuclear reactors were considered in the development of ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2007a) to represent sources of mixed fission and light activation products that were encountered by workers at DOE and contractor sites. Table 5-1 of ORAUT-OTIB-0054, reproduced below, lists the specific reactors that were considered and their corresponding category (ORAUT 2007a).

The fuels and reactors covered a range of enrichments from natural uranium to highly enriched uranium, fission spectra from highly thermalized to fast, and a variety of cladding and structural materials. Both ^{235}U and ^{239}Pu fuels were considered. The two TRIGA reactors at GA were among the reactors ORAUT-OTIB-0054 considered.

Reactor description	Category
Hanford N-Reactor	Plutonium production reactors
Hanford Single-Pass Reactors	Plutonium production reactors
Fast Flux Test Facility	Experimental sodium-cooled reactors
Advanced Test Reactor	Advanced test reactors
TRIGA Reactor (aluminum-clad fuel)	Research reactors
TRIGA Reactor (stainless-steel-clad fuel)	Research reactors
Pressurized waster reactor	Generic reactor

Some of the operations at the HCF and Experimental Critical Facility involved irradiated, experimental HTGR fuels for a thorium fuel cycle and irradiated fuel for the Reduced-Enrichment Research and Test Reactor (RERTR) program (Section 5.2.9). Because the experimental HTGR and RERTR fuels are not specifically mentioned in ORAUT-OTIB-0054, this evaluation of the applicability of ORAUT-OTIB-0054 to GA dose reconstructions is provided. The experimental RERTR fuel was a prototype based on that for TRIGA reactors and was essentially the same as the TRIGA reactor fuel that was considered in ORAUT-OTIB-0054 (ORAUT 2007a), as evidenced by information in Tables 2.3 and 2.4 of GA (2002) and in Section 4.1 of GA (1994). The experimental RERTR fuel is therefore not considered further in this evaluation.

The experimental thorium-bearing HTGR fuels are described in Tables 2.1 and 2.2 of GA (2002) and in Section 4.1 of GA (1994). A more detailed description of these fuels is provided in Section 7.3 of the second edition of *Nuclear Chemical Engineering* (Benedict, Pigford, and Levi 1981). The HTGR design employs two types of fuel, both in the physical form of microspheres. The microspheres are embedded in a graphite matrix to form fuel rods. One type of microsphere consists of enriched uranium carbide; the other type is thorium dioxide.

It appears from Section 4.1 of GA (1994) that other, similar uranium or thorium compounds were being evaluated, but the overall fuel cycle is unchanged: initially the reactor is powered solely by ^{235}U, with the fissile material transitioning from ^{235}U to ^{233}U as the ^{235}U is consumed and ^{233}U is bred from the thorium.

It is not known if the experimental fuel samples that were handled at GA were irradiated for long enough for there to be significant breeding and fission of ^{233}U, but this has no impact on the applicability ORAUT-OTIB-0054. The thermal fission yield curves for ^{235}U and ^{233}U do not differ to a significant degree, and differ less than those for ^{235}U and ^{239}Pu (for example, see Benedict, Pigford, and Levi 1981, Table 2.9 or Figure 2.12). The latter consideration is already addressed in ORAUT-OTIB-0054, so the fact the irradiated fuel materials that were handled at GA might have included fission products from the fission of ^{233}U does not proscribe the use of ORAUT-OTIB-0054 for estimating intakes for workers who were exposed to these materials. The mix of fission products in the irradiated HTGR fuel materials would not have differed to any appreciable extent from those that were considered in ORAUT-OTIB-0054.

The fact that irradiated, experimental HTGR fuel materials were handled in specific facilities at GA does not limit the applicability of ORAUT-OTIB-0054 for the evaluation of gross beta or gross gamma analyses. This is true regardless of the extent to which any contamination that is associated with the experimental HTGR fuels contributed to the overall mixed fission and activation product contamination in the facilities of concern. ORAUT-OTIB-0054 may be applied for these facilities in the same manner as it would be applied for other facilities where the mixed fission and light activation product source term stemmed from irradiated nuclear fuels.

3.3.6.3 Internal Exposure Assessment of Mixed Fission and Activation Products via ORAUT-OTIB-0054 Guidance

To determine chronic intakes from nonspecific radionuclide analyses, each urinalysis result should be multiplied by the fraction of activity from one of the appropriate indicator radionuclides in Table 7-1 or Table 7-2 of ORAUT-OTIB-0054 (ORAUT 2007a), which provides the urine activity ratio of the indicator radionuclide for use in intake analyses. Table 7-1 fractions should be used for beta-counted urine samples that have not been chemically processed. Table 7-2 should be used for beta-counted samples that were chemically processed to remove certain radionuclides.

In the absence of any information about the urine sample processing at GA during the operational period, it is favorable to the claimant to assume that the analysis is strontium; this results in the largest strontium values on which other radionuclides are based. The fractions for this type of analysis are shown in Table 3-7 below.

Table 3-7. Urine activity fraction for bioassay codes PF, MFP and β or gross beta.[a]

Radionuclide	Average fraction of Sr-90 activity in Sr urine sample			
Time since last fuel irradiation	10 d	40 d	180 d	1y
Sr-90	0.045	0.061	0.23	0.75

a. Adapted from ORAUT 2007a, Table 7-1.

b. The presence of radioiodides and naturally occurring beta emitters in urine will lead to an overestimate of intake.

The decay time or time since the last fuel irradiation in Table 3-7 above refers to the mix of contamination to which workers would have been exposed rather than the age of the fuel itself. Table 3-8 below provides additional guidance on the choice of the decay time for various activities at GA.

Table 3-8. Fission and activation product decay times for GA facilities and/or activities.[a]

Activities	GA Facility	Decay time
Reactor operations, fuel examination	Bldg. 21 TRIGA reactors, Bldg. 23 HCF hot cells, Bldg. 2 Science labs, Bldg. 27 EA 1 labs, Bldg. 27-1 # 1 bunker, Bldg. 30 Linac complex, Bldg. 31 HTGR-TCF critical experimental facilities	10d
Fuel dissolution	Bldg. 23 HCF hot cells	180d
Waste management	Bldg. 25 Waste yard/Incinerators	1 y

a. Adapted from ORAUT 2007a, Table 5-12.

3.3.7 Thoron

Specific monitoring and reporting of thoron (^{220}Rn) were not done at GA until after the operational period. The earliest identified thoron measurements are October and November 1975 measurements of thoron concentrations that range from 41.4 to 126 dpm/L in the Thorium Pilot Plant Fume Scrubber 2 effluents during operations and airflows of 30 to 40 lpm (McDonald 1975). The background concentration was reported as <0.05 dpm/L when the Pilot Plant was not operating.

Given the fairly routine references to contamination on air samples from thoron progeny in the fuel fabrication area and the lack of information about what activities took place at the Thorium Pilot Plant, it seems unreasonable to assume that these later Pilot Plant measurements are useful for estimating thoron concentrations during the operational period in areas where large quantities of thorium were stored and handled. However, the following calculations are provided to give a sense of the magnitude of dose for 2,000 hours of exposure to thoron at the reported levels.

To determine exposure due to a specified thoron concentration (assumed to be ^{220}Rn), the concentration *C* in picocuries per liter was converted to potential alpha energy concentration (*PAEC*) in units of working level (WL) using:

$$PAEC = C \times F \div (7.47\ pCi/L\ per\ WL) \tag{3-1}$$

where the equilibrium factor *F* was taken as 0.8, a value recommended by NIOSH (2006b). Using the reported 1975 range of thoron concentrations above that were measured for the Pilot Plant, the exposure would range from 2.0 to 6.1 WL. Potential alpha energy exposure (*PAEE*) in units of working level month (WLM) was obtained from *PAEC* and the number of months of exposure *M* using:

$$PAEE = PAEC \times M \tag{3-2}$$

where a month is assumed to be 170 work hours, which results in 24 to 73 WLM/yr using the Pilot Plant 1975 values (background would be less than 0.29 WLMs). Using the factors of 7.61, 1,750, and 2.90 rem/WLM for lung, ET1, and ET2 (NIOSH 2006b) results in dose ranges of 183 to 555, 42,000 to 128,000, and 69.5 to 212 rem/yr, respectively. Background would be <0.22, <51, and <0.084 rem/yr, respectively, for thoron exposure to each organ.

3.4 OTHER RADIONUCLIDES

Although licensing documentation (see Attachment C) indicates that GA was licensed to possess radionuclides such as $^{170/171}$Tm, it is not clear whether GA personnel were ever exposed to $^{170/171}$Tm. Site documentation from the operational period at GA also indicates that ^{228}Ra and ^{224}Ra were off-gassed during thorium heats in the thorium conversion furnace at the FFF (Bold 1967c, Bold 1967d, and Ray 1958b) which were not accounted for in the bioassay program, and it is not clear what the exposure potential was.

For beta or gamma emitting radionuclides other than those discussed elsewhere in Section 3, bioassay results (particularly in vivo) specific to those radionuclides can be used for dose reconstruction. Alternatively, the favorable to claimant estimates in ORAUT-OTIB-0054 (ORAUT 2007a) can be used to assess organ dose due to beta emitters, as described in 3.3.6.3.

NIOSH has determined, and the Secretary of the U.S. Department of Health and Human Services (DHHS) has concurred, that it is not feasible to reconstruct internal radiation doses from exposures to thorium for General Atomics (GA) employees during the period from January 1, 1960, through December 31, 1969 (Leavitt 2007). Consistent with that SEC determination relative to thorium exposures during the operational period, no methodology is proposed for reconstructing doses due to any intakes of alpha emitting radionuclides other than those discussed elsewhere in Section 3.

3.5 INTERNAL DOSIMETRY SUMMARY

Because of the SEC determination (Leavitt 2007) that it is infeasible to adequately reconstruct internal dose during the AWE operational period covered by the SEC (January 1, 1960 through December 31, 1969), dose estimates for this period are considered partial dose estimates.

For workers who had uranium bioassay after September 1963, estimates of intakes can be made. The uranium urinalysis detection threshold is assumed to be 1 dpm/24 hr for gross alpha counting, and 10 µg/L for fluorimetry analyses [8], (Bethard 1961d).

Uranium-235 chest counts were available for some workers beginning in 1966, but are not considered valid for dose reconstruction purposes until 1968 based on an entry on a whole-body monitoring data card (GDC 1966–1970, p. 69), which states that results are valid using a “new curve fitting technique.” If the phrase “not valid” or “not corrected for chest wall thickness” does not appear on the record of a positive in-vivo count performed after 1967, the reported result can be used in the dose reconstruction [10].

For bioassay results in the operational period that were reported as gross alpha, it is favorable to the claimant to assume that the gross alpha result is attributable to the alpha-emitting radionuclide (^{234}U or ^{233}U) that results in the greatest organ dose, unless there is documentation that indicates that the gross alpha result is representative of only one radionuclide (or radioelement). For bioassay results that were reported as gross beta, it is favorable to the claimant to assign radionuclide-specific intakes of mixed fission and activation products using the methods in ORAUT-OTIB-0054 (ORAUT 2007a), using ^{90}Sr as the indicator radionuclide that document describes, as discussed in Section 3.3.6.3 of this document.

For an individual with bioassay data and on a case-by-case basis using individual bioassay data, it might be possible to estimate intakes and associated doses for some periods and some radionuclides. Although NIOSH is unable to adequately estimate total internal exposures for members of the [SEC] class, internal dose due to intake of uranium can be reconstructed for exposures starting in October 1963, and tritium internal doses can be estimated after September 1965. Because of uncertainties in the exposure periods, the source terms, and the analysis methods as documented in the SEC determination (Leavitt 2007), these estimates are to be considered partial dose estimates for the SEC period January 1, 1960, through December 31, 1969.

4.0 ESTIMATING EXTERNAL EXPOSURE

External whole-body dosimetry results are available for monitored GA workers from January 1960 through 1999. Beginning sometime before 1960, film badge dosimetry services were provided by R. S. Landauer, Jr. and Company; in July 1961, Radiation Detection Company replaced Landauer. Site documentation indicated that dosimetry services from Landauer resumed in 1965. Personnel who worked in areas where potential neutron exposure could occur were issued film badges sensitive to beta, gamma, and neutron radiation. Otherwise, personnel were issued film badges sensitive to only beta and gamma radiation. Beginning in 1981, film was replaced by thermoluminescent devices (TLDs) for monitoring external exposure.

Some details on the Radiation Detection Company dosimeters are available from site documentation (i.e., reported nonpenetrating and penetrating doses) which implies that they were comparable to Landauer dosimeters. The nearby location of the Radiation Detection Company in Palo Alto, California, and faster turnaround time for results were the primary factors in the choice to use

Radiation Detection Company. Film badge detection thresholds for Landauer badges were reported in January 1961 as 10 mrem from gamma, 40 mrem from beta and 20 mrem from fast neutrons. Other documentation indicates that film badge detection thresholds in the early days were assumed to be as high as 50 mrem for gamma and 60 to 80 mrem for beta. A detection threshold of 40 mrem for photons has been recommended for this project (ORAUT 2006b). This document assumes a detection threshold of 40 mrem for gamma, beta, and neutrons.

In addition to the film badge dosimeters, personnel used direct or indirect reading dosimeters as early as 1959, such as personal or pocket ionization chambers (PICs). Early on, the procedure required that two PICs be worn in addition to a film badge, later that was cut back to one, and later it was rare to use direct-reading dosimeters (only around low-energy X-rays). Wrist and/or ring dosimeters were also used in some operations, for instance during removal of irradiated samples from the reactor. Results from the dosimeters were logged by employees on a routine basis, but at this time it is not clear if these onsite readings are included in the individual dosimetry records. Through the early 1960s, doses were reported only for betas of energy greater than 1.5 MeV, according to the explanatory notes in the early reports of external dosimetry (e.g., p. 2 of both Landauer 1959 and Landauer 1961).

Many of the various direct- and indirect-reading pocket dosimeters from Landsverk Electrometer Company were used, including Landsverk Models L-49, L-50, L-65, and L-69. The Model L-49 was a direct-reading dosimeter responsive to thermal neutrons and gamma radiation up to 200 mrem. The Model L-50 was a direct-reading dosimeter responsive to gamma radiation up to 200 mrem. There was also a Model L-50 ring dosimeter that employed an indirect-reading chamber designed to measure hand exposures up to 2 R that might have been used at the GA site. The Model L-65 indirect-reading pocket dosimeter was responsive to gamma radiation up to 200 mR (ORAU 2006). According to site documentation, Model L-69 dosimeter was used to measure total gamma plus thermal neutron dose (Ray 1959).

Site documentation indicates that film badges, fast and thermal neutron detection equipment, and direct- and indirect-reading pocket dosimeters were irradiated to evaluate the radiation each device recorded in the 1959 to 1961 period. Beginning in early 1960, indium foils were incorporated into personnel security badges in case of a criticality event (Ray 1960e). By July 1961, all personnel film holders were equipped with a built-in threshold detector for neutron dose evaluations in the event of a criticality (Bold 1961b). As a consequence, the previous practice of routinely issuing Landsverk L-65 and L-69 PICs in addition to a film badge was discontinued by August 1961 because thermal neutron exposures could be determined from the film badge by use of a cadmium filter over the gamma-sensitive film. Nevertheless, GA continued the practice of issuing Model L-49 self-reading dosimeters to visitors and selected workers throughout the 1960s.

Site documentation from 1960 indicates that wrist badges were issued due to a lack of available finger rings (Ray 1960f). Correspondence from April 1966 indicates that finger ring exposures were expected to be at least 5 times greater than self-reading wrist dosimeter indications (Gurren 1966c). A California Division of Industrial Safety inspection of the X-ray equipment and LINAC included mention that X-ray and diffraction spectrograph operators were to wear finger tab badges for 3 weeks each year, presumably to provide some assurance that radiation exposures were not routinely exceeded (Bethard 1961e).

The variety of external radiation sources that were present at the GA site resulted in a wide range of energies from penetrating and nonpenetrating radiations. Little information is available to quantify the precise distributions of photon and neutron energy ranges. For the purpose of expediting dose reconstruction, the assumption favorable to claimants is made that workers were exposed to photon energies from 30 to 250 keV and neutron energies from 0.1 to 2 MeV. Nonpenetrating dose is assumed to consist primarily of electrons or of photons less than 30 keV. For consistent presentation, exposure or dose is reported as:

- Penetrating, assumed to be associated with photons of energies 30 keV or greater,
- Nonpenetrating, assumed to be associated with photons of energies less than 30 keV or with electrons, and
- Neutron, assumed to be associated with neutrons of energies of 0.1 to 2 MeV.

Because a considerable amount of work at GA occurred in gloveboxes and hoods, correction factors from OCAS-TIB-0010 (NIOSH 2005) should be used to determine the dose to organs in the lower abdomen for which the dosimeter measured dose may not be reasonably accurate unless it is clear that an individual's exposure would not have occurred in this manner.

If it is known that a worker only received external dose from working with plutonium, the guidance on correction factors provided in Section 4.2.3 of NIOSH 2007b should be used for dose reconstruction.

Badge exchange frequency at GA in 1960 and 1961 was bimonthly (Bold 1962). The badge exchange frequency from 1962 through 1969 was monthly (Bold 1968c). From 1996 through 1999, the badge exchange was quarterly. The exchange frequency was probably monthly between 1969 and 1996.

A summary of GA personnel dosimetry by type, provider, period of use, exchange frequency, and minimum detection level (MDL) is provided in Table 4-1.

Table 4-1. Dosimeter type, period of use, exchange frequency, MDL, and potential annual missed dose.

Dosimeter type/ provider	Period of use	Exchange frequency	MDL (rem)[a,b]	Max annual missed dose (rem)[c]
Film badge/ Landauer	Pre-1960–July 1961	Bi-monthly (n=24)	0.010 gamma, 0.040 rem beta, 0.020 rem neutron	0.960
Film badge/ Radiation Detection Company	July 1961–1965	Monthly (n=12)	Up to 0.050 rem gamma, Up to 0.080 rem beta	0.480
Film badge/ Landauer	1965–1981	Monthly through 1969, likely monthly through 1981 (n=12)	NA[d]	0.480
Thermoluminescent dosimeter/NA	1981–present	Likely monthly through 1996, Quarterly from 1996	NA	0.480

a. Reported MDLs as summarized in Section 4.0.
b. MDLs assumed for gamma, beta, and neutron: 0.040 rem (ORAUT 2006b).
c. (Annual exchange quantity) × (0.040 rem).
d. NA = not available.

Very few individual monitoring data on either Landauer or Radiation Detection Company original reports have been found. External monitoring data for workers at GA were maintained on an IBM computer system as of January 7, 1961, and these records along with a few AEC Form 5 records in claimant files can be used to reconstruct external dose to workers at GA beginning January 7, 1961. Based on a review of claimant files the only record of external monitoring data before January 7, 1961, are AEC Form 4 records with only the external, whole-body, penetrating dose that was received at GA from the beginning of initial monitoring to January 6, 1961. An approach that is favorable to claimants is to assign the entire dose from before January 7, 1960, to 1960. This dose can be assigned as either gamma or neutron, whichever is more favorable to the claimant. In addition, available records do not always allow determination of whether an individual GA worker was monitored for gamma and beta exposure or gamma, beta, and neutron exposure. Blanks or zeroes in the dosimetry database do not necessarily relate to whether a person wore a neutron badge; as a consequence, missed neutron dose should be assigned when only summary records are available.

Claimant files do not appear to contain any extremity monitoring data for January 1, 1960, through January 6, 1961, and film badge reports from Landauer from January 1961 through June 1961 indicate that the finger ring doses were reported by Landauer in “arbitrary units.” During the use of two-element film badges, the term *arbitrary units* was used to report an open-window dose. This term was an acknowledgement that the open-window reading could not be accurately interpreted without knowledge of the workplace radiation type and energy (ORAUT 2006a). It is possible that the arbitrary units were entered as extremity doses in the database before July 1961; if so, these results should not be directly used to reconstruct dose. Radiation Detection Company badges were used beginning in July 1961, and reported extremity results were in millirem.

Form AEC-190 summaries of GA-reported external penetrating doses for 1960 through 1964 and 1967 are listed in Table 4-2 and provide a general sense of the program size. The reported doses do not include nonpenetrating and extremity doses. These summaries, which were prepared for the AEC Headquarter Health and Safety Division, are expected to include significant beta, photon and/or neutron occupational dose from GA accelerator operations because of the intermingling of radiation sources at GA.

Table 4-2. External penetrating dose summaries for 1960 through 1964.

	Number of workers in each monitoring/dose category								
Year	**Not monitored**	**Monitored**	**0–1 rem**	**1–2 rem**	**2–3 rem**	**3–4 rem**	**4–5 rem**	**5–6 rem**	**6–7 rem**
1960[a]	1,389	209	194	4	6	0	4	0	1
1961[b]	1,388	299	290	7	1	0	1	0	0
1962[c]	1,500	341	336	4	1	0	0	0	0
1963[d]	1,656	585	561	16	4	4	0	0	0
1964[e]	1,559	610	580	22	6	2	0	0	0
1967[f]	–	576	562	8	5	1	0	0	0

a. Hughey (1961).
b. Hughey (1962).
c. Jackson (1963).
d. Jackson (1964).
e. Bird (1965).
f. Bold (1968d, p. 66).

4.1 DOSE RECORD FORMAT

This section discusses the external dose record format in claimant files for monitored GA personnel between January 7, 1961, and December 31, 1969 (the records before this are in summary form). Each page of external monitoring results contains a file number, employee name, Social Security Number, and birth date at the time of data entry. The birth date was used to calculate allowable lifetime dose as shown in the PERMISS column in accordance with the regulations in effect at the time. The interpretation of the remaining columns is based on correspondence from the current GA Radiation Safety Officer. Table 4-3 lists the codes for the DOS columns. Each page contains only one DOS code; as a consequence, there are three different pages for each monitoring period that document (1) whole-body, (2) skin, and (3) extremity doses. For doses after the third quarter of 1961, each page contains the quarterly monitoring results for whole-body or skin or extremity doses. Even though the doses in the last quarter of 1961 could be presented as monthly, site documentation indicates that the badge exchange frequency was weekly in 1960 and 1961.

Table 4-3. DOS column code.[a] [12]

1	**Whole body**
2	Skin
3	Extremity

a. There is only one DOS designation per page.

The DATE column contains either the month (01 through 12) or a 2-week monitoring period in 1961 followed by the last digit of the year. From January 7, 1961, to January 5, 1962, the first two-digit numeral in the DATE column is interpreted according to Table 4-4, which presents the 2-week monitoring periods for 1961 according to site documentation (GA ca. 1961). A review of claimant files indicates that by the fourth quarter of 1961 monitoring data was reported quarterly rather than weekly, as listed in Table 4-4. After January 5, 1962, the first two-digit numeral in the DATE column designates the month with 3 months per quarter on each page. After a space, the last digit in the DATE column is the last digit of the year (i.e., 1961 through 1969 are designated by the numerals 1 through 9).

Table 4-4. Exposure periods for monitoring devices for 1961[a] (GA ca. 1961).

1st quarter			**3rd quarter**		
Period no.	**From**	**Through**	**Period no.**	**From**	**Through**
01	1/07/61	1/20/61	14	7/08/61	7/21/61
02	1/21/61	2/3/61	15	7/22/61	8/04/61
03	2/04/61	2/17/61	16	8/05/61	8/18/61
04	2/18/61	3/3/61	17	8/19/61	9/1/61
05	3/4/61	3/17/61	18	9/2/61	9/15/61
06	3/18/61	3/31/61	19	9/16/61	9/29/61
07	4/1/61	4/14/61	20	9/30/61	10/13/61
2nd quarter			**4th quarter**		
Period no.	**From**	**Through**	**Period no.**	**From**	**Through**
08 or 1	4/15/61	4/28/61	21	10/14/61	10/27/61
09 or 2	4/29/61	5/12/61	22	10/28/61	11/10/61
10 or 3	5/13/61	5/26/61	23	11/11/61	11/24/61
11 or 4	5/26/61	6/9/61	24	11/25/61	12/8/61
12 or 5	6/10/61	6/23/61	25	12/9/61	12/22/61
13 or 6	6/24/61	7/07/61	26	12/23/61	1/05/62

a. 2-week periods from Saturday 0001 to Friday 2400.

Table 4-5 lists the MON column codes. It is uncertain what the numeric codes 3, 4, 8, and 9 mean.

Table 4-5. MON column codes [12].

0	**TLD**
1	Film
2	Ring
3	Unknown
4	Unknown
5	Unknown
6	Calculated dose
7	Not monitored (in the 1980s, a 7 was also used to indicate a dose received before monitoring at GA)
8	Unknown
9	Unsure, but there is some indication that 9 was used to indicate mixed information such as recorded dose plus an estimated dose result

The GAMMA, BETA, and NEUTRON columns indicate dose in rem.

4.2 FILM BADGE CORRECTION FACTORS

Because very little neutron spectral information is available, this analysis assumed that reported neutron doses were calculated using the quality factors (QFs) from National Committee on Radiological Protection and Measurements (NCRP) Report 38 (NCRP 1971). The Report 38 QFs were averaged over the ICRP Publication 60 (ICRP 1991) energy groups. A dose multiplier of 2.0 (actually 1.91 in the technical information bulletin ORAUT-OTIB-0055, *Technical Basis for Conversion from NCRP Report 38 Neutron Quality Factors to ICRP Publication 60 Radiation Weighting Factors for Respective IREP Input Neutron Energy Ranges* (ORAUT 2006d)) is used for conversion of the NCRP Report 38 dose equivalent to the ICRP Publication 60 equivalent dose for the 0.1- to 2-MeV energy range (ORAUT 2006a).

