DOE/CD-ETEC-028

CERTIFICATION DOCKET

FOR THE RELEASE OF BUILDING 028 AT THE ENERGY TECHNOLOGY ENGINEERING CENTER

APRIL 1997

U.S. DEPARTMENT OF ENERGY
OAKLAND OPERATIONS OFFICE
ENVIRONMENTAL RESTORATION

LR-66365

CERTIFICATION DOCKET

FOR THE RELEASE OF BUILDING 028 AT THE ENERGY TECHNOLOGY ENGINEERING CENTER

APRIL 1997

UNITED STATES DEPARTMENT OF ENERGY
HEADQUARTERS

OFFICE OF ENVIRONMENTAL RESTORATION

NORTHWESTERN AREA PROGRAMS

19901 GERMANTOWN ROAD

GERMANTOWN, MARYLAND 20585

Foreword

The purpose of this docket is to document the successful decontamination and decommissioning of Building 028 at the Energy Technology Engineering Center (ETEC) at the Santa Susana Field Laboratory, Area IV for unrestricted use. Material in this docket consists of documents supporting the DOE certification that conditions at ETEC Building 028 are in compliance with applicable DOE and proposed Environmental Protection Agency and Nuclear Regulatory Commission standards and criteria established to protect human health, safety, and the environment. A notice of certification of the radiological condition of the property was published in the Federal Register on April 4, 1997. A copy of the notice, official correspondence, release criteria, project report, radiological surveys, and an independent verification report are compiled in this docket.

CONTENTS

EXHIBIT I

DOCUMENTS SUPPORTING THE CERTIFICATION FOR THE UNRESTRICTED USE OF BUILDING 028 AT THE ENERGY TECHNOLOGY ENGINEERING CENTER

DOE F 1325.8
(8-89)

United States Government

Department of Energy

memorandum

DATE: January 23, 1997

REPLY TO
ATTN OF: DOE Oakland Operations Office/ER

SUBJECT: Release of Decontaminated Building 028 without Radiological Restrictions at the Energy Technology Engineering Center.

TO: Donald Williams, EM-44

The Oakland Operations Office (OAK) has implemented environmental restoration projects at the Energy Technology Engineering Center (ETEC) as part of the Environmental Restoration Program (ERP) per Headquarters Northwestern Area Program Office direction. The objective of the program is to identify and cleanup or otherwise control facilities where residual radioactive contamination remains from activities carried out under contract to the Atomic Energy Commission and the Energy Research and Development Administration during the early years of the Nation's atomic energy program.

The Energy Technology Engineering Center performed testing of equipment, materials, and components for nuclear and energy related programs. These nuclear energy research and development programs began in 1946 and ended in 1995. Numerous buildings and land areas became radiologically contaminated as a result of facility operations and site activities. One such area that has been designated for cleanup under the ERP is Building T028.

Building T028 originally housed the Shield Test Reactor which was used to perform tests on space nuclear test shields. This reactor was operated from 1961 to 1964. After modifications it was renamed the Shield Test and Irradiation Reactor and operated through 1972. Following shutdowns of the test program, the reactor was removed and the facility was decontaminated. From 1977 to 1981 experiments were conducted in the building to investigate the behavior of molten uranium oxide, which resulted in recontamination of the building. A decision to terminate operations at Building T028 was made in 1984.

Decontamination of Building T028 was performed in 1988. Surplus normal and depleted uranium oxide was removed. Equipment, electrical components, and ventilation ducting were also removed, and building surfaces were decontaminated. The above grade portion of the building was demolished in 1989 leaving only the concrete floor, below-grade test vault and stairwell intact.

DOE F 1325.8
(8-89)
E.G. (07-90)

United States Government — Department of Energy

memorandum

DATE: MAR 19 1997

REPLY TO ATTN OF: EM-44 (D. Williams, 301-903-8173)

SUBJECT: Recommendation for Certification of Cleanup at Building 028 at the Energy Technology Engineering Center

TO: Acting Deputy Assistant Secretary for Environmental Restoration, EM-40

I am attaching for your signature a Federal Register Notice concerning the cleanup of contamination associated with the former Atomic Energy Commission and Energy Research and Development Administration (AEC/ERDA) activities at Building 028, at the Energy Technology Engineering Center (ETEC) near Chatsworth, California.

The Oakland Operations Office has implemented a decontamination and decommissioning project at ETEC as part of the Environmental Restoration Program. The objective of the program is to identify and clean up or otherwise control sites where residual radioactive contamination remains from activities carried out under contract to AEC/ERDA during the early years of the Nation's atomic energy program. In October 1987, Building 028 was formally designated by the Department of Energy (DOE) for cleanup under Environmental Restoration.

ETEC Building 028 was constructed in 1961 to support testing of space reactor shields using a fission plate driven by neutrons from the thermal column of a 50-kW and a 1-MW reactor designated the Shield Test Reactor and Shield Test and Irradiation Reactor, respectively. Space reactor shield testing terminated in 1972. In 1977, experiments to investigate the behavior of molten uranium oxide, relative to simulated reactor accidents, on the reactor floor and structural materials were conducted until 1981. The building remained inactive until 1988 when decontamination was completed. Final radiological and independent verification surveys completed in 1993 demonstrated, and DOE's Oakland Operations Office has certified, that the decontamination project resulted in compliance with DOE decontamination criteria and standards established to protect members of the general public and occupants of the building. Further, future use of the property without radiological restrictions will result in no exposure above applicable radiological guidelines to the general public and occupants of the building.

This office is preparing the certification docket for the subject property and Building 029. The completed docket will be provided to the Oakland Operations Office for their use in preparation of similar dockets for future property releases. The Federal Register Notice will be part of the docket.

I recommend that you sign the attached _Federal Register_ Notice, as well as the transmittal memorandum to the Federal Liaison Officer (Clara Barley, GC-75). The documents transmitted with the certification statement and the _Federal Register_ Notice will be compiled in final docket form by the Office of Northwestern Area Programs and will be made available for public review in DOE Reading Rooms and local libraries.

Sally A. Robison

Sally A. Robison, Ph.D.
Director
Office of Northwestern Area Programs
Environmental Restoration

Attachment

DOE F 1325.8
(8-89)
EFG (07-90)

United States Government Department of Energy

memorandum

DATE: FEB 21 1997

REPLY TO
ATTN OF: EM-44 (D. Williams, 903-8173)

SUBJECT: Draft Certification Docket for Building 028 at the Energy Technology Engineering Center

TO: Assistant General Counsel for Environment, GC-51

I am requesting your review and concurrence of the attached package concerning the cleanup of contamination associated with the former Atomic Energy Commission and Energy Research and Development Administration (AEC/ERDA) activities at Building 028 at the Energy Technology Engineering Center (ETEC) near Chatsworth, California.

The Office of Northwestern Area Programs has implemented a decontamination and decommissioning project at ETEC as part of the Environmental Restoration Program. The objective of the program is to identify and clean up or otherwise control sites where residual radioactive contamination remains from activities carried out under contract to AEC/ERDA during the early years of the Nation's atomic energy program. In October 1987, Building 028 was formally designated by the Department of Energy (DOE) for cleanup.

ETEC Building 028 was constructed in 1961 to support testing of space reactor shields using a fission plate driven by neutrons from the thermal column of a 50-KW and a 1-MW reactor designated the Shield Test Reactor and Shield Test and Irradiation Reactor, respectively. Space reactor shield testing terminated in 1972. In 1977, experiments to investigate the behavior of molten uranium-oxide, relative to simulated reactor accidents, on the reactor floor and structural materials were conducted until 1981. The building remained inactive until 1988 when decontamination was completed. Post-decontamination surveys completed in 1993 demonstrated, and DOE's Oakland Operations Office has certified, that the decontamination project resulted in compliance with DOE decontamination criteria and standards established to protect members of the general public and occupants of the building. Further, future use of the property will result in no radiological exposure above applicable radiological guidelines to the general public or the building occupants.

A draft Federal Register Notice has been prepared as part of the docket and will also be transmitted to the Office of Federal Register for approval after we have received your concurrence on the docket.

The final Federal Register Notice and Certification Statement will be compiled in final docket form by the Office of Northwestern Area Programs and will be made available for public review in DOE Reading Rooms and local libraries.

Your review and comments are requested by March 10, 1997. Mr. Don Williams of my staff is the point-of-contact and can be reached at 903-8173.

Sally A. Robison

Sally A. Robison, Ph.D.
Director
Office of Northwestern Area Programs
Environmental Restoration

Attachment

STATEMENT OF CERTIFICATION: Energy Technology Engineering Center, Building 028

The U.S. Department of Energy, Oakland Operations Office, Environmental Restoration Division, has reviewed and analyzed the radiological data obtained following decontamination of the Energy Technology Engineering Center Building 028. Based on this analysis of all data collected, the Department of Energy (DOE) certifies that the following property is in compliance with DOE decontamination criteria and standards. This certification of compliance provides assurance that future use of the property will result in no radiological exposure above applicable guidelines established to protect members of the general public or site occupants. Accordingly, the property specified below is released from DOE's Environmental Restoration Program.

Property owned by Rockwell International Corporation:

Building 028, at the Energy Technology Engineering Center, located in a portion of Tract "A" of Rancho Simi, in the County of Ventura, State of California, as per map recorded in Book 3, Page 7 of Miscellaneous Records of Ventura County.

CERTIFICATION:

Roger H. Liddle

Roger Liddle, Director, ERD

1/23/97

Date

DOE F 1325.8
(8-89)
EG. (07-90)

United States Government Department of Energy

memorandum

DATE: March 27, 1997

REPLY TO
ATTN OF: EM-44 (D. Williams, 301-903-8173)

SUBJECT: Federal Register Notice for Certification of Cleanup of Building 028 at the Energy Technology Engineering Center

TO: Clara Barley, GC-75

Attached are the original and three copies of the signed Federal Register Notice certifying the completion of remedial action at Building 028 located at the Energy Technology Engineering Center. This surplus building was decontaminated by the Department's Environmental Restoration Program. The attached Notice has been reviewed by and concurred in by the Office of General Counsel (GC-51), and a copy of that concurrence is also attached for your information and use.

Also attached for your signature is a transmittal letter to forward the disk containing the Federal Register Notice to the Office of the Federal Register.

Please forward the attached Notice to the Federal Register for publication.

JFiore

James J. Fiore
Acting Deputy Assistant Secretary
for Environmental Restoration

3 Attachments

Department of Energy
Washington, DC 20585

Mr. Raymond A. Mosley
Director, Office of the Federal Register
National Archives and Records Administration
Washington, D.C. 20408

Dear Mr. Mosley:

This letter is to certify that the enclosed disk is a true copy of the Certification of the Radiological Condition of Building 028 at the Energy Technology Engineering Center located near Chatsworth, California. The disk should be used by the Government Printing Office in preparing the document for publication in the Federal Register.

Sincerely,

James J. Fiore
Acting Deputy Assistant Secretary
for Environmental Restoration

Clara Barley
DOE Federal Register Liaison
Officer

Enclosure

U.S. Department of Energy
DOCKET NO. ETEC-028

Certification of the Radiological Condition of Building 028 at the Energy Technology Engineering Center near Chatsworth, California

AGENCY: U.S. Department of Energy, Office of Environmental Restoration

ACTION: Notice of Certification

SUMMARY: The Department of Energy (DOE) has completed radiological surveys and taken remedial action to decontaminate Building 028 located at the Energy Technology Engineering Center (ETEC) near Chatsworth, California. This property previously was found to contain radioactive materials from activities carried out for the Atomic Energy Commission and the Energy Research and Development Administration (AEC/ERDA), predecessor agencies to DOE. Although DOE owns the majority of the buildings and equipment, a subsidiary of Rockwell International, Rocketdyne, owned the land. Rocketdyne has recently been sold to Boeing North American Incorporated.

FOR FURTHER INFORMATION CONTACT:

Don Williams, Program Manager
Office of Northwestern Area Programs
Office of Environmental Restoration (EM-44)
U.S. Department of Energy
Washington, D.C. 20585

SUPPLEMENTARY INFORMATION:

DOE has implemented environmental restoration projects at ETEC (Ventura County, Map Book 3, Page 7, Miscellaneous Records) as part of DOE's Environmental Restoration Program. One objective of the program is to identify and clean up or otherwise control facilities where residual radioactive contamination remains from activities carried out under contract to AEC/ERDA during the early years of the Nation's atomic energy program.

ETEC is comprised of a number of facilities and structures located within Administrative Area IV of the Santa Susana Field Laboratory. The work performed for DOE at ETEC consisted primarily of testing of equipment, materials, and components for nuclear and energy related programs. These nuclear energy research and development programs, conducted by Atomics International under contract to AEC/ERDA, began in 1946. Several buildings and land areas became radiologically contaminated as a result of facility operations and site activities. Building 028 is one ETEC area that has been designated for cleanup under the DOE Environmental Restoration Program. Other areas undergoing decontamination will be released as they are completed and are verified to meet established cleanup criteria and standards for release without radiological restrictions as established in DOE Order 5400.5.

Building 028 is located in the north-central section of ETEC. The above-grade concrete slab is approximately 300 m^2 in area. The below-grade vault measures approximately 60 m^2 with 6 m (20 ft.) ceilings. Construction consists of a concrete slab floor with concrete walls and ceilings.

Building 028 was originally constructed to perform tests of space reactor shields using a fission plate driven by neutrons from the thermal column of a 50-kW swimming pool-type reactor. This reactor was designated the Shield Test Reactor and operated from 1961 to 1964, when it was replaced with another reactor design to operate at 1 MW. This latter configuration was named the Shield Test and Irradiation Reactor (STIR) and operated through 1972.

Following shutdown of the test program and removal of the reactor, the facility was decommissioned and made available for alternate use in March 1976.

In 1977, operations were started to investigate the behavior of molten uranium-oxide relative to simulated reactor accidents, in particular, its reaction with floor and structural materials. These experiments resulted in some recontamination of various parts of the building that were used for preparation and melting of the uranium-oxide. Tests continued intermittently into 1981. Some facility modifications were made, and a decision to terminate operations was made later in 1981. The building remained inactive, under periodic surveillance, until decontamination began in 1988.

To allow the release of Building 028 for use without radiological restriction, all detectable radioactive material/contamination was removed from the facility. This decontamination and decommissioning was performed in two phases, starting in 1975 (STIR facility) with the removal of the core tank, the activated concrete structures surrounding the core tank, thermal column, reactor shield, test vault carriage, water cooling systems, water shield door, and the partially dismantled exhaust system.

The second and final stage of decontamination of Building 028 began in 1988 and required slightly less than five months to complete.

Briefly, the decontamination steps involved in the second stage: (1) removal of surplus normal and depleted uranium oxide; (2) decontamination and removal of equipment and electrical components, including the furnace system used for the uranium-oxide experiments; (3) removal of the radiologically contaminated ducting system; (4) building surfaces decontamination, including scabbling of the concrete floor in Room 101A; (5) final miscellaneous cleanup operations; and (6) final radiological survey of the building (above-grade and basement).

Rockwell/Rocketdyne performed a radiological survey in 1991. The Environmental Survey and Site Assessment Program of the Oak Ridge Institute for Science and Education performed independent verification of the decontamination project in 1993. Post-decontamination surveys have demonstrated that Building 028 is in compliance with DOE decontamination criteria and standards for release without radiological restrictions. The State of California Department of Health Services has concurred that the proposed release guidelines provide adequate assurance for release without further radiological restrictions. In the event of property transfer, DOE intends to comply with applicable Federal, State, and local requirements.

The external radiation exposure of the nine people directly associated with the STIR project, particularly the dismantling operations, during the period of September 23, 1975, through January 31, 1976, averaged 193 mrem, with a maximum individual exposure of 420 mrem. The entire operation was performed with a total radiation exposure of 1.7 man-rem.

None of the engineering or radiation and nuclear safety personnel assigned to the Building 028 decommissioning project received any measurable exposure to ionizing radiation.

Final costs for the decontamination of the STIR project were $134,922.

Final costs for the decontamination of Building 028 were $239,970.

The certification docket will be available for review between 9:00 a.m. and 4:00 p.m., Monday through Friday (except Federal holidays), in the U.S. DOE Public Reading Room located in Room 1E-190 of the Forrestal Building, 1000 Independence Avenue, S.W., Washington, D.C. Copies of the certification docket will also be available at the following locations: DOE Public Document Room, U.S. DOE, Oakland Operations Office, the Federal Building, 1301 Clay Street, Oakland, California; California State University, Northridge, Urban Archives Center, Oviatt Library, Room 4, 18111 Nordhoff, Northridge, California; Simi Valley Library, 2629 Tapo Canyon Road, Simi Valley, California; and the Platt Branch, Los Angeles Public Library, 23600 Victory Boulevard, Woodland Hills, California.

DOE has issued the following statement of certification:

STATEMENT OF CERTIFICATION: Energy Technology Engineering Center, Building 028

The U.S. Department of Energy, Oakland Operations Office, Environmental Restoration Division, has reviewed and analyzed the radiological data obtained following decontamination of Building 028 at the Energy Technology Engineering Center. Based on analysis of all data collected and the results of independent verification, DOE certifies that the following property is in compliance with DOE radiological decontamination criteria and standards as established in DOE Order 5400.5. This certification of compliance provides assurance that future use of the property will result in no radiological exposure above applicable guidelines established to protect members of the general public or site occupants. Accordingly, the property specified below is released from DOE's Environmental Restoration Program.

Property owned by Boeing North American Incorporated:

Building 028, at the Energy Technology Engineering Center (situated within Area IV of the Santa Susana Field Laboratory), located in a portion of Tract "A" of Rancho Simi, in the County of Ventura, State of California, as per map recorded in Book 3, Page 7 of Miscellaneous Records of Ventura County.

Issued in Washington, D.C., on March 27, 1997.

J. Fiore

James J. Fiore
Acting Deputy Assistant Secretary
for Environmental Restoration

DOE F 1325.8
(08-93)

United States Government **Department of Energy**

memorandum

DATE: APR 02 1997

REPLY TO ATTN OF: EM-44 (D. Williams, 903-8173)

SUBJECT: Release of Decontaminated Building 028 without Radiological Restrictions at the Energy Technology Engineering Center

TO: R. Liddle, Oakland Operations Office

We have completed our review of all documents related to the remediation, final survey, certification, release limits, and independent verification of Building 028 at the Energy Technology Engineering Center (ETEC). We have determined that decontamination of this property has been completed in compliance with the established criteria and standards as required by the Department of Energy (DOE) guidelines and Orders, is consistent with other appropriate Nuclear Regulatory Commission guidelines, and is protective of public health and the environment. Therefore, approval is granted to release subject property to Boeing North American Incorporated without radiological controls pursuant to DOE Order 5400.5, Chapter IV. This property should be removed from the DOE Real Property Inventory in accordance with DOE Order 4300.

In accordance with DOE Order 5820.2A, Section V, the data package compiled for this project must be retained permanently in the Oakland Operations Office (OAK) files.

We recommend that a letter be forwarded to Boeing North American Incorporated requiring prior DOE-OAK notification of any activity which could potentially recontaminate the subject property until final release of the remaining ETEC properties has been completed. Please provide us with a copy of the letter, as well as the distribution list, for our files.

Sally A. Robison

Sally A. Robison, Ph.D.
Director
Office of Northwestern Area Programs
Environmental Restoration

DATES: Written objections must be filed not later June 3, 1997.

ADDRESSES: U.S. Army Waterways Experiment Station, 3909 Halls Ferry Road, Vicksburg, MS 39180–6199. ATTN: CEWES–OC.

FOR FURTHER INFORMATION CONTACT: Mr. Phil Stewart (601) 634–4113, e-mail stewarp@exl.wes.army.mil

SUPPLEMENTARY INFORMATION: The Concrete Armor Unit was invented by Jeffrey A. Melby and George F. Turk. Rights to the patent applications identified above have been assigned to the United States of America as represented by the Secretary of the Army. The United States of America as represented by the Secretary of the Army intends to grant an exclusive license for all fields of use, in the manufacture, use, and sale in the territories and possessions, including territorial waters of each of the listed countries to SOGELREG-SOGREAH, 8P 172, 38042, Grenoble Cedex 9, France.

Pursuant to 37 CFR 404.7(b)(1)(i), any interested party may file a written objection to this prospective exclusive license agreement.

Gregory D. Showalter,

Army Federal Register Liaison Officer.

[FR Doc. 97–8603 Filed 4–3–97; 8:45 am]

BILLING CODE 3710–92–M

DEPARTMENT OF ENERGY

Office of Arms Control and Nonproliferation; Proposed Subsequent Arrangements

AGENCY: Department of Energy.

ACTION: Subsequent arrangements.

Pursuant to Section 131 of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2160), notice is hereby given of a proposed "subsequent arrangement" under the Agreement for Cooperation between the Government of the United States of America and the Government of the Federative Republic of Brazil concerning Civil Uses of Atomic Energy.

The subsequent arrangement to be carried out under the above-mentioned agreement involves approval of the following retransfer: RTD/BR(EU)–10, for the transfer from the Republic of Germany to Brazil of 54,658 pieces of zircaloy-4 cladding tubes, weighing 42,852 kilograms, to be incorporated into uranium fuel assemblies, with an enrichment level between 1.9% and 3.2% of uranium-235, for ultimate use in the Angra-2 reactor.

In accordance with Section 131 of the Atomic Energy Act of 1954, as amended, it has been determined that these subsequent arrangements will not be inimical to the common defense and security.

This subsequent arrangement will take effect no sooner than fifteen days after the date of publication of this notice.

Issued in Washington, D.C. on March 31, 1997.

Cherie P. Fitzgerald,

Director, International Policy and Analysis Division, Office of Arms Control and Nonproliferation.

[FR Doc. 97–8638 Filed 4–3–97; 8:45 am]

BILLING CODE 6450–01–P

Atomic Energy Agreements

AGENCY: Department of Energy.

ACTION: Subsequent arrangement.

SUMMARY: Pursuant to Section 131 of the Atomic Energy Act of 1954, as amended (42 U.S.C. 2160), notice is hereby given of a proposed "subsequent arrangement" under the Agreement for Cooperation in the Peaceful Uses of Nuclear Energy between the United States of America and the European Atomic Energy Community (EURATOM) and the Agreement for Cooperation between the Government of the United States of America and the Government of Canada concerning Civil Uses of Atomic Energy, as amended.

The subsequent arrangement to be carried out under the above-mentioned agreements involves approval of the following retransfer: RTD/EU(CA)–13, for the transfer of 127.8 kilograms of unirradiated low enriched uranium fuel fabrication scrap, containing 25.241 kilograms of the isotope uranium-235 (19.75% enrichment), from AECL in Chalk River, Canada, to UKAEA in Dounreay, United Kingdom, for the purpose of recovering the uranium for return to Canada in the form of uranium metal pieces.

In accordance with Section 131 of the Atomic Energy Act of 1954, as amended, it has been determined that this subsequent arrangement will not be inimical to the common defense and security.

This subsequent arrangement will take effect no sooner that fifteen days after the date of publication of this notice.

Dated: March 31, 1997.

For the Department of Energy.

Cherie Fitzgerald,

Director, International Policy and Analysis Division, Office of Arms Control and Nonproliferation.

[FR Doc. 97–8639 Filed 4–3–97; 8:45 am]

BILLING CODE 6450–01–P

[Docket No. ETEC–028]

Certification of the Radiological Condition of Building 028 at the Energy Technology Engineering Center Near Chatsworth, California

AGENCY: U.S. Department of Energy, Office of Environmental Restoration.

ACTION: Notice of certification.

SUMMARY: The Department of Energy (DOE) has completed radiological surveys and taken remedial action to decontaminate Building 028 located at the Energy Technology Engineering Center (ETEC) near Chatsworth, California. This property previously was found to contain radioactive materials from activities carried out for the Atomic Energy Commission and the Energy Research and Development Administration (AEC/ERDA), predecessor agencies to DOE. Although DOE owns the majority of the buildings and equipment, a subsidiary of Rockwell International, Rocketdyne, owned the land. Rocketdyne has recently been sold to Boeing North American Incorporated.

FOR FURTHER INFORMATION CONTACT: Don Williams, Program Manager, Office of Northwestern Area Programs, Office of Environmental Restoration (EM–44), U.S. Department of Energy, Washington, D.C. 20585.

SUPPLEMENTARY INFORMATION: DOE has implemented environmental restoration projects at ETEC (Ventura County, Map Book 3, Page 7, Miscellaneous Records) as part of DOE's Environmental Restoration Program. One objective of the program is to identify and clean up or otherwise control facilities where residual radioactive contamination remains from activities carried out under contract to AEC/ERDA during the early years of the Nation's atomic energy program.

ETEC is comprised of a number of facilities and structures located within Administrative Area IV of the Santa Susana Field Laboratory. The work performed for DOE at ETEC consisted primarily of testing of equipment, materials, and components for nuclear and energy related programs. These nuclear energy research and development programs, conducted by Atomics International under contract to AEC/ERDA, began in 1946. Several buildings and land areas became radiologically contaminated as a result of facility operations and site activities. Building 028 is one ETEC area that has been designated for cleanup under the DOE Environmental Restoration Program. Other areas undergoing decontamination will be released as they are completed and are verified to meet established cleanup criteria and standards for release without radiological restrictions as established in DOE Order 5400.5.

Building 028 is located in the north-central section of ETEC. The above-grade concrete slab is approximately 300 m^2 in area. The below-grade vault measures approximately 60 m^2 with 6 m (20 ft.) ceilings. Construction consists of a concrete slab floor with concrete walls and ceilings.

Building 028 was originally constructed to perform tests of space reactor shields using a fission plate driven by neutrons from the thermal column of a 50-kW swimming pool-type reactor. This reactor was designated the Shield Test Reactor and operated from 1961 to 1964, when it was replaced with another reactor design to operate at 1 MW. This latter configuration was named the Shield Test and Irradiation Reactor (STIR) and operated through 1972.

Following shutdown of the test program and removal of the reactor, the facility was decommissioned and made available for alternate use in March 1976.

In 1977, operations were started to investigate the behavior of molten uranium-oxide relative to simulated reactor accidents, in particular, its reaction with floor and structural materials. These experiments resulted in some recontamination of various parts of the building that were used for preparation and melting of the uranium-oxide. Tests continued intermittently into 1981. Some facility modifications were made, and a decision to terminate operations was made later in 1981. The building remained inactive, under periodic surveillance, until decontamination began in 1988.

To allow the release of Building 028 for use without radiological restriction, all detectable radioactive material/contamination was removed from the facility. This decontamination and decommissioning was performed in two phases, starting in 1975 (STIR facility) with the removal of the core tank, the activated concrete structures surrounding the core tank, thermal column, reactor shield, test vault carriage, water cooling systems, water shield door, and the partially dismantled exhaust system.

The second and final stage of decontamination of Building 028 began in 1988 and required slightly less than five months to complete.

Briefly, the decontamination steps involved in the second stage: (1) Removal of surplus normal and depleted uranium oxide; (2) decontamination and removal of equipment and electrical components, including the furnace system used for the uranium-oxide experiments; (3) removal of the radiologically contaminated ducting system; (4) building surfaces decontamination, including scabbling of the concrete floor in Room 101A; (5) final miscellaneous cleanup operations; and (6) final radiological survey of the building (above-grade and basement).

Rockwell/Rocketdyne performed a radiological survey in 1991. The Environmental Survey and Site Assessment Program of the Oak Ridge Institute for Science and Education performed independent verification of the decontamination project in 1993. Post-decontamination surveys have demonstrated that Building 028 is in compliance with DOE decontamination criteria and standards for release without radiological restrictions. The State of California Department of Health Services has concurred that the proposed release guidelines provide adequate assurance for release without further radiological restrictions. In the event of property transfer, DOE intends to comply with applicable Federal, State, and local requirements.

The external radiation exposure of the nine people directly associated with the STIR project, particularly the dismantling operations, during the period of September 23, 1975, through January 31, 1976, averaged 193 mrem, with a maximum individual exposure of 420 mrem. The entire operation was performed with a total radiation exposure of 1.7 man-rem.

None of the engineering or radiation and nuclear safety personnel assigned to the Building 028 decommissioning project received any measurable exposure to ionizing radiation.

Final costs for the decontamination of the STIR project were $134,922.

Final costs for the decontamination of Building 028 were $239,970.

The certification docket will be available for review between 9:00 a.m. and 4:00 p.m., Monday through Friday (except Federal holidays), in the U.S. DOE Public Reading Room located in Room 1E–190 of the Forrestal Building, 1000 Independence Avenue, S.W., Washington, D.C. Copies of the certification docket will also be available at the following locations: DOE Public Document Room, U.S. DOE, Oakland Operations Office, the Federal Building, 1301 Clay Street, Oakland, California; California State University, Northridge, Urban Archives Center, Oviatt Library, Room 4, 18111 Nordhoff, Northridge, California; Simi Valley Library, 2629 Tapo Canyon Road, Simi Valley, California; and the Platt Branch, Los Angeles Public Library, 23600 Victory Boulevard, Woodland Hills, California.

DOE has issued the following statement of certification:

Statement of Certification: Energy Technology Engineering Center, Building 028

The U.S. Department of Energy, Oakland Operations Office, Environmental Restoration Division, has reviewed and analyzed the radiological data obtained following decontamination of Building 028 at the Energy Technology Engineering Center. Based on analysis of all data collected and the results of independent verification, DOE certifies that the following property is in compliance with DOE radiological decontamination criteria and standards as established in DOE Order 5400.5. This certification of compliance provides assurance that future use of the property will result in no radiological exposure above applicable guidelines established to protect members of the general public or site occupants. Accordingly, the property specified below is released from DOE's Environmental Restoration Program.

Property owned by Boeing North American Incorporated:

Building 028, at the Energy Technology Engineering Center (situated within Area IV of the Santa Susana Field Laboratory), located in a portion of Tract "A" of Rancho Simi, in the County of Ventura, State of California, as per map recorded in Book 3, Page 7 of Miscellaneous Records of Ventura County.

Issued in Washington, DC, on March 27, 1997.

James J. Fiore,

Acting Deputy Assistant Secretary for Environmental Restoration.

Statement of Certification: Energy Technology Engineering Center, Building 028

The U.S. Department of Energy, Oakland Operations Office, Environmental Restoration Division, has reviewed and analyzed the radiological data obtained following decontamination of the Energy Technology Engineering Center Building 028. Based on this analysis of all data collected, the Department of Energy (DOE) certifies that the following property is in compliance with DOE decontamination criteria and standards. This certification of compliance provides assurance that future use of the property will result in no radiological exposure above applicable guidelines established to protect members of the general public or site occupants. Accordingly, the property specified below is released from DOE's Environmental Restoration Program.

Property owned by Rockwell International Corporation:

Building 028, at the Energy Technology Engineering Center, located in a portion of Tract "A" of Rancho Simi, in the County of Ventura, State of California, as per map recorded in Book 3, Page 7 of Miscellaneous Records of Ventura County.

Certification:

Dated: January 23, 1997.

Roger Liddle,

Director, ERD.

[FR Doc. 97–8640 Filed 4–3–97; 8:45 am]

BILLING CODE 6450–01–P

Office of Energy Efficiency and Renewable Energy

[Case No. F–089]

Energy Conservation Program for Consumer Products: Granting of the Application for Interim Waiver and Publishing of the Petition for Waiver of Rheem Manufacturing Company From the DOE Furnace Test Procedure

AGENCY: Office of Energy Efficiency and Renewable Energy, Department of Energy.

ACTION: Notice.

SUMMARY: Today's notice grants an Interim Waiver to Rheem Manufacturing Company (Rheem) from the existing Department of Energy (DOE or Department) test procedure regarding blower time delay for the company's GFD upflow residential, modulating type, gas-fired furnaces.

Today's notice also publishes a "Petition for Waiver" from Rheem. Rheem's Petition for Waiver requests DOE to grant relief from the DOE furnace test procedure relating to the blower time delay specification. Rheem seeks to test using a blower delay time of 20 seconds for its GFD upflow residential, modulating type, gas-fired furnaces instead of the specified 1.5-minute delay between burner on-time and blower on-time. The Department is soliciting comments, data, and information respecting the Petition for Waiver.