According to GA site documentation, the nuclear track emulsion, type A (NTA) Landauer badges that were used at GA in the 1960s had a neutron energy threshold of 1 MeV. Other documentation indicates that NTA film was a poor detector of neutron energies below 500 to 800 keV (ORAUT 2006a). Table 4-6 contains neutron dose correction factors for NTA film that was developed for dose reconstruction at Argonne National Laboratory-East (ANL-E) to account for <1-MeV neutrons. The facilities and activities at ANL-E appear to encompass the activities and facilities at GA. For NTA film: an unmonitored neutron dose correction factor of 1.5 is recommended for GA due to the presence of the LINAC, uranium-fueled critical assemblies, and the TRIGA reactors; if work handling plutonium is indicated (e.g., certain hot cell work), an unmonitored neutron dose correction factor of 4 is recommended. As a consequence, the recorded neutron doses from film for personnel at GA should be multiplied by a factor of 3 or 8 (as indicated by work location) to account for changes in QFs and NTA film energy threshold.

Table 4-6. Neutron dose correction factors.[a]

Facility	NTA film neutron dose correction factors[a]	ICRP 38 to ICRP 60 adjustment factor	Total neutron dose correction factor
Accelerators	1.5	2	3
Reactors and critical assemblies	1.5	2	3
Pu handling facilities	4	2	8

a. Adapted from ORAUT (2006a).

GA site documentation also indicates that the Landauer badges used in the early 1960s at GA had a beta energy reporting threshold on the order of 1.5 MeV (Landauer 1959, Landauer 1961), which would have resulted in an underreporting of shallow dose equivalent to lower energy beta radiation. An estimated fraction (0.49) of reported dose has been calculated for similar ANL-E two-element badge designs that were used from 1945 to 1959 (ORAUT 2006a). As such, for exposures to MFPs, which contain mixed photon and beta radiation, an upper bound to the recorded nonpenetrating dose from film badge data can be estimated by multiplying the beta dose by 2.04 based on Table 6-19 of ORAUT (2006a), which indicates fraction of reported dose as 0.49. However, if a GA employee worked in the early 1960s with a source term that included primarily beta-emitting radionuclides with a maximum beta energy of less than 1.5 MeV (e.g., $^{170/171}$Tm), the personal beta dose would not have been accurately reported.

4.3 OCCUPATIONALLY REQUIRED MEDICAL X-RAYS

According to 1961 correspondence, “Special tests for Employees Working around BeO. Special Radiation Tests, X-rays and Spirograms,” were conducted for employees who worked around BeO (Bethard 1961a,b,c,d). No other definitive information was found that indicated the type or frequency of occupationally required medical X-ray examinations, although an assumption of an annual chest X-ray examination appears to be a reasonable minimum, and some workers (probably BeO workers) indicated that chest X-ray examinations occurred quarterly. The type of X-ray should be based on current ORAU Team guidance. Organ doses can be obtained from the current revision of ORAUT-OTIB-0006, *Technical Information Bulletin: Dose Reconstruction from Occupationally Related Diagnostic X-Ray Procedures* (ORAUT 2005). There is no documentation that indicates that photofluorography was performed on workers at GA.

4.4 MISCELLANEOUS INFORMATION ON EXTERNAL EXPOSURE

This section includes external dose information that could be of value for specific dose reconstructions. This analysis did not consider such information generically because of its limited applicability or because of limited details.

Site documentation from May 1968 on an extremity overexposure at the LINAC indicates that even though the modulator high voltage and RF were turned off following each tuning and during sample changes, there was still an emission of <1 mA of direct current at 56-kV accelerating potential because of a malfunctioning electron gun (Cockle 1968b). A subsequent dose reconstruction using badges and finger rings indicated that the X-ray energy was approximately 20 keV and that the actual exposure was 7.700 rem, in comparison to the 7.650 rem gamma plus 35.870 rad beta Landauer originally determined. Procedures were changed at the LINAC to prevent reoccurrence.

Employees such as those working with or in the areas of the accelerators and the reactor target irradiation areas might have received nonuniform radiation exposures, and no factors are available to adjust these exposures. It was also noted that plugs were occasionally removed from the HCF walls, which created beams of radiation.

4.5 EXTERNAL DOSIMETRY SUMMARY

Because of the SEC determination (Leavitt 2007) that it is infeasible to adequately reconstruct internal dose during the AWE operational period covered by the SEC (January 1, 1960 through December 31), dose estimates for this period are considered partial dose estimates. External whole-body doses from penetrating photons can be determined from individual dosimetry records. Neutron doses can be estimated by adjusting the reported neutron dose by the appropriate factors in Table 4-6. For exposures to MFPs, the recorded nonpenetrating dose from film can be conservatively estimated by multiplying the beta dose by 2.04 based on guidance that indicates the fraction of reported dose as 0.49 (ORAUT 2006a, Table 6-19). The assumption of initial, annual, and termination chest X-rays appears warranted.

Correction factors for nonuniform radiation doses that might have occurred in the accelerator areas, the reactor target handling areas, or the HCF are not available.

Extremity dose records are not available until about 1961; initially results were reported in arbitrary units.

Sufficiently accurate nonpenetrating dose estimates cannot be made for workers who might have encountered significant quantities for extended periods of beta emitters with energies between 0.03 and 1.5 MeV, which were not <u>mixed</u> fission or activation products.

5.0 <u>ESTIMATION OF EXPOSURE FROM RESIDUAL RADIOACTIVITY</u>

5.1 INTRODUCTION

GA continued to operate after the AWE operational period ended on December 31, 1969. NIOSH (2006c) noted that a significant potential for residual contamination remained after 1969. NIOSH has determined the residual radiation period to be from January 1, 1970 through December 31, 1995, and the DOE Remediation Period to be from January 1, 1996 through December 31, 1999 (NIOSH 2006c). During operations, airborne contamination can be caused directly by operations with radioactive materials as well as by the resuspension of surface contamination. This airborne contamination can be removed by settling on floors or other surfaces, or by ventilation (both engineered and natural building ventilation). The surface contamination levels can be increased directly by radioactive material processing and handling or by the settling of airborne contamination. This level is decreased by various incidental means such as general housekeeping or tracking of contamination out of the area on equipment or personnel.

5.2 ESTIMATION OF INTERNAL EXPOSURE FROM RESIDUAL RADIOACTIVITY

5.2.1 Introduction

The residual source term at GA was identical to the operational period source term that is discussed in Sections 2.3 and 3.1.1 through 3.1.7. It is difficult to estimate exposures to residual activity from workplace measurements, because non-covered operations continued during much of the residual period; however, it is reasonable to estimate exposures to residual contamination from long-lived radionuclides after 1969 by assuming that the residual contamination and or airborne radioactivity levels beginning in 1970 were equivalent to those that GA measured in December of 1969 or earlier depending on workplace measurement availability. Workplace measurements of surface contamination and or general area airborne contamination (non breathing zone air measurements) have been used to bound residual period intakes, because there are insufficient data of worker breathing zones for thorium (SEC determination) and other radionuclides.

Two different methods have been used to estimate bounding intakes for the residual period. The first method utilizes surface contamination data if it was available. For controlled areas, the workplace was assumed to be uniformly contaminated at the maximum operational period contamination level within 1 year from the beginning of the residual period. The maximum level was then assumed to decrease exponentially to a level that was measured some time later in the residual period as shown below [13]:

$$C_t = C_o \times e^{-\lambda t} \tag{5-1}$$

where:

C_t = contamination level at time t measured during residual period
t = time in years between contamination measurements
C_o = contamination level measured at time zero during the operational period
e = base of the natural logarithms
λ = exponential constant

After solving for the exponential constant λ, the daily intake rate was calculated for each year of the residual period (1970 through 1995) by multiplying the contamination level by a resuspension factor of 1×10^{-6}/m (Abu-Eid et al. 2002) and applying a breathing rate of 1.2 m^3/hr for 10 hours a day for 250 workdays per year. This method was used to estimate thorium and uranium intakes (Table 5-1) during the residual period at the GA Fuel Fabrication Facility. For tritium intakes, the highest level of ^3H contamination at the Activation Analysis Facility in 1968 was 6,000 pCi, but no residual period contamination level was available; as a consequence, the tritium radioactive decay constant was assigned for the value of the exponential constant (Table 5-1).

For uncontrolled areas, the GA removable contamination limits for clean areas (6 pCi alpha and 30 pCi beta over 400 cm^2) (Section 2.4) were used to derive intakes assuming a resuspension factor of 1 × 10^{-6}/m (Abu-Eid et al. 2002; ORAUT 2008) and applying a breathing rate of 1.2 m^3/hr for 10 hours a day for 250 workdays per year (Table 5-2).

Note that the controlled area inhalation intake rates (bounding) in Table 5-1 are for areas other than the GA HCF during the residual period. For 1993, 1994, and 1995, the controlled area intake rates for thorium and uranium were lower than the uncontrolled area intake rates due to the exponential removal of radioactivity. As a consequence, the uncontrolled area intake rates have been be used as a lower limit of intake in 1993, 1994, and 1995.

The second method used to estimate bounding intakes for the residual period utilized general area airborne monitoring data from 1968 (gross alpha and gross beta) in the GA HCF along with debris sample analytical data from the High Level Cell obtained in 1994. Based on airborne measurements of plutonium made later in the residual period, the residual period plutonium and other HCF radionuclide intakes are assumed to have remained constant throughout the residual period. The methodology for calculating intakes during the residual period is discussed in detail in the sections below.

5.2.2 Thorium

The Fuel Fabrication Facility (SV-A) at GA used both thorium and uranium in the HTGR fuel fabrication process. Based on the frequency of decontamination crew activities at the Fuel Fabrication Facility (Section 2.4), it appears that the highest (bounding) intakes for uranium and thorium at GA would have occurred at this facility. The highest gross alpha, large area, masslinn wipe result was reported near the beginning of 1970 (in the Main Particle Area) as 70,000 cpm/ft^2 as measured with a PAC-1S (Bold 1970b). A detection efficiency of 18% (Eberline 1977; Bold 1968c) results in the calculation of 419 dpm/cm^2. A survey of the SV-A control area in March 1976 indicates a maximum contamination level of 30 dpm/cm^2 (Bold 1976a). Assuming that all of the contamination was due to thorium, an exponential constant of 0.422 per year was determined. The daily inhalation intake rates for each year during the residual period in controlled areas are shown in Table 5-1. The intake rates in the thorium column of Table 5-1 would be applicable to the sum of the ^{232}Th and ^{228}Th activities. Dose reconstructing methodologies are presented for apportioning ^{232}Th/^{228}Th activities [14].

Thorium intakes during the residual period in uncontrolled (clean) areas of the facility are based on constant resuspension of the clean area alpha contamination limit for removable contamination (6 pCi alpha over an area of approximately 400 cm^2) (Section 2.4). The resultant intake would be 1.23 × 10^{-3} pCi/d from 1970 through 1995 as shown in Table 5-2. This intake rate would be applicable to the sum of the ^{232}Th and ^{228}Th activities. Available worker data and information should be used to determine the likelihood of whether or not a worker would have entered controlled areas.

If claimant information specifically indicates that work was performed involving HTGR fuel fabrication Building 37 or 39, either the thorium or uranium intakes in Table 5-1 (whichever is favorable to the claimant) should be used to reconstruct dose. While evaluating uranium intakes the dose reconstructors should assign uranium intakes from Table 5-1 by assuming that the material is either highly enriched uranium or ^{233}U/^{232}U with an activity ratio of ^{233}U to ^{232}U of 55.7, whichever is favorable to the claimant. If claimant information specifically indicates that work was associated with the GA HCF, the thorium and associated intakes in Tables 5-3 and 5-4 should be assigned. If claimant information specifically indicates that work involving thorium was performed at multiple or unspecified locations, the thorium intakes in Table 5-5 should be assigned.

5.2.3 Uranium

During the fabrication of HTGR fuel, 93.5%-enriched uranium was mixed with thorium at the Fuel Fabrication Facility (SNM-696, pp. 28–66). As previously discussed, ^{233}U was used for fabrication of some of the HTGR fuel compacts that were incorporated into finished HTGR fuel elements. Uranium-233 was also used to conduct neutron yield studies at the LINAC and used for various research and development activities at various GA locations. Assuming that all of the contamination that is discussed in Section 5.2.2 (70,000 cpm/ft^2) was due to 93.5% enriched uranium or ^{233}U/^{232}U with an activity ratio of ^{233}U to ^{232}U of 55.7 (whichever uranium source term is favorable to the claimant) and not thorium, the daily uranium inhalation intake rates for each year during the residual period in controlled areas is shown in Table 5-1.

Uranium intakes during the residual period in uncontrolled (clean) areas of the GA facility are based on constant resuspension of the clean area alpha contamination limit for removable contamination (6 pCi alpha over an area of approximately 400 cm^2) (Section 2.4). The resultant intakes from 1970 through 1995 are shown in Table 5-2.

If claimant information specifically indicates that work involving HTGR fuel fabrication was performed, the uranium intakes in Table 5-1 should be used to reconstruct dose. The dose reconstructors should assign uranium intakes from Table 5-1 by assuming that the material is either highly enriched uranium or ^{233}U/^{232}U with an activity ratio of ^{233}U to ^{232}U of 55.7, whichever is favorable to the claimant. If claimant information specifically indicates that only work related to Triga reactor fuel was performed, only intakes associated 93.5% enriched uranium in Table 5-1 should be assigned. If claimant information specifically indicates that work was associated with the GA HCF, uranium and associated intakes in Tables 5-3 and 5-4 should be assigned. If claimant information specifically indicates that work involving uranium was performed at multiple or unspecified locations, the uranium intakes in Table 5-5 should be assigned.

5.2.4 Plutonium

Attachment E contains a summary of the only workplace monitoring data at GA during plutonium work during the operational period. However there are monthly gross alpha air sample summary results from fixed air sampling stations in HCF areas including the Operating Gallery and Service Gallery from 1965 through 1968. The results for maximum and average concentrations appear to be fairly consistent throughout 1968. The maximum gross alpha air concentration from November 1968 was 7.4×10^{-13} µCi/ml in the HCF Service Gallery (Cockle 1968a). To estimate the fraction of gross alpha airborne radioactivity attributable to plutonium isotopes, the radionuclide mix that was determined from sampling representative debris from the HLC in 1994 (Tables 5-3 and 5-4) is assumed to be representative of the radionuclide mix at the beginning of the residual period.

The bounding daily intake rate was calculated for the residual period (1970 through 1995) by multiplying the gross alpha air concentration of 7.4×10^{-13} µCi/ml by the fraction of activity attributable to ^{238}Pu, $^{239/240}$Pu, ^{241}Am, ^{244}Cm, and other alpha emitters (Table 5-3) and applying a breathing rate of 1.2 m^3/hr for 10 hours a day for 250 workdays per year. The resultant daily intake rates for ^{238}Pu and $^{239/240}$Pu are 3.81 and 0.33 pCi/d, respectively (Table 5-1). These intakes are considered to be constant because the analytical results for ^{238}Pu and $^{239/240}$Pu for air samples from March of 1976 in the HCF Service Gallery during a planned power outage indicated only slightly lower ^{238}Pu and $^{239/240}$Pu air concentrations as a result of activity that escaped the cells (Bold 1976b). Using the fraction of activity attributable to ^{238}Pu and $^{239/240}$Pu from Table 5-3 and the gross alpha air concentration of 7.4×10^{-13} µCi/ml from November 1968 results in ^{238}Pu and $^{239/240}$Pu air concentrations of 4.63×10^{-13} µCi/ml and 4.06×10^{-14} µCi/ml, respectively. Air sampling followed by alpha spectroscopy in 1976 indicated ^{238}Pu and $^{239/240}$Pu air concentrations of 1.25×10^{-13} µCi/ml and 1.81×10^{-14} µCi/ml during a planned power outage during which time the ventilation system in the

HCF was not operating. Given the close agreement between radionuclide-specific air concentrations derived from the 1976 alpha spectroscopy data and those derived from the maximum air concentration that were observed in 1968, the methodology presented above does not underestimate intakes for the residual period.

Intakes of ^{238}Pu, $^{239/240}$Pu, and ^{241}Am during the residual period in uncontrolled (clean) areas of the GA facility are based on constant resuspension of the clean area alpha contamination limit for removable contamination (6 pCi alpha over an area of approximately 400 cm^2) (Section 2.4). The resultant intakes would be 1.23×10^{-3} pCi/d for the sum of ^{238}Pu, $^{239/240}$Pu, ^{241}Pu, and ^{241}Am at the same percentages in Table 3-5 from 1970 through 1995 (Table 5-2).

If claimant information specifically indicates that work was associated with the GA HCF, plutonium and associated intakes in Tables 5-3 and 5-4 should be assigned. If claimant information indicates that work involving plutonium was performed at multiple or unspecified locations, the plutonium intakes in Table 5-6 should be assigned.

It is possible that upon intake some forms of plutonium at GA could be retained in the lung longer than would be predicted by the normal material type S model (commonly referred to as “Type Super S”). Organ doses based on intake of plutonium should be evaluated in accordance with the guidance in ORAUT Team Technical Information Bulletin *Estimating Doses for Plutonium Strongly Retained in the Lung* (ORAUT 2007b).

Summary Tables for Sections 5.2.1, 5.2.2, 5.2.3 and 5.2.6:

Table 5-1. Controlled area inhalation intake rates (pCi/d)[a,d] during the residual period.

Year	Th-232 + Th-228[b]	93.5% enriched uranium	U-233/U-232		H-3
1970	1.55E+01	1.55E+01	1.52E+01	2.73E-01	4.93E+00
1971	1.02E+01	1.02E+01	1.00E+01	1.80E-01	4.66E+00
1972	6.67E+00	6.67E+00	6.55E+00	1.18E-01	4.40E+00
1973	4.37E+00	4.37E+00	4.29E+00	7.71E-02	4.16E+00
1974	2.87E+00	2.87E+00	2.82E+00	5.06E-02	3.93E+00
1975	1.88E+00	1.88E+00	1.85E+00	3.32E-02	3.72E+00
1976	1.23E+00	1.23E+00	1.21E+00	2.17E-02	3.51E+00
1977	8.09E-01	8.09E-01	7.95E-01	1.43E-02	3.32E+00
1978	5.30E-01	5.30E-01	5.21E-01	9.35E-03	3.14E+00
1979	3.48E-01	3.48E-01	3.42E-01	6.14E-03	2.97E+00
1980	2.28E-01	2.28E-01	2.24E-01	4.02E-03	2.80E+00
1981	1.50E-01	1.50E-01	1.47E-01	2.65E-03	2.65E+00
1982	9.81E-02	9.81E-02	9.64E-02	1.73E-03	2.50E+00
1983	6.43E-02	6.43E-02	6.32E-02	1.13E-03	2.37E+00
1984	4.22E-02	4.22E-02	4.15E-02	7.44E-04	2.24E+00
1985	2.77E-02	2.77E-02	2.72E-02	4.89E-04	2.11E+00
1986	1.81E-02	1.81E-02	1.78E-02	3.19E-04	2.00E+00
1987	1.19E-02	1.19E-02	1.17E-02	2.10E-04	1.89E+00
1988	7.81E-03	7.81E-03	7.67E-03	1.38E-04	1.78E+00
1989	5.12E-03	5.12E-03	5.03E-03	9.03E-05	1.69E+00
1990	3.36E-03	3.36E-03	3.30E-03	5.93E-05	1.59E+00
1991	2.20E-03	2.20E-03	2.16E-03	3.88E-05	1.51E+00
1992	1.44E-03	1.44E-03	1.41E-03	2.54E-05	1.42E+00
1993[c]	1.23E-03	1.23E-03	1.21E-03	2.17E-05	1.34E+00
1994	1.23E-03	1.23E-03	1.21E-03	2.17E-05	1.27E+00

Year	Th-232 + Th-228[b]	93.5% enriched uranium	U-233/U-232		H-3
1995	1.23E-03	1.23E-03	1.21E-03	2.17E-05	1.20E+00

a. The results are 365-day intake rate estimates, based on a 250-day work year, 10 hours per work day, in the GA FFF (Th and U) and in the Activation Analysis Facility (H-3) .

b. Th-232/Th-228 activity concentration assumptions for dose reconstruction: 1-to-1 for best estimate for respiratory cancers; 70 to 30 for best estimate for nonrespiratory cancers [14].

c. From 1993 through 1995, the uncontrolled area intakes for Th-232 and uranium were higher than the controlled area intakes; as a consequence the uncontrolled area intakes have been used in Table 5-1 from 1993 through 1995.

d. Use the guidance in OCAS-TIB-009, Estimation of Ingestion Intakes (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake.

Table 5-2. Uncontrolled area inhalation intake rates (pCi/d)[a] during the residual period.

Year	Th-232 + Th-228[b]	93.5% enriched Uranium	Pu-239/240	U-233/U-232		Pu-238	Pu-241	Am-241	H-3	Sr-90
1970	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1971	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1972	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1973	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1974	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1975	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1976	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1977	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1978	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1979	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1980	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1981	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1982	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1983	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1984	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1985	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1986	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1987	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1988	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1989	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1990	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1991	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1992	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1993	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1994	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03
1995	1.23E-03	1.23E-03	1.97E-05	1.21E-03	2.17E-05	2.20E-04	9.63E-04	2.71E-05	2.05E-02	6.16E-03

a. The results are 365-day intake rate estimates, based on a 250-day work year, 10 hours per work day.

b. Th-232/Th-228 activity concentration assumptions for dose reconstruction: 1-to-1 for best estimate for respiratory cancers; 70 to 30 for best estimate for nonrespiratory cancers [14].

c. Use the guidance in OCAS-TIB-009, Estimation of Ingestion Intakes (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake.

5.2.5 Mixed Fission and Activation Products

Mixed fission and activation product intakes have been calculated for two different locations because of the unique nature of contaminants that are associated with the HCF and the TRIGA reactors and critical assemblies. Mixed fission and activation product intakes that are associated with the HCF are based on gross beta air sample summary results from fixed air sampling stations in HCF areas including the Operating Gallery and Service Gallery from 1965 through 1968. The results for maximum and average concentrations appear to be fairly consistent from 1967 through 1968. The highest gross beta air concentration of 1.1×10^{-9} μCi/ml was measured in the same month in 1968 as was the highest gross alpha air concentration at the same Service Gallery sample station (Cockle 1968a). To estimate the fraction of gross beta airborne radioactivity attributable to the airborne radionuclides, the radionuclide mix that was determined from sampling representative debris in the HLC in 1994 (Table 5-4) is assumed to be representative of the radionuclide mix at the beginning of the residual period. It was also assumed that the gross beta measurement was not able to detect the ^{3}H, ^{55}Fe, ^{241}Pu, ^{63}Ni, and ^{106}Ru present on the air sample filter. Concentrations of the latter radionuclides were calculated based on their sample proportion.

The daily intake rate was calculated for the residual period (1970 through 1995) by multiplying the gross beta air concentration of 1.1×10^{-9} μCi/ml by the ratio of the total beta sample activity to the detectable beta activity (1.06) and then multiplying by the fraction of activity attributable the beta emitting radionuclides present (Table 5-4) and applying a breathing rate of 1.2 m^3/hr for 10 hours a day for 250 workdays per year. The intake rates in Table 5-4 are considered to be constant.

Mixed fission and activation product beta-gamma emitter intake rates that are associated with the uranium-fueled critical assemblies, the HTGR and TRIGA reactors can be calculated using the methods in ORAUT-OTIB-0054 (ORAUT 2007a) with use of ^{90}Sr as the indicator radionuclide (Section 3.1.6.2), and removable gross beta-gamma contamination results (352 dpm/100 cm^2) from the TCF, which was shut down in September 1969 (Farney 1973). Applying a resuspension factor of 1×10^{-6}/m (Abu-Eid et al. 2002; ORAUT 2008) results in an estimate of gross beta-gamma airborne concentration of 1.59×10^{-5} pCi/L.

If claimant information specifically indicates that work was associated with the GA HCF the intakes in Tables 5-3 and 5-4 should be assigned. If claimant information indicates that work involving mixed fission and activation product at multiple or unspecified locations, the mixed fission and activation product intakes in Tables 5-6 and 5-7 should be assigned.

Intakes of alpha and beta emitters during the residual period in uncontrolled (clean) areas of the GA facility are based on constant resuspension of the clean area alpha and beta contamination limits for removable contamination (6 pCi alpha or 30 pCi beta over an area of approximately 400 cm^2) (Section 2.4). The resultant intakes would be 1.23×10^{-3} pCi/d for alpha emitters and 6.16×10^{-3} pCi/d for beta emitters from 1970 through 1995.

5.2.6 Tritium

As discussed in Sections 2.3 and 3.1.5, tritium exposures at GA were primarily associated with tritium target changes at the GA Activation Analysis Facility. The only workplace monitoring data available for the Activation Analysis Facility are monthly health physics reports from 1965 through 1968. A review of the reports indicates that by 1968, the number and severity of tritium contamination incidents had decreased significantly. In 1968, the highest level of ^{3}H contamination at the facility was 6,000 pCi, which is assumed to be over an area of 100 cm^2 (Smith 1968). Although the report indicated that the area was decontaminated to <100 pCi, it is assumed that at the beginning of the residual period contamination at a level of 6,000 pCi/100 cm^2 was uniformly distributed at the facility. There are no residual period ^{3}H contamination levels available; as a consequence, the tritium radioactive decay constant (0.05635/year) was assigned for the value of the exponential constant. The daily intake rates in Table 5-1 were calculated for each year of the residual period (1970 through 1995) by multiplying by a resuspension factor of 1×10^{-6}/m (Abu-Eid et al. 2002; ORAUT 2008) and applying a breathing rate of 1.2 m^3/hr for 10 hours a day for 250 workdays per year.

If claimant information specifically indicates that work was associated with the GA HCF, tritium and associated intakes in Tables 5-3 and 5-4 should be assigned. If claimant information indicates that work involving tritium was performed at the Activation Analysis Facility the intakes in Table 5-1 should be assigned. If claimant information indicates tritium work at multiple or unspecified locations, the tritium intakes in Table 5-5 should be assigned.

Intakes of ^{3}H during the residual period in uncontrolled (clean) areas of the GA facility are based on constant resuspension of the detection limit of 100 pCi/100 cm^2 for removable ^{3}H. The resultant intake would be 2.05×10^{-2} pCi/d from 1970 through 1995 (Table 5-2).

5.2.7 Strontium

There are no workplace monitoring data available for strontium thermoelectric generator work during the operational period at GA. However there are monthly gross beta air sample summary results from fixed air sampling stations in HCF areas including the Operating Gallery, Machine Shop, Decon Room, and Service Gallery from 1965 through 1968. These results appear to be fairly consistent from 1967 through 1968. The highest gross beta air concentration in 1968 was 1.1×10^{-9} µCi/ml in the HCF Service Gallery (Cockle 1968a). To estimate the fraction of gross beta airborne radioactivity attributable to ^{90}Sr, the radionuclide mix determined from sampling representative debris from the HLC in 1994 (Table 5-4) is assumed to be representative of the radionuclide mix at the beginning of the residual period. The daily intake rate was calculated for the residual period (1970 through 1995) by multiplying the gross beta air concentration of 1.1×10^{-9} µCi/ml by the fraction of activity attributable to ^{90}Sr and applying a breathing rate of 1.2 m^3/hr for 10 hours a day for 250 workdays per year. The resultant daily intake rate of 2.55×10^3 pCi/d is shown in Tables 5-4 and 5-7.

If claimant information specifically indicates that work was associated with the GA HCF, strontium and associated intakes in Tables 5-3 and 5-4 should be assigned. If claimant information indicates that work involving strontium was performed at multiple or unspecified locations, the strontium intakes in Table 5-7 should be assigned.

Intakes of ^{90}Sr during the residual period in uncontrolled (clean) areas of the GA facility are based on constant resuspension of the clean area beta contamination limit for removable contamination (30 pCi beta over an area of approximately 400 cm^2) (Section 2.4). The resultant intakes would be 6.16×10^{-3} pCi/d from 1970 through 1995 (Table 5-2).

Summary Tables for Sections 5.2.4, 5.2.5, and 5.2.7:

Table 5-3. HLC alpha sampling results and intakes.[a]

Radionuclide	**Sample alpha concentration (μCi/g)**	**Relative alpha fraction**	**Daily total activity intake rate (pCi/d)[b,e]**
U-232	4.50E-01	4.94E-02	3.00E-01
U-233	3.80E-01	4.17E-02	2.54E-01
U-234	7.90E-02	8.68E-03	5.28E-02
U-235	8.00E-04	8.79E-05	5.34E-04
U-236	1.20E-03	1.32E-04	8.02E-04
U-238	5.00E-04	5.49E-05	3.34E-04
Th-228	3.10E-01	3.40E-02	2.07E-01
Th-229	2.40E-02	2.64E-03	1.60E-02
Th-232	2.00E-03	2.20E-04	1.34E-03
Pu-238	5.70E+00	6.26E-01	3.81E+00
Pu-239/240	5.00E-01	5.49E-02	3.34E-01
Pu-241[c]	5.50E-04	6.04E-05	1.67E+01
Pu-242	ND[d]	ND[d]	ND[d]
Am-241	7.10E-01	7.80E-02	4.74E-01
Am-243	1.60E-02	1.76E-03	1.07E-02
Cm-242	2.00E-03	2.20E-04	1.34E-03
Cm-243	4.00E-03	4.39E-04	2.67E-03
Cm-244	9.20E-01	1.01E-01	6.15E-01
Cm-245	ND[d]	ND[d]	ND[d]
Cm-246	ND[d]	ND[d]	ND[d]
Cm-247	1.00E-03	1.10E-04	6.68E-04
Cm-248	5.00E-04	5.05E-05	3.34E-04
Cm-250	1.40E-02	1.23E-04	9.35E-03
Cf-248	8.00E-04	8.79E-05	5.34E-04
Cf-250	3.00E-04	3.30E-05	2.00E-04
Cf-252	1.00E-03	1.07E-04	6.68E-04
Total	9.12E+00	1.00E+00	

a. Adapted from Sills (1994).
b. The results are 365-day intake rate estimates based on a 250-day work year, 10 hours per work day.
c. Based on 0.0022% alpha decay (which would have been detected by gross alpha air sample analysis) from 25 μCi/g of Pu-241 determined by beta analysis (Sills 1994).
d. ND – not detected.
e. Use the guidance in OCAS-TIB-009, Estimation of Ingestion Intakes (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake.

Table 5-4. HLC beta sampling results and intakes.[a]

Radionuclide	Sample concentration (µCi/g)	Relative beta fraction	Daily intake rate (pCi/d)[b,g]
H-3[c]	2.00E-01	1.40E-04	1.34E+00
Fe-55[c]	4.30E+01	3.00E-02	2.88E+02
Ru-106[c]	6.00E+00	4.19E-03	4.02E+01
Ni-63[c]	3.30E+01	2.31E-02	2.21E+02
Rh-106[d]	6.00E+00	4.19E-03	4.02E+01
Co-60	7.50E+01	5.24E-02	5.03E+02
Sr-90	3.80E+02	2.65E-01	2.55E+03
Y-90[e]	3.80E+02	2.65E-01	2.55E+03
Nb-94	4.00E-01	2.79E-04	2.68E+00
Sb-125	2.00E+00	1.40E-03	1.34E+01
Cs-134	2.10E+02	1.47E-01	1.41E+03
Cs-137	2.70E+02	1.89E-01	1.81E+03
Ce-144	2.00E+01	1.40E-02	1.34E+02
Eu-154	4.00E+00	2.79E-03	2.68E+01
Eu-155	1.40E+00	9.78E-04	9.38E+00
Ta-182	3.00E-01	2.10E-04	2.01E+00
Th-231[f]	8.00E-04	5.59E-07	Not applicable
Total	1.43E+03	1.00E+00	

a. Adapted from Sills (1994).
b. The results are 365-day intake rate estimates based on a 250-day work year, 10 hours per work day.
c. These radionuclides are assumed to have been undetected by gross beta air sample counting. Intake rates for H-3, Fe-55, Ni-63, and Ru-106 were calculated based on their proportional sample concentration.
d. Assumed to be in secular equilibrium with Ru-106.
e. Assumed to be in secular equilibrium with Sr-90.
f. Organ doses due to intakes of Th-231 are insignificant in comparison to other assessed intakes [15].
g. Use the guidance in OCAS-TIB-009, Estimation of Ingestion Intakes (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake.

The bounding residual period intakes for GA personnel regardless of location have been tabulated below (Tables 5-5 through 5-7) by combining the results of Tables 5-1, 5-3 and 5-4 (the highest value for each respective radionuclide in Tables 5-1, 5-3, and 5-4 was carried forward into Tables 5-5, 5-6, and 5-7).

Table 5-5. Sitewide bounding controlled area inhalation intake rates (pCi/d)[a] during the residual period.

Year	Th-232 + Th-228[b]	93.5% enriched uranium	U-233/U-232		H-3
1970	1.55E+01	1.55E+01	1.52E+01	3.01E-01	4.93E+00
1971	1.02E+01	1.02E+01	1.00E+01	3.01E-01	4.66E+00
1972	6.67E+00	6.67E+00	6.55E+00	3.01E-01	4.40E+00
1973	4.37E+00	4.37E+00	4.29E+00	3.01E-01	4.16E+00
1974	2.87E+00	2.87E+00	2.82E+00	3.01E-01	3.93E+00
1975	1.88E+00	1.88E+00	1.85E+00	3.01E-01	3.72E+00
1976	1.23E+00	1.23E+00	1.21E+00	3.01E-01	3.51E+00
1977	8.09E-01	8.09E-01	7.95E-01	3.01E-01	3.32E+00
1978	5.30E-01	5.30E-01	5.21E-01	3.01E-01	3.14E+00
1979	3.48E-01	3.48E-01	3.42E-01	3.01E-01	2.97E+00
1980	2.28E-01	2.28E-01	2.54E-01	3.01E-01	2.80E+00
1981	2.08E-01	1.50E-01	2.54E-01	3.01E-01	2.65E+00
1982	2.08E-01	9.81E-02	2.54E-01	3.01E-01	2.50E+00
1983	2.08E-01	6.43E-02	2.54E-01	3.01E-01	2.37E+00
1984	2.08E-01	5.45E-02	2.54E-01	3.01E-01	2.24E+00
1985	2.08E-01	5.45E-02	2.54E-01	3.01E-01	2.11E+00
1986	2.08E-01	5.45E-02	2.54E-01	3.01E-01	2.00E+00
1987	2.08E-01	5.45E-02	2.54E-01	3.01E-01	1.89E+00
1988	2.08E-01	5.45E-02	2.54E-01	3.01E-01	1.78E+00
1989	2.08E-01	5.45E-02	2.54E-01	3.01E-01	1.69E+00
1990	2.08E-01	5.45E-02	2.54E-01	3.01E-01	1.59E+00
1991	2.08E-01	5.45E-02	2.54E-01	3.01E-01	1.51E+00
1992	2.08E-01	5.45E-02	2.54E-01	3.01E-01	1.42E+00
1993	2.08E-01	5.45E-02	2.54E-01	3.01E-01	1.34E+00
1994	2.08E-01	5.45E-02	2.54E-01	3.01E-01	1.34E+00
1995	2.08E-01	5.45E-02	2.54E-01	3.01E-01	1.34E+00

a. The results are 365-day intake rate estimates, based on a 250-day work year, 10 hours per work day.

b. Th-232/Th-228 activity concentration assumptions for dose reconstruction: 1-to-1 for best estimate for respiratory cancers; 70 to 30 for best estimate for nonrespiratory cancers [14].

c. Use the guidance in OCAS-TIB-009, Estimation of Ingestion Intakes (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake.

Table 5-6. Sitewide bounding alpha sampling results and intakes.[a]

Radionuclide	Sample alpha concentration (µCi/g)	Daily total activity intake rate (pCi/d)[b]
Th-229	2.40E-02	1.60E-02
Pu-238	5.70E+00	3.81E+00
Pu-239/240	5.00E-01	3.34E-01
Pu-241[c]	5.50E-04	1.67E+01
Pu-242	ND[d]	ND[d]
Am-241	7.10E-01	4.74E-01
Am-243	1.60E-02	1.07E-02
Cm-242	2.00E-03	1.34E-03
Cm-243	4.00E-03	2.67E-03
Cm-244	9.20E-01	6.15E-01
Cm-245	ND[d]	ND[d]
Cm-246	ND[d]	ND[d]
Cm-247	1.00E-03	6.68E-04
Cm-248	5.00E-04	3.34E-04
Cm-250	1.40E-02	9.35E-03
Cf-248	8.00E-04	5.34E-04
Cf-250	3.00E-04	2.00E-04
Cf-252	1.00E-03	6.68E-04

a. Adapted from Sills (1994).
b. The results are 365-day intake rate estimates based on a 250-day work year, 10 hours per work day.
c. Based on 0.0022% alpha decay (which would have been detected by gross alpha air sample analysis) from 25 µCi/g of Pu-241 determined by beta analysis (Sills 1994).
d. ND – not detected.
e. Use the guidance in OCAS-TIB-009, Estimation of Ingestion Intakes (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake.

Table 5-7. Sitewide bounding beta sampling results and intakes.[a]

Radionuclide	Sample concentration (µCi/g)	Relative beta fraction	Daily intake rate (pCi/d)[b]
Fe-55[c]	4.30E+01	3.00E-02	2.88E+02
Ru-106[c]	6.00E+00	4.19E-03	4.02E+01
Ni-63[c]	3.30E+01	2.31E-02	2.21E+02
Rh-106[d]	6.00E+00	4.19E-03	4.02E+01
Co-60	7.50E+01	5.24E-02	5.03E+02
Sr-90	3.80E+02	2.65E-01	2.55E+03
Y-90[e]	3.80E+02	2.65E-01	2.55E+03
Nb-94	4.00E-01	2.79E-04	2.68E+00
Sb-125	2.00E+00	1.40E-03	1.34E+01
Cs-134	2.10E+02	1.47E-01	1.41E+03
Cs-137	2.70E+02	1.89E-01	1.81E+03
Ce-144	2.00E+01	1.40E-02	1.34E+02
Eu-154	4.00E+00	2.79E-03	2.68E+01
Eu-155	1.40E+00	9.78E-04	9.38E+00
Ta-182	3.00E-01	2.10E-04	2.01E+00
Th-231[f]	8.00E-04	5.59E-07	Not applicable

a. Adapted from Sills (1994).
b. The results are 365-day intake rate estimates based on a 250-day work year, 10 hours per work day.
c. These radionuclides are assumed to have been undetected by gross beta air sample counting. Intake rates for H-3, Fe-55, Ni-63, and Ru-106 were calculated based on their proportional sample concentration.
d. Assumed to be in secular equilibrium with Ru-106.
e. Assumed to be in secular equilibrium with Sr-90.
f. Organ doses due to intakes of Th-231 are insignificant in comparison to other assessed intakes [15].
g. Use the guidance in OCAS-TIB-009, Estimation of Ingestion Intakes (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake.

5.2.8 Other Radionuclides

For radionuclides other than those discussed in this section, dose reconstructors should evaluate intakes and organ doses from bioassay results using current Project guidance.

5.2.9 High-Temperature Gas-Cooled and Reduced-Enrichment Research and Test Reactor Irradiated Fuel Materials Information

In November 1995, DOE irradiated fuel materials (IFM) that were stored in the HCF were remotely inspected, inventoried, and packaged so that the IFM specimens could be removed to allow for the decontamination and decommissioning. The IFM consisted of HTGR and RERTR fuel materials. The IFM samples were associated with DOE-sponsored reactor fuel development programs at GA between 1967 and 1987. Both of these programs involved the in-pile reactor exposure of fuel test capsules or fuel test elements. Radiological characterization of HTGR IFM was performed using the GARGOYLE computer code, while RERTR IFM was characterized using the ORIGEN2 code for fission and activation products and documented actinide content. The results for individual specimens are documented in *HTGR/RERTR Fuel Materials Characterization and Packaging Report* (GA 2002). The initial enrichment of HTGR IFM ranged from 10 to 93.15 wt % of ^{235}U while RERTR IFM enrichment ranged from 20 to 45 wt % (GA 2003). The radioactive content of the consolidated HTGR and RERTR shipment decayed to September 30, 2003, is presented in Attachment J.

5.2.10 Ingestion

When intake estimates are based on other than bioassay results (as in Section 5.0), the ingestion pathway must be considered in addition to the inhalation pathway. Use the guidance in OCAS-TIB-009, *Estimation of Ingestion Intakes* (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake estimates that are determined in Sections 5.2.2 through 5.2.8.

5.3 ESTIMATION OF EXTERNAL EXPOSURE FROM RESIDUAL RADIOACTIVITY

To estimate external exposure to residual activity, external dosimetry results that are available after 1969 are used to provide an upper bound of the dose, and this is assumed to be a constant distribution. It would be reasonable to exclude external doses in this period that are clearly attributable to ongoing operations (e.g., X-ray exposures) rather than activities that might involve exposure to residual radionuclides. It is unlikely that there would be significant nonuniform or extremity exposures from residual contamination. Neutron doses from residual contamination are assumed to be insignificant.

5.4 SUMMARY

To summarize, the information in Table 5-8 is used to account for dose during the residual exposure period.

Table 5-8. Residual radiation exposure summary.

Source	Exposure information	Period of applicability	Annual exposure	IREP[a] distribution
Inhaled residual radioactivity	Select absorption type from information in Section 3.0	Beginning January 1, 1970	Use intake rates determined from guidance in Sections 5.2.2 to 5.2.7 and data in Tables 5.1 to 5.4.[b]	Constant
Ingested residual radioactivity	Select absorption fraction from information in Section 3.0	Beginning January 1, 1970	Apply methodology in NIOSH (2004) to inhalation intake estimates.	Constant
Residual contamination	Penetrating (photons, 30–250 keV, AP, acute)	Beginning January 1, 1970	Use reported dosimetry results.	Constant
Residual contamination	Nonpenetrating (electrons >15 keV, acute)	Beginning January 1, 1970	Use reported dosimetry results multiplied by 2.04.[c]	Constant

a. IREP = Interactive RadioEpidemiological Program.
b. Use controlled area or uncontrolled area intake rate estimates as appropriate for claim documentation.
c. For exposures to MFPs, the recorded nonpenetrating dose from film can be estimated by multiplying the beta dose by 2.04 based on Table 6-19 of ORAUT (2006a), which indicates the fraction of dose reported as 0.49.

6.0 ESTIMATION OF EXPOSURE FROM DOE REMEDIATION

The DOE remediation period is defined as 1996 through 1999 (NIOSH 2006c). Remediation activities occurred before DOE defined *remediation period* and included D&D in Buildings 9, 25, and 31-1 from 1986 to 1996 (Gonzales 2004a). Initial decommissioning of Building 37 began in 1995 and was completed in 1999 (Gonzales 2004a). The decommissioning of the HCF is described below. Additional remediation activities at GA that occurred from 1996 to 1999 included the Group 7, 8A, 8B, 8C, 9, 10, and 11 laboratories in Building 2; Buildings 22, 27,30, 31-1, 35-1, 36, 37, 42, and 45; Room 049B in Building 9, and certain portions of Building 39 (Gonzales 2004a). The L-307 Tank Area was remediated from 1996 to 1999 (Gonzales 2004a).

In December 1994, GA notified the U.S. Nuclear Regulatory Commission (NRC) formally of the intent to cease “principal activities” at the HCF (Building 23) (GA 1998).

The HCF included the EBOR Fuel Fabrication Facility. Floor space totaled approximately 7,400 ft^2 (690 m^2) and consisted of office spaces, three hot cells, and an Operating Gallery; the surrounding fenced service area was 46,740 ft^2 (4,340 m^2). The service area included several concrete pads for material transfer in and out of the HCF. The remaining area of the yard consists of asphalt and soil. There is a small ancillary building of 400 ft^2 (37 m^2) and two potable liquid waste storage tanks of 200-gal capacity. Some areas beyond the yard that had been influenced by HCF operations were included in remediation activities.

The *General Atomics Hot Cell Facility Decommissioning Plan* (GA 1998) describes the work that had been done with the HCF:

> *The primary purpose of the HCF was to facilitate remote inspection of irradiated highly radioactive High Temperature Gas-cooled Reactor (HTGR) fuel. The inspection of other types of irradiated fuel and materials was secondary to this function. The High-Level Cell (HLC) was utilized as a remotely operated “hot” machine shop, to cut open and section irradiated fuel samples according to the inspecting engineer’s direction. The Metallography Cell was used for analytical inspection and research activities,*

> *including the preparation of samples, which entailed grinding, polishing, and mounting. The functions of the Metallography Cell were supported by tasks performed in two other areas, the Warm Metallography (Room 107) and the Tritium Extraction Lab (formerly the Physical Test Lab) (Room 109), during certain periods of operation. The Low-Level Cell (LLC) was used as the interface between the other "hot" cells and other areas of the building, and housed all shipping cask off-loading and packaging operations…Other programs were, at times, conducted within the HCF. What is now the Controlled Machine Shop and Weld Area (Room 108/108A) was previously used to develop and manufacture the Experimental Beryllium Oxide Reactor (EBOR) fuel. The Engineering Scale Tritium Extraction System (ESTES) utilizing Rooms 109, 120, and 119 demonstrated the ability to extract tritium from New Production Reactor (NPR) target compacts. The X-ray Lab (Room 116) was formerly used to perform plutonium transport experiments using plutonium oxide.*

The addressed facility and site areas were:

- Restricted Area Outside Yard,
- Restricted Area Outside Yard Ventilation System Restricted Area,
- Restricted Area Outside Yard Waste Tanks Restricted Area,
- Operating Gallery (Room 105),
- Dark Room and Change Room (Rooms 105A and 106),
- Warm Metallography (Room 107),
- Controlled Machine Shop and Weld Area (Rooms 108 and 108A),
- Tritium Extraction Laboratory (Room 109),
- Service Gallery (Room 111),
- Metallography Cell (Room 112),
- LLC and HLC (Rooms 113 and 115),
- Pump (Room 114),
- X-Ray Laboratory (Room 116),
- Decontamination Room (Room 118),
- Tritium Effluent, Tritium Sample Preparation (Rooms 119 and 120),
- Boiler Room (Room 121), and
- Manipulator Repair (Room 122).

From April 1993 through October 1995, plans and procedures to support the decontamination and dismantlement of the HCF were completed. In addition, the HCF was characterized and legacy waste was packaged for disposal. The DOE-owned legacy IFMs were relocated for onsite storage. According to the HCF Closeout Report (Davis undated), "much of the remote-handled, Category 3 Low-Level Waste stored in the cells was shipped in July 1995." The GA decommissioning plan received interim approval by the NRC in May 1996 (GA 2004).

Based on a review of the HCF closeout report (Davis undated), most of the HCF was shipped as radioactive waste with only a very small fraction of material being released for unrestricted release after dismantlement. According to the closeout report dismantlement activities at the HCF appear to have been complete by October 1998 and from December 1998 through December 1999, waste from with the dismantlement was shipped for disposal or relocated to another building. Other site documentation about the termination of whole-body counting at GA (Gonzales 1998) supports the chronology in the closeout report. According to Gonzales (1998), a whole-body counter was no longer required because the HCF was expected to be dismantled and packaged and air sample concentrations were not expected to exceed 10% of the derived air concentration (DAC). Additional information on HCF D&D is provided in Attachment K.

6.1 SOURCE TERM

The HCF characterization report (Aycock 1995) and D&D plan (GA 1998) relied on gamma spectroscopy results and gross beta-gamma measurements although the gamma spectroscopy results for some sample media in the HCF characterization report do indicate the presence of ^{241}Am at some locations. As a consequence, the source term for the remediation period is based on the radionuclide mix that was determined from the results of sampling representative debris from the HLC in 1994 (Tables 5-3 and 5-4). GA used the results of the debris sampling and analysis to develop a composite DAC specifically for work in the HCF even though the evaluation of smears and air samples from the facility indicated that the ^{90}Sr content in the debris from the HLC was higher than the ^{90}Sr present in the general facility (Sills 1994). According to the characterization report, 21 samples from the HCF were analyzed for ^{137}Cs and ^{90}Sr content and the mean ^{137}Cs-to-^{90}Sr ratio was 3.51; however, none of the samples were from inside the HLC, LLC, or MC.

The survey and sampling of underlying and surrounding soil areas after HCF dismantlement identified elevated activity that consisted primarily of ^{137}Cs, ^{60}Co, and ^{134} Cs. The ^{90}Sr concentrations were estimated using a ^{137}Cs-to-^{90}Sr ratio of 2.1:1 (GA 2000). The ratio of ^{137}Cs to ^{90}Sr in HCF debris in Table 5-4 is 0.71:1. The results for approximately 1,000 soil samples that were analyzed by gamma spectroscopy during post remediation surveys contained <0.3 pCi/g of ^{241}Am and typically <0.3 pCi/g of ^{235}U. No detectable activity was identified in 10 final survey soil samples that were selected for isotopic plutonium and tritium analysis (GA 2000).

6.2 SAFETY

Access to the HCF was limited to a specific group of individuals during the remediation period (1996 to 1999). Authorization for work was controlled by the use of WAs that identified the proposed work scope and activities (GA 1998). Work was to be performed in strict accordance with the methods and precautions provided in approved WAs. The GA Radiation Safety Officer compiled a list all personnel who had worked at the HCF from 1993 through the remediation period along with the WA that they worked under and a description of the activity (Gonzales 2004b). The GA Radiation Safety Officer also compiled a list all personnel who had worked at the Nuclear Waste Processing Facility from 1993 through the remediation period (Gonzales 2004c).

RWPs were used at the HCF as a means to define specific radiation safety requirements for specific activities. When work area conditions were such that the use of engineering controls alone were not enough to control airborne radioactivity, an analysis of alternatives was performed before a decision was made to place workers in respiratory protective equipment. In any situation in which workers were allowed access to an airborne radioactivity area or allowed to perform work that had a high degree of likelihood to generate airborne radioactivity in excess of 0.1 DAC, the decision to allow access was to be accompanied by the performance of representative airborne radioactivity measurements to assess worker intakes (GA 1998).

Workers at the HCF or frequent visitors were provided with a TLD when working in the HCF restricted area or in accordance with a prospective external exposure evaluation. Prospective external exposure evaluations were performed annually at a minimum or whenever changes in worker exposures warranted (GA 1998). Internal monitoring was conducted in accordance with a prospective internal exposure evaluation. Prospective external exposure evaluations were performed annually at a minimum or whenever significant changes in planned work evolutions warranted. Facility workers and frequent visitors were required to participate in the whole-body counting program (GA 1998).

Site documentation includes *in vivo* and *in vitro* monitoring results beginning in 1989 through June 1999 (GA 1999). Personnel who were authorized to work in the HCF and Nuclear Waste Processing Facility from 1993 through the remediation period (1996 to 1999) are included in the *in vivo* and in some of the *in vitro* data. It also appears that personnel under WAs related to the HCF remediation were participating in the GA whole-body counting program (GA 1999).

6.3 ESTIMATION OF INTERNAL EXPOSURE FROM REMEDIATION

Bounding intakes during the remediation period (1996 to 1999) have been calculated assuming that the maximum gross alpha and gross beta air concentrations during remediation activities would be bounded by the latter half of the operational period (1965 to 1969) gross alpha and gross beta air concentrations.