DATES: DOE will accept comments, data, and information not later than May 5, 1997.

ADDRESSES: Written comments and statements shall be sent to: Department of Energy, Office of Codes and Standards, Case No. F–089, Mail Stop EE–43, Room 1J–018, Forrestal Building, 1000 Independence Avenue, SW, Washington, D.C. 20585–0121, (202) 586–7140.

FOR FURTHER INFORMATION CONTACT: Mr. Cyrus H. Nasseri, U.S. Department of Energy, Office of Energy Efficiency and Renewable Energy, Mail Station EE–43, Forrestal Building, 1000 Independence Avenue, SW., Washington, D.C. 20585–0121, (202) 586–9138, or Mr. Eugene Margolis, Esq., U.S. Department of Energy, Office of General Counsel, Mail Station GC–72, Forrestal Building, 1000 Independence Avenue, SW., Washington, D.C. 20585–0103, (202) 586–9507.

SUPPLEMENTARY INFORMATION: The Energy Conservation Program for Consumer Products (other than automobiles) was established pursuant to the Energy Policy and Conservation Act, as amended, (EPCA) which requires DOE to prescribe standardized test procedures to measure the energy consumption of certain consumer products, including furnaces.

The intent of the test procedures is to provide a comparable measure of energy consumption that will assist consumers in making purchasing decisions. These test procedures appear at Title 10 CFR Part 430, Subpart B.

The Department amended the test procedure rules to provide for a waiver process by adding Section 430.27 to Title 10 CFR Part 430. 45 FR 64108, September 26, 1980. Subsequently, DOE amended the waiver process to allow the Assistant Secretary for Energy Efficiency and Renewable Energy (Assistant Secretary) to grant an Interim Waiver from test procedure requirements to manufacturers that have petitioned DOE for a waiver of such prescribed test procedures. Title 10 CFR Part 430, Section 430.27(a)(2).

The waiver process allows the Assistant Secretary to waive temporarily test procedures for a particular basic model when a petitioner shows that the basic model contains one or more design characteristics which prevent testing according to the prescribed test procedures, or when the prescribed test procedures may evaluate the basic model in a manner so unrepresentative of its true energy consumption as to provide materially inaccurate comparative data. Waivers generally remain in effect until final test procedure amendments become effective, resolving the problem that is the subject of the waiver.

An Interim Waiver will be granted if it is determined that the applicant will experience economic hardship if the Application for Interim Waiver is denied, if it appears likely that the Petition for Waiver will be granted, and/or the Assistant Secretary determines that it would be desirable for public policy reasons to grant immediate relief pending a determination on the Petition for Waiver. Title 10 CFR Part 430, Section 430.27 (g). An Interim Waiver remains in effect for a period of 180 days or until DOE issues its determination on the Petition for Waiver, whichever is sooner, and may be extended for an additional 180 days, if necessary.

On January 29, 1997, Rheem filed an Application for Interim Waiver and a Petition for Waiver regarding blower time delay. Rheem's Application seeks an Interim Waiver from the DOE test provisions that require a 1.5-minute time delay between the ignition of the burner and starting of the circulating air blower. Instead, Rheem requests the allowance to test using a 20-second blower time delay when testing its GFD upflow residential, modulating type, gas-fired furnaces. Rheem states that the 20-second delay is indicative of how these furnaces actually operate. Such a delay results in an average of approximately 2.0 percent increase in AFUE. Since current DOE test procedures do not address this variable blower time delay, Rheem asks that the Interim Waiver be granted.

The Department has published a Notice of Proposed Rulemaking on August 23, 1993, (58 FR 44583) to amend the furnace test procedure, which addresses the above issue.

Previous Petitions for Waiver for this type of time blower delay control have been granted by DOE to Coleman Company, 50 FR 2710, January 18, 1985; Magic Chef Company, 50 FR 41553, October 11, 1985; Rheem Manufacturing Company, 53 FR 48574, December 1, 1988, 56 FR 2920, January 25, 1991, 57 FR 10166, March 24, 1992, 57 FR 34560, August 5, 1992; 59 FR 30577, June 14, 1994, and 59 FR 55470, November 7, 1994; Trane Company, 54 FR 19226, May 4, 1989, 56 FR 6021, February 14, 1991, 57 FR 10167, March 24, 1992, 57 FR 22222, May 27, 1992, 58 FR 68138, December 23, 1993, and 60 FR 62835, December 7, 1995; Lennox Industries, 55 FR 50224, December 5, 1990, 57 FR 49700, November 3, 1992, 58 FR 68136, December 23, 1993, and 58 FR 68137, December 23, 1993; Inter-City Products Corporation, 55 FR 51487, December 14, 1990, 56 FR 63945, December 6, 1991 and 61 FR 27057, May 30, 1996; DMO Industries, 56 FR 4622, February 5, 1991, and 59 FR 30579, June 14, 1994; Heil-Quaker Corporation, 56 FR 6019, February 14, 1991; Carrier Corporation, 56 FR 6018, February 14, 1991, 57 FR 38830, August 27, 1992, 58 FR 68131, December 23, 1993, 58 FR 68133, December 23, 1993, 59 FR 14394, March 28, 1994, and 60 FR 62832, December 7, 1995; Amana Refrigeration Inc., 56 FR 27958, June 18, 1991, 56 FR 63940, December 6, 1991, 57 FR 23392, June 3,

EXHIBIT II

SITEWIDE RELEASE CRITERIA FOR REMEDIATION OF FACILITIES AT THE SANTA SUSANNA FIELD LABORATORY (INCLUDES ENERGY TECHNOLOGY ENGINEERING CENTER) AND ASSOCIATED DOCUMENTATION

DOE F 1325.8
(8-89)

United States Government **Department of Energy**

memorandum

DATE: 0 9 SEP 1996

REPLY TO
ATTN OF: DOE Oakland Operations Office(ERD)

SUBJECT: Radiological Site Release Criteria for ETEC

TO: Sally Robison, EM-44

I am requesting the approval of the radiation site release criteria for the Energy Technology Engineering Center. The release criteria are a critical component in the DOE process for releasing facilities for unrestricted use. The California Department of Health Services has approved the site release criteria in a letter dated August 9 (see attachment 1).

The proposed limits were developed in the following way:

1) Annual exposure dose. Rocketdyne proposes to use a dose limit of 15 mrem/yr to comply with the 100 mrem plus ALARA as required by DOE 5400.5). This limit is also consistent with the anticipated rules of the NRC and EPA.

2) Ambient exposure rate. The proposed limit of 5μR/hr above natural background complies with the limit of 20μR/hr, plus ALARA, as stated in DOE Order 5400.5. This proposed limit is consistent with NRC limits for Rocketdyne facilities at the Santa Susana Field Laboratory. This limit would be imposed for accessible, or potentially accessible, structures and land.

3) Surface contamination. Surface contamination limits comply with DOE Order 5400.5 and specify the potential contaminants present in the Rocketdyne facilities.

4) Generic Limits for Soil and Water. The generic limits for soil and water were established using the DOE pathway analysis code RESRAD.

09/16/96

The proposed site release criteria are included in "Proposed Sitewide Release Criteria for Remediation of Facilities at the SSFL", Revision A, N001SRR140127.

Your approval is requested by September 16,1996.

Laurence McEwen
Acting Director
Environmental
Restoration Division

Attachments

cc: R. Liddle, ESO
M. Lopez, ERD
D. Williams, EM-443

96-ER-095/

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1. INTRODUCTION

At several locations at the Santa Susana Field Laboratory (SSFL), low levels of radiological contamination in buildings and in soil have occurred and have been or will be cleaned up for eventual release for use without radiological restrictions. The DOE requirements for allowable residual radioactivity in sites suitable for release without radiological restrictions ("unrestricted release") are established in DOE Order 5400.5 (Ref. 1). Specific guidelines are given in 5400.5 for surface contamination and for direct gamma exposure. However, except for radium and thorium in soil, no specific guidelines are provided for residual contamination in soil or water. It has become clear that a set of DOE-authorized limits for the SSFL would greatly facilitate the process of determining that a facility is acceptably clean, and verifying this with a confirmatory survey. Approval of such a set of authorized limits is provided for in DOE Order 5400.5, Chapter IV, Section 5, and in draft 10 CFR 834.301(c).

The purpose of this report is to develop a set of proposed guideline values for approval by DOE for the release without radiological restriction of DOE facilities at the SSFL. The various categories of release guidelines include; 1) annual expected dose, 2) soil and water concentration guidelines, 3) surface contamination guidelines, and 4) ambient gamma exposure rate. The guidelines presented in this report are for residual radioactivity above background. When feasible, the local background activity of the suspect radionuclides should be determined and these background values subtracted from the measured release survey data.

The goal for these limits is to provide assurance that reasonable future uses of the property will not result in individual doses exceeding 15 millirem per year. This is consistent with current EPA and NRC guidance, and is supported by a generic cost-benefit analysis presented in Reference 2.

3. SOIL AND WATER GUIDELINES

Since there are no federal or state regulatory limits for soil contamination for many of the potential or actual radionuclides of concern at SSFL, site-specific guidelines must be developed. This development is done, as required by the DOE Order, by use of a "pathways" analysis program, which estimates the radiological dose (total effective dose equivalent) that a future user of the property might receive, considering the residual radioactivity and various conditions of use. An effort is made to make these use conditions as reasonable for the use and the local area as can be achieved, without greatly over-estimating or under-estimating potential doses.

To establish these guidelines for cleanup operations at SSFL, the pathways analysis program RESRAD (Ref. 4), developed at Argonne National Laboratory (ANL) for use by DOE, has been used to calculate single radionuclide guidelines for the radionuclides of potential concern at SSFL.

For soil, a dose limit of 15 millirem per year is used. For consideration of radiological contamination in water, which may be collected from wells, sumps, below-grade seepage, or surface water, concentration guidelines were calculated from the Dose Conversion Factors (DCFs) in RESRAD, using the EPA limit of 4 millirem per year for ingested drinking water (Ref. 5), and the EPA assumed intake of water, 2 liters per day. These limits are more restrictive than those imposed on releases from operating facilities, as provided by DOE Order 5400.5 (Ref. 1), NRC (Ref. 6), the State of California (Ref. 7), and EPA for uranium mines and mills (Ref. 8).

3.1 Pathway Analysis

Pathways analysis involves calculating the doses received by a person through several pathways: direct radiation exposure; inhalation of airborne radioactivity; drinking water containing radioactivity; eating foods that have accumulated radioactivity, through uptake of water with radioactivity from the soil, or with airborne radioactivity deposited on the foliage; and ingestion of small amounts of contaminated soil.

The pathways analysis program RESRAD, now in Version 5.61, was developed in the late 1980's for DOE by Argonne National Laboratory for the purpose of performing pathways analysis for a broad range of applications. Considerable flexibility is provided in the program for representing the site-specific conditions of exposure, to permit making the calculation as reasonable for the application as is possible.

Four general types of use may be considered for land for the purpose of calculating dose, other than the obvious zero-dose case of non-use. These may be identified as the industrial scenario, the wilderness scenario (or recreational, such as a park or golf course), the residential scenario, and the family farm scenario. Within these general use scenarios, choices are made for occupancy time (indoors and outdoors), water use, and food sources. Further choices are made to represent the contamination situation, geology, and hydrology. The program comes with a

part of several earlier efforts at the SSFL, a number of screening evaluations were performed using the RESRAD code to determine which of the approximately 80 input parameters required by RESRAD were of significance to the general SSFL area. These screening evaluations also were useful in determining conservative site-specific values for input to the code, when the default values were not used. In general, changes to most of the parameters were found to have a negligible effect on the final results because certain dose pathways were either not applicable or negligible for the given scenarios.

Contaminated Zone Parameters: Default values for the area of contamination (10,000 m^2) and the length parallel to aquifer flow (100 m) were assumed. For the depth of contamination, a conservative value of 1 meter is assumed. Measurements conducted at the site have indicated historical maximum values ranging from about 0.4 to 0.6 m for this parameter.

Occupancy Parameters: The default RESRAD values for occupancy of a residence on an affected site are 50% of the time spent indoors and 25% of the time spent outdoors, on the site. Thus, 25% of the time the occupancy is assumed to be off site. For the residential scenario, assuming 8,760 hours in a year, this translates into 4,380 hours spent indoors, 2,190 hours spent outdoors on the site, and 2,190 hours spent off site. For the industrial scenario, the corresponding percentages are assumed to be 20%, 4%, and 76% respectively. For the wilderness scenario, the corresponding percentages are 0%, 10%, and 90%.

Shielding Factors: The annual dose estimates calculated by RESRAD from either direct exposure or by inhalation (dust) are functions of two "structural" shielding parameters and the fraction of time an individual is assumed to spend inside a structure built on the site. Both shielding factors range from 0 to 1, and may be changed by the user to more appropriately match actual site conditions. For inhalation, the RESRAD default is 0.4, and this value is assumed for the present evaluations. For direct gamma exposure, the RESRAD default is 0.7, which is a rather conservative estimate of gamma shielding by a structure. For the present calculations, this latter value was adjusted from the default, for both the industrial and residential scenarios, to account for local construction practice which dictate a minimum 4-inch (0.1 m) concrete slab under the structure.

The gamma shielding factor used as input to RESRAD was calculated by modeling a typical two-story residential structure, and a single story industrial structure using the computer code MicroShield[1]. MicroShield is a point-kernel gamma shielding code developed for IBM-compatible personal computers, based on the mainframe code ISOSHLD. For the residential structure, a conservative lower bound footprint (area) value of 93 m^2 (1,000 ft^2) was assumed. For the industrial structure, a 186 m^2 (2,000 ft^2) area was assumed. A circular area was used with MicroShield to obtain maximum code accuracy with minimum computational time.

[1] MicroShield, Version 4.0, Grove Engineering, Inc., 15215 Shady Grove Road, Suite 200, Rockville, MD 20850.

It should be noted, that these values do not take into account any out-structures such as garages and patios, both of which would result in additional gamma shielding, and both of which would almost certainly be part of any residences built on the site.

Dietary Parameters: Default RESRAD input values for food and water consumption are based on the family farm scenario, where a significant portion of the diet is grown or raised on the site. For the three credible scenarios considered here, these parameters were adjusted as follows: for the residential scenario, it is conservatively assumed that a small fraction (10% of that grown on a family farm) of the fruit and leafy vegetables consumption would be from material grown on site. The values used are 16 kg/year per person and 1.4 kg/year per person, respectively. It was further assumed that water for the residence would be obtained from a well on the site (510 liters/year per person).

For the industrial and wilderness scenarios, it was assumed that no water would be used that was taken from the site; thus, all water pathways were suppressed with the exception of a secondary pathway via plant ingestion. In the industrial case, bottled drinking water is supplied. Since essentially all surface water at present is a result of the current industrial operations, no surface water would be available in the wilderness scenario. It is also assumed that perhaps 1% of the family farm fruit consumption value might be collected from wild sources, thus, 0.14 kg/year is used for these scenarios.

Contaminated Zone Hydrology Data: The SSFL facility is located in the Simi Hills in eastern Ventura County, California. The Simi Hills are in the northern part of the Transverse Range geomorphic province, and are composed primarily of exposures of the Upper Cretaceous Chatsworth Formation. This formation is a marine turbidite sequence of sandstone with interbedded siltstone/mudstone and minor conglomeratic lenses. The Chatsworth Formation is at least 1,800 m thick in locations east and north of the Facility.

The principal geologic units at the SSFL are the Chatsworth Formation and the shallow alluvium which overlies the Chatsworth Formation in some parts of the Facility, notably in Area IV of the SSFL where the decommissioning and decontamination of nuclear sites is taking place. This layer is Quaternary alluvium consisting of mixtures of unconsolidated sand, silt, and clay, and would include the contaminated zone. Drill holes indicate that the layer may be as thick as 6 meters in some locations.

The density of this alluvium layer is approximately 1.5 g/cm^3. The total and effective porosity of the contaminated zone are assumed to be 0.43 and 0.20 based on the average of data for sand, silt, and clay as given in the RESRAD manual. Precipitation at the facility is measured annually by a rain gauge located in the northeastern portion of the SSFL (Ventura County Rain Gauge Number 249). Based on measured data since 1959, the mean annual precipitation at the SSFL is approximately 18.6 inch, or 0.47 meters. In general, the majority of the precipitation occurs during the months of January through March.

3.5 Proposed Soil and Water Guidelines

Based on the data in Table 3, proposed conservative guidelines, consistent with the several applicable regulations governing residual radioactivity discussed above, are listed in Table 4. With the exception of uranium, radium, and thorium, the proposed soil guidelines are those calculated from RESRAD for the residential use scenario. For uranium, proposed guidelines are those adopted by the NRC (30, 30, and 35 pCi/g for U-234, U-235, and U-238, respectively, see

Table 3. RESRAD-Calculated Single Isotope Guidelines Values

Radionuclide	Soil Guidelines (pCi/g) Industrial	Wilderness	Residential	Water (pCi/l)[a]
Am-241	120	162	5.44	1.50
Co-60	10.9	9.83	1.94	204
Cs-134	18.7	16.9	3.33	74.7
Cs-137	51.9	46.7	9.20	110
Eu-152	25.3	22.8	4.51	845
Eu-154	23.0	20.7	4.11	573
Fe-55	2,370,000	4,780,000	629,000	9,020
H-3	129,000	129,000	31,900	85,600[b]
K-40	162	147	27.6	294
Mn-54	34.4	30.9	6.11	1,980
Na-22	13.0	11.7	2.31	476
Ni-59	1,390,000	1,560,000	151,000	26,100
Ni-63	511,000	572,000	55,300	9,490
Pu-238	140	192	37.2	1.71
Pu-239	127	175	33.9	1.55
Pu-240	127	175	33.9	1.55
Pu-241	4,740	6,430	230	79.9
Pu-242	133	183	35.5	1.63
Ra-226	0.520	13.6	0.199	4.12[b]
Sr-90	370	376	36.0	35.8[b]
Th-228	14.8	14.7	2.81	6.78
Th-232	7.94	7.98	1.53	2.01
U-234	519	647	106	19.3[b]
U-235	163	160	32.1	20.5[b]
U-238	399	445	90.9	20.4[b]

[a]Water guidelines calculated from RESRAD ingestion dose conversion factors, assuming the EPA dose limit of 4 mrem/year (see text).

[b]For these radionuclides, the EPA Safe Drinking Water Act or the State of California CCR Title 22 limits should be used (see Table 4).

The proposed site release criteria are included in "Proposed Sitewide Release Criteria for Remediation of Facilities at the SSFL", Revision A, N001SRR140127.

Your approval is requested by September 16,1996.

Laurence McEwen
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Occupancy Parameters: The default RESRAD values for occupancy of a residence on an affected site are 50% of the time spent indoors and 25% of the time spent outdoors, on the site. Thus, 25% of the time the occupancy is assumed to be off site. For the residential scenario, assuming 8,760 hours in a year, this translates into 4,380 hours spent indoors, 2,190 hours spent outdoors on the site, and 2,190 hours spent off site. For the industrial scenario, the corresponding percentages are assumed to be 20%, 4%, and 76% respectively. For the wilderness scenario, the corresponding percentages are 0%, 10%, and 90%.

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For the industrial and wilderness scenarios, it was assumed that no water would be used that was taken from the site; thus, all water pathways were suppressed with the exception of a secondary pathway via plant ingestion. In the industrial case, bottled drinking water is supplied. Since essentially all surface water at present is a result of the current industrial operations, no surface water would be available in the wilderness scenario. It is also assumed that perhaps 1% of the family farm fruit consumption value might be collected from wild sources, thus, 0.14 kg/year is used for these scenarios.

Contaminated Zone Hydrology Data: The SSFL facility is located in the Simi Hills in eastern Ventura County, California. The Simi Hills are in the northern part of the Transverse Range geomorphic province, and are composed primarily of exposures of the Upper Cretaceous Chatsworth Formation. This formation is a marine turbidite sequence of sandstone with interbedded siltstone/mudstone and minor conglomeratic lenses. The Chatsworth Formation is at least 1,800 m thick in locations east and north of the Facility.

The principal geologic units at the SSFL are the Chatsworth Formation and the shallow alluvium which overlies the Chatsworth Formation in some parts of the Facility, notably in Area IV of the SSFL where the decommissioning and decontamination of nuclear sites is taking place. This layer is Quaternary alluvium consisting of mixtures of unconsolidated sand, silt, and clay, and would include the contaminated zone. Drill holes indicate that the layer may be as thick as 6 meters in some locations.

The density of this alluvium layer is approximately 1.5 g/cm^3. The total and effective porosity of the contaminated zone are assumed to be 0.43 and 0.20 based on the average of data for sand, silt, and clay as given in the RESRAD manual. Precipitation at the facility is measured annually by a rain gauge located in the northeastern portion of the SSFL (Ventura County Rain Gauge Number 249). Based on measured data since 1959, the mean annual precipitation at the SSFL is approximately 18.6 inch, or 0.47 meters. In general, the majority of the precipitation occurs during the months of January through March.

3.5 Proposed Soil and Water Guidelines

Based on the data in Table 3, proposed conservative guidelines, consistent with the several applicable regulations governing residual radioactivity discussed above, are listed in Table 4. With the exception of uranium, radium, and thorium, the proposed soil guidelines are those calculated from RESRAD for the residential use scenario. For uranium, proposed guidelines are those adopted by the NRC (30, 30, and 35 pCi/g for U-234, U-235, and U-238, respectively, see

Table 3. RESRAD-Calculated Single Isotope Guidelines Values

	Soil Guidelines (pCi/g)			
Radionuclide	**Industrial**	**Wilderness**	**Residential**	**Water (pCi/l)[a]**
Am-241	120	162	5.44	1.50
Co-60	10.9	9.83	1.94	204
Cs-134	18.7	16.9	3.33	74.7
Cs-137	51.9	46.7	9.20	110
Eu-152	25.3	22.8	4.51	845
Eu-154	23.0	20.7	4.11	573
Fe-55	2,370,000	4,780,000	629,000	9,020
H-3	129,000	129,000	31,900	85,600[b]
K-40	162	147	27.6	294
Mn-54	34.4	30.9	6.11	1,980
Na-22	13.0	11.7	2.31	476
Ni-59	1,390,000	1,560,000	151,000	26,100
Ni-63	511,000	572,000	55,300	9,490
Pu-238	140	192	37.2	1.71
Pu-239	127	175	33.9	1.55
Pu-240	127	175	33.9	1.55
Pu-241	4,740	6,430	230	79.9
Pu-242	133	183	35.5	1.63
Ra-226	0.520	13.6	0.199	4.12[b]
Sr-90	370	376	36.0	35.8[b]
Th-228	14.8	14.7	2.81	6.78
Th-232	7.94	7.98	1.53	2.01
U-234	519	647	106	19.3[b]
U-235	163	160	32.1	20.5[b]
U-238	399	445	90.9	20.4[b]

[a]Water guidelines calculated from RESRAD ingestion dose conversion factors, assuming the EPA dose limit of 4 mrem/year (see text).

[b]For these radionuclides, the EPA Safe Drinking Water Act or the State of California CCR Title 22 limits should be used (see Table 4).

The proposed site release criteria are included in "Proposed Sitewide Release Criteria for Remediation of Facilities at the SSFL", Revision A, N001SRR140127.

Your approval is requested by September 16,1996.

Laurence McEwen
Acting Director
Environmental
Restoration Division

Attachments

cc: R. Liddle, ESO
M. Lopez, ERD
D. Williams, EM-443

96-ER-095/

DOE F 1325.8
(08-93)

United States Government **Department of Energy**

memorandum

DATE: SEP 17 1996

REPLY TO ATTN OF: EM-44 (D. Williams, 903-8173)

SUBJECT: Sitewide Limits for Release of Facilities Without Radiological Restriction

TO: R. Liddle, Oakland Operations Office

We have reviewed Rocketdyne's proposed sitewide limits for release of facilities at the Santa Susana Field Laboratory (SSFL) without radiological restriction and are satisfied that our previous concerns and comments have been addressed.

The proposed limits are consistent with the Department of Energy (DOE) Order 5400.5 requirement for a Total Effective Dose Equivalent limit of 100 mrem/yr plus As low As Reasonably Achievable (ALARA) for future occupants, the Nuclear Regulatory Commission proposed a radiological guideline of 15 mrem/yr ALARA, and the Environmental Protection Agency proposed a guideline of 15 mrem/yr for release of properties.

Corrective actions taken by Rocketdyne for the sampling and statistical approach to final survey data validation for DOE projects are now comparable to methodologies or standard practices used at other DOE sites and the requirements of Nuclear Regulatory Commission Nuclear Regulation (NUREG)/CR-5489 (Manual for Conducting Radiological Surveys in Support of License Termination).

We also received a copy of the letter from the California Department of Health Services stating concurrence with the proposed release guidelines and the intent to incorporate these guidelines into Rocketdyne's California Radioactive Material License.

Based upon the above information, the proposed sitewide release criteria for remediation of facilities at the SSFL are hereby approved for use.

If you have any questions, please call Mr. Don Williams of my staff at 301-903-8173.

Sally A. Robison, Ph.D.
Director
Office of Northwestern Area Programs
Environmental Restoration

STATE OF CALIFORNIA—HEALTH AND WELFARE AGENCY PETE WILSON, *Governor*

DEPARTMENT OF HEALTH SERVICES
714/744 P STREET
P.O. BOX 942732
SACRAMENTO, CA 94234-7320

96ETEC-DRF-0455

(916) 323-2759

August 9, 1996

Ms. Majelle Lee, Program Manager
Environmental Management
Rocketdyne Division
Rockwell International Corporation
P. O. Box 7930
Canoga Park, CA 91309-7930

Subject: Authorized Sitewide Radiological Guidelines for Release of Unrestricted Use

Dear Ms. Lee:

This letter is to acknowledge the receipt of your letter dated June 28, 1996 requesting concurrence of the above subject. The above mentioned letter and its attachments have been reviewed by the staff of this office. The Radiologic Health Branch (RHB) concurs that the proposed release guidelines provide adequate assurance for the release of the facilities and properties at Rocketdyne's Santa Susana Field Laboratory (SSFL) and DeSoto sites without further radiological restrictions. Your letter dated June 28, 1996 with attachments will be incorporated into Rocketdyne's California Radioactive Material License # 0015-70 upon receipt of a commitment letter signed by Mr. Phil Rutherford.

If you have any questions concerning this matter, please feel free to call Mr. Stephen Hsu of this office at (916) 322-4797.

Sincerely,

Gerard Wong

Gerard Wong, Ph.D., Chief
Radioactive Material Licensing Section
Radiologic Health Branch

Rockwell International
Rocketdyne Division

SUPPORTING DOCUMENT

GO NO.	S/A NO.	PAGE 1 OF	TOTAL PAGES	REV LTR/CHG NO. SEE SUMMARY OF CHG	NUMBER
96469	69100	25	26	A	N001SRR140127

PROGRAM TITLE
Radiological Remediation

DOCUMENT TITLE
Proposed Sitewide Release Criteria for Remediation of Facilities at the SSFL

DOCUMENT TYPE
Safety Review Report

KEY NOUNS
Release, Criteria, Guidelines, Soil, Contamination

ORIGINAL ISSUE DATE: 3-11-96
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APPROVALS / DATE
P. D. Rutherford 3/7/96
M. E. Lee 3/8/96
C. M. Jones 3/11/96

PREPARED BY/DATE	DEPT	MAIL ADDR
B. M. Oliver 3/7/96	641	T100
R. J. Tuttle	641	T100

IR&D PROGRAM? YES ☐ NO ☒
IF YES, ENTER TPA NO.

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ABSTRACT

At several locations at the Santa Susana Field Laboratory (SSFL), low levels of radioactive contamination in buildings and in soil have occurred and have been or will be cleaned up for eventual release for use without radiological restrictions. For this purpose, a complete set of proposed guideline values for approval by DOE for DOE facilities at the SSFL has been developed, and are presented in this report. The various categories of release guidelines include; 1) annual expected dose, 2) soil and water concentration guidelines, 3) surface contamination guidelines, and 4) ambient gamma exposure rate. The proposed guidelines were obtained from regulatory values where available. Where not available, for example for soil, guidelines were calculated by use of the DOE computer code, RESRAD. For these calculations, the proposed annual dose is 15 mrem/year, which is consistent with proposed EPA and NRC guidelines and ALARA principles. The radionuclide guidelines presented in this report are applicable to DOE-optioned facilities and land and other radiological areas.

RESERVED FOR PROPRIETARY/LEGAL NOTICES

12-84

REV	SUMMARY OF CHANGE	APPROVALS AND DATE
A	Section 2: Section reworded to include a reference to ALARA. Dose limit changed to 15 mrem/yr, with new justification. Reference to EPA ALARA analysis included. All references to "without consideration of costs" have been removed. Section 3.2: Reference to topography of region included as additional justification for exclusion of the family farm scenario. Section 3.3 - Shielding Parameter: Shielding calculations revised to reflect a two story residential structure (of the same total floor area), and an effective dose point location midway from the center to the edge of the structure for each story. Residential occupancy realistically apportioned between the first and second stories. Sections 3.4 and 3.5: DOE values for Radium and Thorium are specified instead of the more restrictive RESRAD values. Tables 3 and 4 values have been updated to reflect the new shielding calculations and the 15 mrem/y annual dose limit. Section 6.0: First paragraph revised and combined with second paragraph. Sections 6.1, 6.2, and 6.3: Words added to explain the sampling procedure. Specifically, that sample locations are biased towards areas of known higher readings, or areas of potential contamination. Appendix A: Updated.	B M Oliver 8/13/96 B. M. Oliver R J Tuttle 8/14/96 R. J. Tuttle Phil Rutherford 8/14/96 P. D. Rutherford M E Lee 8/14/96 M. E. Lee C M Jones 8/14/96 C. M. Jones MAG Rel: 8-22-96

TABLE OF CONTENTS

LIST OF TABLES

1. INTRODUCTION

At several locations at the Santa Susana Field Laboratory (SSFL), low levels of radiological contamination in buildings and in soil have occurred and have been or will be cleaned up for eventual release for use without radiological restrictions. The DOE requirements for allowable residual radioactivity in sites suitable for release without radiological restrictions ("unrestricted release") are established in DOE Order 5400.5 (Ref. 1). Specific guidelines are given in 5400.5 for surface contamination and for direct gamma exposure. However, except for radium and thorium in soil, no specific guidelines are provided for residual contamination in soil or water. It has become clear that a set of DOE-authorized limits for the SSFL would greatly facilitate the process of determining that a facility is acceptably clean, and verifying this with a confirmatory survey. Approval of such a set of authorized limits is provided for in DOE Order 5400.5, Chapter IV, Section 5, and in draft 10 CFR 834.301(c).

The purpose of this report is to develop a set of proposed guideline values for approval by DOE for the release without radiological restriction of DOE facilities at the SSFL. The various categories of release guidelines include; 1) annual expected dose, 2) soil and water concentration guidelines, 3) surface contamination guidelines, and 4) ambient gamma exposure rate. The guidelines presented in this report are for residual radioactivity above background. When feasible, the local background activity of the suspect radionuclides should be determined and these background values subtracted from the measured release survey data.

The goal for these limits is to provide assurance that reasonable future uses of the property will not result in individual doses exceeding 15 millirem per year. This is consistent with current EPA and NRC guidance, and is supported by a generic cost-benefit analysis presented in Reference 2.

2. ANNUAL DOSE LIMITATION

DOE Order 5400.5 specifies a base Total Effective Dose Equivalent (TEDE) limit of 100 millirem per year for any potential future occupant of a remediated site. The Order also requires the use of the As Low As Reasonably Achievable (ALARA) principle to establish Authorized Limits at a level that is below the base limit. Rocketdyne is proposing to apply a value of 15 millirem per year for the calculation of derived limits for the cleanup of DOE sites at the SSFL, consistent with EPA and NRC guidance. A limit of 15 millirem per year (mrem/year) is adopted to assure that future uses will contribute small doses compared to natural background doses, which are in the range of 250-400 mrem/year (Ref. 3). This limit is considered to be as low as reasonably achievable below the basic DOE dose limit of 100 mrem/year. The 15 mrem/year value corresponds to a calculated increased lifetime cancer risk to a potential future user of the site of 3×10^{-4}.

For any reasonable assigned cost per person-rem, further reduction of anticipated dose due to exposure to residual radioactivity at the site is difficult to justify. For example, the EPA proposed TEDE of 15 mrem/year was arrived at after extensive ALARA analysis of cleanup costs and benefits at sixteen "Reference Sites" representing a wide range of conditions found at contaminated sites throughout the United States. Their analyses assumed a residential use of the decontaminated sites, and their conclusions were that the 15 mrem/year limit represented the most effective value considering all the technical and socio-political issues involved.

Furthermore, at the SSFL, conservative choices in the development, measurement, and interpretation of limits and final surveys provide a firm bias towards overestimation of the remaining risk. These include, 1) a conservative residential scenario for the pathway analyses, 2) use of calibration sources that tend to underestimate the detector efficiency for the likely contaminants, and 3) both qualitative and quantitative tests that provide assurance that the decommissioned facility is suitable for release without radiological restrictions.

3. SOIL AND WATER GUIDELINES

Since there are no federal or state regulatory limits for soil contamination for many of the potential or actual radionuclides of concern at SSFL, site-specific guidelines must be developed. This development is done, as required by the DOE Order, by use of a "pathways" analysis program, which estimates the radiological dose (total effective dose equivalent) that a future user of the property might receive, considering the residual radioactivity and various conditions of use. An effort is made to make these use conditions as reasonable for the use and the local area as can be achieved, without greatly over-estimating or under-estimating potential doses.

To establish these guidelines for cleanup operations at SSFL, the pathways analysis program RESRAD (Ref. 4), developed at Argonne National Laboratory (ANL) for use by DOE, has been used to calculate single radionuclide guidelines for the radionuclides of potential concern at SSFL.

For soil, a dose limit of 15 millirem per year is used. For consideration of radiological contamination in water, which may be collected from wells, sumps, below-grade seepage, or surface water, concentration guidelines were calculated from the Dose Conversion Factors (DCFs) in RESRAD, using the EPA limit of 4 millirem per year for ingested drinking water (Ref. 5), and the EPA assumed intake of water, 2 liters per day. These limits are more restrictive than those imposed on releases from operating facilities, as provided by DOE Order 5400.5 (Ref. 1), NRC (Ref. 6), the State of California (Ref. 7), and EPA for uranium mines and mills (Ref. 8).

3.1 Pathway Analysis

Pathways analysis involves calculating the doses received by a person through several pathways: direct radiation exposure; inhalation of airborne radioactivity; drinking water containing radioactivity; eating foods that have accumulated radioactivity, through uptake of water with radioactivity from the soil, or with airborne radioactivity deposited on the foliage; and ingestion of small amounts of contaminated soil.

The pathways analysis program RESRAD, now in Version 5.61, was developed in the late 1980's for DOE by Argonne National Laboratory for the purpose of performing pathways analysis for a broad range of applications. Considerable flexibility is provided in the program for representing the site-specific conditions of exposure, to permit making the calculation as reasonable for the application as is possible.

Four general types of use may be considered for land for the purpose of calculating dose, other than the obvious zero-dose case of non-use. These may be identified as the industrial scenario, the wilderness scenario (or recreational, such as a park or golf course), the residential scenario, and the family farm scenario. Within these general use scenarios, choices are made for occupancy time (indoors and outdoors), water use, and food sources. Further choices are made to represent the contamination situation, geology, and hydrology. The program comes with a

complete set of generally conservative default values, and these may be changed as appropriate to reflect local reality in terms of usage practices and physical conditions, to produce a realistic pathways analysis for the specific site. The default values and the values actually used by the program in the analysis are listed in the output for each calculation, so departures from the default set are well recorded. The printed results from the calculations described in this report are stored in the Environmental Remediation (ER) library file.

The family farm, on which family members spend 100% of their time, drinking water from the surface or from wells, eating vegetables and fruit grown on the land and irrigated with the same water, raising their meat, milk, and fish on that land, is not a reasonable scenario for the site. Although commercial farming is practiced in low-lying valley and coastal areas west of the facility, the rugged nature and topography of the SSFL, combined with poor soil quality, would reasonably preclude a family farm activity on the site. Further, recent land use trends in the area have been to conversion of previous farming property to other non-farming uses. Thus, the industrial, wilderness, and residential scenarios are all perhaps equally probable for the future of the site, and should be the scenarios considered.

3.2 Property Usage Scenarios

The basic usage conditions (per year) modeled in these calculations, for each of the three realistic scenarios, are summarized in Table 1. A complete listing of all RESRAD input data, for the three scenarios, is given in Appendix A. Discussion on specific RESRAD input parameters is given below in Section 3.3

Table 1. Property Usage Conditions for Three Realistic Scenarios

	Industrial	Wilderness	Residential
Occupancy, indoors (hours/year)	1752	0	4380
Occupancy, outdoors (hours/year)	350	876	2190
Occupancy, off site (hours/year)	6664	7890	2190
Drinking water (liters/year)	0	0	510
Fruit, vegetables, grain (kg/year)	1.6	1.6	16
Leafy vegetables (kg/year)	0	0	1.4
Cover thickness (meters)	0	0	0
Contamination area (m^2)	10000	10000	10000
Contamination thickness (meters)	1	1	1
Depth to water table (meters)	5	5	5

3.3 RESRAD Input Parameters

Default values provided in RESRAD are considered to be conservative estimates intended for use when no site-specific information is available. Users of the program are encouraged, however, to use input data that most closely reflects actual conditions existing on their site. As

part of several earlier efforts at the SSFL, a number of screening evaluations were performed using the RESRAD code to determine which of the approximately 80 input parameters required by RESRAD were of significance to the general SSFL area. These screening evaluations also were useful in determining conservative site-specific values for input to the code, when the default values were not used. In general, changes to most of the parameters were found to have a negligible effect on the final results because certain dose pathways were either not applicable or negligible for the given scenarios.

Contaminated Zone Parameters: Default values for the area of contamination (10,000 m^2) and the length parallel to aquifer flow (100 m) were assumed. For the depth of contamination, a conservative value of 1 meter is assumed. Measurements conducted at the site have indicated historical maximum values ranging from about 0.4 to 0.6 m for this parameter.

Occupancy Parameters: The default RESRAD values for occupancy of a residence on an affected site are 50% of the time spent indoors and 25% of the time spent outdoors, on the site. Thus, 25% of the time the occupancy is assumed to be off site. For the residential scenario, assuming 8,760 hours in a year, this translates into 4,380 hours spent indoors, 2,190 hours spent outdoors on the site, and 2,190 hours spent off site. For the industrial scenario, the corresponding percentages are assumed to be 20%, 4%, and 76% respectively. For the wilderness scenario, the corresponding percentages are 0%, 10%, and 90%.

Shielding Factors: The annual dose estimates calculated by RESRAD from either direct exposure or by inhalation (dust) are functions of two "structural" shielding parameters and the fraction of time an individual is assumed to spend inside a structure built on the site. Both shielding factors range from 0 to 1, and may be changed by the user to more appropriately match actual site conditions. For inhalation, the RESRAD default is 0.4, and this value is assumed for the present evaluations. For direct gamma exposure, the RESRAD default is 0.7, which is a rather conservative estimate of gamma shielding by a structure. For the present calculations, this latter value was adjusted from the default, for both the industrial and residential scenarios, to account for local construction practice which dictate a minimum 4-inch (0.1 m) concrete slab under the structure.

The gamma shielding factor used as input to RESRAD was calculated by modeling a typical two-story residential structure, and a single story industrial structure using the computer code MicroShield[1]. MicroShield is a point-kernel gamma shielding code developed for IBM-compatible personal computers, based on the mainframe code ISOSHLD. For the residential structure, a conservative lower bound footprint (area) value of 93 m^2 (1,000 ft^2) was assumed. For the industrial structure, a 186 m^2 (2,000 ft^2) area was assumed. A circular area was used with MicroShield to obtain maximum code accuracy with minimum computational time.

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[1] MicroShield, Version 4.0, Grove Engineering, Inc., 15215 Shady Grove Road, Suite 200, Rockville, MD 20850.

Screening calculations indicated no significant differences between the results for circular and square areas of the same volume.

In all cases the contaminated soil was assumed to have a density of 1.5 g/cm^2, and a thickness of 1 meter. Dose calculations were performed for two vertical distances (1m for the ground floor and 3.6 m for the second story) and for three radial distances (center, midpoint, and edge of structure). The isotopic mix input to MicroShield was the same as that used for the present RESRAD calculations, with a concentration of 1 pCi/g for each isotope. Resulting gamma energy groups for this isotope mix ranged from 0.1 to 1.5 MeV. A factor of 0.89 was used to account for gamma shielding from a typical structural wall composed of approximately 1 inch of stucco and 5/8 inch of drywall, and a window area of approximately 10% of the wall area.

Effective gamma shielding factors obtained from the MicroShield calculations are given in Appendix A. For the residential scenario (the most credible), it is assumed that 12 hours are spent inside the structure per day. If it is further assumed that 8 of these hours are spent upstairs in a bedroom, 4 hours are spent downstairs in a family room, and that a person (on average) is located at the midpoint between the center and the edge of the structure, then the effective gamma shielding factor would be: (0.67)(0.61) + (0.33)(0.31) = 0.51. For the industrial scenario, the value is 0.25, which is the shielding value at the midpoint location for the single story structure.

Table 2. Gamma Shielding Factor Calculations for Typical SSFL Structure

	Gamma Shielding Factor	
Radial Location	**1st Floor**	**2nd Floor**
Residential Structure (93 m^2 footprint, two story)		
Center	0.27	0.57
Midpoint[a]	0.31	0.61
Perimeter[b]	0.57	0.71
Industrial Structure (186 m^2 footprint, single story)		
Center	0.22	-
Midpoint[a]	0.25	-
Perimeter[b]	0.58	-

[a]Midpoint between the center and the perimeter of the structure
[b]Edge of the structure.

It should be noted, that these values do not take into account any out-structures such as garages and patios, both of which would result in additional gamma shielding, and both of which would almost certainly be part of any residences built on the site.

Dietary Parameters: Default RESRAD input values for food and water consumption are based on the family farm scenario, where a significant portion of the diet is grown or raised on the site. For the three credible scenarios considered here, these parameters were adjusted as follows: for the residential scenario, it is conservatively assumed that a small fraction (10% of that grown on a family farm) of the fruit and leafy vegetables consumption would be from material grown on site. The values used are 16 kg/year per person and 1.4 kg/year per person, respectively. It was further assumed that water for the residence would be obtained from a well on the site (510 liters/year per person).

For the industrial and wilderness scenarios, it was assumed that no water would be used that was taken from the site; thus, all water pathways were suppressed with the exception of a secondary pathway via plant ingestion. In the industrial case, bottled drinking water is supplied. Since essentially all surface water at present is a result of the current industrial operations, no surface water would be available in the wilderness scenario. It is also assumed that perhaps 1% of the family farm fruit consumption value might be collected from wild sources, thus, 0.14 kg/year is used for these scenarios.

Contaminated Zone Hydrology Data: The SSFL facility is located in the Simi Hills in eastern Ventura County, California. The Simi Hills are in the northern part of the Transverse Range geomorphic province, and are composed primarily of exposures of the Upper Cretaceous Chatsworth Formation. This formation is a marine turbidite sequence of sandstone with interbedded siltstone/mudstone and minor conglomeratic lenses. The Chatsworth Formation is at least 1,800 m thick in locations east and north of the Facility.

The principal geologic units at the SSFL are the Chatsworth Formation and the shallow alluvium which overlies the Chatsworth Formation in some parts of the Facility, notably in Area IV of the SSFL where the decommissioning and decontamination of nuclear sites is taking place. This layer is Quaternary alluvium consisting of mixtures of unconsolidated sand, silt, and clay, and would include the contaminated zone. Drill holes indicate that the layer may be as thick as 6 meters in some locations.

The density of this alluvium layer is approximately 1.5 g/cm^3. The total and effective porosity of the contaminated zone are assumed to be 0.43 and 0.20 based on the average of data for sand, silt, and clay as given in the RESRAD manual. Precipitation at the facility is measured annually by a rain gauge located in the northeastern portion of the SSFL (Ventura County Rain Gauge Number 249). Based on measured data since 1959, the mean annual precipitation at the SSFL is approximately 18.6 inch, or 0.47 meters. In general, the majority of the precipitation occurs during the months of January through March.

Saturated Zone Hydrology Data: There are two groundwater systems at the SSFL: 1) a shallow system in the surficial alluvium and the underlying zones of weathered sandstone and siltstone/claystone, and isolated shallow fracture systems; and 2) a deeper regional system in the fractured Chatsworth Formation. The shallow zone is discontinuous, with depths to groundwater ranging from land surface to over 9 m. For the present study, we assume that this shallow region most conservatively represents the saturated zone, with an average depth to the water table of about 5 m. Hydraulic conductivity in the saturated zone generally ranges from about 30 to 3,000 m/year. Here, the higher value has been assumed.

Typical pumping rates for deep wells in the Chatsworth Formation (rock) range from 60 to 70 m^3/year up to a maximum of about 300 m^3/year. For the shallow (alluvium) region, however, pumping rates are significantly lower, typically about 35 m^3/year. Further, in the shallow region, many wells would be dry for a good fraction of the year as the replenishment rate is generally low. Water table drop rates, therefore, would range up to 10 m as a result of on-site pumping. Without pumping, however, no data is available on any inherent lowering of the water table. For conservatism, therefore, the default value of 0.001 m/year has been assumed.

Radon Pathway: Two default values were modified for the radon pathway. The thickness of the foundation was set at 0.1 m (4 inches) to correspond to the gamma shielding calculations discussed above. Also, the depth below ground surface was also set at 0.1 m, as basement structures are not typical for the local area.

3.4 Calculated Soil and Water Guidelines from RESRAD

The guidelines calculated from the RESRAD code for various single radionuclides are listed in Table 3 for comparison of the three scenarios. Values for each of the scenarios were determined from separate RESRAD calculation runs using the input parameters given in Appendix A. Water guideline values in Table 3 were calculated from the dose conversion factors used in RESRAD for ingestion, using an EPA value of 2 liters/day total water consumption (per person) from the site, and an EPA dose limit of 4 mrem/year (Ref. 5).

For radionuclides specifically regulated by the EPA (and the State of California), the Safe Drinking Water Act (and CCR Title 22) limits were used. These are (in pCi/l):

H-3	20,000
Combined Ra-226 and Ra-228	5
Sr-90	8
Gross alpha (not including radon and uranium)	15
Gross beta	50
Uranium (U-234 + U-235 + U-238)	20

For U-234, U-235, and U-238, DOE imposes the EPA regulations in 40 CFR 192 (and parts 190 and 440). Similarly, for Ra-226, Th-228 and Th-232, DOE imposes the limits in DOE Order 5400.5.

3.5 Proposed Soil and Water Guidelines

Based on the data in Table 3, proposed conservative guidelines, consistent with the several applicable regulations governing residual radioactivity discussed above, are listed in Table 4. With the exception of uranium, radium, and thorium, the proposed soil guidelines are those calculated from RESRAD for the residential use scenario. For uranium, proposed guidelines are those adopted by the NRC (30, 30, and 35 pCi/g for U-234, U-235, and U-238, respectively, see

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Radionuclide	**Industrial**	**Wilderness**	**Residential**	**Water (pCi/l)[a]**
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Cs-137	51.9	46.7	9.20	110
Eu-152	25.3	22.8	4.51	845
Eu-154	23.0	20.7	4.11	573
Fe-55	2,370,000	4,780,000	629,000	9,020
H-3	129,000	129,000	31,900	85,600[b]
K-40	162	147	27.6	294
Mn-54	34.4	30.9	6.11	1,980
Na-22	13.0	11.7	2.31	476
Ni-59	1,390,000	1,560,000	151,000	26,100
Ni-63	511,000	572,000	55,300	9,490
Pu-238	140	192	37.2	1.71
Pu-239	127	175	33.9	1.55
Pu-240	127	175	33.9	1.55
Pu-241	4,740	6,430	230	79.9
Pu-242	133	183	35.5	1.63
Ra-226	0.520	13.6	0.199	4.12[b]
Sr-90	370	376	36.0	35.8[b]
Th-228	14.8	14.7	2.81	6.78
Th-232	7.94	7.98	1.53	2.01
U-234	519	647	106	19.3[b]
U-235	163	160	32.1	20.5[b]
U-238	399	445	90.9	20.4[b]

[a]Water guidelines calculated from RESRAD ingestion dose conversion factors, assuming the EPA dose limit of 4 mrem/year (see text).

[b]For these radionuclides, the EPA Safe Drinking Water Act or the State of California CCR Title 22 limits should be used (see Table 4).

Table 4. Proposed Soil and Water Guidelines for SSFL Facilities

Radionuclide	Soil Guidelines (pCi/g)	Water (pCi/l)
Am-241	5.44	1.5
Co-60	1.94	200
Cs-134	3.33	75
Cs-137	9.20	110
Eu-152	4.51	840
Eu-154	4.11	570
Fe-55	629,000	9,000
H-3	31,900	20,000[a]
K-40	27.6	290
Mn-54	6.11	2,000
Na-22	2.31	480
Ni-59	151,000	26,000
Ni-63	55,300	9,500
Pu-238	37.2	1.7
Pu-239	33.9	1.6
Pu-240	33.9	1.6
Pu-241	230	80
Pu-242	35.5	1.6
Ra-226	5[c] and 15[c]	4.1
Sr-90	36.0	8[a]
Th-228	5[c] and 15[c]	6.8
Th-232	5[c] and 15[c]	2.0
U-234	30[b]	
U-235	30[b]	total uranium 20[a]
U-238	35[b]	
Gross alpha (not including radon and uranium)		15[a]
Gross beta		50[a]

[a]State of California Maximum Contaminant Levels, CCR Title 22
[b]Generally more conservative NRC limits for uranium isotopes are proposed.
[c]DOE Order 5400.5 limits are proposed (5 pCi/g averaged over first 15 cm of soil depth and 15 pCi/g averaged over 15 cm layers below the top 15 cm).

Ref. 9). For radium and thorium, DOE Order 5400.5 limits are proposed (5 pCi/g averaged over first 15 cm of soil depth and 15 pCi/g averaged over 15 cm layers below the top 15 cm, see Ref. 1). Guidelines established from the residential use scenario are the most restrictive of the three scenarios considered.

The choice of a basic dose limit of 15 mrem/year for all pathways combined leads to lower limits than would result from the use of the dose limits established by the EPA for the uranium fuel cycle (Ref. 10) and by DOE for unrestricted release of contaminated property (Ref. 1). The water guidelines are those calculated from the RESRAD dose conversion factors, using the EPA values for the basic dose limit and daily water intake, with the Maximum Contaminant Levels (MCL) specified for certain radionuclides by the State of California (Ref. 11).

4. SURFACE CONTAMINATION GUIDELINES

Surface contamination limits are specified in Figure IV-1 of Chapter IV in DOE Order 5400.5. For SSFL facilities, these limits have been modified by specifying the potential contaminants present in the Rockwell facilities, and eliminating those that are not pertinent. The proposed guidelines are given in Table 5. As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

Table 5. Proposed Surface Contamination Guidelines for SSFL Facilities

Radionuclide	Average over 1 m^2 (dpm/100 cm^2)	Maximum in 100 cm^2 (dpm/100 cm^2)	Removable (dpm/100 cm^2)
Plutonium, Radium	100	300	20
Thorium	1,000	3,000	200
Uranium	5,000	15,000	1,000
Mixed fission products	5,000	15,000	1,000
Activation products	5,000	15,000	1,000
Tritium	-	-	10,000

As included in Table 5, Pu, Ra, U, Th, mixed fission products, and activation products, refer to those forms of radioactive material that comprise the residual activity at the SSFL. Plutonium is predominately Pu-239; Radium is Ra-226. It is assumed that thorium is sufficiently aged that all daughters are in equilibrium, Th-natural. Uranium will occur in depleted, normal, or enriched forms; U-233 is not present. Mixed fission products include Sr-90 and Cs-137 as components of the mixture. Possible activation products include Co-60, Fe-55, Mn-54, Eu-152, Eu-154, Al-26, and similar radionuclides.

Tritium contamination limits are based on interim guidelines for removable surface contamination (Ref. 12). This level of removable contamination insures that any non-removable or volumetric contamination will not cause unacceptable exposures.

These guidelines would be imposed for accessible (or potentially accessible) surfaces and structures.

5. AMBIENT GAMMA EXPOSURE RATE

A guideline of 5 μR/hr above natural background, measured at 1 meter above the surface, is proposed. This value has been imposed by the NRC for decommissioning research reactors (Ref. 13). It is as low as reasonably measurable, due to variations in background, and is significantly lower than the guideline of 20 μR/hr stated in DOE Order 5400.5, Chapter IV, Section 4.c. This guideline would be imposed for accessible (or potentially accessible) structures and land. Our experience has been that this level can be achieved and verified in facilities that would be suitable for continued use.

6. APPLICATION OF GUIDELINES

The guidelines presented above should be used in planning any decontamination effort at the SSFL. Analytical capability for detection of each radionuclide should be, if possible, less than one-tenth of the guideline values. That is, the Minimum Detectable Activity (MDA, our LLD) should be less than 0.1 x guideline. Field measurements used to direct removal of contaminated soil should be capable of practical measurements below the guideline value. Survey measurements and sample analyses should be corrected for the local background activity of each radionuclide.

6.1 Soil Guidelines

Sample analysis is necessary to demonstrate the successful decontamination of soil areas. A qualitative scan will be performed using gamma-sensitive and/or beta-sensitive detectors to identify any significant areas of residual contamination. Soil samples will be taken from locations based on a 3x3 meter master grid. One sample will be taken from within a 1x1 meter grid location in each 3x3-meter section, based either on the qualitative scan survey indications at the area of maximum readings or, if no noticeable readings were found, at the location most likely to have residual contamination, by the surveyor's judgment. This selection assures a reasonably uniform sampling of the ground areas, at a sample density of approximately 11 samples per 100 m^2.

Results from individual samples will be compared with the limit for hotspots of 9-m^2 area, that is, 3.3 x the adopted concentration limit. Averages of adjacent samples, covering 100 m^2, will be compared with the average limit. The overall average, assuming that the individual and 100-m^2 area averages satisfy the applicable limits, will be used for a RESRAD confirmatory calculation. This calculation will be performed to demonstrate that the maximum expected annual dose for the indicated reasonable use scenario for the facility *does not exceed* the proposed 15 mrem/year guideline value.

For mixtures of radionuclides in soil, the "Sum of Fractions" rule is used. The sum of the ratios of concentration of each radionuclide to the corresponding guideline must not exceed 1. This value must be satisfied when samples are averaged over each 100-m^2 region. For cases in which the relative concentrations are known or assumed, this method is used to generate combined radionuclide guidelines for each radionuclide in the mixture.

The guidelines are not intended to be spot limits, and should not be applied to individual measurements. If the specific sampling provides only (or fewer than) one measurement per 100-m^2 area, each measurement becomes, by default, the "average" for that 100-m^2 area, and the guidelines have the effect of acting as spot limits. In cases where an individual sample exceeds the guideline value, additional samples should be taken from within the same 100-m^2 area, and used to define the average contamination in this area.

The maximum concentrations remaining as "hot spots" must have contamination less than that calculated by the hot-spot rule presented in DOE Order 5400.5, Chapter IV, page 4. The average contamination within any area not exceeding 25 m^2 shall not be greater than $\sqrt{100/A}$ guideline, where A is the area in m^2. Reasonable efforts shall be made to remove any soil with contamination that exceeds 30 x guideline (Ref. 4).

6.2 Surface Contamination Guidelines

The proposed surface contamination guidelines would be applied to all accessible surfaces and structures. This would include ceilings, floors, and walls, and other potentially accessible locations such as attics. Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the guidelines established for alpha- and beta-gamma-emitting radionuclides should apply independently. Measurements of average contamination are averaged over an area of 1 m^2. For objects of less surface area, the average should be derived for each such object. The maximum contamination level applies to an area of not more than 100 cm^2. Surfaces of facilities which are likely to be contaminated, but are inaccessible for purposes of measurement, shall be presumed to be contaminated in excess of the applicable limits.

Following a complete qualitative scan of the facility, quantitative surface contamination measurements will be made over a fraction of the structural surfaces, as determined by the designation of the area as affected or unaffected. Affected areas will be surveyed at a nominal fraction of 11%. Unaffected areas will be surveyed at lesser fractions. Locations for the quantitative survey measurements will be based on a 3x3 meter master grid. One sample will be taken from within a 1x1 meter grid location in each 3x3-meter section, based either on the qualitative scan survey indications at the area of maximum readings or, if no noticeable readings were found, at the location most likely to have residual contamination, by the surveyor's judgment. Results from individual locations will be compared with the applicable limits.

Total surface contamination is measured by use of detectors primarily or exclusively sensitive to alpha or beta-gamma radiation. After a qualitative survey of the surfaces of the entire subject area, quantitative measurements are made on 1-m^2 areas selected uniformly throughout the area. These measurements are made with the detectors connected to a scaler set to accumulate counts for a 5-minute period. The detector is slowly scanned over the 1-m^2 grid location and the numerical result, after correction for background, count time, and detector efficiency, yields the 1-m^2 average surface activity. These detectors are calibrated against Th-230 for alpha activity and Tc-99 for beta activity. The emission energies of these radionuclides is generally less than those radionuclides found as contamination at SSFL. This results in an underestimate of the efficiency of the detectors for the actual contaminant radioactivity and hence an overestimate of the actual measurement.

The amount of removable activity per 100 cm^2 of surface area is determined by wiping an area of that size with dry filter or soft absorbent paper, applying moderate pressure, and measuring the amount of radioactive material on the wiping with an appropriate instrument of known efficiency. Typically at Rocketdyne, a low background gas flow proportional counter is used. When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area should be based on the actual area and the entire surface should be wiped. It is not necessary to use wiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the guidelines for removable contamination.

Smear methods for tritium detection are similar to that described above, with the exception that a wet swipe or piece of Styrofoam should be used. If the property has been recently decontaminated, a follow-up measurement (smears) should be conducted to ensure that there is no build-up of contamination with time.

6.3 Ambient Gamma Exposure

Measurements of the ambient gamma exposure rate provides a useful determination of residual volumetric radioactivity that may not be as easily detected by surface measurements or sampling and analysis. For the purpose of demonstrating suitability for release, this measurement provides an additional test.

The DOE established a limit of 20 μR/hr above natural background for screening radium-contaminated property. The NRC has imposed a 10μR/hr limit on the decommissioning of radioactive materials licensees, and a 5μR/hr limit on the decommissioning of research reactors. The 5 μR/hr limit above natural background is proposed for use at Rocketdyne. Because of the variability and differences in natural background, the limit of 5 μR/hr is about as low as can be reasonably implemented.

Quantitative measurements of the ambient gamma exposure rate will be made over a fraction of the structural surfaces, as determined by the designation of the area as affected or unaffected. Affected areas will be surveyed at a nominal fraction of 11%. Unaffected areas will be surveyed at lesser fractions. Locations for the quantitative survey measurements will be based on a 3x3-meter master grid. One measurement, covering one 1-m^2 grid location, will be made at each grid location chosen for the surface contamination measurements. Results from individual locations will be compared with the applicable limits.

At Rocketdyne, gamma exposure rate is generally measured by use of a 1x1 inch NaI(Tl) detector/photomultiplier probe, connected to a scaler to provide objective numerical values. The detector is placed 1 meter above the local (ground or floor) surface. This instrument is calibrated by reference to a High Pressure Ion Chamber (HPIC) in a background area.

6.4 Statistical Validation of Survey Data

The statistical approach employed at Rocketdyne/ETEC for establishing that survey data meets guideline values is a method referred to as Sampling Inspection by Variables (Ref. 14). This method has been widely applied in industry and the military and is essential where the lot size is impractically large. Application of this method to the remediation of contaminated sites has been discussed in detail elsewhere (see for example, Ref. 15).

In sampling inspection by variables, the number of data points on which measurements are obtained is first chosen to be large so that the parameters of the distribution are likely to have a normal distribution (i.e., Gaussian). The mean of the distribution, $\bar{x}$, and its standard deviation, s, are then related to a "test statistic", TS, as follows:

$$TS = \bar{x} + ks$$

where $\bar{x}$ = average (arithmetic mean of measured values)
s = observed sample standard deviation
k = tolerance factor calculated from the number of samples to achieve the desired sensitivity for the test

TS and $\bar{x}$ are then compared with an authorized acceptance limit, U, to determine acceptance or other plans of action, including rejection of the area as contaminated and requiring further remediation.

The sample mean and standard deviation are easily calculable quantities; the value of k, the tolerance factor, bears further discussion. Of the various criteria for selecting plans for acceptance sampling by variables, the most appropriate is the method of Lot Tolerance Percent Defective (LTPD), also referred to as the Rejectable Quality Level (RQL). The LTPD is defined as the poorest quality that should be accepted in an individual lot. Associated with the LTPD is a parameter referred to as consumer's risk (β), the risk of accepting a lot of quality equal to or poorer than the LTPD (or 10%). NRC Regulatory Guide 6.6 (Ref. 16) states that the value for the consumer's risk should be 0.10. Conventionally, the value assigned to the LTPD has been 10%.

The State of California, Department of Radiological Health Branch, has stated that the consumer's risk of acceptance (β) at 10% defective (LTPD) must be 0.1 (Ref. 17). For those choices of β and LTPD, $K_\beta = K_2 = 1.282$. The number of samples is n. Values of k for each sample size are calculated in accordance with the following equations:

$$k = \frac{K_2 + \sqrt{K_2^2 - ab}}{a}; \quad a = 1 - \frac{K_\beta}{2(n-1)}; \quad b = K_2^2 - \frac{K_\beta^2}{n}$$

where k = tolerance factor,

K_β = the normal deviate exceeded with probability of β, 0.10 (from tables, $K_2 = 1.282$, see Ref. 18),

K_2 = the normal deviate exceeded with probability equal to the LTPD, 10% (from tables, $K_\beta = 1.282$, see Ref. 18)^2, and

n = number of samples.

The statistical criteria for acceptance of a remediated area are presented below.

a) Acceptance: If the test statistic ($\bar{x}$ + ks) is less than or equal to the guideline (U), accept the area as clean. If any single measured value exceeds 80% of the limit, decontaminate that location to as near background as is possible, but do not change the value in the analysis.

b) Collect additional measurements: If the test statistic ($\bar{x}$ + ks) is greater that the limit (U), but $\bar{x}$ itself is less than U, independently resample and combine all measured values to determine if $\bar{x}$ + ks ≤ = U for the combined set; if so, accept the area as clean. If not, the area is contaminated and must be remediated.

c) Rejection: If the test statistic ($\bar{x}$ + ks) is greater than the limit (U) and $\bar{x}$ > = U, the region is contaminated and must be remediated.

Thus, based on sampling inspection, we are willing to accept the hypothesis that the probability of accepting an area as not being contaminated which is, in fact, 10% or more contaminated is 0.10. Or in other words, the final survey acceptance criteria corresponds to assuring with 90% confidence that 90% of an area has residual contamination below 100% (a 90/90/100 test) of the authorized limit.

7. REFERENCES

1. "RADIATION PROTECTION OF THE PUBLIC AND ENVIRONMENT," DOE Order 5400.5, U. S. Department of Energy, February 8, 1989.

2. "Regulatory Impact Analysis for Radiation Site Cleanup Proposed Rule - Draft", U. S. Environmental Protection Agency, January 17, 1996.

3. "Health Effects of Exposure to Low Levels of Ionizing Radiation -- BEIR V," Committee on the Biological Effects of Ionizing Radiations, Board on Radiation Effects Research, Commission on Life Sciences, National Research Council. National Academy Press, Washington, 1990.

4. "A Manual for Implementing Residual Radioactive Material Guidelines," DOE/CH/8901, U. S. Department of Energy, June 1989.

5. "National Primary Drinking Water Regulations; Radionuclides," 40 CFR 141.15 and .16, U. S. Environmental Protection Agency, July 18, 1991.