Based on available site documentation for the remediation period, the presence of alpha emitters in the HCF cannot be ruled out; as a consequence, remediation period intakes have been calculated based on the radionuclide mix that was determined from the results of sampling representative debris from the HLC in 1994 (Tables 5-3 and 5-4) and multiplying the residual period alpha emitter intakes (Table 5-3) by a factor of 3.11 and residual period beta emitter intakes (Table 5-4) by a factor of 1.36. The factor of 3.11 is the ratio of the average, maximum gross alpha air concentration of 2.3×10^{-12} µCi/ml (Table 6-1) to the gross alpha air concentration of 7.4×10^{-13} µCi/ml from November 1968 that is used for residual period intakes. The factor of 1.36 is the ratio of the average of the monthly maximum gross beta air concentrations (1.5×10^{-9} µCi/ml) measured in the Service Gallery (Table 6-1) to the maximum monthly gross beta air concentration of 1.1×10^{-9} µCi/ml from November 1968 that is used for residual period intakes. The resultant intakes are presented in Tables 6-2 and 6-3.

Table 6-1. HCF air sampling results from 1965 through 1968.[a].

	Operating gallery		**Service gallery**		**Operating gallery**		**Service gallery**	
	Monthly average concentration (µCi/ml)				**Maximum concentration (µCi/ml)**			
Month and year	**alpha**	**beta**	**alpha**	**beta**	**alpha**	**beta**	**alpha**	**beta**
Nov-68	1.20E-14	6.30E-14	1.70E-13	1.15E-10	2.90E-14	7.90E-14	7.40E-13	1.10E-09
Oct-68	6.60E-14	2.10E-13	1.30E-13	2.60E-11	2.00E-14	3.50E-12	1.50E-13	2.20E-10
Sep-68	1.20E-14	6.00E-14	1.10E-13	3.30E-12	1.50E-14	8.00E-14	3.60E-13	2.25E-10
Aug-68	1.50E-14	7.00E-14	1.30E-13	6.50E-11	3.20E-14	1.00E-13	4.00E-13	2.20E-10
Jul-68	1.10E-14	7.00E-14	1.80E-13	2.20E-10	1.40E-14	1.20E-13	1.30E-12	8.60E-10
Jun-68	1.80E-14	8.00E-14	1.90E-13	1.30E-10	8.20E-14	1.20E-13	4.80E-13	5.20E-10
May-68	1.10E-14	1.00E-13	1.10E-13	3.20E-11	2.60E-14	1.70E-13	4.30E-13	2.90E-10
Apr-68	1.80E-14	2.20E-13	1.20E-13	8.60E-12	6.00E-14	1.20E-12	3.80E-13	2.40E-10
Mar-68	1.10E-14	1.00E-13	1.50E-13	1.10E-11	3.70E-14	1.50E-13	6.50E-13	6.80E-11
Feb-68	2.00E-14	1.00E-13	1.90E-13	2.00E-11	2.70E-13	2.50E-13	1.80E-12	4.60E-10
Jan-68	1.50E-14	1.50E-13	2.00E-13	1.60E-11	7.30E-14	5.00E-13	6.60E-13	2.10E-10
Dec-67	1.30E-14	6.00E-14	1.60E-13	2.84E-12	2.40E-14	9.00E-14	1.90E-13	1.30E-11
Nov-67	2.10E-14	1.00E-13	1.70E-13	5.50E-12	3.40E-14	7.10E-13	4.90E-13	4.80E-11
Oct-67	1.50E-14	1.50E-13	1.10E-13	6.70E-12	8.90E-14	1.40E-12	5.70E-13	3.60E-11
Sep-67	1.20E-14	1.80E-13	1.40E-13	1.50E-11	1.90E-14	1.70E-12	1.90E-13	9.80E-11
Aug 19–Sep 1, 67	1.80E-14	4.40E-14	2.00E-13	1.80E-10	4.80E-14	1.30E-13	3.60E-13	1.40E-09
Aug 1–Aug 18, 67								1.07E-09
Jul-67								5.70E-11
Jun-67								6.10E-10
Apr-67								1.20E-09
Mar-67							Negligible	1.40E-10
Feb-67							Negligible	9.80E-11
Jan-67	4.00E-15	2.30E-14	4.80E-13	1.40E-11	4.90E-15	3.00E-14	2.60E-12	8.60E-11
Dec-66								1.30E-10
Nov-66			4.10E-12	6.20E-11			3.30E-11	2.00E-10
Oct-66	2.00E-15	2.70E-14						8.80E-10
Sep-66	3.00E-15	3.50E-14	1.20E-13	2.90E-12	7.00E-14	1.50E-13	1.40E-12	4.30E-09
Aug-66	2.00E-16	2.40E-14	1.30E-13	5.90E-12	6.90E-15	3.40E-14	5.10E-13	8.30E-11
Jul-66	4.00E-14	3.60E-13	1.50E-13	5.00E-13	5.00E-14	4.00E-13	1.30E-11	6.70E-10
Jun-66	3.46E-14	3.65E-13	5.26E-13	2.69E-11	7.90E-14	1.20E-12	2.50E-12	2.25E-10

	Operating gallery		Service gallery		Operating gallery		Service gallery	
	Monthly average concentration (μCi/ml)				Maximum concentration (μCi/ml)			
Month and year	**alpha**	**beta**	**alpha**	**beta**	**alpha**	**beta**	**alpha**	**beta**
May-66	1.46E-11	1.35E-10	1.20E-12	4.3E-11	1.6E-10	1.6E-09	1.9E-11	6.80E-10
Apr-66	1.20E-14	1.40E-13	1.40E-13	2.90E-09	3.60E-14	5.40E-13	1.60E-13	4.80E-08
Mar-66	6.30E-15	3.70E-14	7.90E-13	5.90E-11	6.80E-14	4.70E-14	1.00E-11	9.30E-10
Feb-66	4.00E-15	3.50E-14	1.70E-13	6.50E-12	4.80E-15	5.70E-14	3.40E-13	4.20E-11
Jan-66	6.00E-15	5.00E-14	2.50E-13	5.90E-12	1.10E-14	1.50E-13	2.00E-12	5.10E-11
Dec-65	6.00E-15	2.80E-14	1.70E-13	5.66E-12	1.00E-13	5.50E-14	5.40E-13	2.50E-11
Nov-65	6.10E-15	2.20E-14	3.10E-13	2.20E-12	1.20E-14	3.10E-14	2.70E-12	1.30E-11
Oct-65	3.70E-15	2.70E-14	1.60E-13	1.28E-11	4.80E-15	3.10E-14	2.70E-13	8.10E-11
Sep-65	8.10E-15	4.10E-14	2.10E-13	2.10E-11	3.40E-14	6.30E-14	7.40E-13	1.80E-10
Aug-65	1.30E-14	7.30E-14	2.40E-13	1.20E-10	2.40E-13	3.30E-13	1.10E-12	1.10E-09
Jul-65	2.00E-14	1.10E-13	1.90E-13	3.50E-11	1.10E-13	4.70E-13	5.50E-13	1.50E-10
Jun-65	1.50E-14	1.70E-13	1.43E-13	2.40E-11	7.00E-14	7.90E-13	2.40E-13	2.20E-10
May-65	1.40E-14	2.40E-13	2.90E-13	4.23E-11	6.00E-14	1.30E-12	1.90E-12	3.90E-10
Apr-65	7.00E-15	9.30E-14	2.10E-13	9.30E-11	1.00E-14	2.70E-13	8.10E-13	7.50E-10
Mar-65	9.20E-15	8.50E-14	4.10E-13	1.68E-10	2.00E-14	2.90E-13	3.70E-12	2.20E-09
Feb-65	1.40E-14	6.20E-13	1.49E-13	8.49E-12	7.00E-14	6.20E-13	2.50E-13	3.30E-11
Jan-65	2.00E-14	3.40E-13	2.40E-13	5.49E-11	1.40E-13	3.40E-13	1.90E-12	4.70E-10
Dec-64	1.45E-14	1.24E-13	5.08E-13	9.76E-11	2.00E-14	1.60E-12	5.08E-13	9.76E-11
Count	3.90E+01	3.90E+01	3.90E+01	3.90E+01	3.80E+01	3.80E+01	3.90E+01	4.70E+01
Average	1.41E-14	1.24E-13	3.22E-13	1.19E-10	5.51E-14	5.03E-13	2.30E-12	1.50E-09
Max	6.60E-14	6.20E-13	4.10E-12	2.90E-09	2.70E-13	3.50E-12	3.30E-11	4.80E-08
Median	1.20E-14	8.50E-14	1.70E-13	2.10E-11	3.50E-14	2.10E-13	5.70E-13	2.20E-10

a. Source: Hughes et al. (1964–1968).

Table 6-2. HCF debris sampling results and remediation period alpha intakes.[a]

Radionuclide	Sample concentration (μCi/g)	Relative alpha fraction	Daily intake rate (pCi/d)[b]
U-232	4.50E-01	4.94E-02	9.35E-01
U-233	3.80E-01	4.17E-02	7.89E-01
U-234	7.90E-02	8.68E-03	1.64E-01
U-235	8.00E-04	8.79E-05	1.66E-03
U-236	1.20E-03	1.32E-04	2.49E-03
U-238	5.00E-04	5.49E-05	1.04E-03
Th-228	3.10E-01	3.40E-02	6.44E-01
Th-229	2.40E-02	2.64E-03	4.99E-02
Th-232	2.00E-03	2.20E-04	4.16E-03
Pu-238	5.70E+00	6.26E-01	1.18E+01
Pu-239/240	5.00E-01	5.49E-02	1.04E+00
Pu-241[c]	5.50E-04	6.04E-05	5.19E+01
Pu-242	ND[d]		
Am-241	7.10E-01	7.80E-02	1.48E+00
Am-243	1.60E-02	1.76E-03	3.32E-02
Cm-242	2.00E-03	2.20E-04	4.16E-03
Cm-243	4.00E-03	4.39E-04	8.31E-03
Cm-244	9.20E-01	1.01E-01	1.91E+00
Cm-245	ND[d]		
Cm-246	ND[d]		
Cm-247	1.00E-03	1.10E-04	2.08E-03
Cm-248	5.00E-04	5.05E-05	1.04E-03
Cm-250	1.40E-02	1.23E-04	2.91E-02
Cf-248	8.00E-04	8.79E-05	1.66E-03
Cf-250	3.00E-04	3.30E-05	6.23E-04
Cf-252	1.00E-03	1.07E-04	2.08E-03
Total	9.12E+00	1.00E+00	

a. Adapted from Sills (1994).

b. The results are 365-day intake rate estimates based on a 250-day work year, 10 hours per work day.
c. Based on 0.0022% alpha decay (which would have been detected by gross alpha air sample analysis) from 25 µCi/g of Pu-241 determined by beta analysis (Sills 1994).
d. ND = not detected.
e. Use the guidance in OCAS-TIB-009, Estimation of Ingestion Intakes (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake.

Table 6-3. HCF debris sampling results and remediation period beta intakes.[a]

Radionuclide	Sample concentration (µCi/g)	Relative beta fraction detected on air sample	Daily intake rate (pCi/d)[b]
H-3[c]	2.00E-01	1.40E-04	1.82E+00
Fe-55[c]	4.30E+01	3.00E-02	3.92E+02
Ru-106[c]	6.00E+00	4.19E-03	5.47E+01
Ni-63[c]	3.30E+01	2.31E-02	3.01E+02
Rh-106[c,d]	6.00E+00	4.19E-03	5.47E+01
Co-60	7.50E+01	5.24E-02	6.84E+02
Sr-90	3.80E+02	2.65E-01	3.46E+03
Y-90[e]	3.80E+02	2.65E-01	3.46E+03
Nb-94	4.00E-01	2.79E-04	3.65E+00
Sb-125	2.00E+00	1.40E-03	1.82E+01
Cs-134	2.10E+02	1.47E-01	1.91E+03
Cs-137	2.70E+02	1.89E-01	2.46E+03
Ce-144	2.00E+01	1.40E-02	1.82E+02
Eu-154	4.00E+00	2.79E-03	3.65E+01
Eu-155	1.40E+00	9.78E-04	1.28E+01
Ta-182	3.00E-01	2.10E-04	2.73E+00
Th-231[f]	8.00E-04	5.59E-07	Not applicable
Total	1.43E+03	1.00E+00	

a. Adapted from Sills (1994).
b. The results are 365-day intake rate estimates based on a 250-day work year, 10 hours per work day.
c. These radionuclides are assumed to have been undetected by gross beta air sample counting. Intake rates for H-3, Fe-55, Ni-63, and Ru-106 were calculated based on their proportional sample concentration.
d. Assumed to be in secular equilibrium with Ru-106.
e. Assumed to be in secular equilibrium with Sr-90.
f. Organ doses due to intakes of Th-231 are insignificant in comparison to other assessed intakes [15].
g. Use the guidance in OCAS-TIB-009, Estimation of Ingestion Intakes (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake.

If no uranium detection threshold is available with the bioassay for this period, the detection threshold can be assumed to be 2 times the 2-sigma error. If no error is reported, the detection thresholds in the internal dose section can be used.

6.3.1 Ingestion

When intake estimates are based on other than bioassay results (as in Section 5.0), the ingestion pathway must be considered in addition to the inhalation pathway. Dose reconstructors should use

the guidance in OCAS-TIB-009, *Estimation of Ingestion Intakes* (NIOSH 2004), to derive estimates of ingestion intakes from the inhalation intake estimates in Tables 6-1 and 6-2.

6.4 ESTIMATION OF EXTERNAL EXPOSURE FROM REMEDIATION

Interpretation of external dosimetry records should be straightforward for the remediation period. The assumptions for external exposures in Table 5-5 can be used to estimate external doses for this period.

7.0 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database.

[1] Olsen, Bernard M. Senior Health Physicist. ORAU Team. August 2007.
This information is based on a telephone conversation former ORAU Team member Kenneth Fleming had with GA employee John Greenwood on March 10, 2006.

[2] Olsen, Bernard M. Senior Health Physicist. ORAU Team. August 2007.
This information is based on a telephone conversation former ORAU Team member Kenneth Fleming had with GA employee Bill LaBonte on March 10, 2006.

[3] Bloom, Cindy W. Senior Health Physicist. ORAU Team. March 2006. Olsen, Bernard M. Senior Health Physicist. ORAU Team. July 2007.
This information is based on reviews of site personnel bioassay history cards, reports, and monitoring records.

[4] Olsen, Bernard M. Senior Health Physicist. ORAU Team. June 2008.
The "typical" data on individual total body or lung count data sheets, and the in-vivo data typically contained within the "Internal Radiation Monitoring Report" records in a GA dosimetry record were arrived at by observation after reviewing several GA dosimetry records covering several decades of data.

[5] Olsen, Bernard M. Senior Health Physicist. ORAU Team. April 2008.
The specific activities of ^{233}U and ^{232}U in Table 3-2 are 22,000,000 pCi/µg for ^{232}U; and 9,800 pCi/µg for ^{233}U. Uranium-232 impurity levels in the ^{233}U at GA may have been up to 8 ppm (Section 2.3). The ratio of ^{233}U to ^{232}U at that level of ^{232}U impurity is 9,800 pCi/µg divided by the quantity [22,000,000 pCi/µg X (8 ppm / 1 E+06)], which is 55.7 to 1.

[6] Olsen, Bernard M. Senior Health Physicist. ORAU Team. July 2007.
The specific activities and half-lives of the listed radionuclides (presented here only to two significant figures) are listed in any standard handbook of radiation protection quantities [e.g. *The Health Physics and Radiological Health Handbook* (Schleien, Slaback, and Birky 1998)], or as presented in the application, Integrated Modules for Bioassay Analysis (IMBA).

[7] Olsen, Bernard M. Senior Health Physicist. ORAU Team. July 2007.
The data for natural, enriched, and depleted uranium are as presented in IMBA.

[8] Olsen, Bernard M. Senior Health Physicist. ORAU Team. June 2008.
A review of some of the data reported on bioassay cards (GA 1963-1967 and GA 1963-1970) was performed. In that review, it was noted that, in general, when the value of 0.4 pCi/24 hr appeared on the bioassay card for an α analysis, a "<" symbol accompanied that value: if the card also contained a column titled "%MPL," then the "<" symbol appeared in that column; when the bioassay card contained no column titled "%MPL," then the "<" symbol appeared in front of the 0.4. For this attribution, two examples of this demarcation are noted: (a) on p. 1519 of GA 1963-1970, values of 0.6 pCi/24 hr in February 1965 and <0.4 pCi/24 hr in May 1965 are recorded; and (b) on p. 3915 of GA 1963-1970, the March 1969 value of 0.1 pCi/24 hr is indicated as being "<1.4 % MPL," and the June 1969 value of 0.4 pCi/24 hr is indicated as

being “1.0 % MPL.” The value of 0.4 pCi/24 hr is equivalent to 0.89 dpm/24 hr (0.4 pCi/24 hr X 2.22 dpm / pCi).

A review of the data reported in the dosimetry records of some of the claimants was also performed. In that review, it was noted that the “<” notation may not have been always transcribed over from the bioassay cards to the computer printout records. For example, in the dosimetry records of one claim, four uranium urine results in 1968-69 are recorded as a value lower than 0.4 dpm/24 / hr, including one at 0.08 dpm/24 hr. The “<” notation does not appear on this record. It is doubtful that a value as low as 0.08 dpm/24 hr could have been detected; it is assumed that the “<” notation from the bioassay card record did not get transcribed. It is prudent to be favorable to claimants and use the rounded-up value of 1 dpm/24 hr (from 0.89 dpm/24 hr) for detection sensitivity of uranium in urine for that time frame.

[9] Olsen, Bernard M. Senior Health Physicist. ORAU Team. June 2008.
A range of minimum detectable activities (by mass) for ^{235}U in the lung is observed or inferred in the GA records. In January 1968, a 14 μg result is reported as positive after correcting for chest thickness (p. 16 of HNS 1968). That same result of 14 μg appears as a positive result in the bioassay cards for January 17, 1968, with the notation, “Valid Results Using New Curve Fit technique.” (GDC 1966-1970, p. 69). In 1969, the typical counting error was listed as 0.02 mg ^{235}U (p. 56 of HNS 1969a). In 1970, positive results as low as 50 μg ^{235}U in the lungs was reported (HNS 1970). In 1971, positive results as low as 33 μg ^{235}U in the lungs was reported (HNS 1971). The typical numerical values for minimum sensitivities of detection of ^{235}U in the lung was characterized in 1976 as being 30 to 60 micrograms (HNS 1976). With a referenced range of 30 to 60 μg, and with positive values being reported as low as 14 μg since 1968, it is favorable to claimants to assume a nominal minimum sensitivity of 60 μg for ^{235}U in the lung, unless there are reported specific data to use instead of that nominal value.

Since some GA chest count results for ^{235}U are presented in the records as a percentage of MPLB, the following is provided. GA (via HNS) was using 255 μg as 100% of MPLB for ^{235}U in 1968 (p. 16 of HNS 1968). In 1969, the MPLB determination was explained with more detail: “The MPLB is a function of the percentage of uranium-235 in the uranium fuel and the ratio of the uranium-234 to uranium-235. For fully enriched material the MPLB is 0.245 mg.” (Note (2) on p. 51 of HNS 1969b).

[10] Olsen, Bernard M. Senior Health Physicist. ORAU Team. June 2008.
Uranium-235 chest counts were available for some workers beginning in 1966, but are not considered valid for dose reconstruction purposes until 1968 based on an entry on a whole-body monitoring data card (GDC 1966–1970, p. 69), which states that results are valid using a “new curve fitting technique.” A review of approximately 1,000 whole-body monitoring data cards (GDC 1966–1970) indicates that before January 1968, ^{235}U chest counts were “not valid” (e.g. GDC 1966–1970, pp. 564–567). The reason for the “not valid” determination in the pre-1968 counts is explained by HNS as constant bias in the ^{235}U data on p. 45 of HNS 1967. By mid January 1968, HNS had solved the bias issue in counting for ^{235}U (HNS 1968). In addition, a review of the in-vivo data contained within several GA dosimetry records covering several decades of data indicate that the “not valid” phrase does not typically appear for data after January 17, 1968. If the phrase “not valid” or “not corrected for chest wall thickness” does not appear on the record of a positive in-vivo count performed after 1967, the reported result can be used in the dose reconstruction. This conclusion is further supported by the detailed counting and analyses procedures provided in the later Helgeson reports (HNS 1968, HNS 1969c, chest thickness correction is discussed in detail, e.g., beginning on p. 6 of HNS 1976).

[11] Olsen, Bernard M. Senior Health Physicist. ORAU Team. June 2008.
A review of some of the data reported on bioassay cards (GA 1963-1967 and GA 1963-1970) was performed. In that review, it was noted that except for a few entries in 1966, the "<" notation does not appear for a beta ("β" or "MFP") result. Positive results as low as 6 to 7 pCi/24 hour for β analyses were observed throughout these records (e.g., p. 1087 of GA 1963-1967, p. 3849 of GA 1963-1970). The few "<" results were recorded in the 13 to 15 pCi/24 hour range (e.g., p. 3857 of GA 1963-1967, p. 72 and 87 of GA 1963-1970). Similar observations were made in a review of the data reported in the dosimetry records of some of the claimants. In that review, the lowest recorded β result observed was 14 dpm/24 hour (6.3 pCi/24 hour).

A review of the urine sample calculation sheets indicates that various standards (Sr-90 and Cs-137 for example) were used to convert beta count rate results to activity. A January 1964 result (p. 5 of GDC 1965a) indicating no detectable activity (at "<3 net cpm") is converted to "<69 pCi as Cs-137/24 hr" using the ^{137}Cs "standard" for that count of 8.75 pCi/net cpm. An April 1965 result (p. 10 of GDC 1965a) indicating 1.3 net cpm is converted to 12 pCi as Sr-90/24 hr using the Sr-90 "standard" for that count of 1.55 pCi/net cpm. From a review of the calculations in GDC 1965a and 1965b, it appears that ^{90}Sr was used as the calibration standard when converting to a gross beta result, and ^{137}Cs was used as the calibration standard when converting to a mixed fission product result. A nominal range for the minimum sensitivity of a gross beta result (without normalization to time or urine volume) would be 5 to 10 pCi and would be on the order of 70 pCi for a mixed fission product result based on this review.

[12] Bloom, Cindy W. Senior Health Physicist. ORAU Team, March 2006.
This information is based on correspondence with the current GA Radiation Safety Officer.

[13] Olsen, Bernard M. Senior Health Physicist. ORAU Team. July 2007.
The methodology is that used to evaluate residual radioactivity over time as presented in Section 5.0 of OCAS-TKBS-0002, *Technical Basis Document for Atomic Energy Operations at Blockson Chemical Company, Joliet, Illinois* (NIOSH 2007c).

[14] Olsen, Bernard M. Senior Health Physicist. ORAU Team. February 2008.
The ratio of ^{232}Th to ^{228}Th is variable and ranges from equal activity concentrations to the minimum condition of ^{228}Th being present at approximately 40% the activity concentration of ^{232}Th (West 1965; GA 1971b). The relative ratio of ^{232}Th to ^{228}Th is significant for dose reconstruction. Organ dose that is determined by applying the dose conversion factors of ICRP Publication 68 (ICRP 1990) is maximized for nonrespiratory cancers by assuming all thorium is ^{232}Th, while the dose is maximized for respiratory cancers by assuming that all thorium is ^{228}Th. For a nonrespiratory cancer, it is favorable to the claimant to assume that all the thorium is ^{232}Th. For a respiratory cancer, it is favorable to the claimant to assume that all the thorium is ^{228}Th. For a best-estimate evaluation, dose reconstructors should assume that the ration of ^{232}Th to ^{228}Th is 1:1 for a respiratory cancer, or 7:3 (the minimum condition of ^{228}Th present at 40% that of ^{232}Th) for a nonrespiratory cancer.

[15] Olsen, Bernard M. Senior Health Physicist. ORAU Team. February 2008.
Thorium-231 adds less than 1 millirem total dose to both the inhalation and ingestion calculations for the residual period, and it is not applied for the remediation period. Based on Tables 5-2 and 6-2, the maximum 1-year inhalation intakes of ^{231}Th would be less than 2 pCi during the residual period and less than 250 pCi during the remediation period. Applying the maximum inhalation and ingestion dose coefficients from ICRP Publication 68 to those estimates on intake results in less than 1 mrem each to the maximally exposed organ (ICRP 1990).

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ATTACHMENT A
FUEL FABRICATION PROCESS

The following is a general description of the fuel fabrication from License SNM-696 for the Fuel Fabrication Facility, which began operations in 1962. Similar operations were occurring in other GA facilities before 1962. This description does not necessarily include all activities involving fuel fabrication.

A1.0 FABRICATION OF PYROLYTIC URANIUM OXIDE OR CARBIDE OR URANIUM-THORIUM OXIDE OR CARBIDE PARTICLES

Station A1 – Raw Material Storage. Raw material storage was in the raw material storage vault. Any enrichment greater than 93.5% was subject to the same rules as fully enriched uranium. The uranium was stored in either 1-gal or 3.6-L containers in such a way as to avoid criticality (SNM 696, pp. 30–31).

Station A2 – Batch Weighing. Uranium dioxide powder was received in stock containers and weighed into individual portions. The weighing occurred in a glovebox. Horizontal racks held the stock containers on centers of 16 in. or greater. Each rack was limited to four stock containers so the rack held a total of 30 kg of ^{235}U or less, the station's mass limit. The material was then weighed and placed into batch containers with screw caps for removal from the station. These batch containers were placed on another rack with centers at least 16 in. from each other, the scale, and other stock material. Each container was loaded with 7.3 kg or less and was sealed to prevent water entrance (SNM 696, p. 32).