6. "STANDARDS FOR PROTECTION AGAINST RADIATION," 10 CFR 20, U. S. Nuclear Regulatory Commission.

7. "CALIFORNIA RADIATION CONTROL REGULATIONS," CCR 17, State of California Department of Health Services.

8. "ORE MINING AND DRESSING POINT SOURCE CATEGORY," Subpart C--Uranium, Radium and Vanadium Ores Subcategory, 40 CFR 440, U. S. Environmental Protection Agency.

9. "Disposal or Onsite Storage of Thorium or Uranium Wastes from Past Operations," Federal Register . Vol. 46, No. 205, pp. 52061-52063, Friday, October 23, 1981. U. S. Nuclear Regulatory Commission.

10. "ENVIRONMENTAL PROTECTION STANDARDS FOR NUCLEAR POWER OPERATIONS," 40 CFR 190, U. S. Environmental Protection Agency.

11. "Environmental Health, Radioactivity," CCR 22, Article 5. State of California Department of Health Services.

12. "Application of DOE 5400.5 Requirements for Release and Control of Property Containing Residual Radioactive Material", DOE Memorandum, DOE-OAK, January 5, 1996.

13. "Order Authorizing Dismantling of Facility and Disposition of Component Parts", Docket No. 50-375, Enclosure to NRC Letter dated February 22, 1983, D. Eisenhut to M. Remley.

14. DOE/CH/8901, A Manual for Implementing Residual Radioactive Material Guidelines, T. L. Gilbert, et al., June 1989.

15. "Statistical Treatment of Radiological Survey Data for Cleanup Decisions and Regulatory Compliance", Proc. Spectrum '92 Conference, Boise, Idaho, August 23 - 27, 1992.

16. "Acceptance Sampling Procedures for Exempted and Generally Licensed Items Containing By-Product Material", U. S. Nuclear Regulatory Commission Guide 6.6, dated June 1974.

17. DECON-1, State of California for Decontaminating Facilities and Equipment Prior to Release for Unrestricted Use, dated June 1977.

18. MIL-STD-414, Sampling Procedures and Tables for Inspection by Variables for Percent Defective, June 11, 1957.

Appendix A

Input Parameters for RESRAD Calculations (Sheet 1 of 3)

Parameter	Value Used for Scenario			RESRAD
	Industrial	Wilderness	Residential	Default
Area of contaminated zone (m^2)	1.000E+04	1.000E+04	1.000E+04	1.000E+04
Thickness of contaminated zone (m)	1.000E+00	2.000E+00	1.000E+00	2.000E+00
Length parallel to aquifer flow (m)	1.000E+02	1.000E+02	1.000E+02	1.000E+02
Basic radiation dose limit (mrem/yr)	1.500E+01	1.500E+01	1.500E+01	3.000E+01
Time since placement of material (yr)	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Times for calculations (yr)	1.000E+00	1.000E+00	1.000E+00	1.000E+00
Times for calculations (yr)	3.000E+00	3.000E+00	3.000E+00	3.000E+00
Times for calculations (yr)	1.000E+01	1.000E+01	1.000E+01	1.000E+01
Times for calculations (yr)	3.000E+01	3.000E+01	3.000E+01	3.000E+01
Times for calculations (yr)	1.000E+02	1.000E+02	1.000E+02	1.000E+02
Times for calculations (yr)	3.000E+02	3.000E+02	3.000E+02	3.000E+02
Times for calculations (yr)	1.000E+03	1.000E+03	1.000E+03	1.000E+03
Times for calculations (yr)	3.000E+03	0.000E+00	3.000E+03	0.000E+00
Times for calculations (yr)	1.000E+04	0.000E+00	1.000E+04	0.000E+00
Cover depth (m)	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Density of cover material (g/cm^3)	not used	not used	not used	1.500E+00
Cover depth erosion rate (m/yr)	not used	not used	not used	1.000E-03
Density of contaminated zone (g/cm^3)	1.500E+00	1.500E+00	1.500E+00	1.500E+00
Contaminated zone erosion rate (m/yr)	1.000E-03	1.000E-03	1.000E-03	1.000E-03
Contaminated zone total porosity	4.300E-01	4.300E-01	4.300E-01	4.000E-01
Contaminated zone effective porosity	2.000E-01	2.000E-01	2.000E-01	2.000E-01
Contaminated zone hydraulic conductivity (m/yr)	3.000E+03	3.000E+03	3.000E+03	1.000E+01
Contaminated zone b parameter	5.300E+00	5.300E+00	5.300E+00	5.300E+00
Humidity in air (g/cm^3)	8.000E+00	8.000E+00	8.000E+00	8.000E+00
Evapotranspiration coefficient	5.000E-01	5.000E-01	5.000E-01	5.000E-01
Precipitation (m/yr)	4.700E-01	4.700E-01	4.700E-01	1.000E+00
Irrigation (m/yr)	2.000E-01	2.000E-01	2.000E-01	2.000E-01
Irrigation mode	overhead	overhead	overhead	overhead
Runoff coefficient	2.000E-01	2.000E-01	2.000E-01	2.000E-01
Watershed area for nearby stream or pond (m^2)	1.000E+06	1.000E+06	1.000E+06	1.000E+06
Accuracy for water/soil computations	1.000E-03	1.000E-03	1.000E-03	1.000E-03
Density of saturated zone (g/cm^3)	1.500E+00	1.500E+00	1.500E+00	1.500E+00
Saturated zone total porosity	4.300E-01	4.300E-01	4.300E-01	4.000E-01
Saturated zone effective porosity	2.000E-01	2.000E-01	2.000E-01	2.000E-01
Saturated zone hydraulic conductivity (m/yr)	3.000E+03	3.000E+03	3.000E+03	1.000E+02
Saturated zone hydraulic gradient	2.000E-02	2.000E-02	2.000E-02	2.000E-02
Saturated zone b parameter	5.300E+00	5.300E+00	5.300E+00	5.300E+00
Water table drop rate (m/yr)	1.000E-03	1.000E-03	1.000E-03	1.000E-03
Well pump intake depth (m below water table)	1.000E+01	1.000E+01	1.000E+01	1.000E+01

Input Parameters for RESRAD Calculations (Sheet 2 of 3)

Parameter	Value Used for Scenario			RESRAD
	Industrial	Wilderness	Residential	Default
Model: Nondispersion (ND) or Mass-Balance (MB)	ND	ND	ND	ND
Well pumping rate (m^3/yr)	not used	not used	7.000E+01	2.500E+02
Number of unsaturated zone strata	1	1	1	1
Unsat. zone 1, thickness (m)	4.000E+00	4.000E+00	4.000E+00	4.000E+00
Unsat. zone 1, soil density (g/cm^3)	1.500E+00	1.500E+00	1.500E+00	1.500E+00
Unsat. zone 1, total porosity	4.300E-01	4.300E-01	4.300E-01	4.000E-01
Unsat. zone 1, effective porosity	2.000E-01	2.000E-01	2.000E-01	2.000E-01
Unsat. zone 1, soil-specific b parameter	5.300E+00	5.300E+00	5.300E+00	5.300E+00
Unsat. zone 1, hydraulic conductivity (m/yr)	3.000E+03	3.000E+03	3.000E+03	1.000E+01
Inhalation rate (m^3/yr)	8.400E+03	8.400E+03	8.400E+03	8.400E+03
Mass loading for inhalation (g/m^3)	2.000E-04	2.000E-04	2.000E-04	2.000E-04
Dilution length for airborne dust, inhalation (m)	3.000E+00	3.000E+00	3.000E+00	3.000E+00
Exposure duration	3.000E+01	3.000E+01	3.000E+01	3.000E+01
Shielding factor, inhalation	4.000E-01	4.000E-01	4.000E-01	4.000E-01
Shielding factor, external gamma	2.500E-01	7.000E-01	5.100E-01	7.000E-01
Fraction of time spent indoors	2.000E-01	0.000E+00	5.000E-01	5.000E-01
Fraction of time spent outdoors (on site)	4.000E-02	1.000E-01	2.500E-01	2.500E-01
Shape factor flag, external gamma	1.000E+00	1.000E+00	1.000E+00	1.000E+00
Fruits, vegetables and grain consumption (kg/yr)	1.600E+00	1.600E+00	1.600E+01	1.600E+02
Leafy vegetable consumption (kg/yr)	0.000E+00	0.000E+00	1.400E+00	1.400E+01
Milk consumption (L/yr)	not used	not used	not used	9.200E+01
Meat and poultry consumption (kg/yr)	not used	not used	not used	6.300E+01
Fish consumption (kg/yr)	not used	not used	not used	5.400E+00
Other seafood consumption (kg/yr)	not used	not used	not used	9.000E-01
Soil ingestion rate (g/yr)	3.650E+01	3.650E+01	3.650E+01	3.650E+01
Drinking water intake (L/yr)	not used	not used	5.100E+02	5.100E+02
Contamination fraction of drinking water	not used	not used	1.000E+00	1.000E+00
Contamination fraction of household water	1.000E+00	0.000E+00	1.000E+00	1.000E+00
Contamination fraction of livestock water	not used	0.000E+00	not used	1.000E+00
Contamination fraction of irrigation water	1.000E+00	1.000E+00	1.000E+00	1.000E+00
Contamination fraction of aquatic food	not used	not used	not used	5.000E-01
Contamination fraction of plant food	-1	-1	-1	-1
Contamination fraction of meat	not used	not used	not used	-1
Contamination fraction of milk	not used	not used	not used	-1
Livestock fodder intake for meat (kg/day)	not used	not used	not used	6.800E+01
Livestock fodder intake for milk (kg/day)	not used	not used	not used	5.500E+01
Livestock water intake for meat (L/day)	not used	not used	not used	5.000E+01
Livestock water intake for milk (L/day)	not used	not used	not used	1.600E+02
Livestock soil intake (kg/day)	not used	not used	not used	5.000E-01
Mass loading for foliar deposition (g/m^3)	1.000E-04	1.000E-04	1.000E-04	1.000E-04
Depth of soil mixing layer (m)	1.500E-01	1.500E-01	1.500E-01	1.500E-01
Depth of roots (m)	9.000E-01	9.000E-01	9.000E-01	9.000E-01

Input Parameters for RESRAD Calculations (Sheet 3 of 3)

Parameter	Value Used for Scenario			RESRAD Default
	Industrial	Wilderness	Residential	
Drinking water fraction from ground water	1.000E+00	1.000E+00	1.000E+00	1.000E+00
Household water fraction from ground water	not used	not used	1.000E+00	1.000E+00
Livestock water fraction from ground water	1.000E+00	1.000E+00	1.000E+00	1.000E+00
Irrigation fraction from ground water	not used	not used	not used	1.000E+00
C-12 concentration in water (g/cm^3)	not used	not used	not used	2.000E-05
C-12 concentration in contaminated soil (g/g)	not used	not used	not used	3.000E-02
Fraction of vegetation carbon from soil	not used	not used	not used	2.000E-02
Fraction of vegetation carbon from air	not used	not used	not used	9.800E-01
C-14 evasion layer thickness in soil (m)	not used	not used	not used	3.000E-01
C-14 evasion flux rate from soil (1/sec)	not used	not used	not used	7.000E-07
C-12 evasion flux rate from soil (1/sec)	not used	not used	not used	1.000E-10
Fraction of grain in beef cattle feed	not used	not used	not used	8.000E-01
Fraction of grain in milk cow feed	not used	not used	not used	2.000E-01
Storage times of contaminated foodstuffs (days):				
Fruits, non-leafy vegetables, and grain	1.400E+01	1.400E+01	1.400E+01	1.400E+01
Leafy vegetables	1.000E+00	1.000E+00	1.000E+00	1.000E+00
Milk	not used	not used	not used	1.000E+00
Meat and poultry	not used	not used	not used	2.000E+01
Fish	not used	not used	not used	7.000E+00
Crustacea and mollusks	not used	not used	not used	7.000E+00
Well water	1.000E+00	1.000E+00	1.000E+00	1.000E+00
Surface water	1.000E+00	1.000E+00	1.000E+00	1.000E+00
Livestock fodder	not used	not used	not used	4.500E+01
Thickness of building foundation (m)	1.000E-01	not used	1.000E-01	1.500E-01
Bulk density of building foundation (g/cm)	2.400E+00	not used	2.400E+00	2.400E+00
Total porosity of the cover material	not used	not used	not used	4.000E-01
Total porosity of the building foundation	1.000E-01	not used	1.000E-01	1.000E-01
Volumetric water content of the cover material	not used	not used	not used	5.000E-02
Volumetric water content of the foundation	3.000E-02	not used	3.000E-02	3.000E-02
Diffusion coefficient for radon gas (m/sec):				
in cover material	not used	not used	not used	2.000E-06
in foundation material	3.000E-07	not used	3.000E-07	3.000E-07
in contaminated zone soil	2.000E-06	not used	2.000E-06	2.000E-06
Radon vertical dimension of mixing (m)	2.000E+00	not used	2.000E+00	2.000E+00
Average annual wind speed (m/sec)	2.000E+00	not used	2.000E+00	2.000E+00
Average building air exchange rate (1/hr)	5.000E-01	not used	5.000E-01	5.000E-01
Height of the building (room) (m)	2.500E+00	not used	2.500E+00	2.500E+00
Building interior area factor	0.000E+00	not used	0.000E+00	0.000E+00
Building depth below ground surface (m)	1.000E-01	not used	1.000E-01	-1.000E+00
Emanating power of Rn-222 gas	2.500E-01	not used	2.500E-01	2.500E-01
Emanating power of Rn-220 gas	not used	not used	not used	1.500E-01

EXHIBIT III

INDEPENDENT VERIFICATION DOCUMENTATION OF THE RADIOLOGICAL CONDITION OF BUILDING 028 AT ENERGY TECHNOLOGY ENGINEERING CENTER AFTER DECONTAMINATION AND DECOMMISSIONING

VERIFICATION SURVEY OF THE OLD CONSERVATION YARD, BUILDING T064 SIDE YARD, AND BUILDING T028 SANTA SUSANA FIELD LABORATORY ROCKWELL INTERNATIONAL VENTURA COUNTY, CALIFORNIA

T. J. VITKUS

Prepared for the Office of Environmental Restoration
U.S. Department of Energy

ORISE
OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION
Environmental Survey and Site Assessment Program
Energy/Environment Systems Division

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Prepared for the

Office of Environmental Restoration
U.S. Department of Energy

FINAL REPORT

OCTOBER 1993

This report is based on work performed under contract number DE-AC05-76OR00033 with the U.S. Department of Energy.

VERIFICATION SURVEY
OF THE
OLD CONSERVATION YARD, BUILDING T064 SIDE YARD,
AND BUILDING T028
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

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TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

ABBREVIATIONS AND ACRONYMS

ac	acres
AEC	Atomic Energy Commission
cm	centimeter
DOE	Department of Energy
dpm/100 cm^2	disintegrations per minute per 100 square centimeters
EML	Environmental Measurement Laboratories
ER	Office of Environmental Restoration
ERDA	Energy Research and Development Administration
ESG	Energy Systems Group
ESSAP	Environmental Survey and Site Assessment Program
ETEC	Energy Technology Engineering Center
ft	feet
ft^2	square feet
ha	hectares
in	inch
km	kilometer
m	meter
m^2	square meter
mi	mile
mrem	millirem
NIST	National Institute of Standards and Technologies
OCY	Old Conservation Yard
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocurie per gram
STR	Shield Test Reactor
SSFL	Santa Susana Field Laboratory
STIR	Shield Test and Irradation Reactor
μR/h	microroentegens per hour

VERIFICATION SURVEY
OF THE
OLD CONSERVATION YARD, BUILDING T064 SIDE YARD,
AND BUILDING T028
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
VENTURA COUNTY, CALIFORNIA

INTRODUCTION AND SITE HISTORY

Rockwell International's Rocketdyne Division operates the Santa Susana Field Laboratory (SSFL) for the Department of Energy (DOE). The facility, known as the Energy Technology Engineering Center (ETEC), began nuclear energy research and development programs in 1946. Contract work for the Atomic Energy Commission (AEC) and the Energy Research and Development Administration (ERDA), predecessor agencies to the DOE, began in the early 1950's. Specific programs conducted for AEC/ERDA/DOE involved the engineering, development, testing, and manufacturing of nuclear reactor systems and components. Other site activities have also been conducted for the Nuclear Regulatory Commission, the Department of Defense, and other government related or affiliated organizations and agencies.

Numerous buildings and land areas became radiologically contaminated as a result of facility operations and site activities which included ten reactors, seven criticality test facilities, fuel fabrication, reactor and fuel disassembly, laboratory work, and on-site storage of nuclear material. Potential radioactive contaminants identified at the site are uranium (in natural, depleted, and enriched isotopic abundances), plutonium, americium-241, fission products (primarily cesium-137 and strontium-90), activation products (cobalt-60, europium-152, nickel-63, promethium-147, and tantalum-182) and tritium. Chemical contaminants, mainly chlorinated organic solvents, have also been identified in groundwater.

Decontamination and decommissioning of facilities began in the late 1960's and continues as specific DOE-sponsored projects are phased out. In addition to radiological surveys to support current facility decontaminations, Rockwell/Rocketdyne determined that the documentation describing the radiological status for a number of early projects was inadequate; therefore,

SURVEY PROCEDURES: BUILDING T028

Reference Grid

A reference grid, consisting of 1 m^2 grid blocks, was established on the above-grade concrete slab and on the floor and lower walls (up to 2 m) of the below-grade vault (Figures 8 and 9). Upper walls, ceilings and the stairwell were not gridded. Measurements and samples from ungridded surfaces were referenced to the floor or lower wall grid or to prominent building features.

Surface Scans

Surface scans for alpha, beta, and gamma activity were performed on the concrete slab and below-grade floors, walls and overhead surfaces using ZnS scintillation, GM, and NaI detectors coupled to ratemeters or ratemeter-scalers with audible indicators.

Surface Activity Measurements

The primary contaminant within Building T028 was uranium in natural and depleted isotopic abundances. Uranium emits both alpha and beta radiations at approximate ratios of 1:1 and 1:1.6 for natural and depleted uranium, respectively. The surface contamination guidelines for uranium are in units of alpha dpm/100 cm^2; however, because rough, dirty, or damp surfaces selectively attenuate alpha radiation, beta activity was also measured.

Direct measurements for total alpha and total beta activity were performed on a total of ten randomly selected grid blocks located in the vault or on the above-ground concrete slab. One set of five direct measurements was obtained from each grid block. Measurements were performed at the center and four points equidistant from the center and grid block corners. Single-point alpha and beta measurements were performed at six locations on upper walls and ceiling of the vault and at three locations in the stairwell. Direct measurements were made using ZnS and GM detectors coupled to ratemeter-scalers. A smear sample, for determining gross alpha and gross beta activity, was collected from the location within each grid block corresponding to the highest total direct measurement and from each single-point measurement location. Figures 8 through 10 indicate measurement and sampling locations.

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data were returned to ORISE's ESSAP laboratory in Oak Ridge, TN for analysis and interpretation. Soil samples were analyzed by gamma spectrometry for Cs-137 and uranium. Spectra were also reviewed for other identifiable photopeaks. Soil samples were also analyzed by wet chemistry methods for Sr-90. Soil sample results are reported in units of pCi/g. Smear samples were analyzed for gross alpha and gross beta activity using a low background proportional counter. Smear sample results and direct measurement data were converted to units of disintegrations per minute per 100 cm^2 (dpm/100 cm^2).

FINDINGS AND RESULTS

DOCUMENT REVIEW

ESSAP's review of the SSFL decontamination and survey reports identified several procedural, analytical, and data findings where clarification would provide additional support that the sites have been adequately characterized and meet the requirements for release without radiological restrictions. These findings were provided in a June 5, 1992 correspondence.[5]

OCY AND T064 SIDE YARD

Surface Scans

Gamma surface scans of the OCY and T064 Side Yard identified three locations of elevated direct radiation, each measuring less than 1 m^2 in area, within the T064 Side Yard (Figure 6). All other gamma surface scans were within the range of ambient site background.

Radionuclide Concentration In Soil

Radionuclide concentrations in soil samples are summarized in Table 1. Concentrations in samples from two of the locations of elevated direct radiation that were individually sampled were: Cs-137, 35.1 and 210 pCi/g; Sr-90 <0.4 and 2.0 pCi/g; U-235, 0.3 pCi/g; U-238, 0.9 and 1.4 pCi/g. Concentrations in the composite samples, which represent the averages in 100 m^2 areas, were as follows: 0.6 to 27.7 pCi/g Cs-137; <0.5 to 1.9 pCi/g Sr-90; 0.1 to 0.4 pCi/g U-235; and 0.9 to 1.6 pCi/g U-238.

BUILDING T028

Surface Scans

Surface scans of the above-ground concrete slab, below-grade vault, and the stairwell for alpha, beta, and gamma activity did not identify any locations of elevated direct radiation.

Surface Activity Levels

Surface activity levels for Building T028 are summarized in Table 2. The average surface activity levels within surveyed 1 m^2 grid blocks were <83 dpm/100 cm^2 for alpha and ranged from <860 to 1200 dpm/100 cm^2 for beta. Individual direct measurements ranged from <83 to 89 for alpha and <860 to 1400 dpm/100 cm^2 for beta. Removable activity levels were <12 dpm/100 cm^2 for gross alpha and <15 to 25 dpm/100 cm^2 for gross beta.

COMPARISON OF RESULTS WITH GUIDELINES

Rockwell/Rocketdyne identified Cs-137 as the primary contaminant within the Building T064 Side Yard and the OCY and assumed that there was an equivalent concentration of Sr-90 present. Guidelines for these radionuclides are developed on a site-specific basis and Rockwell/Rocketdyne used the RESRAD computer code to determine both a two nuclide and a single nuclide limit for the Building T064 Side Yard and the OCY.[1] The two nuclide guideline limits developed were 60.4 pCi/g and 314 pCi/g each of Cs-137 and Sr-90 for the Building T064 Side Yard and the OCY, respectively.

ESSAP's soil sample analytical results were compared with these guidelines. Samples collected from the OCY verified the Rockwell/Rocketdyne results and conclusions regarding soil status relative to the guidelines. Samples from the Building T064 Side Yard indicated that small area "hot spots" were still present which exceeded the guideline. In addition, the assumption that equivalent concentrations of Sr-90 were present could not be verified.

Subsequent to ESSAP's findings, Rockwell/Rocketdyne remediated the hot spots and revised the Building T064 Side Yard guidelines to meet a more restricive 10 mrem/yr maximum dose rate for the residential scenario. The guidelines were a single nuclide (Cs-137) limit of 7.08 pCi/g average in a 100 m^2 area and a maximum concentration of 70.8 pCi/g in a 100 m^2 area. The final status guidelines and results were provided in a September 22, 1993 transmittal as Appendix F to the original 1990 report.[3] The data provided in the report indicated that the contaminated locations had been remediated to levels below the average guideline limit.

The site characterizations did not identify uranium as a contaminant; therefore, a guideline was not developed. Uranium concentrations in verification soil samples were comparable to the Rockwell/Rocketdyne determined average background levels of 0.7 pCi/g and 0.1 for U-238 and U-235, respectively.

Surface activity levels in Building T064 were compared to the guidelines for residual surface contamination for uranium which are:

<u>Total Activity</u>

5000 α dpm/100 cm^2, averaged over 1 m^2

15,000 α dpm/100 cm^2, maximum in a 100 cm^2 area

<u>Removable Activity</u>

1000 α dpm/100 cm^2

The more conservative 1:1 alpha to beta decay ratio of natural uranium was used to compare beta surface activity levels to the alpha guidelines. All of the ESSAP independent measurement data were well within these guideline levels.

ESSAP reviewed the SSFL exposure rate data for compliance with the DOE guideline of 20 μR/h above background. The site has chosen to use a more conservative criteria of 5 μR/h above background and exposure rates are, therefore, within the DOE guideline.

SUMMARY

At the request of the U.S. Department of Energy, the Environmental Survey and Site Assessment Program of the Oak Ridge Institute for Science and Education performed a verification survey of the Old Conservation Yard, Building T064 Side Yard, and Building T028 located at the Santa Susana Field Laboratory near Chatsworth, California. Activities included document reviews, surface scans, surface activity measurements, soil sampling, and sample analyses.

The documentation prepared by Rockwell/Rocketdyne provides descriptions of characterization, remediation, and post-remedial action survey procedures as well as the current radiological status of each area. ESSAP provided specific comments which suggest modifications to the current procedures and investigative approaches used at SSFL. If the suggested modifications are adapted by Rockwell/Rocketdyne on future projects, a more accurate and complete appraisal of the pre- and post-remedial action site conditions would be possible.

ESSAP's independent investigation supports Rockwell/Rocketdyne's field and analytical data for the Old Conservation Yard and following the additional remediation, the Building T064 Side Yard. ESSAP's independent measurement and sampling data for Building T028 were within the generic surface contamination DOE guidelines. It is, therefore, ESSAP's opinion that these areas meet the requirements for release to unrestricted use.

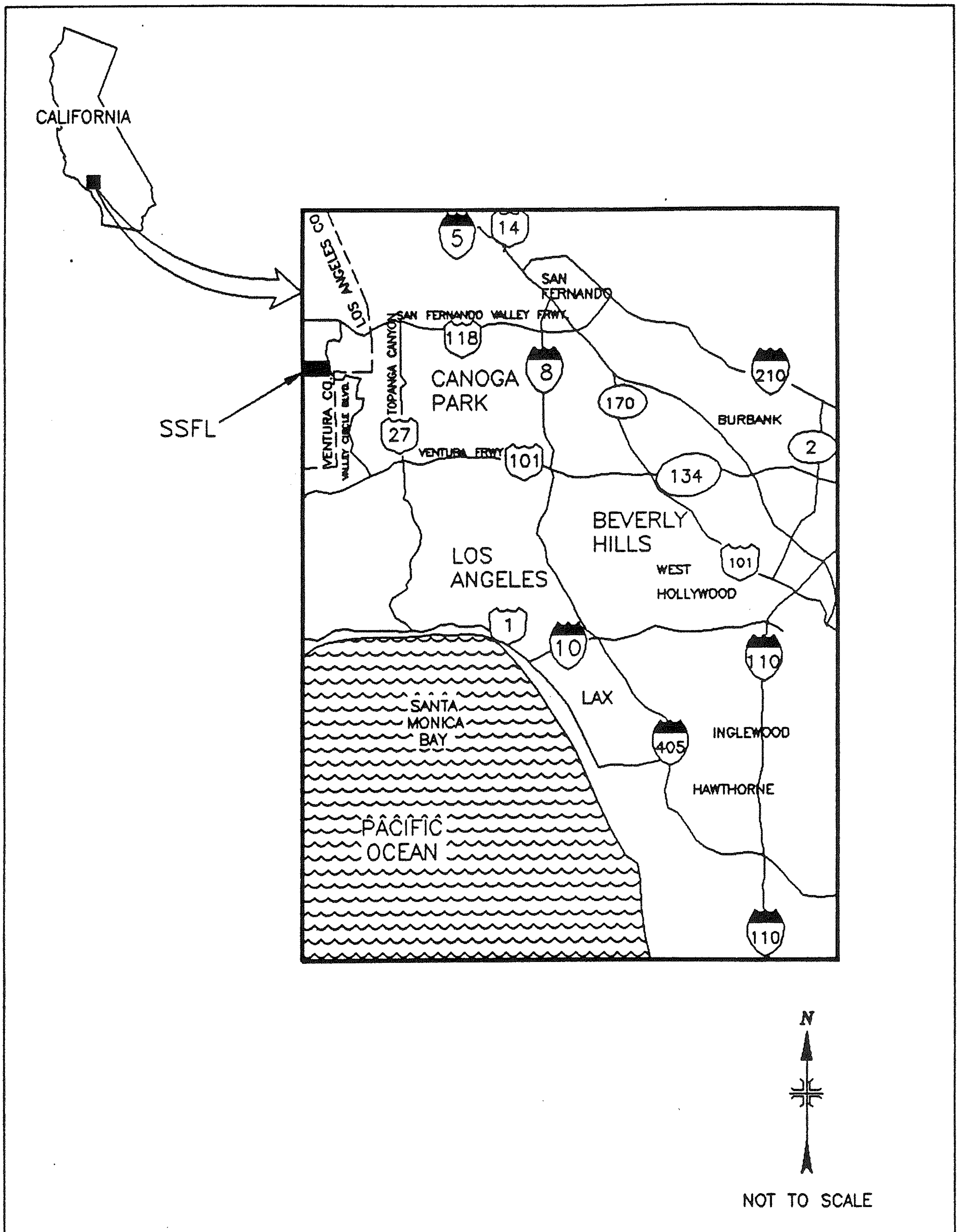

FIGURE 1: Los Angeles California Area, Location of Santa Susana Field Laboratory Site