Station A3 – In-Process Storage. The material was stored in closed containers 5 in. or less in diameter on steel shelves. The containers were placed in an array of 16 and centered horizontally but set as close as possible vertically. The effect was to make a linear array of 5-in.-diameter vertical cylinders on 16 centers with surface-to-surface spacing of 11 in. or more. Each bottle was held firmly in a storage location by a bar or chain across the opening, and intermediate spacers prevented the bottle from tipping over. The ^{235}U mass density was always 1.8 kg/L or less (SNM 696, p. 32).

Station A3A – In-Process Storage. This station was similar to Station A3 except that material was stored in 3.6-L containers with a maximum of 3.6 kg of ^{235}U (SNM 696, p. 32–33).

Station A4 – Weighing. The station was used to weigh all material other than uranium (SNM 696, p. 33).

Station A5 – Blending. A weighed batch of uranium from dioxide from Station A3, carbon, a binder and, if required, thorium dioxide using a solvent weighed at Station A4 were blended at this station. The maximum amount of ^{235}U at this station was 750 g. The material was mixed in an 8-quart blender inside a glovebox (SNM 696, p. 33).

Station A5A – Blending. This station was similar to Station A5 except ^{235}U was limited to less than 5 kg. The blending is done in a blender with a maximum ability to mix 5 L in accordance with the 1 kg/L for ^{235}U (SNM 696, p. 33).

Station A6 – Drying. The mixture of carbon, binder, solvent, uranium dioxide, and thorium dioxide from Station A5 was delivered to Station A6 in trays and heated in an oven at approximately 140°F. The mass limit for this station was the blending batch limit of 750 g of ^{235}U or less split into three trays. The oven had three open shelves, each large enough for only one tray. Unused shelves were covered with aluminum foil (SNM 696, p. 34).

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Station A6A – Drying. The mixture of uranium dioxide, thorium dioxide, if used, carbon, binder and solvent from Station A5 or A5A was brought in trays one at a time and dried in an oven at approximately 150°F. A total of six trays were placed in the end of the oven one after another and run through on a conveyor to the other end. The combined mass of trays could contain no more than 8 kg of ^{235}U, the station mass limit. The trays were removed and taken one at a time to Station A7.

The heating was supplied by an electric radiant heating element that ran the length of the oven. The oven was an insulated, closed-box structure that was connected by a duct to the facility hood ventilation system to remove fumes and dust. Wire cloth restricted the area above the tray conveyor to a height to accommodate only one tray (SNM 696, p. 34).

Station A7 – Sizing and Weighing. The dried mixture of carbon, binder, uranium dioxide, and thorium dioxide, if used, was brought to the station in one oven tray at a time. The material was transferred into a 3.6-L container. Additional trays would be added until the container was full. This container was emptied using a smaller container with a limit of no more than 0.75 L and placed on a set of screens and sized into the required size fractions. The screens were unloaded one at a time and the material fraction was removed from the screen and placed in a separate 3.6-L container. This procedure was repeated until the entire container was emptied. The sized fractions were weighed in their separate containers and put in in-process storage.

The entire station was enclosed in a glovebox connected by a duct to the facility hood ventilation system. A special storage fixture was provided in the glovebox so there was a minimum of 16 in. between centers for the containers. No more than four containers and one screen set were present at the station any one time. The total mass limit was 10 kg of ^{235}U (SNM 696, pp. 34–35).

Station A8 – In-Process Storage. This station was the same is Station A3 or A3A (SNM 696, p. 35).

Station A9 – In-Process Storage. This station was the same is Station A3 or A3A (SNM 696, p. 35).

Station A10 – Batch Weighing. This station was similar to Station A2. The dried mixture of carbon, binder, uranium dioxide, and thorium dioxide sized at Station A7 was weighed for blending at Station A11 or for loading directly into the graphite crucible in Station A11A. The station mass limit was 10 kg of ^{235}U (SNM 696, p. 35).

Station A11 – Blending and Loading the Conversion Crucible. Weighed portions of the dried, sized mix from Station A10 were packed into a crucible. The crucible was a graphite sleeve closed at one end with an inside diameter under a 5 in. or less. The filled crucible was loaded to 800 g or less of ^{235}U, which was a station mass limit. The operation at this station was performed in a glovebox (SNM 696, p. 36).

Station A11A – Conversion Crucible Loading. Individual containers of the dried mix and Station A10 were brought to this glovebox and opened and material was transferred into a single graphite conversion crucible similar to what happened at Station A11. The inside diameter of the conversion crucible was 4.5 in. or less. The mass limit for the crucible was 6.5 kg of ^{235}U, which was also the station mass limit (SNM 696, p. 36).

Station A12 – Conversion. Converter crucibles loaded at Station A11 or A11A were placed in an induction furnace and heated to a maximum temperature between 2,000°C and 2,800°C under a high vacuum or in an inert gas atmosphere. The crucibles were then cooled and removed from the

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converter to an inert atmosphere glovebox for unloading. The converter furnace consisted of a large quartz cylinder heated by water-cooled induction coils. One or more crucibles were placed inside a close-fitting cylindrical graphite susceptor. The susceptor was secured inside the quartz cylinder and insulated for the quart cylinder with carbon insulation.

One or more loaded crucibles were brought to the conversion station with up to 6.5 kg of ^{235}U, which was the mass limit for the station (SNM 696, pp. 36–37).

Station A12A – Conversion. Weighed batches of the dry, sized mix from Station A10 or the dicarbide product from Station A12 was loaded into the hopper of the automatic feeder for the converter at Station A13. The material hopper was a 4-in.-diameter by 6-in. stainless-steel canister. The loaded hopper was sealed and the interior charged with an inert gas atmosphere. The presized material was automatically fed into the furnace section of the converter. The furnace section consisted of an 8-in.-diameter graphite tube surrounded by 4 in. of lampblack and a thick quartz tube. Water-cooled induction heaters were placed around the quartz tubes, and protective screens surrounded the entire furnace section.

A cooling section in series with the furnace section consisted of a 9-in. inside diameter, 0.125-in. walled stainless-steel tube that was surrounded by cooling coils. The converted dicarbide material entered the cooling section continuously from the furnace section and was discharged into a 5-in.-diameter stainless-steel receiver.

The furnace section was heated to between 2,000° C. and 2,800° C. The material hopper, automatic feeder, furnace section, cooling sections, and receiver were continuously purged with an inert gas. The outlet purge gas of the converter consisting of the inert gas and carbon monoxide was conducted to an additional high-efficiency absolute filter. The outlet gas from the filter was diluted with large quantities of air and injected into the glovebox and hood ventilation system (SNM 696, pp. 37–38).

Station A12B – Conversion. This station was the same is Station A12B with the exception that all parts of the furnace, the material feed hopper, furnace section, cooling section, and the material receiver had an inside diameter of 5 in. or less. The SNM was loaded into the sealed inert-gas-filled hopper at Station A13 and brought to this station. The mass limit of this station was 4 kg of ^{235}U (SNM 696, p. 38).

Station A12C – Conversion. The converter at this station was a horizontally oriented furnace that consisted of a resistance preheater a quartz cylinder that was heated by water-cooled induction coils with a water-coil cooling section. The material to be converted was moved through a 5-in. inside diameter cylinder lining the three sections of the converter and was introduced into the preheater end of the converter in individual graphite boats, each with a mass limit of 2 kg of ^{235}U. The boats were pan-shaped (1 in. deep by 4 in. wide by 12 in. long) and flat-bottomed with a vertical piece shaped like a segment of a circle and each end was sized to fill the converter cylinder and partition the boats from each other. The boats were placed in the liner cylinder one behind the other with each subsequent boat pushing the previous boats through the converter. A mechanical feed mechanism propelled the boats through the cylinder. The volume of the boat (0.8 L) and the bulk density (2 g/cm^3) of the material prevented filling a boat in excess of the 2 kg of ^{235}U mass limit (SNM 696, p. 39).

Station A13 – Crucible Unloading, Feed Hopper Loading, and Receiver Unloading. The crucible from Station A12 was brought to this inert-gas-filled glovebox. The material was unloaded, passed through

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a screen to break up any large chunks, and loaded into a special bottle that fit into the separator, Station A14.

The material feed container or the receiver for Station A12A or A12B was brought to this glovebox for loading or unloading. Feed material was transferred directly from individual storage containers into the feed hopper for Station A12A or A12B; the station mass limit was 3.6 kg of ^{235}U (SNM 696, p. 40).

Station A14 – Separation and Weighing. The carbon and uranium-thorium dicarbide or uranium dicarbide mixture from the converter came to this station in a feed bottle that was attached to the separation apparatus. It was screw-fed to the classifying chamber where a flow of inert gas carried away the carbon dust. The product particles of uranium dicarbide and thorium dicarbide fell into the product bottle. The mass limit for the station was 3.6 kg of ^{235}U. Before another batch was charged to the classifier, a material balance check was performed to guard against holdup material in the classifier (SNM 696, p. 40).

Station A15 – In-Process Storage. This station was similar to Station A3 or A3A (SNM 696, p. 40).

Station A16 – In-Process Storage. This station was similar to Station A3 or A3A (SNM 696, p. 41).

Station A17-1 – Weighing, Screen and Loading. This station was a controlled-atmosphere, sealed glovebox of the same kind as that at Station A13 and had a mass limit of 3.6 kg of ^{235}U. Containers were opened and the material was screened, sampled, weighed, and placed in the coater loading fixture. Weighed batches of material with 3.6 kg of ^{235}U or less were placed in a loading fixture that was then sealed and removed from the glovebox (SNM 696, p. 41).

Station A17A – Coating. The coater was an insulated cylinder that was heated by resistance elements or water-cooled induction coils and was vented to the exhaust system. In operation it was heated, flushed with inert gas, and loaded with particles. The coater tube, the portion holding SNM, was 5 in. inside diameter or less. Material was brought to the station in a sealed loading fixture, 5 in. inside diameter or less with up to 3.6 kg of ^{235}U, the station mass limit. All of the uranium-containing material was placed into the coater where appropriate coating gases were introduced (SNM 696, p. 41).

Station A17B – Coating. This dry coater was an insulated cylinder that was heated by resistance elements and was vented to one of two exhaust systems. The coater was cooled using halocarbon rather than water. A series of graphite inner liners was used to control thermal distributions, gas, and product flow parameters.

The material was brought to the station in sealed containers each having up to 3.6 kg of ^{235}U. The maximum quantity of particles in the coater did not exceed 20 kg of ^{235}U, and enrichments did not exceed 93.2%. The form of the materials was one of the following: UC_2 kernels, UC_2 LTI particles, UC_2 SiC coated particles, UC_2 outer LTI particles, or uranium-thorium mixtures of the above. The carbide materials were typically converted from oxide materials in a furnace at 2,200°C to drive off all hydrogen present. The carbide-converted materials were offloaded from the conversion furnace and stored under an inert atmosphere to prevent back-conversion to the oxide form by hydrolyzing.

After coating, the hot particles were transferred by gravity to one or both of 25-in.-diameter, 4-ft.-long cylindrical cooling hoppers, each capable of containing 10 kg of ^{235}U. After cooling the material was meted out of the cooling hoppers into several 3.6-L containers with a limit of 3.6 kg of ^{235}U each. An inert atmosphere was maintained during all material transfer operations (SNM 696, p. 42).

Station A18 – Screening, Weighing, and Packaging. Carbon-coated particles of uranium dicarbide and thorium dicarbide were screened in a hood and the product and oversized materials were packaged for storage. Material arrived in the cylindrical receiving container from the coater. Portions were placed on the screen set and shaken manually. The size fractions were removed to storage cans, sealed, weighed, and removed. A maximum of 10 kg of ^{235}U was allowed at the station at any time (SNM 696, p. 48).

Station A19 – In-Process Storage. This station was similar to Station A3 or A3A (SNM 696, p. 48).

Station A20 – In-Process Storage Vault. The storage vault was a closed installation approximately 30 ft. square with 8-in.-thick reinforced concrete walls and a 6-in.-thick reinforced concrete ceiling. All entrances to the vault were secured by doors with combination locks except the east wall double doors, which were barred on the inside to provide an emergency escape. Storage racks for the material were laid out in arrays in the vault.

The SNM in the vault was stored in steel racks that were securely fastened to the walls or partitions. Individual storage locations were provided by specially installed metal fixtures or spacers that were designed according to the type of storage unit. Vertical shelf spacing was at least 16 in., and material was stored on the shelf was not less than 16 in. center-to-center spacing (SNM 696, pp. 48–49).

Station A21 – Oversized Material Ballmilling, Screening. Oversized dicarbide material from Station A13 was processed at this station by ballmilling to obtain the proper material size fractions. One or two ballmill units were used at this station as well as a screen set and as many as four containers at a time. Each unit had a volume of no more than 3.6 L, and the total station mass limit was 3.6 kg of ^{235}U. Racks were installed in the glovebox for each material container to provide a minimum of 16 in. center-to-center spacing and 12-in. surface for spacing between each container, each ballmill, and the screen set. All operations at the station were done in a closed glovebox to contain any dust.

At any one time, only a sealed container of oversized dicarbide material with up to 3.6 kg of ^{235}U was brought to the station; the container had a volume of 3.6 L or less. The material was transferred from the feed container into one or both of the ballmills. As product was formed it was removed and screened into three size fractions. Each size fraction, one at a time, was removed from the screen and placed in an individual container of 3.6-L volume or less. The material from each ballmill was handled alternately until the entire batch of the material was sized and put in one of the three size fractions bottles. All material of the given batch was removed before any additional material was charged to the station because the mass limit was 3.6 kg ^{235}U (SNM 696, p. 49).

Station A22 – Splitting. Batches of the dicarbide particle material from Station A13 or the coated particle material from Station A18 were blended at this station by successive splitting. Five splitters in a special fixture were set up in a series-parallel arrangement so, as the material passed through the first unit, it was split into two equal fractions. The fractions entered separate units simultaneously and were then split into two more fractions. Each of these fractions entered a splitter to be separated into two more fractions. Each of these fractions was then passed into an individual receiver container. A second batch of material was passed through the splitters and the eight fractions of the second batch were added to those of the first. The splitting and blending step was sometimes repeated for several batches of material with a total ^{235}U content of up to 28.8 kg, the station mass limit. All of the blending

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process occurred in a closed glovebox to contain any dust. As with all gloveboxes, it was connected to the facility ventilation system and absolute filters so when the glovebox material pass-through was open an inward airflow was provided to control air contamination outside the box.

The material was brought to this station in closed containers with a volume of 3.6 L or less and a mass of 2.6 kg ^{235}U or less. The container was opened inside the glovebox and then passed through the splitters (SNM 696, p. 49).

A2.0 FABRICATION OF COMPACTS COMPOSED OF COATED PARTICLES

Station B1-1 – Splitting. Uranium and/or uranium-thorium particles were brought to this station in a container whose volume was limited to 3.6 L and whose mass was limited to 3.6 kg of ^{235}U for blending by a splitting technique. The particles were poured into the splitting fixture such that the material was split into as many as 10 equal fractions. Each of the fractions flowed by gravity through tubing into as many as 10 individual receiver containers, each of which were limited to 3.6 L and 3.6 kg of ^{235}U. The splitter and tubing were emptied before another batch of particles was introduced. Once splitting was accomplished, the filled receiver containers were removed from their positions at the station one at a time.

There were actually two B1-1 splitting stations. The B1-1A station was in the vicinity of the A8 and A17 stations. The B1-1B station was in the vicinity of the A18 and A19 stations (SNM 696, p. 50).

Station B1 – Weighing. Coated uranium-thorium dicarbide particles were brought to this station and split and weighed into individual compact charges. The mass limit of this station was 0.35 kg of ^{235}U (SNM 696, p. 51).

Station B2 – Mixing. Several individual compact charges containing uranium-thorium dicarbide were brought to this station and mixed in several blenders with graphite, a binder, and the hydrogenous solvent. After mixing was completed, the single compact mixes were transferred to boats and placed in a drying oven. The mass limit of this station was 0.35 kg of ^{235}U (SNM 696, p. 51).

Station B3 – Drying. Boats that contained single compact mixes from Station B2 were placed in a drying oven and heated until the solvent evaporated. The mass limit of this station was 0.35 kg of ^{235}U (SNM 696, p. 52).

Station B4 – Transfer Cart. The material was transferred among stations on standard four-wheel carts with a special top to hold the various containers and compacts. Each cart contained 0.35 kg of ^{235}U or less. No more than one cart was allowed in any one station at a time other than Station B9, where there could be a maximum of three (SNM 696, p. 53).

Station B5 – Hot Pressing. Batches of carbon, binder, and uranium-thorium dicarbide were placed in the dies of a hydraulically operated, electrically heated press and heated to approximately 700°C at about 5,000 psi. The mass limit for this station was 0.35 kg of ^{235}U (SNM 696, p. 53).

Station B6 – Sintering Furnace Load Box. Compacts that were to be sintered were brought to this station on a transfer car, and entered an inert-gas-filled glovebox through an airlock. The compacts were then placed on graphite trays for introduction into the sintering furnace, Station B7 (SNM 696, p. 53).

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Station B7 – Sintering. Pressed fuel compacts were heated in a horizontal furnace to temperatures of up to 2,000°C. The furnace consisted of a resistance preheater, a quartz cylinder heated by water-cooled induction coils, and a water coil cooling section.

The compacts were placed on graphite trays that were introduced into the preheater end of the furnace, one behind the other, with each subsequent tray pushing the previous trays through the three sections of the furnace. A feed mechanism propelled the trays. The trays were pan-shaped with a holed insert such that eight compacts were equally spaced on the tray (SNM 696, p. 54).

Station B8 – Sintering Furnace Unload Box. Compacts that were to be sintered were brought to this station on a transfer cart, and entered an inert-gas-filled glovebox through an airlock. The compacts were then placed on graphite trays for introduction into the sintering furnace, Station B7 (SNM 696, p. 54).

Station B9 – Inspection. The loaded transfer cart with sintered compacts was brought to this inspection station. The compacts were removed from the cart one at a time, inspected, and placed on either off-specification compact transfer carts or acceptable compact transfer carts. The mass limit for this station was 0.35 kg of ^{235}U (SNM 696, p. 55).

Station B10 – Tube Loading. Compacts that passed final inspection were brought to this station on transfer carts for loading into special storage tubes. No more than one tube was loaded at a time. This station had a mass limit of 0.35 kg of ^{235}U (SNM 696 p. 55).

Station B11 – In-Process Storage. Storage tubes that were loaded with compacts at Station B10 were transferred, one at a time, to a special storage rack at this station. The storage rack was constructed of steel and secured to the building wall. Special hangers that would hold only one tube were attached to the rack (SNM 696, p. 55).

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Station C1 – Fuel Element Assembly. Compacts were brought to this station on a transfer cart and placed into a fixture that contained three horizontally oriented V-trays in a vertical plane. Each V-tray could contain up to 30 compacts. Three partially assembled fuel elements were placed in the same vertical plane as an extension of the V-trays and the compacts were loaded into the elements.

Each compact contained up to 10 g of ^{235}U and was 3 in. in length and 3 in. in diameter (SNM 696, p. 56).

Station C2 – Element Curing and Inspection. The partially assembled elements were removed from Station C1 one at a time and placed in a horizontal plane array. The remaining graphite components were joined to the elements and the cemented joints were cured in curing ovens. Movable tables were used to accumulate elements while other elements were curing (SNM 696, p. 56).

Station C3 – Storage. For storage, assembled fuel elements were placed into cans inside a specially designed container. The container held 25 fuel elements with a maximum weight of 7.2 kg of ^{235}U. The individual cans had a 4-in. outside diameter with a 0.5-in. wall thickness. The cans were placed in the container in a square array with a 4-in. pitch. The container was constructed of 0.25-in.-thick steel side plates with a plating of cadmium metal 0.0025 in. thick in the inside surfaces (SNM 696, p. 56).

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Station D1 – Weighing. SNM in a container with limits of 3.6 L and 3.6 kg of ^{235}U was sometimes withdrawn from the process or in-process storage for weighing, measuring, and sampling, and inspection (SNM 696, p. 61).

Station D2 – Chemical and Physical Laboratories. SNM was sometimes withdrawn from various steps of the process for chemical and physical analysis and for laboratory-scale processing and process investigation. The total amount of ^{235}U maintained at this station was limited to 350 g (SNM 696, p. 61).

Station E1 – Blending. Uranium dioxide was brought to this station from Station E2 or in-process storage in a batch that contained up to 790 g of ^{235}U, the station mass limit. The uranium was mixed in a batch-type blender with thorium dioxide, graphite flour, ethyl cellulose, and trichloroethylene. The thorium-to-^{235}U ratio was not less than 3.6 to 1 (SNM 696, p. 63).

Station E2 – Rolling. The mixed material was processed through a rolling mill to achieve homogeneity. Adjoining rolls turned in opposite directions and the material moved from the first roller to the last roller where it came off into a container and was returned to the mixing process. The mass limit this station was 790 g of ^{235}U (SNM 696, p. 63).

Station E3 – Drying. The mixture of uranium dioxide, thorium dioxide, graphite flour, and ethyl cellulose from Station E2 was poured into 1-in.-deep trays and the trays were placed into a horizontal drying oven, one behind the other, and conveyed to the opposite end of the oven. The oven when full could contain up to 8 kg of ^{235}U, the station mass limit. The trays were removed from the oven one at a time and taken to Station E4.

Heating was supplied by electric radiant heating that ran the length of the oven. Angle-iron framing on both sides of the tray conveyor restricted the areas such that trays could not be stacked (SNM 696, p. 63).

Station E4 – Unload and Weigh. The dried mixture of uranium dioxide, thorium dioxide, graphite flour, and ethyl cellulose was brought to this station from the drying ovens in individual pans. The pans were unloaded and the material weighed into a container with limits of 3.6 L and 3.6 kg of ^{235}U, the station mass limit. When a container was filled, it was transferred in single units by an elevator that ran through the unload scuttles to the scuttle of the Grinding-Screening Station, E5 (SNM 696, pp. 63–64).

Station E5 – Grinding–Sizing. Individual containers of the dried mixture from Station E4 were brought to this station and placed in a glovebox and a special storage fixture. The station mass limit was 8.5 kg of ^{235}U. The special storage fixture in the box provided for each container to have a minimum of 16 in. center-to-center spacing and 8 in. surface-to-surface spacing from any other container.

The dried material was poured from the containers, one at a time, to the funnel feeder, which was connected by a 3-in. pipe through the bottom of the glovebox to a rotary valve that fed a 5.0-L mechanical grinder. The ground material passed through a flexible tube to a screen separator unit to size the particles. The material was cascaded through a series of screen separators, each connected by 3-in. internal diameter tubes, until the desired material fractions were obtained. The material fractions (fines, oversize, product) were collected in individual containers of 3.6-L capacity or less below the final screen separator (SNM 696, p. 64).

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Station E6 – Conversion Crucible Loading. Individual containers of the dried mixture from in-process storage or Station E5 were brought to this glovebox and opened, and the material was transferred into a conversion crucible. More than one batch of material was sometimes required for a full crucible load; however, only one container was handled at a time at this station. All material transfers took place in the glovebox, so any dusting was carried off in the facility ventilation and ducting system. The station mass limit was 3.6 kg of ^{235}U (SNM 696, pp. 64–65).

Station E7 – Conversion. Converter crucibles that were loaded in Station E6 were placed in a furnace and heated to a maximum temperature of 2,800°C under a high vacuum or an inert gas atmosphere. The crucibles were then cooled and removed from a furnace to the Station A13 inert atmosphere glovebox for unloading. The crucible inside diameters were 5 in. or less. The station mass limit was 3.6 kg of ^{235}U (SNM 696, p. 65).

Station E8 – Screening. Coated particles of uranium-thorium dicarbide were brought to this station in a 5-in. inside diameter receiver that contained up to 790 g of ^{235}U, the station mass limit. The material was placed in a screen set and shaken into sized fractions. The sized fractions were caught in individual containers that were then removed from the station and weighed (SNM 696, p. 65).

Station F-1 – Rod Batching. A batch of coated uranium-thorium fuel particles was brought to the Station in a 5-in. inside diameter containers with a limit of 3.6 kg of ^{235}U. The batch container was connected by a 1-in. inside diameter pipe to a rotating metering drum. The metering drum filled with fuel particles contained up to a maximum of 20 g of ^{235}U. A conveyor below the metering drum extended between two 30-in.-diameter revolving tables. Empty vials were placed on a revolving table and were then fed onto the conveyor in a single line. The single line of vials was moved beneath the metering drum and the fuel particles in the drum orifices were emptied into the vials (up to 0.4 g of ^{235}U per vial) as they passed below the drum in synchronization with the drum movement. The loaded vials continued on the conveyor to the revolving take-up table. The vials were then manually placed into a metal loading board that had 30 locations in a 5-by-6 matrix with 1-in. center-to-center spacing. After the board was loaded with vials, it was placed on one of the transfer carts in the station or into Station F-2. The station mass limit was 4.2 kg of ^{235}U (SNM 696, pp. 70–71).

Station F-1A – Alternate Rod Batching. A batch of coated uranium-thorium fuel particles was brought to this alternate rod batching station in a 5-in. inside diameter container with a limit of 3.6 kg of ^{235}U. The batch container was placed on a fixture with a tube that extended downward into a preset metering, mixing, and splitting device. The splitting portion was connected by tubes that extended downward to vials on a metal loading board similar to the one at Station F-1. Volume of the tubes, device, and board was such that no more than 350 g of ^{235}U were present. When the operation at this station was completed, the loading board was placed in a transfer cart or into Station F-2. The station mass limit was 4.0 kg of ^{235}U (SNM 696, p. 71).

Station F-2 – In-Process Storage. The metal loading boards from Stations F-1, F-1A, and F-3 were placed in layers that had a minimum vertical spacing of 5 in.; the storage rack was constructed of metal and securely attached to a building member (SNM 696, p. 71).