SSL1

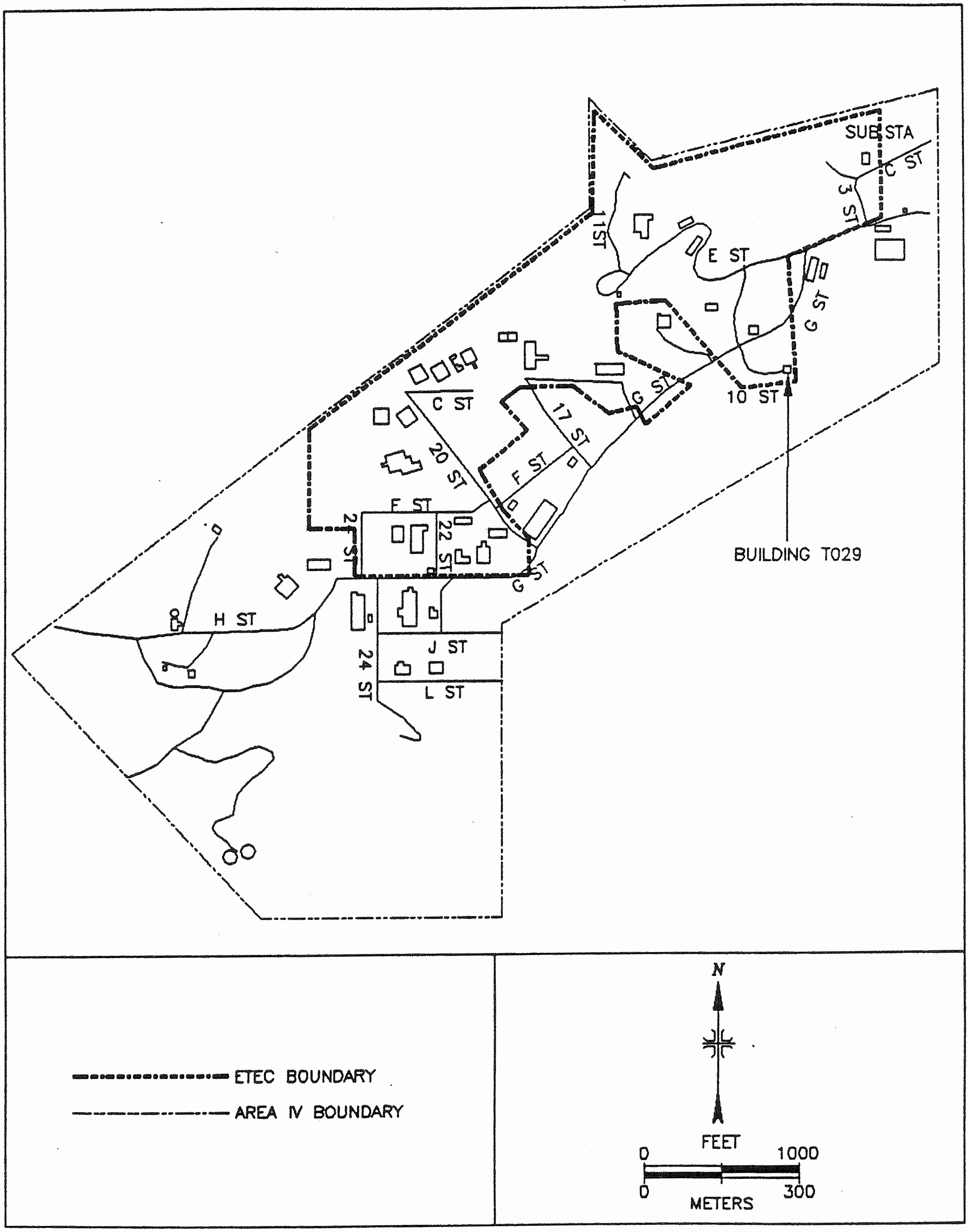

FIGURE 2: Plot Plan of Santa Susana Field Lab Area IV

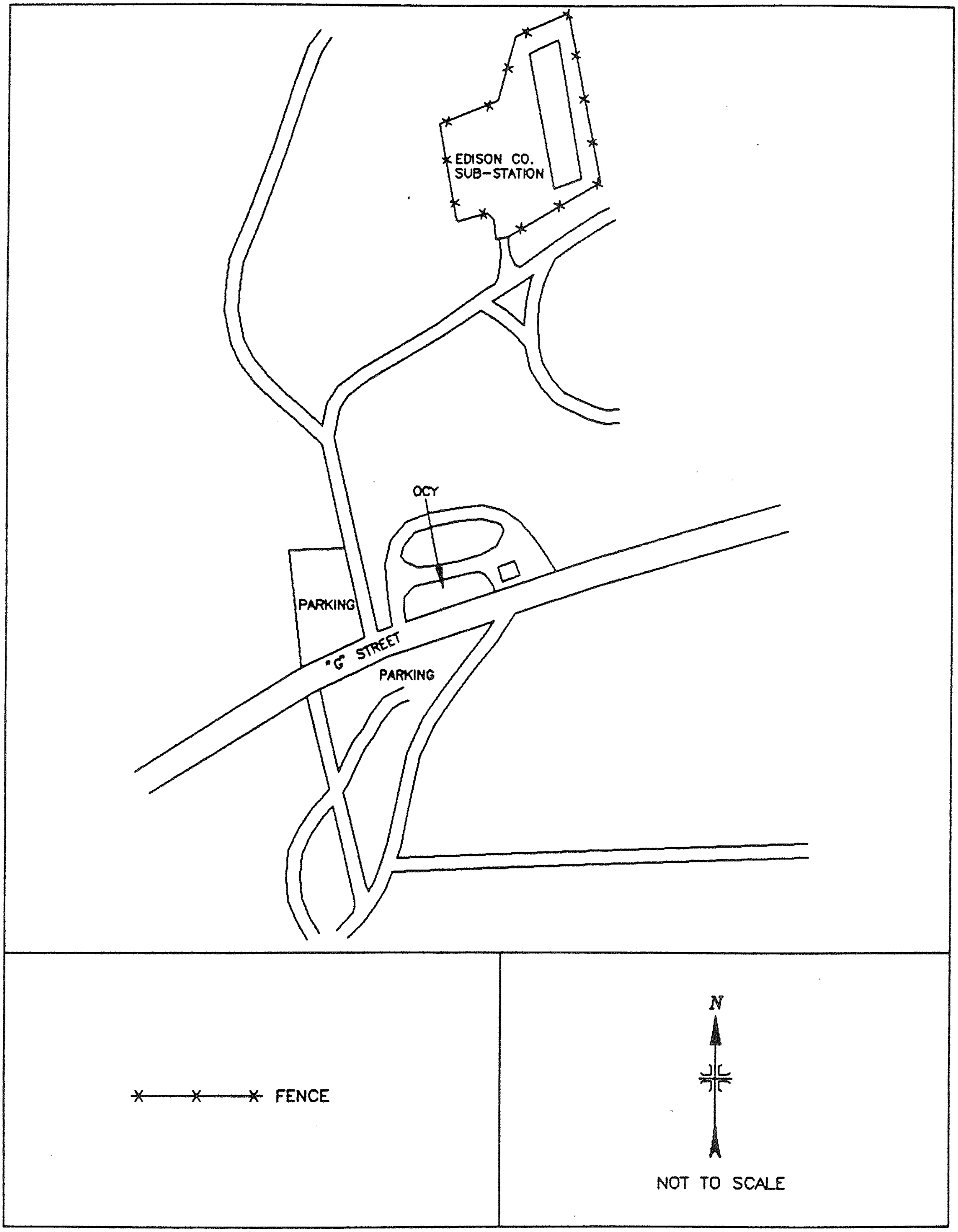

FIGURE 3: Location of the Old Conservation Yard

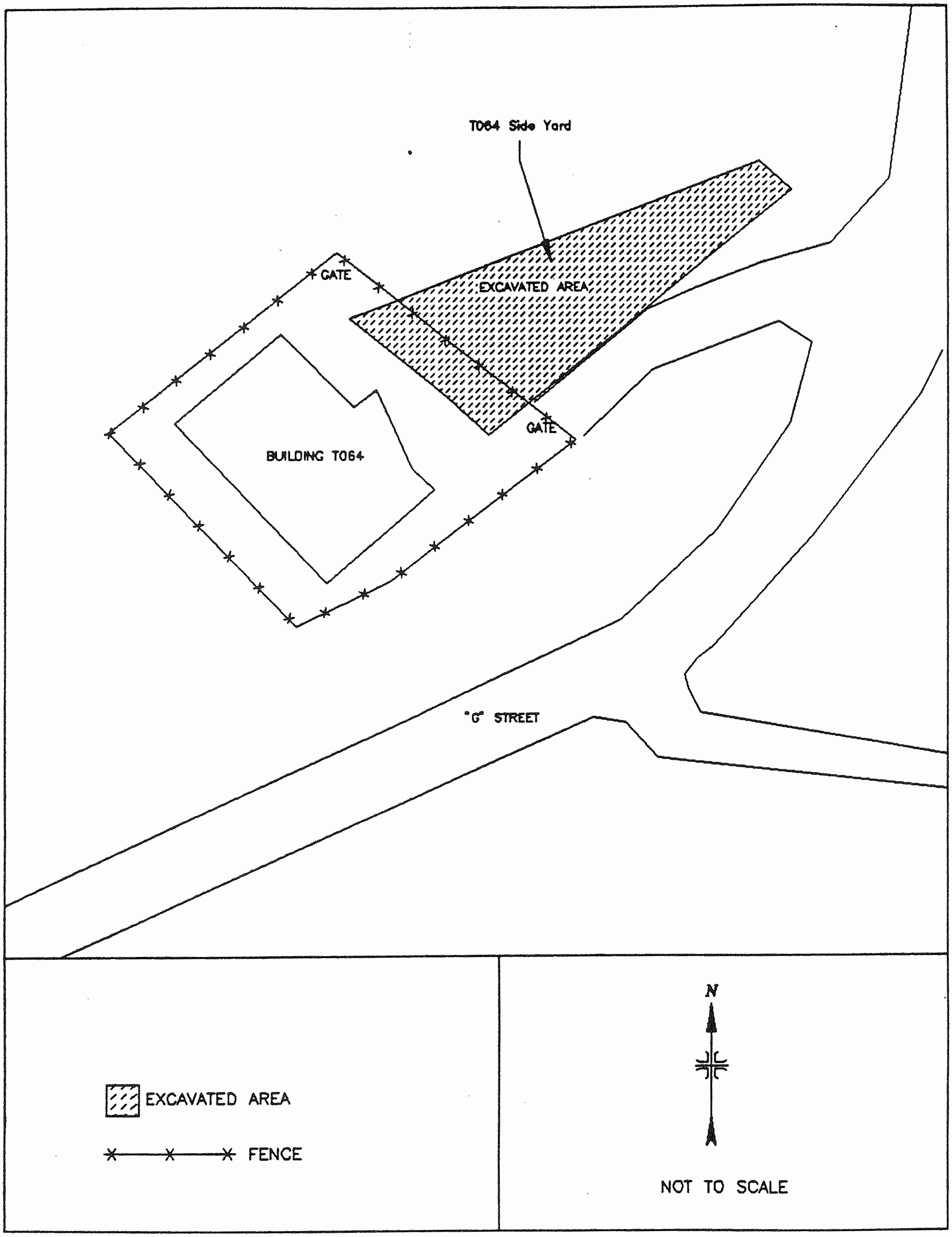

FIGURE 4: Location of Building T064 Side Yard

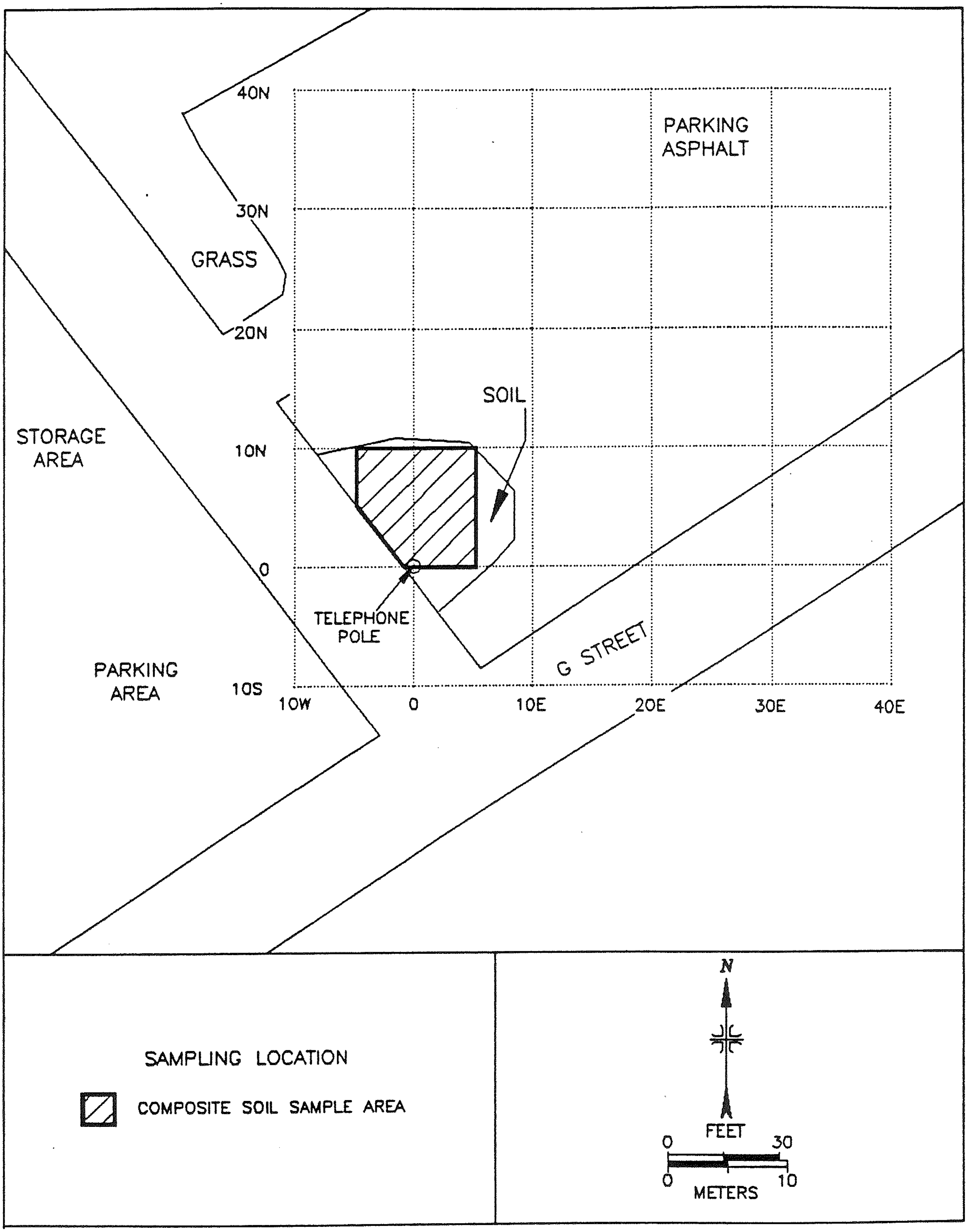

FIGURE 5: Old Conservation Yard – Reference Grid and Sampling Locations

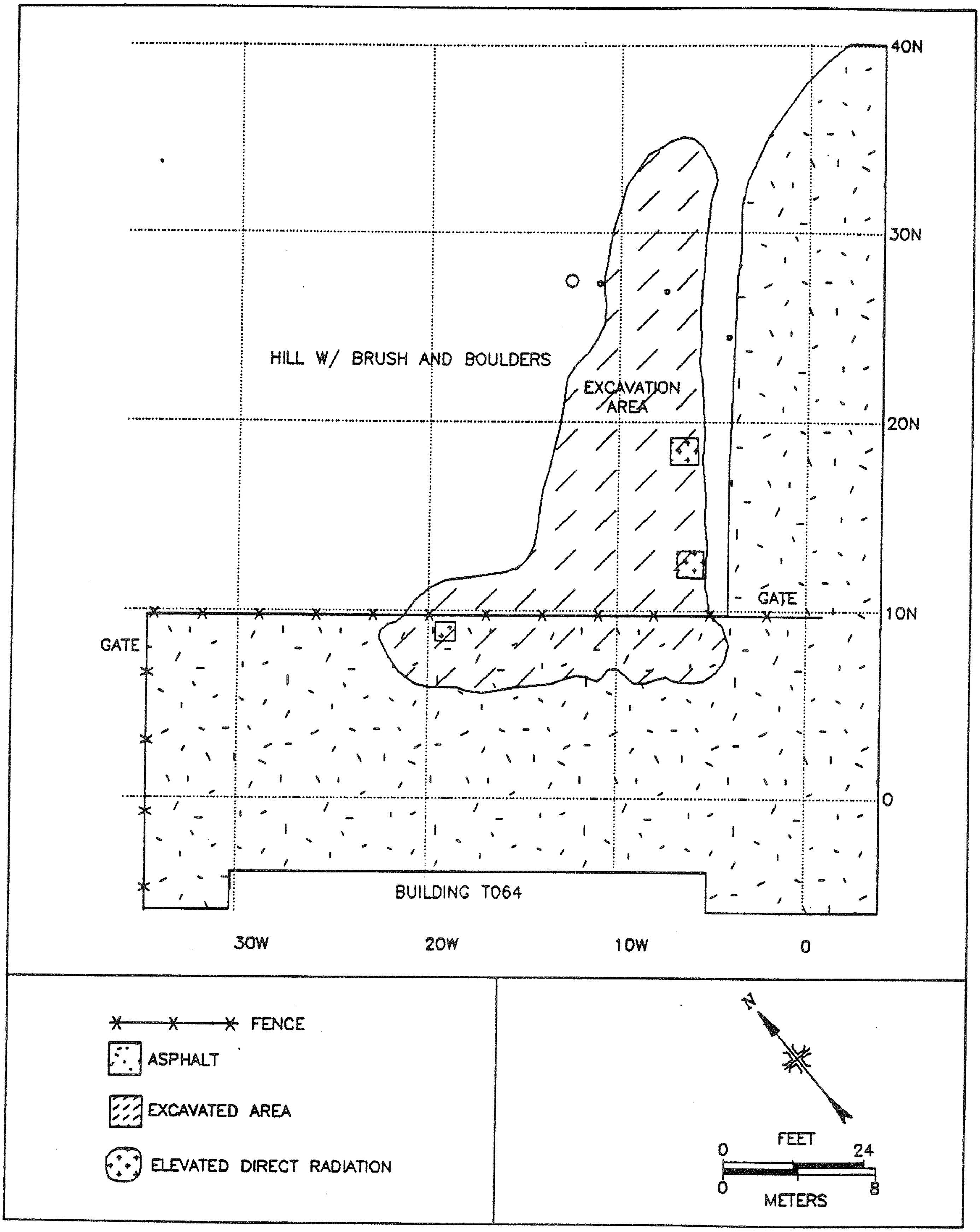

FIGURE 6: Building T064 Side Yard – Locations of Elevated Direct Radiation

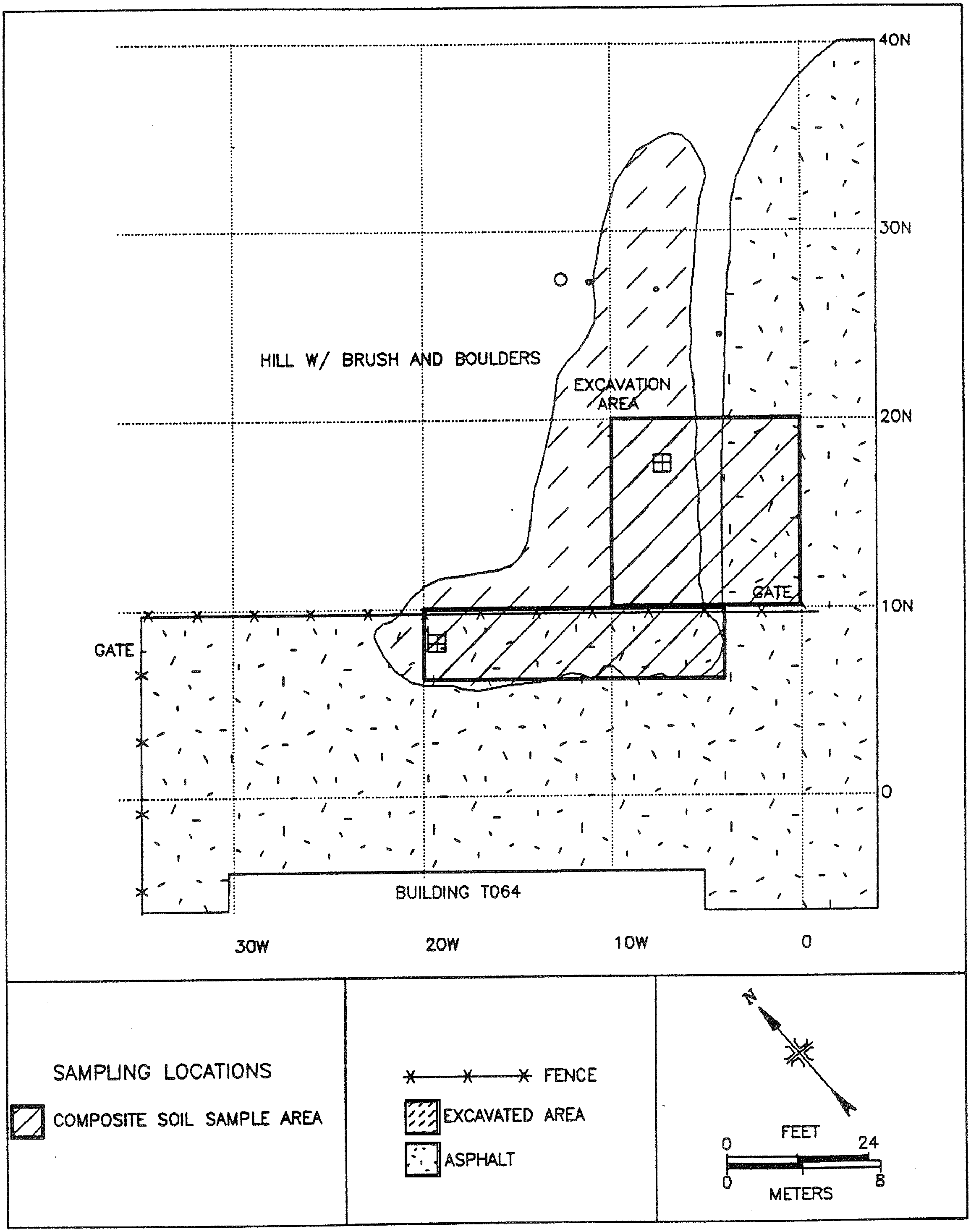

FIGURE 7: Building T064 Side Yard – Reference Grid and Measurement and Sampling Locations

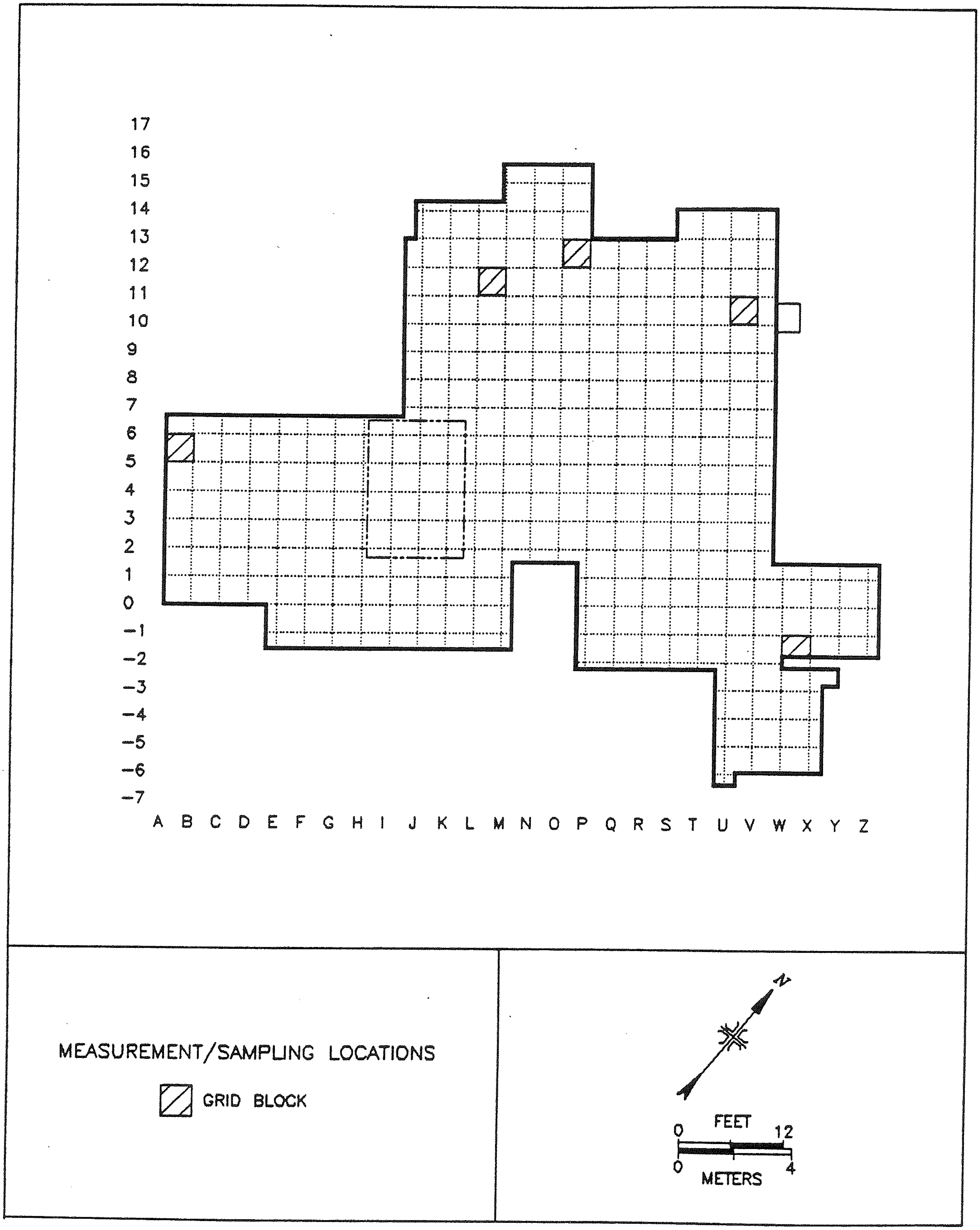

FIGURE 8: Building T028 Above Grade Pad — Reference Grid and Measurement and Sampling Locations

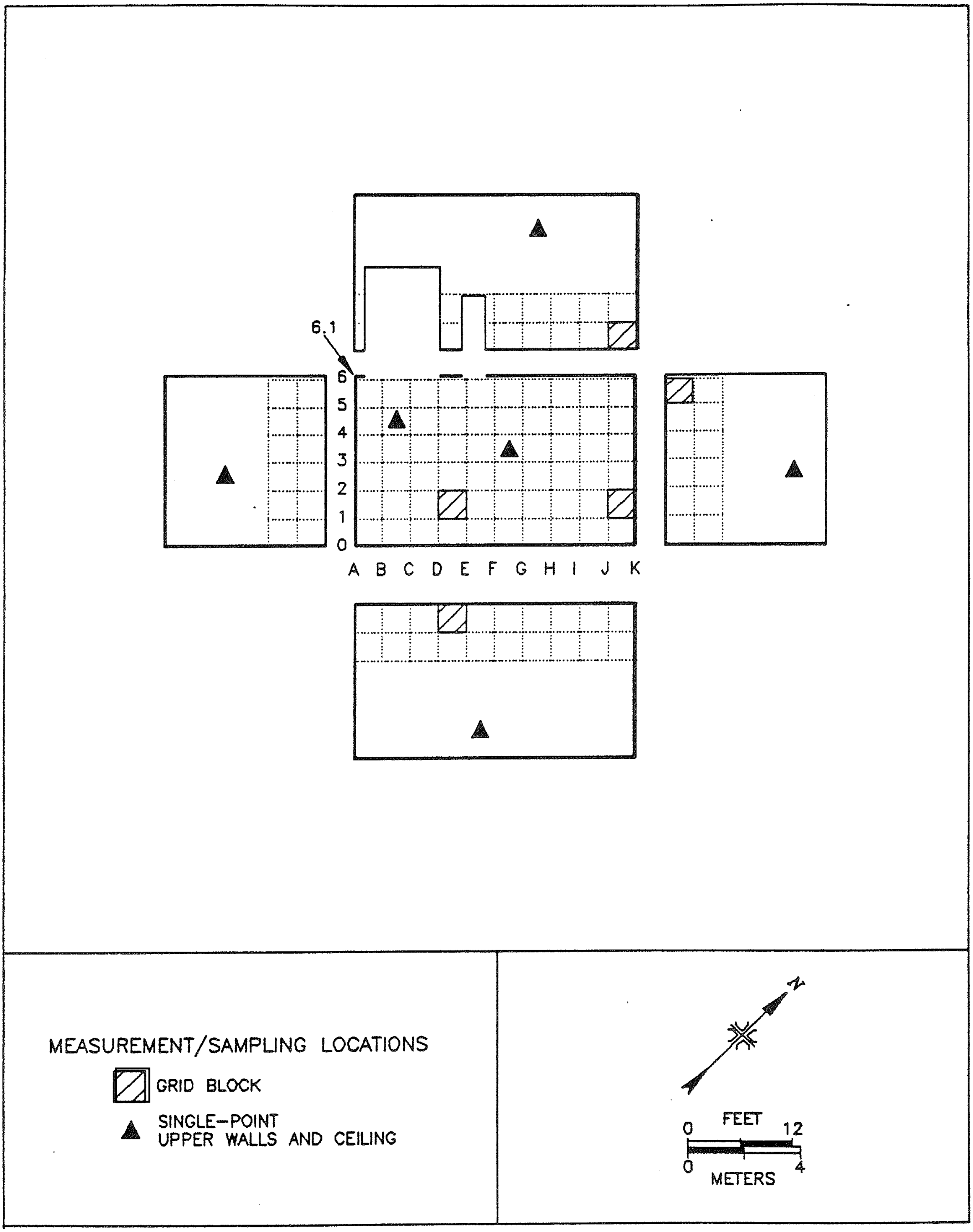

FIGURE 9: Building T028 Vault — Reference Grid and Measurement and Sampling Locations

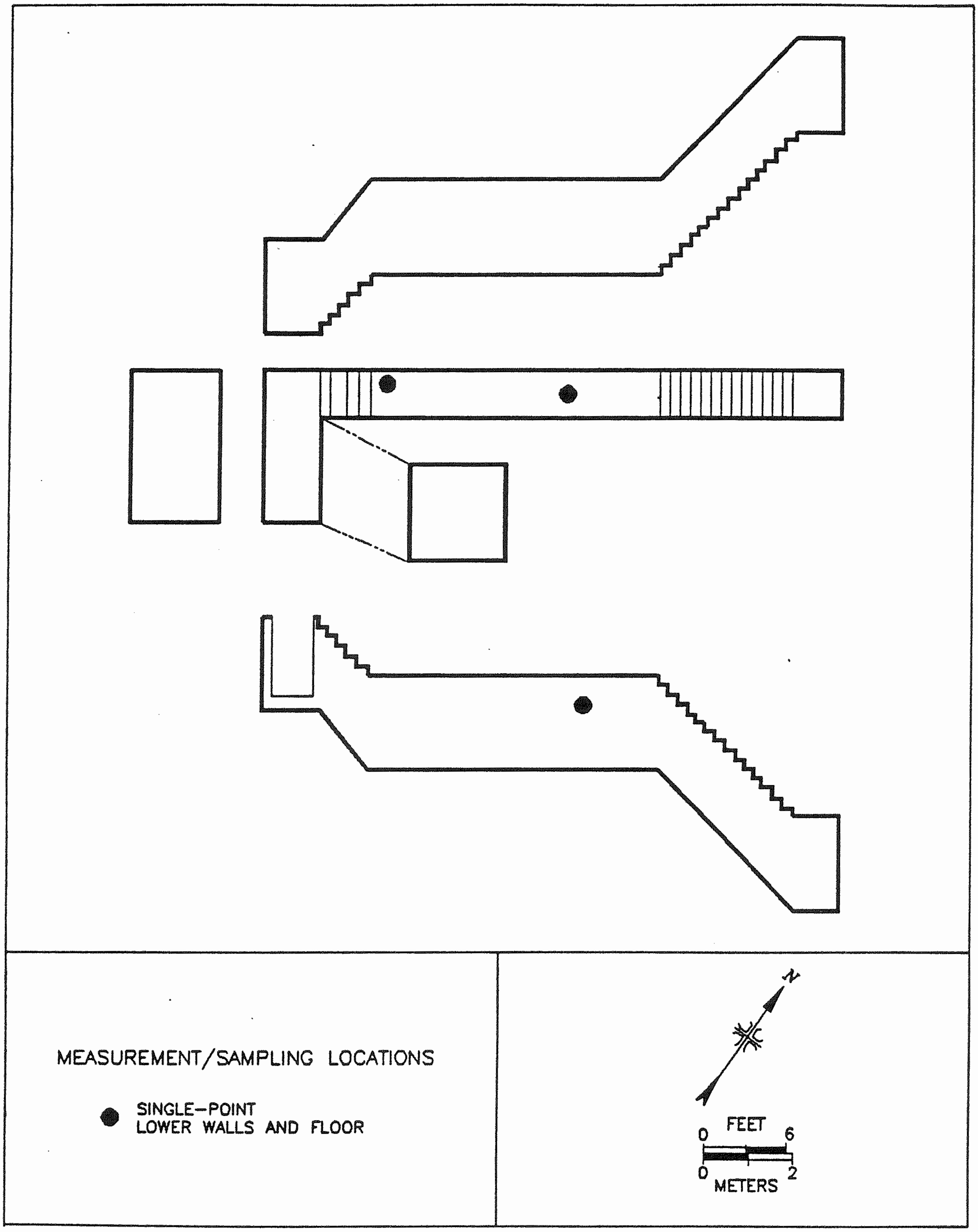

FIGURE 10: Building T028 Stairwell — Measurement and Sampling Locations

TABLE 1

RADIONUCLIDE CONCENTRATIONS IN SOIL
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
CANOGA PARK, CALIFORNIA

Location[a]	Radionuclide Concentration (pCi/g)			
	Cs-137	Sr-90	U-235	U-238
OCY				
0N, 5W[b]	0.6 ± 0.1[c]	<0.6	0.1 ± 0.1	1.4 ± 1.2
T064 Side Yard				
0N, 20W[b]	7.5 ± 0.9	<0.5	<0.2	1.6 ± 1.2
10N, 10W[b]	27.7 ± 3.1	1.9 ± 1.0	0.4 ± 0.2	1.2 ± 1.2
9N, 19.5W[d]	35.1 ± 3.9	<0.4	0.3 ± 0.2	0.9 ± 1.2
19.5N, 8.5W[d]	210 ± 23	2.0 ± 0.3	0.3 ± 0.3	1.4 ± 2.8

[a]Refer to Figures 5 and 6.
[b]Radionuclide concentration levels presented are the averages for a composite sample representing a 100 m^2 area and include "hot-spots".
[c]Uncertainties represent the 95% confidence level, based only on counting statistics.
[d]Radionuclide concentration levels presented are those for a single "hot-spot" location.