Station F-3 – Fuel Rod Injection. The loading boards were brought to this station one at a time and placed in a fixture that transferred the fuel particles from the vials into a fuel rod die-mold. A mixture of pitch and graphite was then injected into the die-mold. The resultant fuel rods were injected into a metal loading board dimensionally like the particle loading board. The boards were placed in transfer carts or into Station F-2 (SNM 696, p. 72).

Station F-4 – In-Process Storage. Containers of fuel rods were placed in the special storage rack such that fuel rods in the containers formed a 4-in. planar array. The storage rack was constructed of metal and securely attached to a building member (SNM 696, p. 72).

Station F-5 – Rod Transfer Cart. Containers of fuel rods were placed on a cart for transfer between stations. The containers of fuel rods formed a 4-in.-thick slab in a horizontal planar array (SNM 696, p. 72).

Station F-6 – Inspection. Fuel rods were brought to the station and subjected to various inspection procedures. The operations of the station were such that fuel density in the station did not exceed 172 g of ^{235}U in any 1 ft^2 of aspect area as viewed perpendicular to the surface of the plane of the array (SNM 696, p. 72).

Station F-7 – Fuel Rod Low-Temperature Heat Treating. Transfer carts from Station F-3 were moved to this process level, where the rod loading boards were removed from the cart one at a time and the fuel rods were placed in graphite boats that were then introduced into a horizontal furnace. The furnace was constructed in such a manner that boats could not stack one on top of the other.

The graphite boats exited the furnace cooling section and were placed in a hood where the rods were removed from the boat, cleaned, and placed in a container. As each container was filled with fuel rods, it was removed from the station and placed in Station F-4, In-Process Storage, or moved to Station F-8, Fuel Rod High-Temperature Heat Treating (SNM 696, p. 73).

Station F-8 – Fuel Rod High-Temperature Heat Treating. Rods in graphite boats were introduced into another horizontal furnace. The furnace was constructed in such a manner that boats could not stand one on top of the other. The boats exited from the furnace cooling section, where the rods were removed from the boats and placed in other containers (SNM 696, p. 74).

Station F-9 – Fuel Element Assembly. Fuel rods in containers were brought to this station on the rod transfer cart. The fuel rods were inserted into the holes in the graphite element block. Single elements were vertically oriented on carts that restricted edge-to-edge spacing between adjacent elements to a minimum of 12 in.

The operations performed at this station included fuel rod loading, whole plugging, plug curing, and inspection (SNM 696, p. 74).

Station F-10 – Shipping Container Loading. Completed fuel elements were prepared for shipping and then loaded into the shipping containers. One element was handled at a time. After the element was loaded into the shipping container, the container was removed to storage to await shipment. Containers were stored in planar arrays no more than three layers thick (SNM 696, p. 76).

Station F-11 – Fuel Rod Cleaning. Fuel rods were brought to this station in graphite crucibles of 5-in. inside diameter or less and placed one crucible at a time, one behind the other, into the conversion furnace described as Station A12B. When loaded with crucibles, the furnace was heated and a mixture of gases that contained chlorine was introduced into the furnace to accomplish fuel rod cleaning through vaporization of heavy metal chlorides. Outlet gas was conducted to a fuel scrubber before it was vented (SNM 696, p. 76).

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Recovery Operations. Recovery of unirradiated ^{235}U was performed on GA-licensed and -contracted SNM, including that for AEC. These operations were performed in the recovery operations areas on the main floor and on the South mezzanine of the Fuel Fabrication Facility (see Figures 2-4 and 2-5). At no time were GA-licensed and -contracted SNM mixed or combined in a single process. The following three recovery processes were performed (SNM 696, p. 160).

- Conversion to Oxides. Recovery of contained SNM mixed with carbonaceous or other materials was done by oxidation. The combustible and volatile materials were burned off and residue oxides were recovered in the high-level furnaces in the fluid bed furnace. The oxides were sometimes further processed by grinding in the ballmill (SNM 696, p. 168).

- Uranium and Uranium-Thorium Dicarbide Size Reduction. Oversize dicarbide particles were reduced to the desired size range by grinding. This size reduction was performed with the ballmill in a controlled-atmosphere glovebox (SNM 696, p. 168).

- Burning Contaminated Carbon. In the fabrication of uranium carbide fuels, the carbonaceous materials and combustible materials associated with the manufacturing became contaminated with uranium. This material was reduced to an ash by burning in the low-level furnace and the ash was collected for recovery of the uranium. Only dry uranium-carbon or uranium-thorium carbon materials were processed in this operation (SNM 696, p. 168).

Station R1 – In-Process Storage. Storage of in-process material was in a metal storage rack section made from module assemblies. Each module consisted of a sheet-metal assembly that was fabricated to form shelf units on a common vertical axis. The shelf units were designed to have a center-to-center spacing of 18 in. A completed storage rack was made by connecting module assemblies to form a two-dimensional plane array with 8-in. surface-to-surface and 16-in. center-to-center spacing between maximum storage units. Nuclear material was stored in a rack in containers with a volume of 3.6 L or less and a mass limit of 3.6 kg of ^{235}U (SNM 696, p. 162).

Station R3 – High-Level Material Transfer. The transfer of all dry in-process material for the high-level furnaces from storage containers to process equipment and from process equipment back into storage containers occurred in this glovebox. The glovebox was coupled directly to the loading flange of the high-level furnace and thus effectively controlled the spread of dry or dusty materials. The glovebox was connected to the facility ventilation duct system and absolute filters. A minimum inward air velocity of 150 ft/min was provided by the ventilation system when the loading door of the glovebox was open. The station had a mass limit of 3.6 kg of ^{235}U (SNM 696, p. 162).

Station R4 – High-Level Uranium Oxidation and Reduction. The oxidation of uranium carbide or uranium-thorium carbide material of 10% to 90% ^{235}U was carried out in a controlled oxygen-air atmosphere. Reduction of the oxidized material was carried out in a hydrogen atmosphere. The nuclear material was placed in five steel material trays that were in the central 18 in. of the furnace-heated zone. A tray rack fixed the trays in the 7-in.-diameter liner. Heat was supplied by the cylindrical electric heating element. The gas atmosphere of the furnace was supplied through the inlet gas tube. Cylindrical blocks of solid silica insulation were placed in each end of the liner to control heat loss.

During operation, the temperature in the furnace was controlled to around 1,800°F during the oxidation stage and 1,200°F during the reduction stage. The inlet gas to the furnace was automatically controlled, and alarms sounded if there was a failure of the heating element, nitrogen gas pressure, or diluting airflow.

The sequence of furnace operation was as follows:

1. Oxidation at 1,800°F with air and excess oxygen

2. Furnace cooling with nitrogen purge gas to room temperature when the oxidation product was desired or to 1,200°F when reduction of the oxidized material was desired

3. Reduction at 1,200°F with hydrogen gas atmosphere

4. Furnace cooling with nitrogen purge gas to room temperature

The furnace was mounted horizontally on a steel table with the centerline of the furnace liner about 36 in. above the floor. A portable hood was provided for servicing the ash filter. The hood was ventilated such than a minimum air velocity of 150 ft/min was provided. The mass limit for the station was 3.6 kg of ^{235}U (SNM 696, pp. 162–163).

Station R4A - Fluid Bed Furnace. An alternative fluid bed furnace was used for the oxidation or reduction of dry material that contained recoverable SNM. The fluid bed furnace was a vertical cylinder with an external heat source. Material to be burnt was brought to the station in a closed container with a 5-in. or less inside diameter that was attached to the feed mechanism of the furnace. As material was fed into the furnace, it was levitated into the hot zone by air-oxygen gas. When the burning was completed, the impure oxide was removed from the furnace or in some cases reduced by the introduction of hydrogen gas to the furnace. The station mass limit was 4.0 kg of ^{235}U (SNM 696, p. 164).

Station R4B – Calcining. Calcination of uranium precipitate took place at this station. The precipitate was changed to each furnace action in a boat with the capacity of no more than 1.6 L and a maximum uranium capacity as precipitate of no more than 750 g. This precipitate was heated to 900°C in air to oxidize the uranium to U_3O_8. The product was cooled and removed from the boats and placed in polyethylene bottles with a volume of 3.6 L or less. Each bottle held a maximum of 3 kg of unscreened U_3O_8. The station mass limit was 3.6 kg of ^{235}U. The boats were loaded, charged to the furnace, unloaded, and emptied one at a time to preclude the possibility of stacking loaded boats on top of one another (SNM 696, p. 165).

Station R5 - Low-Level Uranium Oxidation. Oxidation of uranium dicarbide material of 10% ^{235}U or less was carried out in an excess oxygen atmosphere similar to that at Station R4. The material charged to the low-level furnace was limited to 300 g of ^{235}U or less. A combustible charged container was loaded and placed in the furnace. No external source of furnace heating or cooling was required at the station. The charge material was ignited inside the furnace, the furnace closed, and oxygen supplied through the nozzle to support combustion and provide the excess oxygen atmosphere. The low-level furnace was mounted inside the hood box. The discussions at Station R4 for hood ventilation, inlet gas control, and only gas handling were directly applicable to this station.

After each batch of material was burned, the furnace was cooled and the ash was removed and placed into a polyethylene bottle with a volume of 3.6 L or less. The furnace was checked visually and with the counter for any accumulation of ash. If the counter indicated an accumulation of 50 g of

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^{235}U in the furnace, the furnace was torn down and rebuilt. The relationship between accumulated ^{235}U and the counter reading was established by calibration. Mass limit for the station was 350 g of ^{235}U or less (SNM 696, p. 165).

Station R6 - Size Reduction. The reduction of the particle size of dry uranium-carbon material was carried out in a sealed glovebox provided with a controlled gas atmosphere. Nuclear material was brought to the station a single sealed polyethylene bottle with the limit of 3.6 kg of ^{235}U or less and a capacity of 3.6 L or less.

The nuclear material was ground in a standard ballmill. Product from the ballmill was screened and divided into specification and off-specification material. The screened material was placed into separate bottles with limits of less than 3.6 kg of ^{235}U and 3.6 L (SNM 696, pp. 165–166).

Station R7 – Grinding. Dry uranium-bearing material was brought to the station in a container of 3.6-L capacity or less with up to 3.6 kg of ^{235}U, the station mass limit. The material was fed to an enclosed grinding mill with a total volume, including the feed funnel and catch receiver, of less than 3.6 L (SNM 696, p. 166).

Station R17 – Higher Level Uranium Oxidation. An alternate station for burning uranium-bearing materials consisted of four horizontal furnaces in a planar array on a 16-in. center-to-center spacing. The dimension of each furnace was 2.5 in. high by 8 in. wide by 14 in. deep. This was brought to the station in a container with a maximum volume of 3.6 L and up to 3.0 kg of ^{235}U, the station mass limit. The material was charged to each furnace action in a boat with up to 750 g of ^{235}U. No more than one boat that actually contained material was outside a furnace at any one time. After the material was burned and cooled, the boats were removed one at a time and the contents were emptied into a container with a volume of 3.6 L or less (SNM 696, p. 166).

Station R18 – Blending. Dry uranium-bearing material was brought to the station in a container with a volume limit of 3.6 L and containing up to 3.6 kg of ^{235}U, the station mass limit. The material was fed into blender with a volume of 3.6 L or less (SNM 696, p. 167).

Station R19 – Grinding. Dry uranium-bearing material was brought to the station in a container with limits of 3.6 L and 3.6 kg of ^{235}U. The material was fed to a hopper and grinder with a total volume less than 3.6 L. The ground material passed through a pipe through the bottom of the glovebox to a sealed receiver hopper with a volume limit of 3.6 L or less (SNM 696, p. 167).

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Station P1 – In-Process Storage. SNM in-process was stored in containers with limits of 3.6 L and 3.6 kg of ^{235}U. Each storage location of the in-process storage rack was limited to a single container. Storage locations were on 16-in. centers (horizontal and vertical) with a minimum of 8-in. surface spacing between containers and adjacent locations. The storage racks were constructed of metal and securely fastened to the building structure (SNM 696, as renewed, p. 193).

Station P9 – In-Process Storage. Liquids from the various steps of the uranium purification process were stored at the station. These liquids might have been from the dissolution step, organic in aqueous extraction steps, or the precipitation and filtering steps. The uranium content of liquids that were transferred to this station was determined by chemical analysis or gamma accounting before storage.

Storage containers at this station were 5-in. inside diameter cylinders. Each cylinder was secured in a specially constructed metal rack with a minimum 12-in. surface spacing between adjacent cylinders (SNM 696, p. 193).

Station P2 – Weighing. A single closed container of uranium-bearing scrap material was brought to this station and weighed into a batch container. The balance was enclosed in a hood box with a vertical opening front panel. The hood was connected to facility ventilation duct and absolute filters. When the front panel was opened, a minimum inward air velocity of 150 ft/min was provided.

Nuclear material was handled at the station in two containers each with a volume of no more than 3.6 L and up to 3.6 kg of ^{235}U. A fixture was provided to space raw material containers with at least 16-in. center spacing and 12-in. surface spacing from the balance. The batch container was placed on the balance of the raw material transferred and weighed into the batch container. More than one raw material container might have been needed to make up the batch; however, the first container was emptied and removed before a second container was brought to the station (SNM 696, pp. 194–195).

Station P3 – Dissolution. A weighed batch of raw material was brought to the station and placed in a special container on an electric hot plate. The container was made of special acid-resistant material and had a volume no greater than 5 L. While more than one weighed batch of material might have been brought to this station at any time and placed in the dissolution container, the total combined batch size was limited to 3.6 kg of ^{235}U, the station mass limit. An acid solution was slowly added to the raw material in the container. The mixture was heated and allowed to react until all the raw material was dissolved or the solids and liquid reached equilibrium. A special temperature control was provided on the hot plate to prevent damage to the dissolution container or boiling over of the acid solution. However, if either event did occur, a 5-L catch pan in which the dissolution container sat prevented spilling. The dissolution station was fully enclosed in a special acid hood and equipped with ducts and filters suitable for handling any acid fumes that were generated. Tell-tales that indicated the amount of air draft through the hood were installed to indicate plugged duct system filters.

If material residue remained in the dissolution container, the entire contents were filtered to separate the residue and the solution. The filter and filtrate receiver were each limited to a volume of 5 L and there was no more than 3.6 kg of ^{235}U at the station. Filtrate from this station was transferred either to Station P4 or liquid storage in Station P9 (SNM 696, pp. 195–196).

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Station P3A – Dissolution. Weighed batches of uranium-bearing material and up to two 5-in. inner diameter scene hoppers were brought to this alternate dissolution station one at a time and attached to two vertical 5-in. inner diameter cylinders on 17-in. center-to-center spacing and 12-in. surface-to-surface spacing. The total combined batch size was limited to 3.6 kg of ^{235}U, the station mass limit. An acid solution was added to the columns, and heat was applied until dissolution was complete.

The dissolved material was then pumped from the dissolver cylinders to an adjacent 5-in. inner diameter pressure vessel cylinder where it was pressurized to an adjacent filter to remove the solids from the solution. The filter and a volume envelope that was 5 by 5 by 16.5 in.. The filtered solution was accumulated in an adjacent 5-in. inner diameter cylinder, then transferred to Station P9, In-Process Storage (SNM 696, p. 197).

Station P4 – Extraction. The dissolved material from Station P3 was moved to this extraction of station one container at a time and transferred to the first stage of the extraction unit. More than one dissolution batch might have accumulated in the first extraction stage, but the unit and a total volume limit of 5 L and the total extraction batch size was limited to 3.6 kg of ^{235}U, the station mass limit. Up to three steps of extraction were carried out at this station – organic extraction of dissolved uranium-bearing scrap, aqueous extraction of the impurities from the organic phase, and an aqueous extraction of the uranium from the organic phase. For each type of extraction several stages of contacting and separation might have been required.

Each extraction unit was identical and had a total unit volume of 5 L. Contacting of solutions in the extraction unit was accomplished by mechanical or other comparable means of agitation. Phases were transferred from one unit to another or two liquid waste containers by gravity flow. Individual extraction units were securely mounted to a metal rack that was securely mounted to the building.

Dissolved material was transferred into the first unit, which contained a premeasured amount of the organic extraction agent. The solutions were contacted in the mixture and allowed to stand and separate. The uranium selectively went into the organic phase. Separation was accomplished by draining off the aqueous phase and leaving the uranium-containing organic phase in the first unit.

As the aqueous phase flowed from the first extraction unit, it flowed directly into a second unit that contained a second premeasured amount of organic extracting agent. The second extraction was completed with the aqueous phase sometimes being transferred to a third extraction unit with a premeasured amount of the organic extracting agent. This step was repeated as required. When the final extraction step was completed, the aqueous phase stripped of uranium was sampled; the sample was gamma-counted and the aqueous solution was transferred to liquid storage.

From each extraction stage, the organic phase that contained the uranium was transferred and accumulated in a single extraction unit. At this point in the process, all of the medium that was initially charged to the extraction station, except for a trace amount that might have been present in the waste aqueous solution, should have been present in a single extraction unit. The second type of extraction began at this step with the addition of a premeasured amount of an aqueous scrubbing solution. The solution was contacted in the mixture and was then allowed to stand and separate into the light organic phase and the denser aqueous phase. The uranium material remained in the organic phase. Separation was performed by draining the aqueous phase into a second extraction unit.

The organic phase that remained in the first extraction unit was contacted a second time with another charge of the aqueous scrubbing agent. The aqueous phase was drained into a second extraction

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unit and added to the material obtained from the first extraction stage. Sometimes a third extraction was performed on the organic solution with the aqueous phase being combined with that of the previous extractions. The accumulated aqueous solution was transferred from the second extraction unit into the third unit and a premeasured amount of organic extraction agent was added. The solution was contacted and allowed to separate. Any uranium that was extracted by the aqueous scrubbing solution selectively went into the organic phase. The aqueous phase, stripped of uranium, was separated and sampled, and the sample gamma counted in the solution was transferred to a liquid storage container at Station P9. The organic phase was drained and added to the organic phase from the first extraction unit.

The clean organic solution in the first extraction unit contained all the uranium that was charged to the extraction station except a small amount that might have been present in the waste aqueous solutions. The aqueous extraction of the uranium from the organic phase began at this step with the addition of a premeasured amount of the aqueous solution. Contacting of the solutions was done and the material was allowed to stand and separate into the light organic phase and the denser aqueous phase. The uranium material selectively went into the aqueous phase. Separation was accomplished by draining the aqueous phase into the second extraction unit. The above steps were repeated until all the uranium, except for trace amounts, was contained in the new aqueous solution. This aqueous solution was then placed in one of the 5-L extraction containers (SNM 696, pp. 199–200).

Station P5 – Precipitation. A single container of the aqueous phase that contained the purified uranium in solution was brought to this station from the final stage of the extraction process. The container of product from Station P4 was limited to 5 L and no more than 3.6 kg of ^{235}U, the station mass limit. The container of solution was placed on a hot plate, heated, and maintained at a temperature of about 90°C. At this point, a basic reagent was added that resulted in the precipitation of the uranium.

A special temperature control was provided on the hot plate to prevent damage to the container or boiling over of the solution. The precipitation station was fully enclosed in a special acid hood with ducts and filters suitable for handling the acid fumes that were generated (SNM 696, p. 200).

Station P6 – Filtration. A container of aqueous solution with a precipitated uranium salt was brought to this filtration station. The uranium was separated from the aqueous solution by vacuum filtration. While more than one container with the precipitated material was allowed at this station at any given time with their contents transferred to the filter, the total batch accumulated in the filter was limited to a volume of 5 L and 3.6 kg of ^{235}U, the station mass limit. The filtrate was transferred into a liquid storage container at Station P9.

The filter cake that contained the uranium precipitate was air dried, removed from the filter, and placed in a close container with a maximum volume of 3.6 L. While it was possible that more than one container was required for a batch of the filter material, no more than one container was handled at this station at a time and it was limited to 3.6 kg of ^{235}U. As each container was loaded, it was closed and placed in a storage location in an in-process storage rack at Station P1. The material was taken from Station P1 to be calcined at Station R4B (SNM 696, pp. 201–202).

Station P10 – Large Batch Extraction-Precipitation. At this station to dissolve batch material from Station P9 with up to 3.6 kg of ^{235}U was pumped to the first of nine vertically oriented columns in a single planar array. The columns had a length of up to 11 ft and inside diameter of 5 in. in a surface-to-surface spacing of 12 in.

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Organic extracting agents were pumped to the first column, and the solution was then circulated between the first and second columns through a 2-in. inner diameter pipe that was connected to the top until extraction was completed.

The organic phase that contained the extracted uranium was pumped from the first two columns to the third column where scrubbing agents were added to remove impurities.

The pregnant organic phase was then pumped from the third column to the fourth column where scrubbing agents were added to separate the uranium from the organic phase.

The resultant pregnant aqueous phase was then pumped from the fourth column to the first column where washing agents were added to remove any residual impurities. Agitation of the solution was accomplished in the third fourth and fifth columns by air sparging.

The pregnant aqueous phase was then pumped from the fifth column to one or more precipitation columns where a basic reagent was added that resulted in the precipitation of the uranium.

The discard solution was drained from the bottom of each column into an 11-L bottle (SNM 696, pp. 202–203).

Station P6A – Filtration. The aqueous solution with a precipitated uranium salt and the precipitation columns was pumped to this alternate filtration station. The uranium was separated from the aqueous solution by a filter method. A total accumulated batch on the filter did not exceed 4.8 L, the station volume limit, or 3.6 kg of ^{235}U, the station mass limit. The aqueous discard was drained to a critically safe container. No more than one batch was processed through the filter at one time and all solids and liquids were removed from the station before a second batch was filtered. The uranium in the filter cake was placed in a container with a maximum volume of 3.6 L and transferred to Station P11, Calcining (SNM 696, p. 203).

Station P11 – Calcining. The filter cake from Station P6A was brought to this station in a container with a maximum volume of 3.6 L and up to 3.0 kg of ^{235}U, the station mass limit. This station consisted of four horizontal furnaces oriented in a planar array on 16-in. center-to-center spacing. The dimensions of each furnace were 3.5-in. high by 8 in. wide by 8 in. long.

The filter cake was charged to each furnace section in a boat with up to 750 g of ^{235}U. No more than one boat that actually contained material was outside the furnace at any time. After the material was burned the boats were removed one at a time and transferred to Station P12, Grinding-Sizing-Blending (SNM 696, p. 204).

Station P12 – Grinding-Sizing-Blending. Uranium-bearing material was brought to this station in a container with a maximum volume of 3.6 L and up to 3.6 kg of ^{235}U, the station mass limit. The material then underwent separate grinding, sizing, blending, or a combination (SNM 696, p. 204).

ATTACHMENT C
ALLOWABLE QUANTITIES OF RADIOACTIVE MATERIAL AT GENERAL ATOMICS

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Table C-1 provides a snapshot of the allowable quantities of radioactivity at GA. The SNM licenses were in effect before 1973. Radionuclides and allowable quantities might have changed over time but, based on site descriptions, health physics reports, and inventory reports, the materials in Table C-1 seem generally representative of the variety and the upper limits on source term during the covered period.

ATTACHMENT C
ALLOWABLE QUANTITIES OF RADIOACTIVE MATERIAL AT GENERAL ATOMICS

Table C-1. Summary of licensed source terms.

Date	License	Material	Description	Form	Quantity	Total	Use
ca. 1960	AEC License No. 4-1611-11 (J-60)	Byproduct materials, atomic numbers from 3 through 83		Any	10 Ci each	500 Ci	
		H-3			200 mCi		
		Sr-90		Sealed	1 Ci		
		Sr-90		Unsealed	100 mCi		
		Po-210		Sealed	50 Ci		
		Po-210		Unsealed	100 mCi		
		Tm-170/171			50,000 Ci		
Dec. 1960	AEC License No. 4-1611-11 (J-60), proposed increase	Byproduct materials, atomic numbers 3 through 83	Any		100,000 Ci each	500,000 Ci	
	Proposed addition	Fission products	x		1,000,000 Ci		
1973	SNM-696, Amendment No. 45	U	Up to fully enriched in U-235		985 kg		Fabrication of fuel materials and fuel elements
		U	Up to 100% enriched in U-235		9.5 kg		Fabrication of fuel materials and fuel elements
		U	Up to fully enriched in U-233		2.5 kg		Fabrication of fuel materials and fuel elements
		Pu			3.0 kg		Fabrication of fuel materials and fuel elements
ca. 1973	SNM-69	U-235	Up to fully enriched	Metal, metallic alloy, or compounds	1,600 kg		
		U-235	Up to 100% enriched	Metal, metallic alloy, or compounds	10 kg		
		U-235	100%	Metal, metallic alloy, or compounds	2.5 kg		
		U-236*	100%	Metal, metallic alloy, or compounds	300 g		*As residue part of material enriched U-235
		Pu	100%	Metal, metallic alloy, or compounds	13 g		
ca. 1973	SNM-696	U-235	Fully enriched	Metal, metallic alloy, or compounds	2,500 kg		
		U-236*	100%	Metal, metallic alloy, or compounds	1 kg		*As residue part of material enriched U-235
		U-233	Fully enriched	Metal, metallic alloy, or compounds	500 g		
		U-233	Encapsulated	Metal, metallic alloy, or compounds	10 g		
		U-235	100%	Metal, metallic alloy, or compounds	10 kg		
c. 1973	State of California	Any radioactive material (except source and SNM)		Any	30 Ci each nuclide	1,500 Ci	

ATTACHMENT C
ALLOWABLE QUANTITIES OF RADIOACTIVE MATERIAL AT GENERAL ATOMICS
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Date	License	Material	Description	Form	Quantity	Total	Use
		Any radioactive material or activated material (except source and SNM)		Any	100,000 Ci each nuclide	500,000 Ci	
		Thorium (natural)		Any	200,000 lb		
		Uranium (natural or depleted)		Any	400,000 lb		
		C-14		Sealed sources	50 mCi		
		Cs-137		Sealed sources	40 Ci		
		Co-60		Sealed sources	20 Ci per source		
		Sr-90		Sealed sources		4,800 Ci	
		Ir		Sealed sources			
		Po-210		Sealed sources		50 Ci	
		Ra-226		Sealed sources		47 mCi	
		U-235		Sealed sources	50 g		
		Pu-238 and 239		Sealed sources	30 g		
		Am-241		Sealed sources		0.1 Ci	
		Cf-252		Sealed sources		70 Ci	
		Co-60		Metal discs	100,000 Ci		
ca. 1973	AEC R-38 (TRIGA Reactor)	U-235		SNM in fuel elements and experiments	5 kg		
ca. 1973	AEC R-67 (TRIGA Reactor)	U-235		SNM in fuel elements	30 kg		
		Pu		SNM in sources	1 g		
		U-235		SNM in experimental elements	300 g		
ca. 1973	AEC R-100 (TRIGA Reactor)	U-235		SNM in fuel elements	10 kg		
		Pu		SNM in fuel elements	30 g		
		Po-210 Be		Sealed sources	15 Ci		
		Sb-124 Be		Sealed sources	150 Ci		
		U-235		SNM in experimental elements	806 g		

ATTACHMENT D
LABORATORY AND FACILITY THROUGHPUT INFORMATION FOR 1972

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Table D-1 provides a snapshot of the SNM that might have been used in the Science Laboratory in 1972. It is assumed that throughput means the amount of material used in 1972, and that the total amount was the amount on hand at the end of 1972.

Table D-1. Science Laboratory 1972 annual inventory information.[a]

Location	1972 throughput (g)			1972 total amount (g)		
Laboratory	U-235	U-233	Pu	U-235	U-233	Pu
L-122	200			90		
L-134	100			100		
L-137	3,000			499		
L-139	250			1,000	10	
L-141	3,000			499		
L-143	3,000			499		
L145	<200			Not to exceed 100		
L-147	4,000			499		
L-154	4,000			499		
L-218	80	1	1	10	0.1	0.1
L-230	90			500		
L-232	4,000			499		
	1,000	0.20	0.02	360	0.20	0.02
L-234	4,000			<1		
	1,150			499		
	1,000			499		
		0.20	0.02	350	0.20	0.02
L-236	4,000			499		
	1,150			499		
	1,000	0.20	0.02	350	0.20	0.02
L-240	1	1		10^{-3}	10^{-4}	
	3,000			499		
L-242	7			100		
	1,000	100		<1	10	
L-307	<200			Not to exceed 100		
	7			1,000		
	250	3		350	10	
	1,000	0.20	0.02	<1	0.20	0.02
L-311	250			1,000	10	
L-313	1		0.1	10^{-3}	10^{-4}	
	3,000					
L-315	4,000			499		
L-317	1	0.1		10^{-3}	10^{-4}	
	3,000			499		
L-331						
L-333						
L-347 through L-361			80			80
L-359 and L-361						
L-401 and L-403	4,000			499		
L-405	4,000			499		

ATTACHMENT D
LABORATORY AND FACILITY THROUGHPUT INFORMATION FOR 1972

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Location	**1972 throughput (g)**			**1972 total amount (g)**		
Laboratory	**U-235**	**U-233**	**Pu**	**U-235**	**U-233**	**Pu**
L-407	4,000			499		
	1,000	0.20	0.02	350	0.20	0.02
L-409	4,000			499		
L-411	4,000			499		
L-413	3,500			350		
	250	0.500		1,000	10	
L-415	250			1,000	10	
L-417 L-419 L-421	500			500		
L-425 L-427	500			6,000		
L-429	4,000			500		
				<1		
L-441						
L-443	700			350		
L-530						
L-540	7			<1		
	5					
L-543	3,000			350		
L-545	50			10		
L-548	50			10		
L-550	50			10		
L-552	50			10		
L-554	3,000			300		
L-556	3,000			300		
	7					
	25			300		
L-558	3,000			300		
L-635 L-637 L-639	300			600		
L-647	3,000			350		
Activation Analysis						160 g
	270 kg	20 g	220 g	270 kg	20 g	220 g
	8,000 g			350 g		
	10 kg		5 g	400 g		5 g
Experimental Building	50,000g			50,000 g		
			40 curies			40 curies
E-141	3,000 g			350 g		
TRIGA	<200 g			Not to exceed 100 g		
	3 kg			350 g		
	3,000 g			499 g		
	190 g	1 g	1 g	900 g	5 g	5 g
	190 g			<190 g		

ATTACHMENT D
LABORATORY AND FACILITY THROUGHPUT INFORMATION FOR 1972
Page 3 of 3

Location	1972 throughput (g)			1972 total amount (g)		
LINAC			10^{-7} g			
	0.15 g			600 kg		
Bldg. 512			<1.5 g	725 g		<1.5 g
	15 kg	1000 g	1,000 g	725 g	299 g	299 g
			604 g			604 g
Waste Yard	24 kg	50 g	10 g			
	24 kg					
	5,000 g			4,000 g		
	15,000 g			500 g		
SVA-Tunnel Glovebox	3.6 kg			NA[b]		
SVA-Quality Control/Fuels	NA			350 g 350 g		
SVB	1,000 g			350 g		
	2,500 g			700 g		
	3,600 g			1,750 g		
	500 g			5,000 g		
	5 kg			500 g		
	350 g			350 g		
Hot Cell	2,000 g	100 g	100 g	2.1 kg	100 g	100 g
	<200 g			Not to exceed 100 g		
Hot Cell Radiography Room	5 g			100 g		
HTGR	87 kg			2.52 kg		
Convoy Court	350 g			10 kg		
Machine Shop	350 g			10 kg		
Barrel Storage Yard	700 kg			NA		

a. GA (1972).
b. NA = not available.

ATTACHMENT E
EXAMPLES OF ACTIVITIES AND AIR MONITORING IN THE PLUTONIUM LABORATORY

Table E-1 provides information on work and associated monitoring in the Plutonium Laboratory. No evidence of plutonium bioassays have been found to date. Well after 1970, lung counts might have been able to detect plutonium or ^{241}Am.

Table E-1. Plutonium Laboratory workplace air sample results.

Period	Alpha (μCi/ml)	Beta (μCi/ml)	Comments
November 1968	<5E-15	<6.4E-13	On November 8, a survey following classified Pu-contaminated fuel particle work indicated <15 pCi beta removable and <100 cpm alpha on a PAC-1A air proportional survey meter, but on November 21, a masslinn sweep detected 800 pCi beta on the floor in the room behind the LLC.[a]
October 1968	<3.6E-14	<7E-15	A masslinn sweep on October 16 detected 4,000 cpm tactile contamination on the floor that was subsequently decontaminated to <15 pCi beta and <3 pCi alpha.[b]
January to September 1968			No operations involving Pu were performed.
December 1967			No operations involving Pu. Continuous air samples showed no detectable airborne radioactivity.
November 1967			Limited operations in the Plutonium Laboratory.
October 1967			Sintering, sectioning, and counting graphite Pu samples in glovebox complex continued throughout month. No samples were removed from “enclosures” in October. Air and wipe samples indicated “no detectable” contamination.[c]
September 1967			Sintering, sectioning, and graphite Pu sample counting was performed in glovebox complex. No samples were removed from “enclosures” in October. Air and wipe samples indicated “no detectable” contamination.[d]
August 19 to September 1, 1967			No operations involving Pu were performed.
August 1 to August 18, 1967			Very little work was performed. Air and wipe sampling indicated no “significant airborne or tactile” contamination.
April 1967			No Pu work was performed during this period.
February 1967			No Pu work was performed during this period. Two g of Pu were still in storage in one of the gloveboxes. Periodic masslinn wipes indicated no tactile contamination.
January 1967	2.1E-13	8.8E-14	Four g of Pu-239 O_2 was separated into 2-g lots with 1 g going to Science Building. Material removed from the gloveboxes was wiped and found free from significant contamination.[e]

ATTACHMENT E
EXAMPLES OF ACTIVITIES AND AIR MONITORING IN THE PLUTONIUM LABORATORY

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Period	Alpha (µCi/ml)	Beta (µCi/ml)	Comments
December 1966			PuO_2 was dissolved in acid. Wipe and air samples collected during and following the operation indicated no significant transfer of activity. 100 ml of Pu-239 solution was removed and double-bagged and checked for contamination. A wipe sample of the container taken inside No. 2 glovebox showed an activity level of 200 cpm gamma through the envelope containing the filter paper. The container was removed from glovebox No. 3 into two double bags and placed inside a heavy receiving bag. The pass-through bag was then removed and placed inside the Chemistry hood. The flask inside the container was then removed in the hood and placed inside double bags. The outside of all bags removed from the plutonium facility were wipe-checked and found free of alpha contamination. The gamma level was <1 mR/hr at the surface.[f]
November 1966			The Plutonium Laboratory was activated for storage of 115 mg of PuO_2 in No. 1 glovebox. Both inside and outside containers were checked by wipe sampling and found free of contamination.[g]
October 1966			All gloveboxes were pressure-tested, absolute filters were in the exhaust system, and radiation detection instruments were in place. The area was free of alpha and beta contamination. U-232 would be the first isotope used in the facility.[h]

a. Cockle (1968a).
b. Cockle (1968c).
c. Kesting (1967d).
d. Kesting (1967e).
e. Gurren (1967c).
f. Gurren (1967d).
g. Gurren (1966d).
h. Gurren (1966a).

ATTACHMENT F
LINAC RADIOACTIVE MATERIAL PRODUCTION INFORMATION

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The source term in the LINAC was as varied as the targets that were irradiated. Table F-1 shows estimated production rates for the 45-MeV accelerator operating at 25-MeV electron energy (Author unknown, undated). The following was assumed:

- Accelerator is operating at 25 MeV.
- Tungsten converters are used to produce bremsstrahlung.
- Estimates, except where noted, are based on a 1-g sample 1.5 in. in front of the photon converter.
- Average dose in the 1-g sample is ~80% of the peak surface dose.
- Activities are given as of shutdown.
- Actual production conditions had not yet been explored.

Table F-1. LINAC radioactive material production information (Author unknown 1970, pp. 4–7).

Radionuclide	Half-life	Irradiation time[a]	Activity produced	Accuracy[b]	Pile isotope available	Sample material[c]
H-3	12.46 yr	1 hr	104 pCi	1	Yes[d]	He
C-11	20.4 m	S	415 mCi	1	No	C
N-13	9.93 m	S	208 µCi	1	No	N_2
N-17	4.14 s	S	3.6 µCi	1	No	O_2
O-15	118 s	S	156 mCi	1	No	O_2
F-17	70 s	S	3.6 mCi	1	No	LiF
F-18	112 m	1 hr	78 mCi	1	No	LiF
Ne-19	18.2 s	S	386 µCi	1	No	Ne
Na-22	2.60 yr	1 hr	21 µCi	1	No	Na
Na-24	15.1 hr	1 hr	2.4 mCi	1	Yes[d]	Na
Na-25	60 s	S	47 mCi	1	No	Na
Mg-23	11.9 s	S	300 mCi	1	Yes	Mg
Al-26	6.5 s	S	47 mCi	1	No	Al
Al-28	2.27 m	S	47 mCi	1	Yes	Si
Al-29	6.56 m	S	26 mCi	1	No	Si
Si-27	4.9 s	S	31.2 mCi	1	No	Si
P-30	2.55 m	S	571 mCi 10 mCi	1	No	P S
S-31	3.2 s	S	364 mCi	1	No	S
Cl-39	55.5 m	S	2 mCi	1	No	Ar
Ar-39	265 yr	1 hr	520 pCi	1	Yes	Ar
Ca-39	1.06 s	S	200 mCi	1	No	Ca
Sc-47	3.43 d	1 hr	67 µCi	1	No	V
V-52	2.6 m	S	26 mCi	1	Yes	Cr
Cr-49	419 m	1 hr	31 mCi	1	No	Cr
Mn-54	310 d	1 hr	182 µCi	1	No	Mn
Fe-53	8.9 m	S	83 mCi	1	No	Fe
Fe-55	2.94 yr	1 hr	47 µCi	1	Yes[d]	Fe
Co-57	270 d	1 hr	26 µCi	1	No	Ni
Co-58	72 d	1 hr	780 µCi	1	No[d]	Co
Co-61	99 m	1 hr	2 mCi	1	No	Cu
Ni-57	36 h	1 hr	16 mCi	1	No	Ni

ATTACHMENT F
LINAC RADIOACTIVE MATERIAL PRODUCTION INFORMATION
Page 2 of 2

Radionuclide	Half-life	Irradiation time[a]	Activity produced	Accuracy[b]	Pile isotope available	Sample material[c]
Cu-61	3.33 hr	1 hr	7 mCi	1	No	Cu
Cu-62	10.1 m	S	1.2 Ci	1	No	Cu
Cu-64	12.8 hr	1 hr	68 mCi	1	Yes[d]	Cu
Zn-63	38.3 m	S	1.3 Ci	1	No	Zn
As-73	76 d	1 hr	125 µCi	1	No	As
As-74	17.5 d	1 hr	3.9 mCi	1	No	As
As-77	38 hr	1 hr	18 µCi	1	No	Br
			16 µCi	1	Yes[d]	U
Br-78	6.4 m	S	830 mCi	1	no	Br
Br-80	18 m	S	935 mCi	1	Yes	Br
Br-80m	4.6 hr	1 hr	57 mCi	1	Yes	Br
Br-83	2.33 hr	1 hr	208 µCi	1	No	Rb
			1.9 mCi	1	Yes	U
Rb-86	29.5 d	1 hr	1.5 mCi	1	Yes[d]	Rb
Sr-85	65 d	1 hr	93 µCi	1	Yes	Sr
Sr-89	53 d	1 hr	39 µCi	1	Yes[d]	U
Y-88	2.0 hr	1 hr	52 mCi	1	No	Y
Zr-89	79.3 hr	1 hr	21 mCi	1	No	Zr
Zr-89m	4.4 m	S	2.1 Ci	1	No	Zr
Zr-97	17 hr	1 hr	3.8 mCi	1	Yes	U
Nb-92	21.6 hr	1 hr	88 mCi	1	No	Nb
Nb-97	72 m	1 hr	52 mCi	3	No	Mo
Nb-99	2.5 m	S	10 mCi	1	No	Mo
Mo-91	75 s	S	42 mCi	1	No	Mo
Mo-91m	15.5 m	S	210 mCi	1	No	Mo
Mo-99	67 hr	1 hr	4.2 mCi	1	Yes	Mo
			880 µCi	1	Yes[d]	U
Ru-103	39.8 d	1 hr	36 µCi	1	Yes”	U
Ru-105	4.5 hr	1 hr	5.5 mCi	1	Yes	U
Rh-102	210 d	1 hr	390 µCi	1	No	Rh
Rh-105	36.5 hr	1 hr	0.21 µCi	1	No	Ag
Ag-105	40 d	1 hr	100 µCi	3	No	Ag
Ag-106	24 m	S	1.6 Ci	1	No	Ag
Ag-108	2.3 m	S	1.6 Ci	1	Yes	Ag
Ag-111	7.6 d	1 hr	21 µCi	1	Yes[d]	U
Ag-112	3.2 hr	1 hr	10 mCi	3	No	Cd
In-113m	104 m	1 hr	286 mCi	1	No	In
In-114	72 s	S	730 mCi	1	Yes[d]	In
In-114m	49 d	1 hr	2.9 mCi	1	Yes[d]	In
In-115m	4.5 hr	1 hr	2.3 mCi	1	No	In
In-117	2.5 hr	1 hr	5.5 mCi	3	No	Sn
Sb-120	16.4 m	S	1.6 Ci	2	No	Sb
Sb-122	13.0 d	1 hr	21 mCi	1	Yes[d]	Sb
I-126	33 m	1 hr	5.7 mCi	1	No	I
Cs-138	85 m	S	94 mCi	1	Yes	U
Ba-139	33 hr	1 hr	31 mCi	1	Yes	U
Ce-143	33 hr	1 hr	1.6 mCi	1	Yes	U
Ta-179	600 d	1 hr	21 µCi	1	No	Ta
Ta-180	8.15 hr	1 hr	310 mCi	1	No	Ta
Au-195	180 d	1 hr	36 µCi	3	No	Au
Au-196	5.55 d	1 hr	16 mCi	2	No	Au
Au-197m	7.4 s	S	21 mCi	2	No	Au
Tl-207	4.8 m	S	2.1 mCi	1	No	Pb
Bi-207	50 yr	1 hr	0.6 µCi	1	No	Bi
Th-231	25.64 hr	1 hr	114 mCi	1	No	Th
U-237	6.75 d	1 hr	16 mCi	1	No	U

a. Either 1 hour (1 hr) or Saturation (S)
b. 1 = ± 50%, 2 = within factor of 3, 3 = within factor of 10
c. 1 g of natural element unless noted
d. Listed in Oak Ridge Isotope catalog, January 1957

ATTACHMENT G
RADIOLOGICAL LABORATORY HAZARD CLASSIFICATION SCHEME

Table G-1 lists the classifications from “United States of America Standards Institute (USASI) Design Guide for Radioisotope Laboratory,” which provided guidelines for radiochemistry laboratories based on the four classifications of relative radiotoxicity: very high, high, moderate, and slight (Bold 1970a). Recommended maximum activities for each laboratory were given. These guidelines were used at GA to assess conditions in the Science Laboratory.

Table G-1. Classification of isotopes according to relative radiotoxicity per unit of activity.

Class 1 **(very high radiotoxicity**	**Sr-90 + Y-90, Pb-210 + Bi-210 (Ra D + E), Po-210, At-211, Ra-226 + 55% progeny, Ac-227, U-233, Pu-239, Am-241, Cm-242, plus other transuranic isotopes**
Class 2 **(high radiotoxicity)**	Ca-45, Fe-59, Sr-89, Y-91, Ru-106, I-131, Ba-140 + La-140, Ce-144 + Pr-144, Sm-151, Eu-154, Tm-170, Th-234 + Pa-234, natural uranium
Class 3 **(moderate radiotoxicity)**	Na-22, Na-24, P-32, S-35, Cl-36, K-42, Sc-46, Sc-48, V-48, Mn-52, Mn-54, Mn-56, Fe-55, Co-58, Co-60, Ni-59, Cu-64, Zn-65, Ga-72, As-74, As-76, Br-82, Rb-86, Z5-95 + Nb-95, Mo-99, Tc-98, Rb-105, Ru-103 + Rh-103, Ag-105, Ag-111, Cd-108 + Ag-109, Sn-113, Te-127, Te-129, I-132, Cs-137 + Ba-137, La-140, Pr-143, Pm-147, Ho-166, Lu-177, Ta-182, W-181, Re-183, Ir-192, Pt-191, Pt-193, Au-196, Au-198, Au-199, Tl-200, Tl-202, Tl-204, Pb-203
Class 4 **(slight radiotoxicity)**	H-3, Be-7, C-14, F-18, Cr-51, Ge-71, Tl-201

Radiotoxicity here refers to the potential of a radionuclide to cause damage to living tissue by absorption of energy from the disintegration of the radioactive material introduced into the body (Table G-2).

Table G-2. Radioisotope laboratory classification.

Relative radiotoxicity of isotopes	**Minimum significant quantity (μc)**	**Type of laboratory or working place required**		
		C. Good chemical laboratory	**B. Radioisotope laboratory**	**A. High-level laboratory**
Very high	0.1	10 μCi or less	10 μCi to 10 mCi	10 mCi or more
High	1.0	100 μCi or less	100 μCi to 100 mCi	100 mCi or more
Moderate	10	1 mCi or less	1 mCi to 1 Ci	1 Ci or more
Slight	100	10 mCi or less	10 mCi to 10 Ci	10 Ci or more

ATTACHMENT H
INSTRUMENTATION ON HAND AND DESIRED AS OF SEPTEMBER 1960

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Table H-1. Instrumentation on hand and desired as of September 1960.[a]

Instrument	Function	On hand	Total needed	HP surveyors	Field crews	Emergency truck	Experimental building	Fusion	Science buildings	LINAC	HTGR	MGCR	TRIGA	FLAIR	Hot cell	Be facility	Waste yard	Trans-portation	Service loaner	Janitors
GM[b] counter with headphones	Beta-gamma source locating	50	37	9	1	1	1	0	26	2	*	*	1	*	*	1	2	2	1	2
Alpha proportional counters with phones	Alpha contamination detecting	26	3	7	0	0	2	0	8	1	*	1	0	0	*	1	2	2	1	1
Alpha-detecting ion chamber (Samson)	Alpha contamination detecting	15	14	5	0	0	2	0	5	0	0	0	0	0	0	0	0	0	1	2
Beta-gamma ion chamber (Samson)	Beta dose rate measuring	24	10	9	0	1	*	0	7	1	*	*	*	*	2	*	1	2	1	0
Gamma ion chamber (Radetector)	Gamma dose rate measuring	50	38	9	4	1	*	0	16	1	2	1	2	1	5	1	3	2	1	1
Gamma ion chamber (Qtpi)	Gamma dose rate measuring	4	4	0	0	0	0	0	0	2	1	1	0	0	0	0	0	0	0	0
Fast neutron survey meter	Fast neutron dose rate measuring	4	2	2	2	0	0	0	0	*	*	*	*	*	0	*	0	0	0	0
Slow neutron survey meter	Thermal neutron flux locating	4	0	2	2	0	0	0	0	*	*	*	*	*	0	*	0	0	0	0
Anemometer	Measuring air velocity	4	2	4	0	0	*	0	*	*	*	*	*	*	*	*	*	0	0	0
Air samplers (Staplex)	Collecting airborne particulates	9	7	9	0	1	*	0	*	*	*	*	*	*	*	*	*	0	0	0
Air samplers (Giraffe)	Collecting airborne particulates	14	10	1	0	0	3	0	4	*	1	1	0	0	1	2	1	0	0	0
Continuous air monitors (Mobile)	Detecting airborne beta particulates	9	4	0	0	0	0	0	2	1	1	1	1	1	2	0	0	0	0	0
Continuous air monitors (Station)	Recording beta activity of atmosphere	3	1	2	0	0	0	0	*	*	*	*	*	*	1	*	0	0	0	0

ATTACHMENT H
INSTRUMENTATION ON HAND AND DESIRED AS OF SEPTEMBER 1960

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Instrument	Function	On hand	Total needed	HP surveyors	Field crews	Emergency truck	Experimental building	Fusion	Science buildings	LINAC	HTGR	MGCR	TRIGA	FLAIR	Hot cell	Be facility	Waste yard	Trans-portation	Service loaner	Janitors
GM frisking rate meters	Hand, clothing and shoe check station	8	5	0	0	0	0	0	3	0	0	0	1	1	3	0	0	0	0	0
GM foot counters	Foot checking station	7	4	0	0	0	0	0	2	0	1	0	0	0	3	0	0	0	0	0
Alpha-beta probe friskers	Hand, clothing and shoe check station	12	12	0	0	0	2	0	4	1	1	2	0	0	1	1	1	0	1	0
Alpha proportional hand and foot counters	Alpha hand and foot portal check	4	0	0	0	0	2	0	1	0	0	1	0	0	0	0	0	0	0	0
Criticality alarms	Warn of radiation pulse	22	12	0	0	0	5	3	2	1	1	2	1	1	1	1	2	1	1	0
Criticality alarm cameras	Locate nuclear accident victim	7	0	0	0	0	3	1	0	0	1	2	0	0	0	0	0	0	0	0
Accident dosimeters	Evaluate nuclear excursions	60	0	0	0	0	12	5	12	3	3	5	1	1	5	2	6	4	0	0
Thin-window GM counters	Beta counting in laboratory	2	1																	
Automatic sample changers	To feed beta counters	2	0																	
Low background counter	Count alpha and beta activity of sewage, etc.	2	1																	
Alpha scintillation counters	Alpha counting in laboratory	5	5																	
Analytical balance	Weigh samples in laboratory	1	1																	
Minometers	Read PICs	3	3																	
PICs	Basic personnel monitoring	1,600	1,400																	

ATTACHMENT H
INSTRUMENTATION ON HAND AND DESIRED AS OF SEPTEMBER 1960
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Instrument	Function	On hand	Total needed	HP surveyors	Field crews	Emergency truck	Experimental building	Fusion	Science buildings	LINAC	HTGR	MGCR	TRIGA	FLAIR	Hot cell	Be facility	Waste yard	Trans-portation	Service loaner	Janitors
Radiation dosimeters	Permit workers to measure own dose	80	57																	
Calibration facility	Calibrate radiation and anemometer instrument	1	0																	
Portable generator	Emergency and environmental air sampling power supply	2	0																	
Recording weather vane and anemometer	Record wind history	1	1																	
Remote recording thermo-meters	Record inversion history	6	0																	
Oscilloscope	Test and maintain laboratory equipment	1	0																	

a. Source: Ray (1960g).
b. GM = Geiger-Müller.

ATTACHMENT I
SUMMARY OF SOME GA ACCIDENTS AND INCIDENTS
Page 1 of 2

Table I-1. Summary of some GA accidents and incidents.