TABLE 2

SUMMARY OF SURFACE ACTIVITY LEVELS
BUILDING T028
SANTA SUSANA FIELD LABORATORY
ROCKWELL INTERNATIONAL
CANOGA PARK, CALIFORNIA

Location[a]	Number of Measurement Locations		Range of Total Activity (dpm/100 cm^2)				Range of Removable Activity (dpm/100 cm^2)	
			Single Measurement		Grid Block Average			
	Single Pt.	Grid Blocks	Alpha	Beta	Alpha	Beta	Alpha	Beta
Foundation	N/A	5	<83-89	<860-1400	<83	<860-1200	<12	<15
Vault, Floor and Lower Wall	N/A	5	<83	<990-1000	<83	<990	<12	<15-25
Vault, Upper Wall and Ceiling	6	N/A	<83	<990	N/A	N/A	<12	<15
Vault, Stairwell	3	N/A	<83	<990-1000	N/A	N/A	<12	<15

[a]Refer to Figures 7, 8, and 9.

REFERENCES

1. Argonne National Laboratory, "A Manual for Implementing Residual Radioactive Material Guidelines", DOE/CH/8901, June 1989.

2. Rockwell International, "Final Decontamination and Radiological Survey of the Old Conservation Yard," Document No. N704SRR990030, 1990.

3. Rockwell International, "Final Decontamination and Radiological Survey of the Building T064 Side Yard," Document No. N704SRR99031, 1990.

4. Rockwell International "Final Decontamination and Radiological Survey of Building T028," Document No. N704SRR990033, 1991.

5. Vitkus, T.J., Oak Ridge Associated Universities. Letter to Anthony Kluk, Ph.D, Director, San Francisco Operations, U.S. Department of Energy, June 5, 1992.

APPENDIX A

MAJOR INSTRUMENTATION

APPENDIX A

MAJOR INSTRUMENTATION

The display of a specific product is not to be construed as an endorsement of the product or its manufacturer by the author or their employers.

DIRECT RADIATION MEASUREMENT INSTRUMENTATION

Instruments

Eberline Pulse Ratemeter
Model PRM-6
(Eberline, Santa Fe, NM)

Eberline "Rascal" Ratemeter-Scaler
Model PRS-1
(Eberline, Santa Fe, NM)

Detectors

Eberline GM Detector
Model HP-260
Effective Area, 15.5 cm^2
(Eberline, Santa Fe, NM)

Eberline ZnS Scintillation Detector
Model AC-3-7
Effective Area, 59 cm^2
(Eberline, Santa Fe, NM)

Victoreen NaI Scintillation Detector
Model 489-55
3.2 cm x 3.8 cm Crystal
(Victoreen, Cleveland, OH)

LABORATORY ANALYTICAL INSTRUMENTATION

High-Purity Germanium Detector
Model GMX-23195-S, 23% Eff.
(EG&G ORTEC, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G-16
(Gamma Products, Palos Hills, IL) and
Multichannel Analyzer
3100 Vax Workstation
(Canberra, Meriden, CT)

Low Background Gas Proportional Counter
Model LB-5110
(Tennelec, Oak Ridge, TN)

APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

SURVEY PROCEDURES

Surface Scans

Surface scans were performed by passing the probes slowly over the surface; the distance between the probe and the surface was maintained at a minimum—nominally about 1 cm. Surfaces were scanned using portable gamma scintillation and small area (15.5 cm^2 or 59 cm^2) hand-held detectors. Identification of elevated levels was based on increases in the audible signal from the recording and/or indicating instrument. Combinations of detectors and instruments used for the scans were:

Alpha	-	ZnS scintillation detector with ratemeter-scaler
Beta	-	GM detector with ratemeter-scaler
Gamma	-	NaI scintillation detector with ratemeter

Surface Activity Measurements

Measurements of total alpha and total beta activity levels were performed using ZnS scintillation and GM detectors with ratemeter-scalers. Count rates (cpm), which were integrated over 1 minute in a static position, were converted to activity levels (dpm/100 cm^2) by dividing the net rate by the instrumentations 4 π efficiency, determined at calibration, and correcting for the active area of the detector. The alpha activity background countrates for the ZnS scintillation detectors averaged approximately 1 cpm for each detector. The alpha efficiency factor was 0.19 for the ZnS scintillation detectors. The beta activity background count rate for the GM detectors averaged 52 cpm. Beta efficiency factors ranged from 0.24 to 0.27 for the GM detectors. The effective windows for the ZnS scintillation and GM detectors were 59 cm^2 and 15.5 cm^2, respectively.

Removable Activity Measurements

Removable activity levels were determined using numbered filter paper disks, 47 mm in diameter. Moderate pressure was applied to the smear and approximately 100 cm^2 of the surface was wiped. Smears were placed in labeled envelopes with the location and other pertinent information recorded.

Soil Sampling

Approximately 1 kg of soil was collected at each sample location. Collected samples were placed in a plastic bag, sealed, and labeled in accordance with ESSAP survey procedures.

ANALYTICAL PROCEDURES

Removable Activity

Smears were counted on a low background gas proportional system for gross alpha and gross beta activity.

Gamma Spectrometry

Soil Samples

Samples of soil were dried, mixed, crushed, and/or homogenized as necessary, and a portion sealed in 0.5-liter Marinelli beaker or other appropriate container. The quantity placed in the beaker was chosen to reproduce the calibrated counting geometry and ranged from 800 to 900 g of material. Net material weights were determined and the samples counted using intrinsic germanium detectors coupled to a pulse height analyzer system. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyzer system. Energy peaks used for determination of radionuclides of concern were:

Cs-137	0.662 MeV
U-235	0.143 MeV (or 0.186 MeV)
U-238	0.063 and 0.093 MeV from Th-234*
	(or 1.001 MeV from Pa-234 ^{m})*

*Secular equilibrium assumed.

Spectra were also reviewed for other identifiable photopeaks.

Strontium-90

Soil Samples

Soil samples were dried, mixed, crushed and then aliquots of the soil were dissolved using a potassium fluoride pyrosulfate fusion in which strontium was precipitated as a sulfate. Successive treatments with EDTA preferentially removed lead and excess calcium and returned the strontium to solution. Ferric and other insoluble hydroxides were precipitated at a pH of 12 to 14. Strontium was reprecipitated as a sulfate and barium was removed as a chromate using DTPA. The final precipitate of strontium carbonate was counted using a low-background gas proportional counter and the activity calculated using an in-house algorithm specifically designed for strontium analyses.

UNCERTAINTIES AND DETECTION LIMITS

The uncertainties associated with the analytical data presented in the tables of this report represent the 95% confidence level for that data. These uncertainties were calculated based on both the gross sample count levels and the associated background count levels. When the net sample count was less than 95% statistical deviation of the background count, the sample concentration was reported as less than the detection limit of the measurement procedures. Because of variations in background levels, measurement efficiencies, and contributions from other radionuclides in samples, the detection limits differ from sample to sample and instrument to instrument. Additional uncertainties, associated with sampling and measurement procedures, have not been propagated into the data presented in this report.

CALIBRATION AND QUALITY ASSURANCE

Analytical and field survey activities were conducted in accordance with procedures from the following documents:

-Survey Procedures Manual Revision 7 (June 1992)
-Laboratory Procedures Manual Revision 6 (April 1991)
-Quality Assurance Manual Revision 5 (June 1992)

The procedures contained in these manuals were developed to meet the requirements of DOE Order 5700.6B and 5700.6C for Quality Assurance and contain measures to assess processes during their performance.

Calibration of all field and laboratory instrumentation was based on standards/sources, traceable to NIST, when such standards/sources were available. In cases where they were not available, standards of an industry recognized organization were used.

Quality control procedures include:

- Daily instrument background and check-source measurements to confirm that equipment operation is within acceptable fluctuations.
- Participation in EPA and EML laboratory Quality Assurance Programs.
- Training and certification of all individuals performing procedures.
- Periodic internal and external audits.

APPENDIX C

RESIDUAL RADIOACTIVE MATERIAL GUIDELINES SUMMARIZED FROM DOE ORDER 5400.5

BASIC DOSE LIMITS

The basic limit for the annual radiation dose (excluding radon) received by an individual member of the general public is 100 mrem/yr. In implementing this limit, DOE applies as low as reasonable achievable principles to set site-specific guidelines.

STRUCTURE GUIDELINES

Surface Contamination Guidelines

	Allowable Total Residual Surface Contamination (dpm/100 cm^2)[1]		
Radionuclides[2]	Average[3,4]	Maximum[4,5]	Removable[4,6]
Transuranics, I-125, I-129, Ra-226, Ac-227, Ra-228, Th-228, Th-230, Pa-231	Reserved	Reserved	Reserved
Th-Natural, Sr-90, I-126, I-131, I-133, Ra-223, Ra-224, U-232, Th-232	1,000	3,000	200
U-Natural, U-235, U-238, and associated decay product, alpha emitters	5,000α	15,000α	1,000α
Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.[7]	5,000β-γ	15,000β-γ	1,000β-γ

External Gamma Radiation

The average level of gamma radiation inside a building or habitable structure on a site that has no radiological restriction on its use shall not exceed the background level by more than 20 μR/h and will comply with the basic dose limits when an appropriate-use scenario is considered.

SOIL GUIDELINES

Radionuclides	Soil Concentration (pCi/g) Above Background[8,9]
Cs-137 and Sr-90	Soil guidelines are calculated on a site-specific basis, using the DOE manual developed for this use.

[1] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[2] Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

[3] Measurements of average contamination should not be averaged over an area of more than 1 m^2. For objects of less surface area, the average should be derived for each such object.

[4] The average and maximum dose rates associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h and 1.0 mrad/h, respectively, at 1 cm.

[5] The maximum contamination level applies to an area of not more than 100 cm^2.

[6] The amount of removable material per 100 cm^2 of surface area should be determined by wiping an area of that size with dry filter or soft absorbent paper, applying moderate pressure, and measuring the amount of radioactive material on the wiping with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area should be based on the actual area and the entire surface should be wiped. It is not necessary to use wiping techniques to measure removable contamination levels if direct scan surveys indicate that total residual surface contamination levels are within the limits for removable contamination.

APPENDIX C

RESIDUAL RADIOACTIVE MATERIAL GUIDELINES SUMMARIZED FROM DOE ORDER 5400.5

[7] This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90, which has been separated from the other fission products, or mixtures where the Sr-90 has been enriched.

[8] These guidelines represent allowable residual concentrations above background averaged across any 15-cm-thick layer to any depth and over any contiguous 100 m^2 surface area.

[9] If the average concentration in any surface or below-surface area, less than or equal to 25 m^2, exceeds the authorized limit of guideline by a factor of $(100/A)^{1/2}$, where A is the area or the elevated region in square meters, limits for "hot spots" shall also be applicable. Procedures for calculating these hot spot limits, which depend on the extent of the elevated local concentrations, are given in the DOE Manual for Implementing Residual Radioactive Materials Guidelines, DOE/CH/8901. In addition, every reasonable effort shall be made to remove any source of radionuclide that exceeds 30 times the appropriate limit for soil, irrespective of the average concentration in the soil.

EXHIBIT IV

BUILDING 028 AND STIR FACILITY FINAL REPORTS

ENERGY TECHNOLOGY ENGINEERING CENTER	No. 028-AR-0001 Rev. ____
OPERATED FOR THE U.S. DEPARTMENT OF ENERGY	Page 1 of 20
ROCKETDYNE DIVISION, ROCKWELL INTERNATIONAL	Orig. Date 03-18-96
	Rev. Date ____

FINAL REPORT

TITLE: BUILDING 028 AND STIR FACILITY DECONTAMINATION AND DECOMMISSIONING

- APPROVALS -

Originator	A. L. Pascolla	Fac. Mgr.	P. H. Horton
		Prog. Mgr.	M. E. Jensen
		Environ.	R. Rutherford Phil Rutherford
		H&S	J. Erman C. Fontanella for J. Erman
		QA	S. Reeder

REV. LTR.	REVISION	APPROVAL/DATE

Form 735-A-6 Rev 6-91

1022-0007

CONTENTS

TABLES

FIGURES

1. INTRODUCTION AND BACKGROUND

1.1 LOCATION

Building T028 is located within Rockwell International's Santa Susana Field Laboratory (SSFL) in the Simi Hills of southeastern Ventura County, California, adjacent to the Los Angeles County Line and approximately 29 miles northwest of downtown Los Angeles. Location of the SSFL relative to Los Angeles and vicinity is shown in Figure 1-1. An enlarged map of neighboring SSFL communities is shown in Figure 1-2. Figure 1-3 is a SSFL layout showing location of Building T028. A drawing (plan view) of Building T028, as it existed prior to above-grade demolition, is shown in Figure 1-4.

Figure 1-5 shows the above grade portion of Building T028 after demolition. Using USGS terminology, the description for Building T028 is: Section 25 of Township T2N: Range R18W: Calabasas Quadrangle.

1.2 AREA CHARACTERISTICS

Figure 1-6 shows the remaining below-grade structure, consisting of the original test vault area.

The terrain throughout most of the SSFL areas is uneven due to rock outcroppings. Rock outcroppings are prevalent east upslope from the facility to the north, and to the south and west. Water runoff is primarily to the west at the western end of the facility. Surrounding the facility in all directions is asphalt paving. The minimum distance to the SSFL boundary is approximately 300 ft. This boundary lies in a northeasterly direction (Simi Valley direction). Grade floor elevation is approximately 1,800 ft above sea level.

1.3 OPERATING HISTORY

Building T028 was originally constructed to perform tests of space reactor shields using a fission plate driven by neutrons from the thermal column of a 50-kW swimming pool-type reactor. This reactor was designated the Shield Test Reactor (STR) and operated from 1961 to 1964, when it was replaced with another reactor design to operate at 1 MW. This latter configuration was named the Shield Test and Irradiation Reactor (STIR) and operated through 1972. Following shutdown of the test program and removal of the reactor, the facility was decommissioned and made available for alternate use in March 1976 (Ref. 1).

In 1977, operations were started to investigate the behavior of molten UO_2, relative to simulated reactor accidents, in particular, its reaction with floor and structural materials. These experiments resulted in some recontamination of various parts of the building that were used for the preparation and the melting of the UO_2. Tests continued intermittently into 1981. Some facility modifications were made after that, and a decision to terminate operations was made later in 1981. The building remained inactive, under periodic surveillance, until 1988 when cleanout and decontamination began.

In April 1989, it was determined that there was no remaining radioactive contamination in the above-grade portion of the building and that part of the structure was demolished. Only the concrete floor and the below-grade test vault and stairway currently remain.

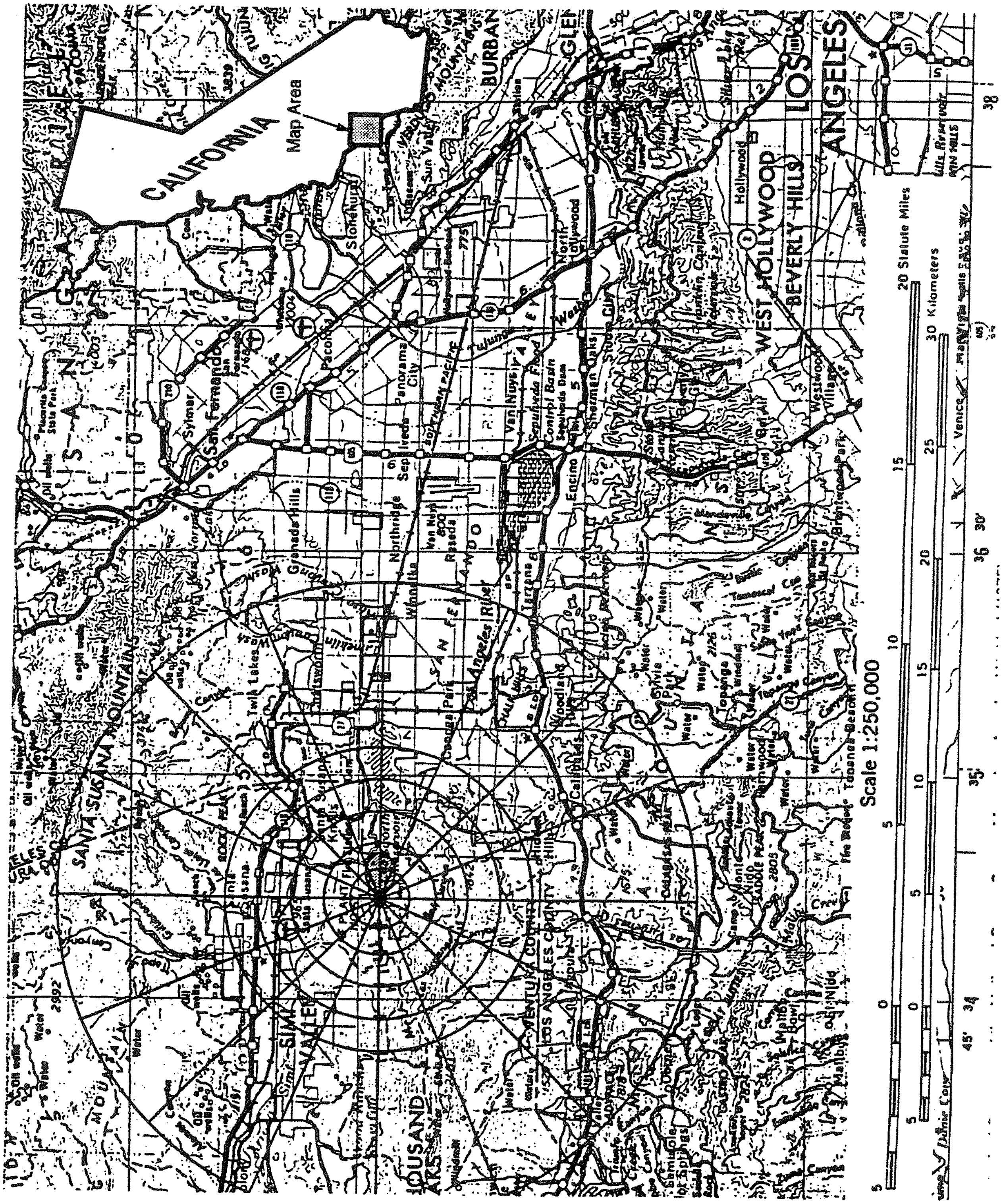

Figure 1-1. Map of Los Angeles Area

Figure 1-2. Map of Neighboring SSFL Communities

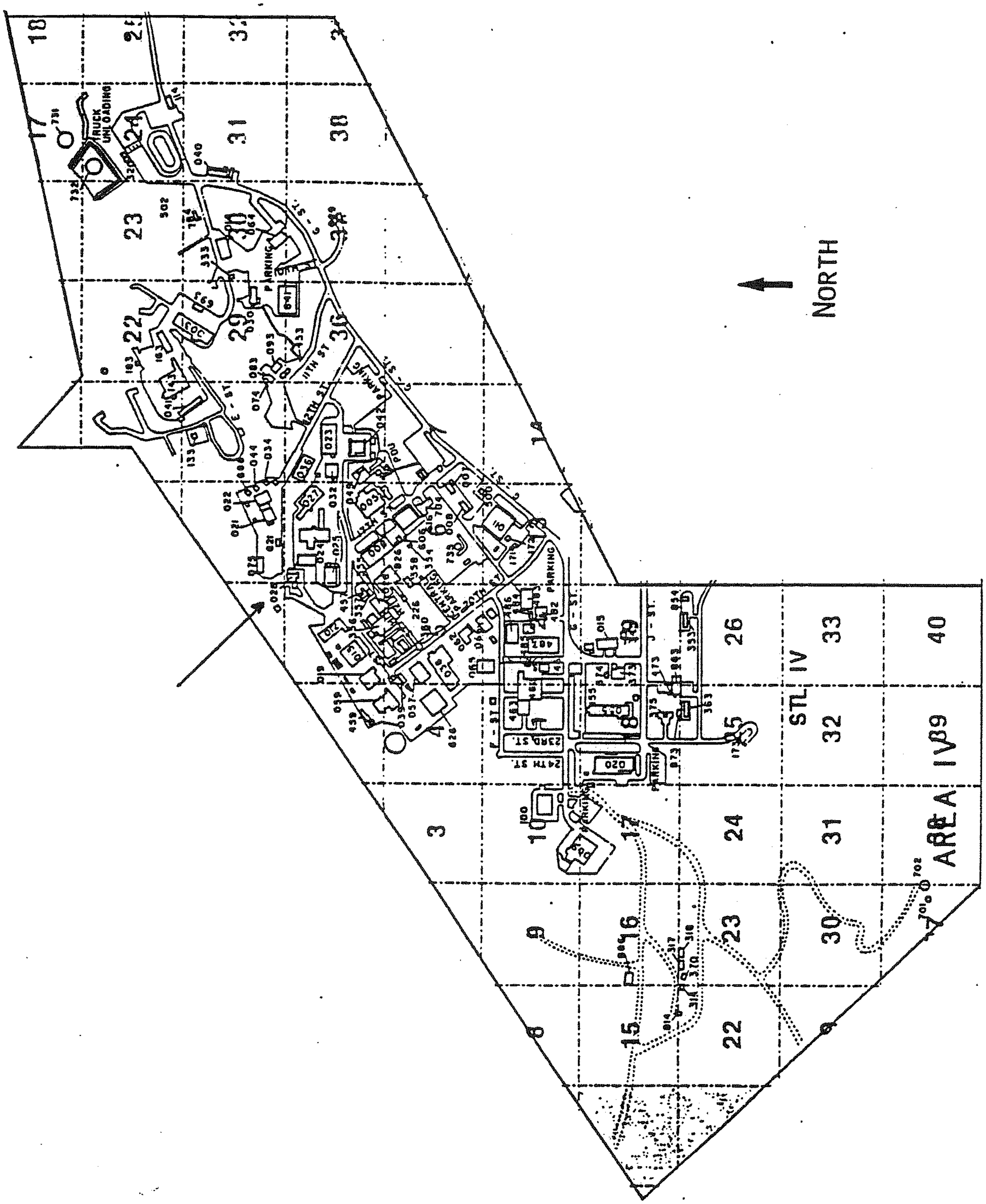

Figure 1-3. SSFL Layout Showing Location of Building T028

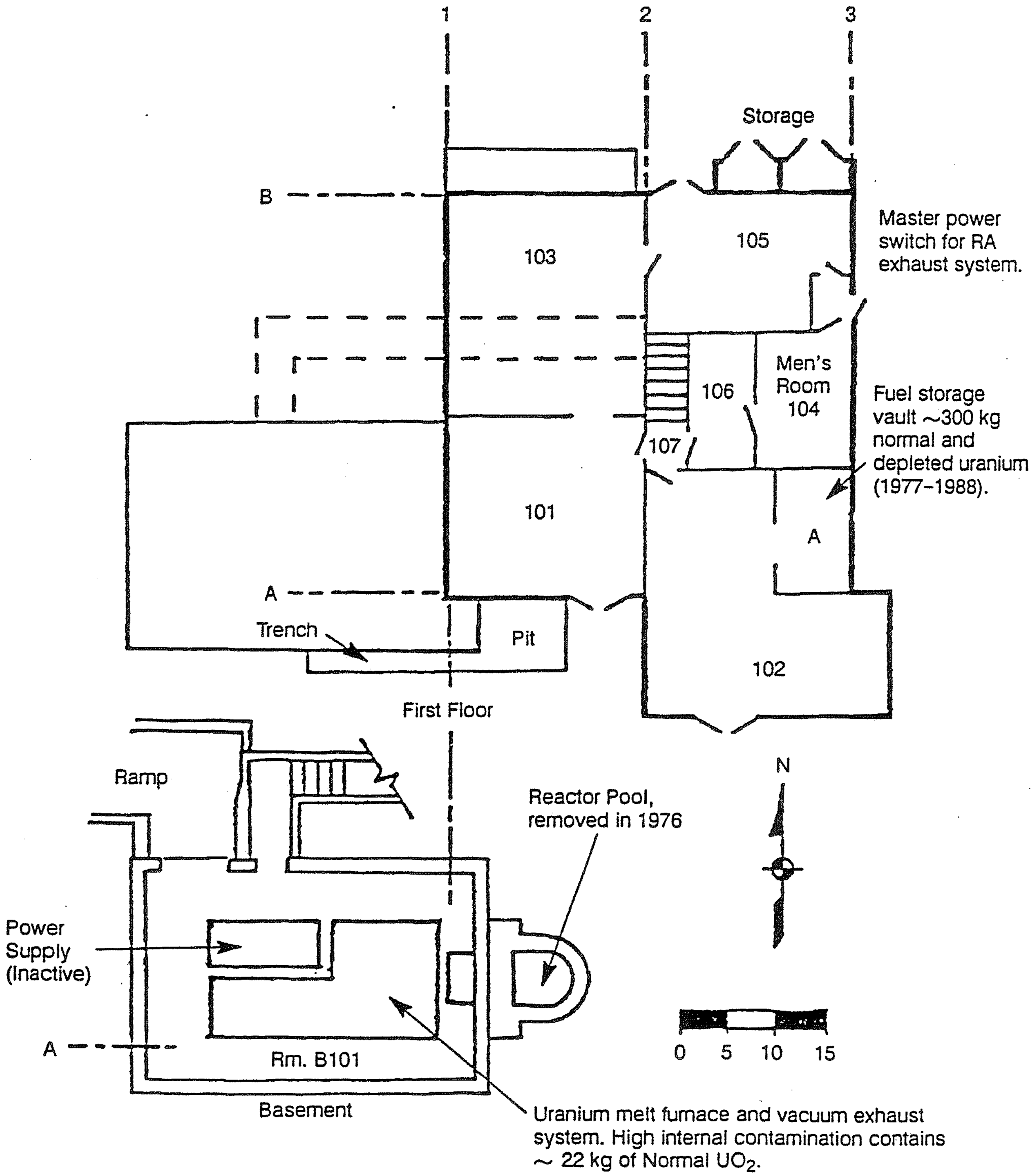

Figure 1-4. Plan View of Building T028 Prior to Decontamination and Above-Grade Demolition (1977-1988)

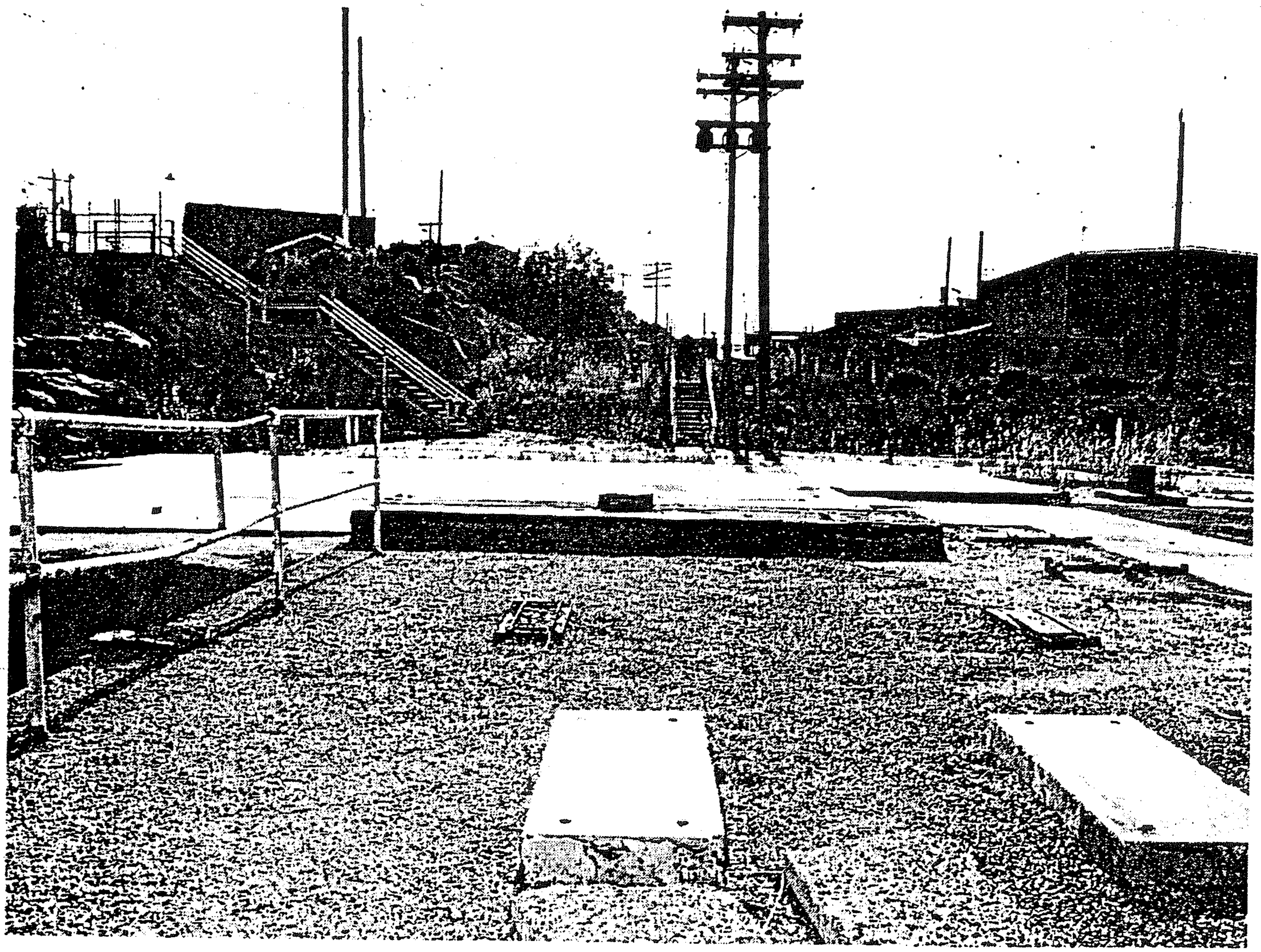

Figure 1-5. Above-Grade Portion of Building T028 After Demolition

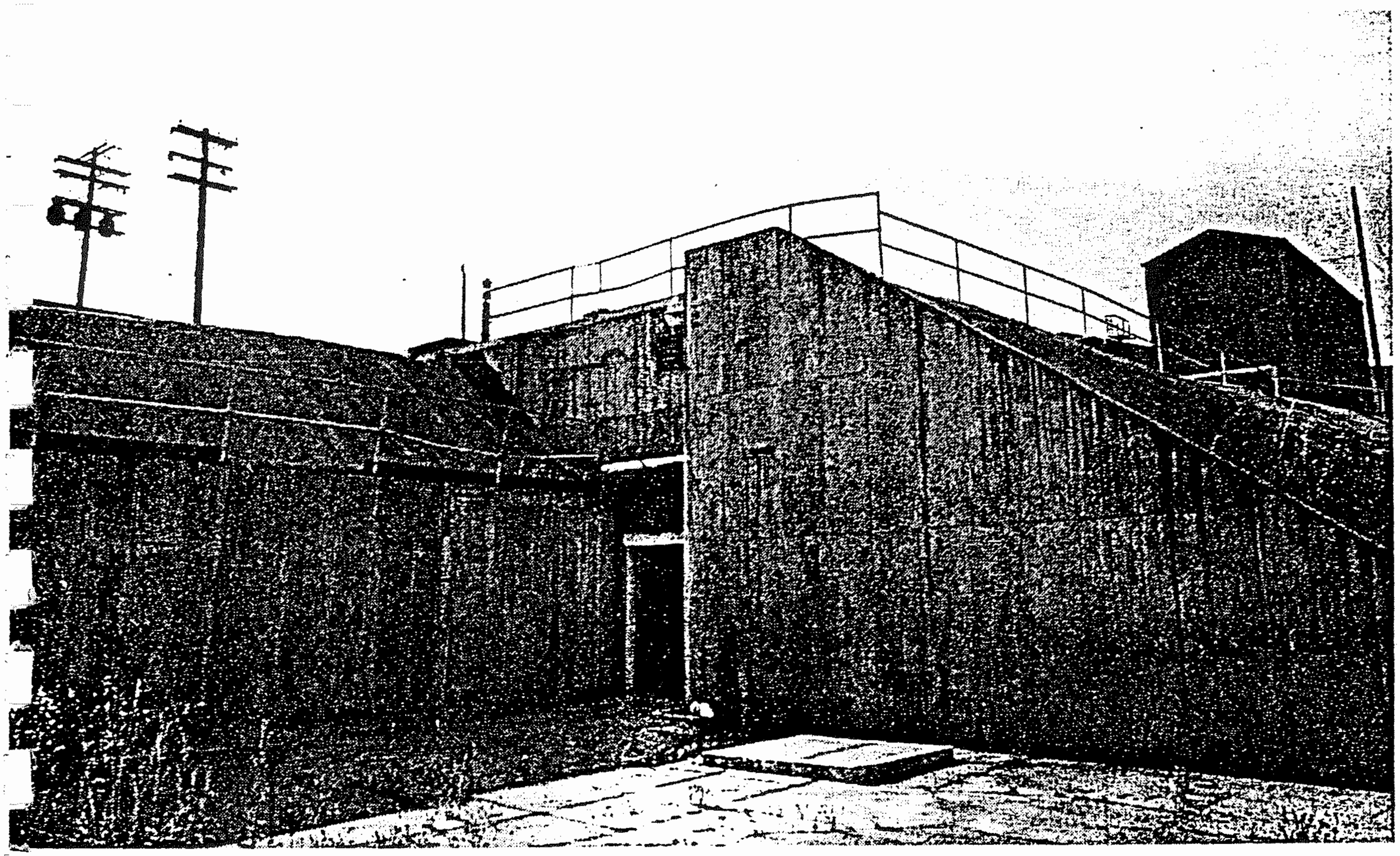

Figure 1-6. Existing Below-Grade Portion of Building T028

2. PRIOR DECONTAMINATION EFFORTS

The Shield Test Irradiation Reactor (STIR) facility was declared excess, and the dismantling proceeded as described in the "Decontamination and Disposition (D&D) of Facilities Program Plan," PP-704-990-002 (Ref. 8). The dismantling of STIR was estimated to begin on October 1, 1975 and be completed on October 1, 1976 (12-month time frame). The actual dismantling of STIR began on September 24, 1975 and was completed March 26, 1976. The fuel elements were removed, and the pool water was drained in June 1973. Contaminated and irradiated components and structures associated with the reactor, water cooling system, thermal column, test carriage, and facility exhaust system were removed, packaged, and shipped to Beatty, Nevada for disposal by land burial. Nonradioactive peripheral equipment such as the cooling tower, shield door, and film conveyor were removed as salvage. Floor and wall openings resulting from the D&D operations were filled and covered with concrete. This was required to restore the facility to a safe condition.