Date	Location	Description	Source
January 1, 1960	Hot cell	β exposure – 2 individuals, 136-150 mR gamma	Cockle 1960a
February 11, 1960	Hot Chem. Lab No. 164	Radioactive spill	Cockle 1960b
April 1, 1960		Radioactive waste spill – 7 individuals, 35,000 cpm	Ray 1960h
April 14, 1960	TRIGA reactor	Release of airborne radioactivity, Ru-88 – 5 individuals maximum	Bold 1960b
September 2, 1960	Science Building B	K-85 release – 5 Ci from September 2 to September 3	Ray 1960i
October 31, 1960		Fire in hot furnace room, Clothing contamination – 150-1,200 cpm alpha	Bold 1960a
August 30, 1961	Counting Room L-236	TRIGA reactor incident – 900 rep	Ray 1961
November 2,01961	Hot machine shop	Fire – 10 individuals, 8% uranium alloy, 20% enriched	Cockle 1961
March 6, 1962	Metallurgical Development Building	3 incidents, including uranium fires and flask of fission products explosion, from March 6 to March 12, 1962. No radiation exposure was associated with the explosion.	Bonner 1963
March 27, 1962	Mark F reactor	Release, Iodine-131,-133,-135 – 0.016 mCi released	Stern 1962
September 10, 1962	Hot suite	Fire – 8% uranium alloy, 20% enrichment	Cockle 1962
October 5, 1962	Hot cell	Personnel exposure – 1 individual, <2 rem	Bedsaul 1962
May 1, 1963	Experimental Building	Fire in the hot furnace room, uranium sludge	Keith 1963a
May 22, 1963	L-540	Fire – 150 g normal uranium	Bold 1963b
May 23, 1963	LINAC Facility	Fire – 40 mR	Bold 1963c
June 4, 1963	L-441	Spill – 9 individuals, 50,000 cpm alpha	Keith 1963b
September 19, 1963	L-403	Radioactive spill – 1 individual, MFPs, 7 mR/hr	Lepper 1963
December 4, 1963	Mark-F Reactor Room	Radiation release, xenon and krypton	Bold 1963d
January 29, 1964	LINAC	Accidental irradiation – 2 individuals, 25 mR	Kesting & Cockle 1964
March 11, 1964	Service Gallery	Urine bioassays – 2 individuals, max 2,200 pCi/24 hr beta in urine, clothing contamination	Hughes 1964b
April 23, 1964	L-307	Bioassay for Cs-134 – 1 individual, 20,000 pCi	Andrews 1964b
May 17, 1964	Experimental Building	Filter fire	Bold 1964c
December 30, 1964	Experimental Building	Fire in hot machine shop – 1.5 kg of 8.5% U alloy	Bold 1965f
January 1, 1965	Hot cell	From January 1 to January 15, release of 100 mCi of I-131 from cutting of a P-9 capsule	Andrews 1965
July 14, 1965		X-ray intensities from X-ray diffraction machine	Lepper 1965
May 17, 1966	Station No. 5	Contamination of hot cell operating gallery, Ta-182, Co-58, Cr-51	Bold 1966b
May 17, 1966	Console Room	Operational error at the 17-MEV accelerator – 9 individuals, 0-90 mrem	Cockle 1966
June 27, 1966	TRIGA	Overexposure incident – cadmium, 1.8 R gamma, 18.7 rad, beta	Gurren 1966c
July 16, 1966	L-427	Fire – UZr pellet	Bold 1966c
December 6, 1966	B area of LINAC	Report stated, “Near fatal radiation accident.” – 1 individual involved. Recorded dose was 3 mrem. Individuals working in LINAC area did not necessarily require radiation safety training or escort at this time.	Bold 1966d
December 28, 1966	Customer Particle Line Area	Fire in controlled area – 3 individuals, 150 pCi alpha	Hughes 1967
March 6, 1967	LINAC	Contamination incident – W-185	Friesenhahn 1967
April 1, 1967	LINAC	Incident at the 16-m neutron detection station – 20 mR	Bold 1967b
April 13, 1967	Hot suite machine room	Fire - UZr fire, 20% enriched	Smith 1967
April 27, 1967	Column contractor station	Column contractor accident – 26 g uranium	Quimby 1967
July 5, 1967	L-441	Uranium fire – 1 individual, 127 grams U-235	Moore 1967a
August 9, 1967		Exposed shoe and home, Ce-144, Cs-134, Cs-137, Zr-95	Maschka 1967
November 1, 1967	Experimental Critical Facility	Skin contamination – 6,000 cpm alpha	Bedsaul 1967
November 22, 1967	Fuel Fabrication Building	Explosion – 60 g uranium	Moore 1967b
April 25, 1968	L-236	Hand exposure, Sr-89, Sr-90 – 2.67 rem	Jordan 1968
May 14, 1968	LINAC	Overexposure of extremities, Finger ring – 43 rem total	Cockle 1968b

ATTACHMENT I
SUMMARY OF SOME GA ACCIDENTS AND INCIDENTS
Page 2 of 2

Date	Location	Description	Source
August 5, 1968	LINAC	Radiation incident – radiation dose while machine is off	Trimble 1968
September 11, 1968	Experimental Building	Uranium fire – 325 g U-235, 8.5% U, 70% enriched	Thompson 1968
January 17, 1969	L-218	Ta-182 contamination – 4 individuals	Kesting 1969a
January 21, 1969	Clean High Bay in Fuel Fabrication Facility	Thorium container explosion – nose wipes showed no activity, average air concentration for week of Jan. 20, 1969, 2.39×10^{-11} µCi/ml (reported as alpha beta)	Hughes 1969a
February 28, 1969	L-407	Contamination incident – Ce-144, Pr-144, 0.1 µCi	Kesting 1969b
April 2, 1969	C-D Room adjoining the Fast Spectrum Cell in LINAC area	High personal radiation exposure – 1 individual, 1.8 rem	Young and Jordan 1969a
June 22, 1969	South Mezzanine	Spill of U-235 solution – 80 g U-235 unrecovered	Hughes 1969b
June 29, 1969	Fast Spectrum Cell	Sub-Critical Time of Flight Spectrum Facility incident – 6 individuals, I-133, 8 to 140 mrem	Young and Jordan 1969b
August 13, 1969	TRIGA	Release of Kr-85 – 1.5 mCi	Bedsaul 1969
August 29, 1969	TRIGA	Fission product release in reactor room – 40 µCi released	Trimble 1969

ATTACHMENT J
HTGR AND RERTR IFM INFORMATION

Table J-1. Radionuclide content of HTGR and RERTR IFM, September 30, 2003.

Radionuclide	HTGR IFM Activity (Ci)	RERTR IFM Activity (Ci)	Total HTGR+RERTR Activity (Ci)	Activity fraction
H-3	1.96E-01	1.61E+00	1.81E+00	6.78E-04
Mn-54	0.00E+00	2.14E-05	2.14E-05	8.03E-09
Fe-55	0.00E+00	4.27E+00	4.27E+00	1.60E-03
Co-60	0.00E+00	8.84E-01	8.84E-01	3.32E-04
Ni-59	0.00E+00	3.30E-01	3.30E-01	1.24E-04
Ni-63	0.00E+00	3.75E+01	3.75E+01	1.41E-02
Kr-85	5.57E+00	3.55E+01	4.11E+01	1.54E-02
Sr-90	1.27E+02	6.32E+02	7.59E+02	2.85E-01
Y-90	1.27E+02	6.32E+02	7.59E+02	2.85E-01
Tc-99	0.00E+00	1.40E-01	1.40E-01	5.26E-05
Ru-106	0.00E+00	3.43E-03	3.43E-03	1.29E-06
Sb-125	1.64E-02	5.97E-01	6.13E-01	2.30E-04
Cs-134	2.70E-02	1.71E+00	1.74E+00	6.52E-04
Cs-137	1.31E+02	6.91E+02	8.22E+02	3.09E-01
Pm-147	3.34E-01	1.22E+01	1.25E+01	4.70E-03
Sm-151	1.20E+00	3.16E+00	4.36E+00	1.64E-03
Eu-154	8.11E-01	1.28E+01	1.36E+01	5.11E-03
Eu-155	4.78E-02	2.15E+00	2.20E+00	8.25E-04
Th-232	2.10E-04	0.00E+00	2.10E-04	7.88E-08
U-233	2.92E-01	1.71E-07	2.92E-01	1.10E-04
U-234	3.13E-02	3.91E-04	3.17E-02	1.19E-05
U-235	2.27E-04	7.39E-04	9.66E-04	3.63E-07
U-236	1.04E-03	5.61E-03	6.65E-03	2.50E-06
U-238	3.84E-06	8.58E-04	8.62E-04	3.24E-07
Np-237	0.00E+00	2.48E-03	2.48E-03	9.31E-07
Pu-238	2.74E+00	0.00E+00	2.74E+00	1.03E-03
Pu-239	1.70E-02	1.30E+00	1.32E+00	4.94E-04
Pu-240	1.91E-02	1.35E+00	1.37E+00	5.14E-04
Pu-241	2.16E+00	1.95E+02	1.97E+02	7.40E-02
Pu-242	1.08E-04	3.35E-03	3.46E-03	1.30E-06
Total	3.98E+02	2.27E+03	2.66E+03	1.00E+00

Source: GA (2003).

ATTACHMENT K
HOT CELL FACILITY REMEDIATION INFORMATION

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Table K-1. Sequence of D&D approach (GA 1998)

Room	Tasks
Controlled Machine Shop and Weld Area (108, 108A)	Decontamination of walls, floor, and ceiling. Removal of added floor. Removal of filler material between added floor and original floor. Removal of original floor.
MC (112)	Removal of equipment inside cell. Decontamination of walls, floor, and ceiling.
Ladies' and Men's Restrooms, Men's Change Room (116)	Removal of all equipment from rooms. Decontamination of walls, floor, and ceiling. Removal of internal nonsupporting walls. Removal of floor.
Met Cell (112)	Complete removal of equipment inside cell. Decontamination of walls, floor, and ceiling.
Dark Room (105A) Counting Room (106)	Removal of all equipment from room. Decontamination of walls, floor, and ceiling. Removal of internal nonsupporting walls. Removal of floor, showers, and drop ceiling. Removal of hot drains under floor. Capping of hot drain line going to Room 121.
MC (112)	Begin removal of walls.
Counting Room (106) Warm Metallography Room (107)	Complete Room 106. Decontamination of Room 107 walls, floor, and ceiling. Removal of floor. Removal of hot drain line from under floor. Capping of hot drain as it leaves Room 121 and enters Room 122.
Hot Cells (113, 115)	Complete removal of south and west walls of Met Cell. Remove Koll-Morgan [periscope] from HLC and begin decontamination.
Warm Metallography Room (107) Manipulator Repair Room (122)	Complete work in Room 107, remove abandoned underground duct. Removal of hot sink from Room 122. Decontamination of walls, floor, and ceiling. Removal of steel liner from walls. Removal of floor sink. Removal of floor. Removal of nonsupporting wall. Removal of hot drain lines from under floor. Capping of drain line leading to Room 109.
Hot Cells (113, 115)	Complete decontamination of HLC, begin decontamination of LLC.
Tritium Extraction Laboratory (Room 109) Corridor, Ladies Change Room	Removal of equipment from rooms. Decontamination of walls, floor, and ceiling. Removal of floor. Removal of hot drain lines from under the floor. Capping of drain line leading to Room 119. Removal of buried ventilation under Corridor and Dark Room.
LLC (Room 113)	Complete decontamination of LLC. Remove Manipulators, PaR, and remaining Hot Cell windows.
Tritium Sample Prep. (120) Tritium Effluent (119), and 117A and 116A	Decontamination of walls, floor, and ceiling. Removal of nonsupporting walls. Removal of slab from Rooms 120 and 119 and abandoned ventilation duct under Room 120.
Hot Cells	Removal of steel liner and decontamination of concrete walls.
X-ray laboratory (116) Tool Room (117) Pump Room (114) Decontamination Room (118) Service Gallery (111)	116 & 117 – Removal of fume hood. Decontamination of walls, floor, and ceiling. Removal of ceiling. 114 – Removal of equipment. Decontamination of walls, floor, and ceiling. 118 – Decontamination of walls, floor, and ceiling. Removal of nonsupporting walls. 111 – Begin decontamination of walls, floor, and ceiling.
Yard	Shed- Decontamination of walls, floor, and ceiling. Decontamination of exterior of shed. Dismantle shed. Removal of concrete slab. Service yard – Old stack monitor room; decontamination of walls, floor, and ceiling. Decontamination and removal of concrete slabs.
Service Gallery (111) Boiler Room (121) Air Ducts Building Roof	111 – Complete decontamination. 121 – Decontamination floor, walls, and ceiling. Remove floor. Remove hot drain lines. Air Ducts – Remove air ducts from Facility. Roof – Begin removal of roof.
Yard	Remove hot drain line. Begin decontamination of liquid waste storage area.
Boiler Room (121) Building Roof Building Walls	121 – Complete demolition of room Roof – Finish roof removal Walls – Begin removal of building walls (except for old offices).
Hot Cells	Begin removal of cell roof.
Yard Outside HEPA System	Complete removal of hot drain lines and clearing of waste storage area. Begin removal of HEPA system.
Hot Cells	Complete removal of Hot Cell roof. Remove Hot Cell doors. Begin removal of cell walls.
Walls	Complete removal of building walls.
HEPA System	Finish removing HEPA System and surrounding concrete slabs. Decontamination and removal of Hot Cell HEPA pit.

ATTACHMENT K
HOT CELL FACILITY REMEDIATION INFORMATION

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Room	Tasks
Building 23	Begin removal of remainder of slab.
Hot Cells	Finish removing Hot Cell walls. Remove Hot Cell floor and hot drain beneath Hot Cells.
Hot Drain Service yard	Remove exterior hot drains. Begin soil remediation.
Building 23	Complete removal of building slab.
Hot Cells	Begin removal of Hot Cell foundation, pits, and HEPA ducts.
Service Yard	Continue with soil remediation.
Building	Remove pit from under Room 111.
Hot Cells	Continue removal of HEPA ducts. Begin removal of Hot Cell walls.
100, 102, 103, 104	Demolish remaining rooms.
Hot Cells	Complete removal of HEPA ducts. Complete removal of Hot Cell wells.
Yard	Begin soil remediation.
Yard	Complete soil remediation.
Site	Postdecontamination surveys, additional remediation.

Table K-2. Concentrations of radionuclides found in soil, asphalt, and concrete before remediation of the HCF (GA 1998)

Sample location	Depth interval [cm (in.)]	Cs-137 (pCi/g)	Cs-134 (pCi/g)	Co-60 (pCi/g)
A-4[a]	0.0–5.0 (0.0–2.0)	1.8	<LLD[b]	<LLD
	5.0–30.0 (2.0–12.0)	<LLD	<LLD	<LLD
	60.0–87.5 (24.0–35.0)	<LLD	<LLD	<LLD
A-5[c]	0.0–7.5 (0.0–3.0)	<LLD	<LLD	<LLD
	7.5–27.5 (3.0–11.0)	4.6	<LLD	0.56
A-6[a]	0.0–7.5 (0.0–3.0)	1.52	<LLD	<LLD
	7.5–32.5 (3.0–13.0)	0.62	<LLD	<LLD
	32.5–40.0 (13.0–16.0)	<LLD	<LLD	<LLD
A-7[a]	0.0–6.2 (0.0–2.5)	0.34	<LLD	<LLD
	6.2–23.8 (2.5–9.5)	0.37	<LLD	0.25
B-1[a]	0.0–15.0 (0.0–6.0)	<LLD	<LLD	<LLD
	15.0–42.5 (6.0–17.0)	<LLD	<LLD	0.29
B-2[c]	1.2–23.8 (0.5–9.5)	0.68	<LLD	0.53
	23.8–35.0 (9.5–14.0)	0.22	<LLD	<LLD
B-3	0.0–7.5 (0.0–3.0)	<LLD	<LLD	<LLD
	7.5–23.8 (3.0–9.5)	<LLD	<LLD	<LLD
B-4[c]	0.0–15.0 (0.0–6.0)	0.22	<LLD	1.26
	15.0–30.0 (6.0–12.0)	<LLD	<LLD	<LLD
B-6	0.0–5.0 (0.0–2.0)	<LLD	<LLD	<LLD
	5.0–22.5 (2.0–9.0)	<LLD	<LLD	<LLD
B-7[c]	0.0–3.8 (0.0–1.5)	2.97	<LLD	1.44
	5.0–25.0 (2.0–10.0)	0.77	<LLD	0.57
B-8[d]	0.0–12.5 (0.0–5.0)	38.3	0.24	21
	12.5–30.0 (5.0–12.0)	0.43	<LLD	<LLD
B-9	0.0–12.5 (0.0–5.0)	<LLD	<LLD	<LLD
	12.5–30.0 (5.0–12.0)	<LLD	<LLD	<LLD
C-1[c]	2.5–13.8 (1.0–5.5)	0.81	<LLD	1.75
	13.8–30.0 (5.5–12.0)	0.07	<LLD	<LLD
C-5[c]	0.0–3.8 (0.0–1.5)	2.92	<LLD	0.92
	3.8–17.5 (1.5–7.0)	3.47	<LLD	4.03
C-6	0.0–13.8 (0.0–5.5)	0.62	<LLD	<LLD
	13.8–28.2 (5.5–11.5)	0.38	<LLD	<LLD
C-7	N/A–Not sampled			
C-8[a]	0.0–15.0 (0.0–6.0)	0.46	<LLD	<LLD
	21.2–42.5 (8.5–17.0)	<LLD	<LLD	<LLD

ATTACHMENT K
HOT CELL FACILITY REMEDIATION INFORMATION
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Sample location	Depth interval [cm (in)]	Cs-137 (pCi/g)	Cs-134 (pCi/g)	Co-60 (pCi/g)
C-9	0.0–12.5 (0.0–5.0)	0.28	<LLD	<LLD
	15.0–40.0 (6.0–16.0)	<LLD	<LLD	<LLD
C-10	0.0–20.0 (0.0–8.0)	0.17	<LLD	<LLD
	20.0–40.0 (8.0–16.0)	<LLD	<LLD	<LLD
D-1[a]	1.2–18.8 (0.5–7.5)	0.67	<LLD	<LLD
	18.8–45.0 (7.5–18.0)	0.26	<LLD	<LLD
D-5	N/A–Not sampled			
D-6[c]	0.0–7.5 (0.0–3.0)	8.4	<LLD	<LLD
	7.5–20.0 (3.0–8.0)	2.22	<LLD	<LLD
D-7[d]	0.0–13.8 (0.0–5.5)	4.53	<LLD	0.44
	15.0–35.0 (6.0–14.0)	0.12	<LLD	<LLD
D-8[a]	0.0–5.0 (0.0–2.0)	2.71	<LLD	0.52
	5.0–21.2 (2.0–8.5)	0.11	<LLD	0.27
D-9	0.0–5.0 (0.0–2.0)	7.11	0.65	<LLD
	5.0–17.5 (2.0–7.0)	0.23	<LLD	<LLD
D-10	0.0–5.0 (0.0–2.0)	<LLD	<LLD	<LLD
	10.0–30.0 (4.0–12.0)	<LLD	<LLD	<LLD
E-1[c]	0.0–16.2 (0.0–6.5)	<LLD	<LLD	<LLD
	16.2–32.5 (6.5–13.0)	1.89	<LLD	1.47
E-2[a]	0.0–13.8 (0.0–5.5)	<LLD	<LLD	<LLD
	13.8–30.0 (5.5–12.0)	0.79	<LLD	0.22
	52.5–65.0 (21.0–26.0)	<LLD	<LLD	<LLD
	65.0–72.5 (26.0–29.0)	<LLD	<LLD	<LLD
	75.0–82.5 (30.0–33.0)	<LLD	<LLD	<LLD
	90.0–100.0 (36.0–40.0)	<LLD	<LLD	<LLD
E-3	0.0–13.8 (0.0–5.5)	<LLD	<LLD	<LLD
	13.8–25.0 (5.5–10.0)	0.2	<LLD	<LLD
E-4	0.0–11.2 (0.0–4.5)	<LLD	<LLD	0.76
	13.8–35.0 (5.5–14.0)	<LLD	<LLD	<LLD
E-5	0.0–7.5 (0.0–3.0)	2.28	0.28	<LLD
	11.2–37.5 (4.5–15.0)	<LLD	<LLD	<LLD
E-6	0.0–15.0 (0.0–6.0)	0.54	<LLD	<LLD
	0.0–30.0 (0.0–12.0)	<LLD	<LLD	<LLD
	30.0–55.0 (12.0–22.0)	<LLD	<LLD	<LLD
	55.0–87.5 (22.0–35.0)	<LLD	<LLD	<LLD
E-7	0.0–7.5 (0.0–3.0)	1.58	<LLD	<LLD
	7.5–27.5 (3.0–11.0)	0.21	<LLD	<LLD
E-8	N/A–Not sampled			
E-9	0.0–5.0 (0.0–2.0)	0.19	<LLD	<LLD
	22.5–37.5 (9.0–15.0)	<LLD	<LLD	<LLD
E-10[a]	0.0–5.0 (0.0–2.0)	0.64	<LLD	<LLD
	10.0–30.0 (4.0–12.0)	<LLD	<LLD	<LLD
F-2	0.0–11.2 (0.0–4.5)	0.65	<LLD	<LLD
	11.2–30.0 (4.5–12.0)	<LLD	<LLD	<LLD
F-3	0.0–8.8 (0.0–3.5)	<LLD	<LLD	<LLD
	8.8–37.5 (3.5–15.0)	<LLD	<LLD	<LLD
F-4[a]	0.0–5.0 (0.0–2.0)	0.6	<LLD	0.19
	5.0–15.0 (2.0–6.0)	<LLD	<LLD	<LLD
F-5[c]	0.0–5.0 (0.0–2.0)	2.1	<LLD	1.5
	15.0–22.5 (6.0–9.0)	<LLD	<LLD	<LLD
F-6[c]	0.0–5.0 (0.0–2.0)	2.9	<LLD	2.4
	10.0–22.5 (4.0–9.0)	<LLD	<LLD	<LLD
F-7[c]	0.0–5.0 (0.0–2.0)	1.1	<LLD	0.36
	5.0–17.5 (2.0–7.0)	0.6	<LLD	0.12

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Sample location	Depth interval [cm (in)]	Cs-137 (pCi/g)	Cs-134 (pCi/g)	Co-60 (pCi/g)
F-8[a]	0.0–5.0 (0.0–2.0)	0.7	<LLD	0.25
	7.5–27.5 (3.0–11.0)	0.08	<LLD	<LLD
F-9[a]	0.0–5.0 (0.0–2.0)	0.45	<LLD	<LLD
	22.5–30.0 (9.0–12.0)	<LLD	<LLD	<LLD
F-10[c]	0.0–15.0 (0.0–6.0)	2.1	<LLD	0.58
	15.0–30.0 (6.0–12.0)	0.07	<LLD	<LLD
	30.0–45.0 (12.0–18.0)	0.47	<LLD	<LLD
	75.0–90.0 (30.0–36.0)	<LLD	<LLD	<LLD
Judgment locations				
General trestle area R-11A[d]	0.0–10.0 (0.0–4.0)	217	158	9.38
	10.0–22.5 (4.0–9.0)	28.4	6.71	4.62
	45.0–60.0 (18.0–24.0)	0.69	0.26	<LLD
	65.0–90.0 (26.0–36.0)	<LLD	<LLD	<LLD
	90.0–105.0 (36.0–42.0)	<LLD	<LLD	<LLD
General trestle area R-11B[d]	0.0–6.2 (0.0–2.5)	41.2	<LLD	5.38
	6.2–18.8 (2.5–7.5)	25.8	0.66	9.9
	41.2–50.0 (16.5–20.0)	0.84	<LLD	0.19
	67.5–80.0 (27.0–32.0)	0.79	<LLD	0.2
General trestle area R11C[c]	0.0–5.0 (0.0–2.0)	163	6.56	23.2
	6.2–22.5 (2.5–9.0)	5.88	0.39	1.18
	60.0–67.5 (24.0–27.0)	0.23	<LLD	<LLD
	75.0–82.5 (30.0–33.0)	0.18	<LLD	<LLD
Liquid waste tanks Area R13[d]	0.0–17.5 (0.0–7.0)	84	2.49	18.4
	35.0–50.0 (14.0–20.0)	7.81	0.88	2.58
	55.0–97.5 (22.0–39.0)	0.58	<LLD	0.16
Liquid waste tanks Area R16[d]	0.0–20.0 (0.0–8.0)	126.8	<LLD	107.3
	45.0–57.5 (18.0–23.0)	0.78	<LLD	1.16
	65.0–100.0 (26.0–40.0)	0.7	<LLD	0.92
Liquid waste tanks Area R35[d]	0.0–17.5 (0.0–7.0)	110.4	1.32	10.9
	35.0–60.0 (14.0–24.0)	2.36	<LLD	0.6
	62.5–100.0 (25.0–40.0)	0.31	<LLD	<LLD
Drainage path from liquid waste tanks Area S06A[d]	0.0–7.5 (0.0–3.0)	17.1	0.15	4.3
	7.5–22.5 (3.0–9.0)	107.2	<LLD	14
	35.0–60.0 (14.0–24.0)	4.07	<LLD	0.92
	82.5–100.0 (33.0–40.0)	1.47	<LLD	0.45
Drainage path from liquid waste tanks Area S06B[d]	0.0–18.8 (0.0–7.5)	73.5	<LLD	13.3
	30.0–50.0 (12.0–20.0)	5.6	<LLD	1.53
	50.0–90.0 (20.0–36.0)	1.59	<LLD	0.58
Drainage path from liquid waste tanks Area S06C[d]	0.0–25.0 (0.0–10.0)	34.5	6.29	4.77
	25.0–50.0 (10.0–20.0)	5.59	0.69	1.6
	57.5–97.5 (23.0–39.0)	0.6	<LLD	<LLD

a. The location soil was positive for licensed material in excess of background values.

b. <LLD means that measured concentration was less than the lower limit of detection for the radionuclide.

c. The location soil was positive for licensed material and exceeded 10% of proposed release criteria.

d. The location soil was positive for licensed material and exceeded proposed release criteria.