3. SUMMARY

3.1 STIR FACILITY

The decontamination and disposition (D&D) of Building 028, STIR facilities, are complete. The core tank, the activated concrete structures surrounding the core tanks, the thermal column, the reactor shield, the test vault carriage, the water cooling systems, and the water shield door were removed, and the facility exhaust system was partially dismantled. The facilities were decontaminated to levels which were as low as practicable, but in all cases to levels below the limits described as acceptable for future unrestricted use. The more significant D&D activities are summarized, and special techniques are noted in Section 4.0. Results of the radiological monitoring in support of the D&D operations and of the final radiological survey are presented in (Ref. 1)

3.2 BUILDING T028

The overall schedule for the D&D of Building T028 facility was estimated to require 6 months, excluding the demolition. The actual time required was slightly less than 5 months (July through December 1988), including disposal of an unexpected amount of oil found within the vacuum systems. The demolition required 3 months and was completed by mid July 1989.

Briefly, the D&D steps involved were (1) removal of surplus normal and depleted uranium oxide; (2) decontamination and removal of equipment and electrical components, including the furnace system used for the uranium-oxide experiments; (3) removal of the radioactive ducting system; (4) building surfaces decontamination, including scabbling of Room 101A concrete floor; (5) final miscellaneous cleanup operations; and (6) final radiological survey of the T028 building facility (above-grade and basement).

Following analysis of the final radiological survey data, which showed no residual radionuclide contamination above acceptable levels (Ref. 9), the building was released to Taylor Wrecking Co. for demolition and removal of the above-grade structures. The structure demolition and removal work was completed in July 1989.

All radioactive waste from the facility D&D was sent to the RMDF for packaging and shipment to Hanford, Washington. A total of about 1,200 ft^3 of waste was shipped to Hanford.

At the request of the U.S. Department of Energy (DOE), the Environmental Survey and Site Assessment Program (ESSAP) of the Oak Ridge Institute for Science and Education performed a verification survey of Building T028. Activities included document reviews, surface scans, surface activity measurements, soil sampling, and sample analyses.

ESSAP's independent measurement and sampling data for Building T028 were within the generic surface contamination DOE guidelines. It is, therefore, ESSAP's opinion that these areas meet the requirements for release to unrestricted use.

4. PROJECT ACTIVITIES AND RESULTS

All of the activities discussed below were performed in accordance with approved, written procedures. The procedures employed are cited in references 4,6 and 10 and presented in Table 4-1. The details of the day-by-day activities, identification of crews, and other information are contained in the operational log book titled "Building T028 Decontamination — August 1988," which is located in the Atomics International Library as R001410. Copies of the Health and Safety Analysis Reports citing the activity levels of pallets 1, 2, 3, and 5 prior to shipment from Building T028, and electrical equipment prior to wire removal are contained in Appendix A, Ref. 10.

Table 4-1. Building T028 Decontamination, Decommissioning, and Demolition Procedures

1.	173DWP000010, Structural Surfaces Decontamination, Rev New. Revised 8/11/88 for use at T028.
2.	173DWP000019, Known and Suspect Contaminated Support Areas Decontamination, Rev New. Revised 8/11/88 for use at T028.
3.	N001OP160007, Decontamination and Size Reduction of Low Level R/A Materials, Rev New. Approved for use 8/11/88.
4.	4173DWP000020, radioactive Waste Handling Procedure, Rev New. Revised for use at Building T028, and approved 8/11/88.
5.	N704DWP990082, High Volume Exhaust Removal, Rev A. Revised for use at Building T028 and approved 8/11/88.
6.	N001DWP000019, Size Reduction and Removal of Vacuum Furnace System, Rev New. Approved for use 8/3/88.
7.	173DWP000021, Bldg T028 Radiological Survey Procedure, Rev New. Revised for use at T028 and approved 8/11/88.
8.	094QAP-00, Inspection Requirements for the Shipment of Radioactive Materials, Rev E, Approved 8/11/88.
9.	089QPP000001, Radioactive Material Packaging and Shipping Quality Assurance Program Plan, Rev A, Approved 8/11/88.
10.	N704DWP990094, Solidification of TRU-Contaminated Oil, Rev New. Approved 10/4/88.

4.1 SURPLUS URANIUM OXIDE DISPOSAL

The surplus uranium oxide was assembled, packaged and palletized for disposal. The total inventory removed was 278,671 gm of normal uranium oxide and 22,405 gm of depleted uranium oxide as detailed in Appendix D, Ref. 10. The work was performed from July 14, 1988 to August 1, 1988. This material was shipped to Hanford, Washington as radioactive waste.

4.2 EQUIPMENT DECONTAMINATION

Equipment, piping, hardware and electrical components were disconnected, disassembled and packaged for disposal. The health physicist monitored the waste continuously as it was being removed and packaged for shipment to salvage if clean, and to the Radioactive Material Disposal Facility (RMDF) if contaminated. This activity encompassed both rooms 102A and B-101. This effort was performed from August 1, 1988 to August 19, 1988.

4.3 BUILDING SURFACES DECONTAMINATION

Room 101A concrete floor was scabbled and the walls were decontaminated over the period of August 22, 1988 through August 24, 1988. Radiological release surveys showed the room to be acceptable for release.

4.4 FILTER SYSTEM DECONTAMINATION, REMOVAL AND DISPOSAL

Removal of radioactive ducting began with the attic and continued through Room 102A, the change room and the rest room during the period from August 25, 1988 through August 30, 1988. The effort was stopped while the furnace and appurtenances were examined and work started to achieve the disposal site's schedule target for furnace shipment. It was necessary to repair and have the radioactive filter system operational for the furnace cleanup and removal work. Following removal of the furnace the remaining ventilation ducting was removed. This activity was performed over the period of October 10, 1988 through October 20, 1988.

4.5 FURNACE DECONTAMINATION, REMOVAL AND SHIPMENT

Vacuum pump flushing was completed on September 1, 1988, before a 2-week hiatus was called for other site work. Decontamination, monitoring, appurtenance removal and sealing of the arc furnace was performed over the period of October 10, 1988 through October 13, 1988 during which an oily substance was found to be leaking from the filter box. Delay of the shipment of the furnace to RMDF until November 14, 1988 resulted from the resolution of this problem. A special procedure was prepared and implemented (Ref. 3). The oil was solidified with Petroset and the surfaces wiped.

During the period of October 17, 1988 through October 18, 1988, surveys were conducted, the furnace placed on a pallet and its exterior cleaned. The furnace was loaded with LSA waste and diatomaceous earth, sealed and prepared for shipment. The furnace was shipped to Hanford, Washington, for burial as radioactive waste and the equipment was struck from the property accountability rolls per Ref. 5.

4.6 MISCELLANEOUS CLEANUP

Over the period of October 31, 1988 through November 22, 1988, miscellaneous cleanup and surveys were done. The prefilter, the HEPA filter components and the stack was removed from the building exterior, the sump was pumped out and the furnace power transformer removed. The balance of cleanup, decontamination and disposal activities were conducted at the RMDF and completed by December 7, 1988.

4.7 FINAL SURVEYS

The final radioactive survey was conducted beginning November 14, 1988, and the radiological status of the facility, reported in Ref. 6, was that all portions of the above ground structure may be disposed of as conventional waste. Radiological survey overchecks were performed on demolished materials. Below-grade concrete portions met the criteria for release for unrestricted use, and remain in place. A site water runoff analysis was done on September 15, 1988, and determined that there was no detectable activity. (Ref. 10, Appendix E)

4.8 BUILDING DEMOLITION

Reference 7 is the demolition specification that was used by Taylor's Wrecking Company for the demolition of the above-ground portions of the building, under Purchase Order No. R 95NJZ89-09-6030. The work was performed over the period of April 17, 1989 through July 26, 1989.

5. WASTE

5.1 STIR FACILITY

All radioactive waste generated from the STIR D&D activities was sent to the RMDF. Contaminated water from the concrete coring and Hoe-Ram operations was evaporated at the RMDF. Solid waste was packaged in containers and shipped in three shipments to Beatty, Nevada for land burial. A total of 1,500 ft^3 of waste was shipped.

5.2 BUILDING T028

All radioactive waste resulting from the Building T028 D&D activities was sent to RMDF for packaging and shipment, and ultimately sent to Hanford, Washington, for land burial. A total of 1,183.7 ft^3 for the arc furnace, 690 ft^3 of boxed waste, and 22.2 ft^3 of material in drums.

For the types of waste generated at the STIR facility and T028 see reference Section 3.0 of this report.

Two separate waste disposal sites were used, Beatty, Nevada (1976), and Hanford, Washington (1988), as noted above.

6. PERSONNEL EXPOSURE

Monitoring of internal and external radiation exposure to personnel, as prescribed in the Operational Safety Plan, was conducted throughout the STIR dismantling operations.

Personnel were periodically evaluated, by urinalysis, for internal exposure to mixed fission products, activation products, and nonspecific gross alpha emitters. All results were at or below the appropriate minimum detection limits for the analysis performed.

The external radiation exposure of the nine persons directly associated with the dismantling operations, during the period of September 23, 1975 through January 31, 1976, when the reactor vessel internals, and reactor shielding were removed, averaged 193 mrem, with a maximum individual exposure of 420 mrem. The entire operation was performed with a total radiation exposure of 1.7 man-rem (Ref. 1).

Monitoring of internal and external radiation exposure to personnel, as prescribed in the Rocketdyne Health & Safety manual, was conducted throughout the Building T028 D&D operations.

Film badges were worn by all persons entering the radiologically posted areas. These badges, which contained beta-gamma-sensitive film packets with the appropriate shields for radiation quality assessment, were processed quarterly by an independent laboratory and provided the legally documented record of external exposure.

None of the Engineering or Radiation and Nuclear Safety personnel assigned to the T028 decommissioning activity received any measurable exposure to ionizing radiation during the decommissioning (Ref. 10).

Rockwell International
Rocketdyne Division

SUPPORTING DOCUMENT

GO NO.	S/A NO.	PAGE 1 OF	TOTAL PAGES	REV LTR/CHG NO. SEE SUMMARY OF CHG	NUMBER
96075	32100	36	36	NEW	N001TI000322

PROGRAM TITLE

Decommissioning Surplus Facilities

DOCUMENT TITLE

Building T028 Decontamination and Demolition Final Report

DOCUMENT TYPE: Technical Information	KEY NOUNS: Decommission, Building 028
ORIGINAL ISSUE DATE: 6-6-90	REL. DATE: 06-6-90

APPROVALS — DATE

W. R. McCurnin 5/30/90

PREPARED BY/DATE	DEPT	MAIL ADDR
A. Klein	635	T020

IR&D PROGRAM? YES ☐ NO ☒
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ABSTRACT

Building 028 previously housed an irradiation reactor and was known as the STIR facility. It was decommissioned and decontaminated, and released for unrestricted use in 1976. A uranium oxide melting experiment was later conducted in the facility which contaminated the experimental equipment and other portions of the facility. In order to reduce the potential environmental risks and avoid the costs of maintenance and surveillance; the contaminated systems and equipment were removed, the above ground structure demolished, and the site prepared for release for unrestricted use in April 1989.

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FORM 734-C REV. 12-84

CONTENTS

TABLE

surveys or resurveys of selected sites were initiated in 1985. Sites surveyed in these recent investigations included the Old Conservation Yard (OCY), Building T064 Side Yard, and Building T028.

From 1952 until 1977, the OCY and surrounding land areas were used for the storage of excessed equipment some of which was contaminated with either uranium or mixed fission products. The 1988 radiological survey of the OCY identified elevated concentrations of Cs-137 in soil, with assumed equivalent concentrations of Sr-90. Although there is no available confirming documentation, the source of the contamination is believed to be the result of a contaminated liquid spill. The area was further investigated to delineate the areal extent of contamination. This investigation identified a 37 m^2 (400 ft^2) area with contamination to a depth of 15 cm (6 in). A Cs-137 clean-up guideline was established through the use of the DOE computer code RESRAD.[1] Contaminated soil was excavated, and post-remedial action measurements and sampling were performed and documented.

Building T064, which was formerly known as the Source and Special Nuclear Material Storage Facility, was used for the storage of packaged items of source and special nuclear materials prior to 1980; it is currently used to store non-nuclear components and equipment and metal boxes containing low-level contaminated soil. Site history indicates that the area around the building and the side yard was occasionally used for storage of recoverable uranium scrap, irradiated fuel elements, and miscellaneous radioactive wastes, which included in the early 1960's a lead-pig cask containing irradiated "Seawolf" fuel and contaminated water. The drain plug in the cask failed, allowing the water to leak onto the Side Yard. A 65 m^2 area was excavated immediately following the incident; however, a 1988 comprehensive radiological survey of the area around Building T064 identified elevated soil concentrations of Cs-137 (assumed equivalent amount of Sr-90). Further investigations determined that a 47 m^2 area of contamination was located within the northeast fence line and extended in a northeast direction past the fence line over an additional area of 370 m^2. A Cs-137 guideline was developed and the top 41 cm of soil was subsequently excavated from the area and a post-remedial action survey performed and documented.

Building T028 housed the Shield Test Reactor (STR) from 1961 until 1964, at which time STR was modified and renamed the Shield Test and Irradiation Reactor (STIR) which operated until 1972. The reactor was dismantled and the building decontaminated. From 1977 to 1981, experiments were conducted in the building to investigate the behavior of molten uranium oxide, which resulted in recontamination of building and equipment surfaces. Decontamination of the building was performed in 1988 and the above-grade portion demolished in 1989, leaving only the concrete slab floor, below-grade concrete test vault, and stairwell intact.

DOE's Office of Environmental Restoration (DOE/ER), Northwestern Area Programs, San Francisco Operations Division is responsible for oversight of a number of remedial actions that have been or will be conducted at the SSFL. It is the policy of DOE to perform independent (third party) verification of remedial action activities conducted within Office of Environmental Restoration programs. The Environmental Survey and Site Assessment Program (ESSAP) of the Oak Ridge Institute for Science and Education (ORISE) has been designated as the organization responsible for this task at SSFL. This report describes the results of the verification surveys.

SITE DESCRIPTION

The SSFL is located near Chatsworth in the Simi Hills of southeastern Ventura County, California, approximately 47 km (29 mi) northwest of downtown Los Angeles (Figure 1). The site is comprised of a total of approximately 1090 hectares (2700 acres) and is divided into four administrative areas (Areas I - IV) and a Buffer Zone. DOE operations are conducted in Rockwell International-owned and DOE-owned facilities located within the 117 ha Area IV. The ETEC portion of Area IV consists of government-owned buildings that occupy 36 ha. The Area IV plot plan is provided in Figure 2 and indicates the locations of those areas addressed by this report.

The OCY is located in the northeast quadrant of Area IV and is a portion of adjacent land groupings totaling 2 ha, termed the Old Energy Systems Group (ESG) Salvage Yard, Rocketdyne Barrel Storage Yard and the New Salvage yard (also known as T583). The OCY occupies an area at the corner of G Street and the Old Salvage Yard Road (Figure 3). The surface is paved with asphalt and is currently used for trailer storage.

Building T064 is in the northeast quadrant of Area IV, north of and above G Street (Figure 4). The Side Yard is located to the east of T064 and includes an area of approximately 0.8 ha.

Building T028 is located in the north-central portion of Area IV. The above-grade concrete slab is approximately 300 m^2 in area. The below-grade vault measures approximately 60 m^2 with 6 m (20 ft) ceilings. Construction consists of a concrete slab floor with concrete walls and ceiling.

OBJECTIVE

Through document reviews and independent surveys, an independent evaluation is performed. The purpose of the evaluation is to validate that cleanup procedures and survey methods utilized by Rockwell/Rocketdyne were adequate. In addition, independent verification provides assurance that the post-remediation data is sufficient, accurate, and demonstrates that remedial actions were accomplished in accordance with appropriate standards and guidelines, and that authorized limits were met.

DOCUMENT REVIEW

The final decontamination and survey reports for the OCY, Building T064 Side Yard, and Building T028 were reviewed for general thoroughness, accuracy, and completeness.[2,3,4] The procedures used and data developed for area characterization and post-remedial action monitoring were evaluated to determine if surveys had been adequately performed, areas of contamination were identified and remediated, and that the DOE guidelines had been met.

PROCEDURES

ESSAP personnel conducted independent measurement and sampling activities at SSFL on June 9 and 10, 1992. Survey activities were performed in accordance with a site specific survey plan, using procedures and instruments described in the ESSAP Survey Procedures Manual and summarized in Appendices A and B.

SURVEY PROCEDURES: OCY AND T064 SIDE YARD

Reference Grid

A reference grid, consisting of 10 m x 10 m grid blocks, was established on outdoor areas associated with the OCY and T064 Side Yard (Figures 5 and 6). The remaining 2 ha and 0.8 ha land areas were not gridded. Measurements and samples from ungridded surfaces were referenced to prominent site features.

Surface Scans

Gamma surface scans were performed over the remediated portions of the OCY and T064 Side Yard. In addition, portions of the respective 2 ha and 0.8 ha adjacent areas were also surface scanned. Scans were performed with NaI detectors, coupled to ratemeters with audible indicators. Locations of elevated direct radiation identified by surface scans were marked for further investigation.

Soil Sampling

Composite surface (0-15 cm) soil samples were collected from three 100 m^2 areas within the OCY and T064 Side Yard. Two additional soil samples were collected from the T064 Side Yard at locations of elevated direct radiation detected during surface scans. Figures 5 and 7 show soil sampling locations.

1.0 INTRODUCTION

Building 028 originally housed the Shield Test and Irradiation Reactor (STIR). The reactor was also used for neutron radiography. The reactor was removed and the facility decontaminated in late March 1976, as reported in reference 1. A uranium oxide melting experiment was conducted in the facility in support of a reactor safety program sponsored by Department of Energy. Normal and depleted uranium oxide was processed and melted under controlled conditions. Following completion of the experiment, in September 1982, the equipment was sealed, the building closed, and routine maintenance and surveillance performed awaiting DOE funding for D&D under the Strategic Facilities Initiative Program. GFY 1988 funds were allocated and the work scope expanded to include demolition of the facility based upon its reported poor condition, references 2 and 3.

The uranium oxide melting experimental equipment was comprised of a vacuum arc furnace, the vacuum equipment, the associated electrical power systems and a ventilation system. A plan view of the facility is presented in Figure I-1. Room B101, the basement, housed the experimental equipment while Room 102A, at ground level, contained normal and depleted uranium which was surplus at the end of the experiment.

The highlights of the overall activity plan, as extracted from reference 4, are as follows:

1. Package and ship the surplus normal and depleted uranium oxide.
2. Size reduce, package and remove the contaminated hood and lab cabinet in room 102A.
3. Survey and decontaminate room 102A.
4. Isolate, seal, package and dispose of the arc furnace intact.
5. Disassemble, size reduce and package the furnace, peripherals (vacuum pump, HEPA filter, plumbing and electrical equipment) for disposal.
6. Remove, size reduce, and package the radioactive exhaust system ducting and plenum, filters and blowers for disposal.
7. Decontaminate and survey Room B101.
8. Survey facility for unrestricted usage.
9. Demolish the above ground structure.

1 2 3

2.0 FACILITY DESCRIPTION

2.1 GENERAL

The facility is thoroughly described in reference 1 and not repeated here. The following discussion relates to the special experimental equipment and activities conducted after the STIR D&D activities. Figure I–1 shows plan views of the facility and locates key items. The ventilation system was left intact after the prior D&D work and distributed air throughout the facility through ducting and plenums; and contains HEPA filter elements and the air drive system.

2.2 ROOM 102A

The ground level room contained approximately 300 kg of depleted and normal uranium oxide, declared surplus in reference 8. The fume hood and laboratory cabinet were also located in this room. Due to the test operations, the room floor and walls required decontamination.

2.3 ROOM B–101

The basement room contained the arc melting furnace whose characteristics were: 5.5 ft long, 5 ft wide by 6 ft high, representing a disposal volume of 165 cubic ft^3, with a maximum gross weight of 6,000 lb. It was a model NCCND 4157331 manufactured by Vacuum Specialties, Inc., Somerville, Mass. It was a steel vacuum furnace containing internal induction melting capabilities and included the associated wiring and controls. The power conditioning equipment was located adjacent to the furnace.

The room also contained the vacuum pumping equipment, associated plumbing and controls, and approximately 22 kg of normal uranium oxide were retained within the furnace and vacuum systems.

3.0 SUMMARY OF ACTIVITIES

All of the activities discussed below were performed in accordance with approved, written procedures. The procedures employed are cited in references 4, 6 and 10 and presented in Table III-1. The details of the day-by-day activities, identification of crews and other information are contained in the operational log book titled "Building T028 Decontamination—August 1988", which is located in the Atomics International Library as R001410. Copies of the Health and Safely Analysis Reports citing the activity levels of pallets 1, 2, 3, and 5 prior to shipment from building T028, and electrical equipment prior

Table III-1. Building T028 Decontamination, Decommissioning, and Demolition Procedures

1. 173DWP000010, Structural Surfaces Decontamination, Rev New. Revised 8/11/88 for use at T028.
2. 173DWP000019, Known and Suspect Contaminated Support Areas Decontamination, Rev New. Revised 8/11/88 for use at T028.
3. N0010P160007, Decontamination and Size Reduction of Low Level R/A Materials, Rev New. Approved for use 8/11/88
4. 4173DWP000020, R/A Waste Handling Procedure, Rev New. Revised for use at Building T028, and approved 8/11/88.
5. N704DWP990082, High Volume Exhaust Removal, Rev A. Revised for use at Building T028 and approved 8/11/88.
6. N001DWP000019, Size Reduction and Removal of Vacuum Furnace System, Rev New. Approved for use 8/3/88.
7. 173DWP000021, Bldg T028 Radiological Survey Procedure, Rev New, Revised for use at T028 and approved 8/11/88.
8. 094QAP-00, Inspection Requirements for the Shipment of Radioactive Materials, Rev E, Approved 8/11/88.
9. 089QPP000001, Radioactive Material Packaging and Shipping Quality Assurance Program Plan, Rev A, Approved 8/11/88.
10. N704DWP990094, Solidification of TRU-Contaminated Oil, Rev New, Approved 10/4/88.
11. Vacuum Furnace Packaging (procedure), Created new within reference 10.
12. Procedures for Removing Residual Oil . . . from Exhaust System . . . from the Vacuum System. Created new within reference 6.

D635-0109

to wire removal are contained in Appendix A. A listing of the Government Owned property removed from building T028 is provided in Appendix B. R/A contaminated equipment was packaged and disposed of as contaminated waste. Clean equipment was excessed through the procedures of property administration. All of the equipment listed was removed from the accountability listings.

Photographs of the equipment, the facility and miscellaneous items are contained in Appendix C.

3.1 SURPLUS URANIUM OXIDE DISPOSAL

The surplus uranium oxide was assembled, packaged and palletized for disposal. The total inventory removed was 278,671 gm of normal uranium oxide and 22,405 gm of depleted uranium oxide as detailed in Appendix D. The work was performed from July 14, 1988 to August 1, 1988. This material was shipped to Hanford, Washington as R/A waste.

3.2 EQUIPMENT DECONTAMINATION

Equipment, piping, hardware and electrical components were disconnected, disassembled and packaged for disposal. The HP monitored the waste continuously as it was being removed and packaged for shipment to salvage if clean and to the Radioactive Material Disposal Facility (RMDF) if contaminated. This activity encompassed both rooms 102A and B–101. This effort was performed from August 1, 1988 to August 19, 1988.

3.3 BUILDING SURFACES DECONTAMINATION

Room 101A concrete floor was scabbled and the walls dusted, from August 22, 1988 through August 24, 1988. Survey showed the room to be acceptable.

3.4 FILTER SYSTEM DECONTAMINATION, REMOVAL AND DISPOSAL

Removal of R/A ducting began with the attic and continued through Room 102A, the change room and the rest room during the period from August 25, 1988 through August 30, 1988. The effort was stopped while the furnace and appurtenances were examined and work started to achieve the disposal site's schedule target for furnace shipment. It was necessary to repair and have the R/A filter system operational for the furnace cleanup and removal work. Following removal of the furnace the remaining ventilation ducting was removed. This activity was performed over the period of October 10, 1988 through October 20, 1988.

3.5 FURNACE DECONTAMINATION, REMOVAL AND SHIPMENT

Vacuum pump flushing was completed on September 1, 1988, before a 2-week hiatus was called for other site work. Decontamination, monitoring, appurtenance removal and sealing of the arc furnace was performed over the period of October 10, 1988 through October 13, 1988 during which an oily substance was found to be leaking from the filter box. Delay of the shipment of the furnace to RMDF until November 14, 1988 resulted from the resolution of this problem. A special procedure was prepared and implemented (reference 6.). The oil was solidified with Petroset and the surfaces wiped. During the period of October 17, 1988 through October 18, 1988, surveys were conducted, the furnace placed on a pallet and its exterior cleaned. The furnace was loaded with LSA waste and diatomaceous earth, sealed and prepared for shipment. The furnace was shipped to Hanford, Washington for burial as R/A waste and the equipment struck from the property accountability rolls per reference 9.

3.6 MISCELLANEOUS CLEANUP

Over the period of October 31, 1988 through November 22, 1988, miscellaneous cleanup and surveys were done. The prefilter, the HEPA filter components and the stack was removed from the building exterior, the sump was pumped out and the furnace power transformer removed. The balance of cleanup, decontamination and disposal activities were conducted at the RMDF and completed by December 7, 1988.

3.7 FINAL SURVEYS

The final R/A survey was conducted beginning November 14, and the radiological status of the facility, reported in reference 11, was that all portions of the above ground structure may be disposed of as conventional waste. Below grade portions meet the criteria for release for unrestricted use, and may remain in place. A site water runoff analysis was done on September 15, 1988, and determined that there was no detectable activity. Appendix E is a copy of the report.

3.8 BUILDING DEMOLITION

Reference 12 is the demolition specification that was used by Taylor's Wrecking Company, who demolished the above ground portions of the structure, under purchase order number R 95NJZ89-09-6030. The work was performed over the period of April 17, 1988 through July 26, 1989.

3.9 DISPOSAL OF RADIOACTIVE WASTE

All radioactive waste resulting from the Building T028 D&D activities was sent to RMDF for packaging and shipment, and ultimately sent to Hanford, Washington for land burial. A total of 1183.7 ft^3 of waste was shipped; comprised of 276 ft^3 normal and depleted uranium oxide, 195.5 ft^3 for the arc furnace, 690 ft^3 of boxed waste, and 22.2 ft^3 of material in drums.

3.10 PERSONNEL DOSIMETRY

Monitoring of internal and external radiation exposure to personnel, as prescribed in the Rocketdyne Health & Safety Manual, was conducted throughout the building T028 D&D operations.

Film badges were worn by all persons entering the radiologically posted areas. These badges, which contained beta–gamma–sensitive film packets with the appropriate shields for radiation quality assessment, were processed quarterly by an independent laboratory and provided the legally documented record of external exposure.

None of the Engineering or Radiation and Nuclear Safety personnel assigned to the T028 decommissioning activity received any measurable exposure to ionizing radiation during the decommissioning.

4.0 COSTS

DOE was the funding source for the entire costs of decontamination, decommissioning and demolition. The project was identified as SAN–1–89–1405 (reference 3) and estimated the total cost at $241,000 to be available in GFY 1988.

4.1 FINAL ACTUALS

The actual total costs of the project were $239,970 as recorded under ETEC's General Order number 95943 (reference 7). It was comprised of approximately $150,000 for in–house labor of disassembly, decontamination, cleaning and packaging; $52,000 for demolition of the building above ground structure by a contractor; $28,000 for off–site burial and disposal costs of contaminated materials; and the balance (approximately $10,000) was for miscellaneous items (crane rental, materials, etc.).

5.0 REFERENCES

1. AI–ERDA 13168, STIR Facility, Decontamination and Disposition, Final Report, B. F. Ureda, Rockwell International, August 26, 1976
2. 87 ETEC–DRF–0871 "Site Consolidation," H. Wieseneck, ETEC, April 20, 1987
3. DOE FAX SFI Project, SAN–1–09–1405 , Demolition of Bldg 028, LMFBR Fuel Safety Bldg. J. F. Saidi, September 24, 1987
4. IL, "Decontamination Plan for Building T028," C. H. Knox, August 1, 1988
5. IL, "Building 028 D&D (Scope, Approach and Cost Estimate)., F. C. Schrag, December 9, 1987.
6. IL, "Procedures for Removing Residual Oil . . .T028, (Cost Estimate), R. J. Tuttle, October 10, 1988.
7. Telecon, T028 D&D and Demolition Costs, G.O. 95943 Klein (AI) to Lucci (ETEC), 18 April 1990.
8. Telecon, Fricken/Wiley (DOE–SAN) to Schaubert (AI), Excess DOE Owned Normal and Depleted Uranium., August 31, 1987
9. IL, "Furnace, Melting 20189543", Hirasuna, December 19, 1989
10. IL, " Vacuum Furnace Packaging," F. G. Schmidt, November 11, 1988
11. IL 495 ll.rjt, "Radiological Status of T028," R. J. Tuttle, April 17, 1989
12. M028–69180–T1, "Building T028 Demolition Specification," January 17, 1988

APPENDIX A

N001TI000322
Page 14

Rockwell International
Rocketdyne Division

HEALTH AND SAFETY ANALYSIS REPORT

ANALYZED BY V.B. Saba
DATE ANALYZED 8-2-88
FILM NO. (DO NOT WRITE IN THIS BOX)

SUBMITTED BY V. B. Saba
DATE SAMPLED 8-2-88
BLDG. AND ROOM NO. T028

SAMPLE NUMBER	DESCRIPTION AND LOCATION	RESULTS dpm/100 cm² α	β	Direct Reading dpm/100cm² α	β
	Pallet #1 Misc. ITEMS on Pallet outside T028				
1	Drum # 075030	< 20	< 50	Non detectable Activity	Non detectable activity
2	Drum # 8160175				
3	Plastic molding compound in CAN				
4	Box of Zirconia Bulk Fiber				

COMMENTS: Canberra # 354916 - due 8-22-88
Ludlum 12 # 327794 - due 9-19-88
Ludlum 12 # 381726 - due 9-19-88

TYPE OF SAMPLE: SOIL ___ WATER ___ AIR ___
SMEAR ✓ OTHER Direct-Reading (IDENTIFY)
TYPE OF ANALYSIS:
RADIOMETRIC ✓ BERYLLIUM ___
OTHER ___ (IDENTIFY)

LEDGER ACCOUNT 29841 CONTRACT OR ORDER 95943 SUB-ACCOUNT 43100 WORK RELEASE ___
LOG BOOK NO. ___ PAGE 1

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N001TI000322
Page 15

Rockwell International
Rocketdyne Division

HEALTH AND SAFETY ANALYSIS REPORT

ANALYZED BY: K.B. Saba
DATE ANALYZED: 8-3-88
FILM NO.: (DO NOT WRITE IN THIS BOX)

SUBMITTED BY: V.B. Saba
DATE SAMPLED: 8-3-88
BLDG. AND ROOM NO. T028

SAMPLE NUMBER	DESCRIPTION AND LOCATION Pallet # 2 Misc Items on Pallet outside T028	dpm / 100 cm² α	dpm / 100 cm² β	RESULTS Direct Reading dpm / 100 cm² α	RESULTS Direct Reading dpm / 100 cm² β
1	stainless steel container	< 20	< 50	N.D.A.	N.D.A.
2	small box with cables inside				
3	aluminum				
4	DC Amperes indicator plate				
5	Two color pyrometer				
6	Transducer indicator model CA12				
7	Metal container				
8	wood lid				
9	Metal basket				
10	Iron magnetic weights (3)				
11	various switches				
12	small piece of mtl				
13	lid for stainless steel container				

ANALYZED BY V.B. SABA
DATE ANALYZED 8-3-88
FILM NO. ______ (DO NOT WRITE IN THIS BOX)

Rockwell International
Rocketdyne Division

HEALTH AND SAFETY ANALYSIS REPORT

SUBMITTED BY V.B. SABA
DATE SAMPLED 8-3-88
BLDG. AND ROOM NO. T028

SAMPLE NUMBER	DESCRIPTION AND LOCATION	RESULTS dpm/100 cm² α	β	Direct Reading dpm/100 cm² α	β
	Pallet # 5				
	Misc. Items on Pallet outside T028				
1	Esterline Angus graphic voltmeter prop #173592	∠ 20	∠ 50	700 *	N.D.A.
2	Honeywell graphic chart Indicator prop # 173594			1120 *	
3	Oven # 2				

COMMENTS: Canberra #354916 - due 8-22-88
Ludlum 12 Scintillator #327794 - due 9-19-88
Ludlum 12 #381726 - due 9-19-88
: Radioactive waste

LEDGER ACCOUNT 29841 CONTRACT OR ORDER 95943 SUB-ACCOUNT 43100 WORK RELEASE ______
LOG BOOK NO. ______ PAGE 2
732-A REV 4-87

TYPE OF SAMPLE: SOIL ____ WATER ____ AIR ____
SMEAR ✓ OTHER Direct Reading (IDENTIFY)
TYPE OF ANALYSIS:
RADIOMETRIC ✓ BERYLLIUM ______
OTHER ______ (IDENTIFY)

ANALYZED BY K.B. SABA
DATE ANALYZED 8-5-88
FILM NO. (DO NOT WRITE IN THIS BOX)

Rockwell International
Rocketdyne Division

HEALTH AND SAFETY ANALYSIS REPORT

SUBMITTED BY V. B. SABA
DATE SAMPLED 8-5-88
BLDG. AND ROOM NO. T028

SAMPLE NUMBER	DESCRIPTION AND LOCATION PANEL # 3 Misc Items on Pallet outside T028	RESULTS dpm/100 cm² α	dpm/100 cm² β	Direct Reading dpm/100 cm² α	Direct Reading dpm/100 cm² β
1	Beta Timer	< 20	< 50	N.D.A	N.D.A
2	Function generator prop. # 677255				
3	Daytronic model 9010 prop. # 251393				
4	Nuclear chicago model 181A prop. # 153347				
5	Wood box				
6	Reactor power pool type prop. # 157041				
7	Vacum tube voltmeter model 400 H # 116-19367				
8	Modline - automatic optical pyrometer prop. # 252531				
9	Dual channel DC amp-recorder prop. # 670968				
10	320 dual channel DC amp-recorder prop. # 676273				
11	Various cables				

COMMENTS: Canberra # 354916 - due 8-22-88
Ludlum 12 Scintillator # 327794 - due 9-19-88
Ludlum 12 # 381726 - due 9-19-88

TYPE OF SAMPLE: SOIL ___ WATER ___ AIR ___
SMEAR ✓ OTHER Direct Reading (IDENTIFY)
TYPE OF ANALYSIS:
RADIOMETRIC ✓ BERYLLIUM ___
OTHER ___ (IDENTIFY)

LEDGER ACCOUNT 29841 CONTRACT OR ORDER 95943 SUB-ACCOUNT 43100 WORK RELEASE ___
LOG BOOK NO. ___ PAGE 1

FORM 732-A REV 4-87

N001TI000322
Page 18

ANALYZED BY V. B. Saba
DATE ANALYZED 8-4-88
FILM NO. (DO NOT WRITE IN THIS BOX)

Rockwell International
Rocketdyne Division

HEALTH AND SAFETY ANALYSIS REPORT

SUBMITTED BY V. B. Saba
DATE SAMPLED 8-4-88
BLDG. AND ROOM NO. T028 - Rm 101

SAMPLE NUMBER	DESCRIPTION AND LOCATION	RESULTS dpm/100 cm² α	dpm/100 cm² B	Direct Reading dpm/100 cm² α	Direct Reading dpm/100 cm² B
	Surveyed conduit and electrical panel	< 20	< 50	N.D.A	N.D.A.
	on southside of wall in Room 101, T028				
	in order to allow electrician to remove				
	wires. I found no detectable activity				

COMMENTS: Canberra # 354916 - due 8-22-88
Ludlum 12 Scintillator #327794 - due 9-19-88
Ludlum 12 # 381726 - due 9-19-88

TYPE OF SAMPLE: SOIL ___ WATER ___ AIR ___
SMEAR ✓ OTHER Direct Reading (IDENTIFY)
TYPE OF ANALYSIS:
RADIOMETRIC ✓ BERYLLIUM ___
OTHER ___ (IDENTIFY)

LEDGER ACCOUNT 29841 CONTRACT OR ORDER 95943 SUB-ACCOUNT 43100 WORK RELEASE ___
LOG BOOK NO. ___ PAGE 1

FORM 732-A REV 4-87

N001TI000322
Page 17

APPENDIX B

Rockwell International

ROCKWELL INTERNATIONAL - ROCKETDYNE DIVISION
FIXED ASSET BUILDING LISTING

DIVISION 051

TAG NO	DESCRIPTION	EXTRA DESCRIPTION	YR MF	B F	BLDG	USE DPT	LOCATION	ACQ MO-YR	ACCT NUBR	ASSET COST	NET BOOK VALUE	DEP DPT
N0879152	-POLISHER,MTL,LAB	JET THINNING INSTRUMENT	75		4024	635	RM110*	02-75	17417	.00	.00	635
N0879744	-ANALYZER,OXYGEN		74		4024		BACK-RM	10-74	17467	.00	.00	635
N0880241	-RECORDER,INK WRTG	TWO PEN 1MV TO 100V	68		4024		INDSIDE	09-68	17462	2,108.89	.00	635
N0880281	-CONSOLE,ELEC INST		69		4024	635	RM110*	09-69	17462	2,808.75	12.95	635
V0701129	-SCAFFOLD	DEVELOPEMENT TEST	82		4024		A-41<<<<<	07-82	17418	2,254.87	.00	635
V0701129	-SCAFFOLD	SCAFFOLD AROUND SPRAY DRYER	82		4024		A-41<<<<<	01-83	17418	-.03	.00	635
BUILDING TOTAL					4024					143,052.20	7,728.16	
N0861453	-PRESS,HYDRAULIC		60		4025		LOWBAYRK	10-60	17412	681.41	.00	577
V0700223	-PUBLIC ADDRESS SY	ETEC TOOL CRIB	71		4025		A-41	03-71	17412	25,446.08	.00	052
BUILDING TOTAL					4025					26,127.49	.00	
N0327283	-WELD MACH,ARC	230/460V 98/49A PH1 60HZ	75		4026		HIGH BAY	10-75	17417	2,357.94	.00	635
BUILDING TOTAL					4026					2,357.94	.00	
N0356187	-TANK,LIQ NITROGEN	CYLINER-TYPE	81		4027		YARD	10-81	17417	1,411.68	347.02	577
BUILDING TOTAL					4027					1,411.68	347.02	
N0327270	-SCALER,RADIATION	DECADE	76		4028	[redacted]	102	01-76	17417	.00	.00	635
N0327905	-WELD MACH,ARC	1500-AMP-BUMBLEBEE-WELDER	76		4028	[redacted]	INSIDE	10-76	17417	5,284.10	.00	635
N0354646	-PROCESSOR,DATA		80		4028	[redacted]	BASEMENT	03-80	17467	10,278.75	1,178.95	635
N0354646	-PROCESSOR,DATA		80		4028	[redacted]	BASEMENT	10-81	17467	1,642.98	326.58	635
N0355618	-POWER SUPPLY,HIVL	10,000AMPERE	80		4028	[redacted]	INSIDE	10-80	17467	28,281.30	5,093.30	635
N0355618	-POWER SUPPLY,HIVL	10,000AMPERE	80		4028	[redacted]	INSIDE	02-81	17467	80.57	14.59	635
N0414610	-POWER SUPPLY,ELTR	2000VA350-450C	59		4028	[redacted]	YARD	09-59	17462	4,295.20	.00	635
N0468373	-MICROCOPE	STEREOSCOPIC,BINOCULAR,6-50X	60		4028	[redacted]	RM102	04-60	17412	638.56	.00	635
N0640953	-POWER SUPPLY,ELTR	R/M,0-36VDC,0-10A	70		4028	[redacted]	RM102	03-70	17462	521.00	.00	635
N0670213	-PYROMETER	OPTICAL 775-2800C 3RANGE	63		4028	[redacted]	RM102	10-63	17462	582.14	.00	635
N0670968	-RECORDER,LT BEAM	OSCILLOGRAPH	64		4028	[redacted]	RM102	02-64	17462	1,728.10	.00	635
N0675524	-POWER SUPPLY,ELEC	PRESSURE LIQUID HELIUM 800MMHG	66		4028	[redacted]	100	10-66	17462	.00	.00	635
N0675558	-CONSOLE,ELEC INST	INDUCTION FURNACE SYSTEM	66		4028	[redacted]	BASEMENT	10-66	17462	3,000.00	.00	635
N0675559	-MOTOR-GENERATOR	100KW 150HP	66		4028	[redacted]	BASEMENT	10-66	17412	20,158.00	41.12	635
N0676273	-RECORDER	OSCILLOGRAPH DC TO 125CPS 2 CH	67		4028	[redacted]	RM102	04-67	17462	2,047.50	.00	635
N0676273	-RECORDER	OSCILLOGRAPH DC TO 125CPS 2 CH	67		4028	[redacted]	RM102	07-67	17462	11.71	.00	635
N0676960	-AMPLIFIER	PREAMPLIFIER PLUG-IN UNIT	68		4028	[redacted]	100	07-68	17462	.00	.00	635
N0676960	-AMPLIFIER	PREAMPLIFIER PLUG-IN UNIT	68		4028	[redacted]	100	08-68	17462	.00	.00	635
N0676961	-AMPLIFIER	PREAMPLIFIER PLUG-IN UNIT	68		4028	[redacted]	100	07-68	17462	.00	.00	635

DIVISION 051

TAG NO	DESCRIPTION	EXTRA DESCRIPTION	YR MF	B F	BLDG	USE DPT	LOCATION	ACQ MO-YR	ACCT NUBR	ASSET COST	NET BOOK VALUE	DEP DPT
N0876976	-AMPLIFIER	PREAMPLIFIER PLUG-IN UNIT	68		4028	[redacted]	100	07-68	17462	.00	.00	835
N0877255	-GENERATOR	FUNCTION 0.001HZ TO 1MHZ	69		4028	[redacted]	RM102	06-69	17462	868.13	.00	835
N0877637	-CALCULATOR	ELECTRONIC,PROGRAMABLE	70		4028	[redacted]	INSIDE	08-70	17752	839.50	.00	835
N0889513	-MONITOR,TV		80		4028	[redacted]	SANTA SU	09-81	17467	.00	.00	835
N0889514	-CAMERA,TV		80		4028	[redacted]	SANTA SU	09-81	17467	.00	.00	835
N0889514	-CAMERA,TV		80		4028	[redacted]	SANTA SU	10-81	17467	.00	.00	835
N0868527	-RECORDER,INK WRTG	5MV-100V 10RNGE	83		4028	[redacted]	RM102	03-83	17462	825.23	.00	835
N0871409	-POWER SUPPLY,ELTR	L SY-18-30	65		4028	[redacted]	RM102	03-65	17462	815.15	.00	835
BUILDING TOTAL					4028	[redacted]				81,277.92	6,654.54	
N0119971	-COMPUTER.PCXT		85		4030	[redacted]	C.MALWITZ	10-86	17737	3,172.53	1,065.98	844
N0131209	-PUMP,LAB	DIFFUSION PUMP	86		4030	631	STORAGE*	06-88	10088	27,257.34	27,257.34	999
N0161885	-FORKLIFT	HYDRAULIC FORKLIFT	87		4030	023	GARA000000	07-87	17417	35,815.29	28,920.85	023
N0325983	-CONTROL UNIT,ADPE		78		4030	065	TRAFFIC*	01-81	17731	12,628.47	.00	065
N0325983	-CONTROL UNIT,ADPE		78		4030	065	TRAFFIC*	05-84	17731	2,878.38	.00	065
N0327868	-TYPEWRITER,ELEC	SELECTRIC II	76		4030		100	10-76	17757	879.80	.00	577
N0354871	-PRINTER SYS,COMPT		79		4030	065	TRAFFIC*	01-80	17731	5,312.15	.00	840
N0356064	-AUTOMOBILE	CHEVY-IMPALA,STATION-WAGON	79		4030		YARD	02-81	17648	3,467.27	.00	577
N0382608	-STATION WAGON	9 PASSANGER	79		4030		YARD	07-82	17648	1,349.81	.00	023
N0382713	-RADIO,ELEC	TWO-WAY,PTBL,BATTERY W/CASE	82		4030		100	10-82	17417	2,083.17	652.77	586
N0383373	-CONTROL UNIT,DSPL	USED W/DISPLAY A KEYBOARD	83		4030		100	01-83	17731	2,803.73	.00	065
N0383376	-TERMINAL,DISPLAY	COLOR W/KEYBOARD A CONTROLLER	83		4030		100	01-83	17731	3,325.08	.00	065
N0383379	-TERMINAL,DISPLAY	COLOR W/KEYBOARD A CONTROLLER	83		4030		100	01-83	17731	3,325.08	.00	065
N0383382	-PRINTER,DOT MATRI		83		4030		100	01-83	17731	5,414.11	.00	065
N0383384	-PRINTER,DOT MATRI		83		4030		100	01-83	17731	5,414.11	.00	065
N0641510	-TYPEWRITER,ELEC	TYPEWRITER,ELEC SELECTRIC II	72		4030		100	08-72	17752	661.50	12.41	525
N0642095	-TYPEWRITER,ELEC	SELECTRIC II	73		4030		100	08-73	17752	667.80	16.19	525
N0678481	-REFRIGERATION UNI	18 CU FT	73		4030		INSIDE	07-73	17412	355.10	8.45	577
N0680438	-CALCULATOR,ELCTRN	8 DIGIT	75		4030		100	02-75	10010	.00	.00	553
N0689874	-CALCULATOR,ELCTRN		76		4030	060	248-1	02-77	17757	.00	.00	577
R0056674	-COPY MACH,ELECSTC		80		4030		100	01-81	10070	.00	.00	999
Y0700303	-LAND IMPROVEMENT	TRAFFIC & WAREHOUSING	74		4030		B-29	07-74	17117	1,984.00	598.34	081
BUILDING TOTAL					4030					118,774.72	58,530.33	
N0327360	-BREATH APPARATUS	AIR MASK	76		4032		DOORWAY	06-76	17467	583.00	.00	635

Rockwell International

AUG 31, 1988
BLDGMPT

ROCKWELL INTERNATIONAL - ROCKETDYNE DIVISION
GOVERNMENT OWNED PROPERTY LISTING

LISTED BY BUILDING

Z28

TAG NO	NAME	DESCRIPTION	MFG NAME	MODEL NO	SERIAL NO	USE DPT	LOC	ASSET COST
Z0183533	RECORDER,INK WRTG	X-Y PLOTTER	MOSLEY ELECTRONIC			635	YARD	2,212.05
Z0173272	RECORDER,INK WRTG	CONTROLLER 0-10	BARBER COLMAN CO	800-2522	95D0794---	635	BASEMENT	1,863.54
Z0173592	RECORDER,INK WRTG	GRAPHIC 0-75 VD	ESTERLINE ANGUS	AW	150295----	635	INSIDE	1,331.57
Z0173594	RECORDER,INK WRTG	MILLIVOLT 0-10	HONEYWELL,MINN	1540183601	4888809001	635	INSIDE	1,024.28
Z0177178	VOLTAGE REGULATOR	115 V 60 HZ 1 K	INTL ELEC RES	C-1000B	124-------	635	YARD	1,427.38
Z0177500	MANIPULATOR,LAB	S		VM3600-S18	**********	635	YARD	2,082.50
Z0177501	MANIPULATOR,LAB	BELLOWS SEALED		VM3600-S18	**********	635	YARD	2,082.50
Z0177502	MANIPULATOR,LAB	BELLOWS SEALED		VM3600-S18	**********	635	YARD	2,082.50
Z0177503	MANIPULATOR,LAB	BELLOWS SEALED		VM3600-S18	**********	635	YARD	2,100.00
Z0177504	MANIPULATOR,LAB	BELLOWS SEALED		VM3600-S18	**********	635	YARD	2,100.00
Z0177505	MANIPULATOR,LAB	BELLOWS SEALED		VM3600-S18	**********	635	YARD	2,089.50
Z0189045	ANALYZER,WAVE	WAVE 2 CPS-5KC	HEWLETT-PACKARD	302A.	724-05686-	635	YARD	2,155.12
Z0189162	RECORDER,INK WRTG	2 CHANNEL 0-10	LEEDS / NORTHRUP	720-30-30	57352-1-1	635	BASEMENT	1,759.48
Z0189543	FURNACE,MELTING	SKULL	GCA/VACUUM INDUST			635	BASEMENT	50,762.54
Z0251347	RECORDER,LAB	OMNISCRIBE,RECO	HOUSTON INST DIV	5222-5	5/6448-120	635	RM102	1,194.20
Z0251393	INDICATOR,DGT DISP		DAYTRONIC CORP	9010		635	RM102	5,582.20
Z0252531	TEMP MEAS SYS		IRCON INC	2-99C30	99762	635	RM102	2,198.00
Z0253031	MAINFRAME	VISIGRAPH		VG-2060-CR-4	5728	635	BASEMENT	5,167.10
Z0253152	PUMP,CENTRIFUGAL		FREDERICK PUMP	83C38	58785-80-1B	564	POND	1,497.00
Z0253153	PUMP,CENTRIFUGUL		FREDERICK PUMP	83C38	58785-80-1A	564	POND	1,497.00
Z0253269	SCANNER,ELECTRONIC		KAYE INSTS INC	128SS	7010	635	BASEMENT	9,750.00
Z0253833	FILTER,WATER	150GPM, 150PSI	FILTERITE CORP	900866	5717-3	635	LOWERYD	3,258.02
BUILDING TOTAL								105,216.48
	COUNT							22

APPENDIX C

...A-30 BUILDING 1-028

NUCLEAR MATERIAL CONSOLIDATION

To Ticket Nos. 5-1667 & 600895

ITEM No	TICKET No	NET WEIGHT	% U	URANIUM WEIGHT	% U235	GRAMS U-235	DESCRIPTION
1	500298	44,666.00	88.15	39,373.08			Poly Bag w/5gal pail of Depleted & Normal UO_2 Slag Waste from Item (MELTWASTE)
	601265	5,665.80	88.07	4,989.87	0.203	10.13	
2	500284	22,561.00	88.15	19,887.52			Poly Bag w/5gal pail of Depleted & Normal UO_2 waste from Item (TST3-HOOD)
	601261	3,522.50	88.07	3,102.27	0.203	6.30	
3	500296	29,314.50	88.15	25,840.73			5Gal pail of Normal UO2 waste in 2Lb Can from Item (UO2-WASTE)
4	601262	9,541.20	88.07	8,402.93	0.203	17.06	5Gal Pail of Depleted UO2 waste in 2lb Can from Item (UO2-COMB)
5	500291	8,825.00	88.15	7,779.24			5Gal Pail of Normal UO_2 Dust in 2Lb Can From Item (DST-CAN17)
6	601264	4,156.00	88.07	3,660.19	0.203	7.43	5Gal Pail of Depleted UO_2 Powder in 2Lb Can From Item (UO2-PWDRO)
7	500282	819.90	88.15	722.74			5Gal Pail of Normal Slag in 2Lb Can From Item (TST2CAN10)
8	500301	9,937.00	88.15	8,759.47			5Gal Pail of Normal UO2 Pellets in Breadpan From Item (U-PELLETS)
9	500268	9,557.00	88.15	8,424.50			5Gal Pail of Normal Furnace Dust in 2Lb Can From Item (DUST-CAN7)
10	500285	13,788.00	88.15	12,154.12			5Gal Pail of Normal Furnace Dust in 2Lb Can From Item (DST-CAN11)
11	500288	12,606.00	88.15	11,112.19			5Gal Pail of Normal Furnace Dust in 2Lb Can From Item (DST-CAN14)
12	500286	15,064.50	88.15	13,279.36			5Gal Pail of Normal Furnace Dust in 2Lb Can From Item (DST-CAN12)
13	500290	10,740.00	88.15	9,467.31			Polybag w/5Gal Pail of Normal Furnace Dust From Item (DST-CAN16)
14	500281	15,418.60	88.15	13,591.50			Poly Bag w/5Gal Pail of Depleted & Normal Slag & UO_2 Waste in 2Lb Coffee Can From Item (TST2-CAN9)
	601257	2,554.50	88.07	2,249.75	0.203	4.57	
15	500287	17,116.10	88.15	15,087.84			
16	500289	13,340.00	88.15	11,759.21			5Gal Pail of Normal UO_2 Dust in 2Lb can From Item (DST-CAN13)
17	500280	8,935.00	88.15	7,876.20			5Gal Pail of Normal UO_2 Dust in 2Lb Can From Item (DST-CAN15)
18	500267	8,768.50	88.15	7,729.43			5Gal Pail of Normal UO_2 Slag in 2Lb Can From Item (TST1-CAN8)
19	500263	7,681.00	88.15	6,770.80			5Gal Pail of Normal UO_2 Dust in 2Lb Can From Item (DUST-CAN6)
20	500264	2,250.00	88.15	1,983.38			5Gal Pail of Normal UO_2 Dust in 2Lb Can From Item (DUST-CAN2)
21	500265	5,625.00	88.15	4,958.44			5Gal Pail of Normal UO_2 Dust in 2Lb Can From Item (DUST-CAN3)
22	500262	5,618.00	88.15	4,952.27			5Gal Pail of Normal UO_2 Dust in 2Lb Can From Item (DUST-CAN4)
23	500266	4,554.00	88.15	4,014.35			5Gal Pail of Normal UO_2 Dust in 2Lb Can From Item (DUST-CAN1)
24	500299	788.00	88.15	694.62			5Gal Pail of Normal UO_2 Dust in 2Lb Can From Item (DUST-CAN5)
25	500300	24,080.00	88.15	21,226.52			EmptyContaminated Pellet Rods (Aluminum) in 55Gal Poly Bag
	500300	24,080.00	88.15	21,226.52			Poly Bag of 20 Al Pellet Rods Loaded With Normal UO_2 Pellets
							Poly Bag of 20 Al Pellet Rods Loaded With Normal UO_2 Pellets
TOTALS		316,133.10	88.15	278,671.34			Total Normal Uranium From Above Shown on Ticket No. 5 1667
TOTALS		25,440.00	88.07	22,405.01	0.203	45.49	Total Depleted Uranium From Above Shown on Ticket No. 600895

1 55 Gallon Bag Containing Stainless Cans, Cake Tins, Oil Can, Balance Weights, Balance Parts, Bread Pans, 5Gal Pail & Cardboard

1 55 Gallon Bag Containing Cardboard, Pre-Filter, Hardware, Glass, and Glass Vacuum Jar.

1 55 Gallon Bag Containing a Metal Stool, Balance, 5 Gallon Pail, and Steel Rod

10 5 Gallon Bags Containing 12"x12"x4" Fire Bricks

Assorted Plastic Sheeting, Rubber Gloves, Tape, and Paper Wipes

N001TI00032?

APPENDIX D

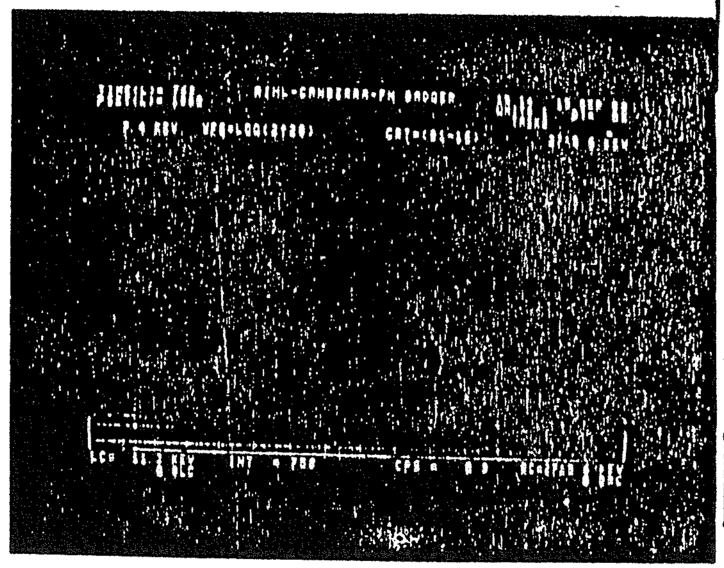
```
*************************************************************************************************************************
     BLDG 28  FIRELINE BREAK WATER SAMPLE FROM RUNOFF AS IT LEFT THE FACILITY------500 ML SAMPLE--------------------BADGER
                         ABOUT 1440 HOURS  9/15/88
*************************************************************************************************************************
```

```
TAG NO. =         28          RIHL-CANBERRA-FH BADGER              PAGE    1
ADC#    1    LIVE TIME =       766 TRUE TIME =       766   15 SEP 88  15:14
COLLECTED AT:  15:01:17.7     15 SEP 88

ENERGY(KEV)=  0.94551E-07 *CH^2 + 0.66095E00 *CH +0.73735E01
PEAK STATISTICS= 1.86            MIN WIDTH=  6           MAX WIDTH= 15
AREA BACKGROUND=     3           %ERROR=  1.86
ISO ID WINDOW=     2.0 KEV       ISO ID LIBRARY=  2
AGE (DAYS)=  0.29968E-01         TABLE   1


ROI#    1  FROM  235.4 KEV TO  246.7 KEV
           PEAK AT  238.9 KEV  FWHM=    2.2 KEV
           INTEGRAL=          16  RATE=    0.0 CPS
           AREA=              16  ERR=    46.5%

             PB-212  AT  238.5 KEV =  0.50128E-04 UCI
```

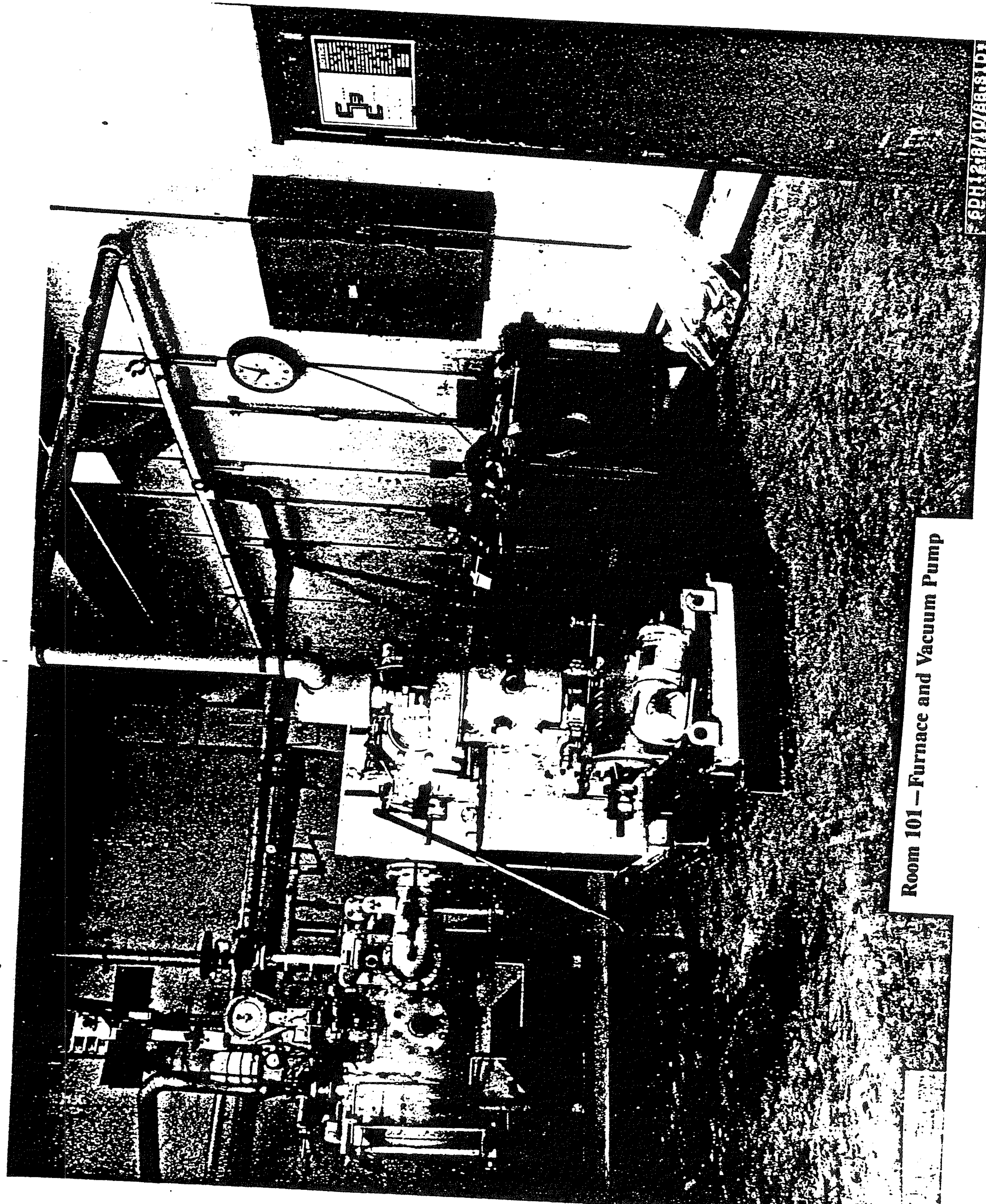


CHANNEL #	DATA							
345		0	0	0	0	3	4	2
352	2	0	0	0	1	0	1	1
360	0	2	0					

No detectable activity

N001TI000322
Page 26

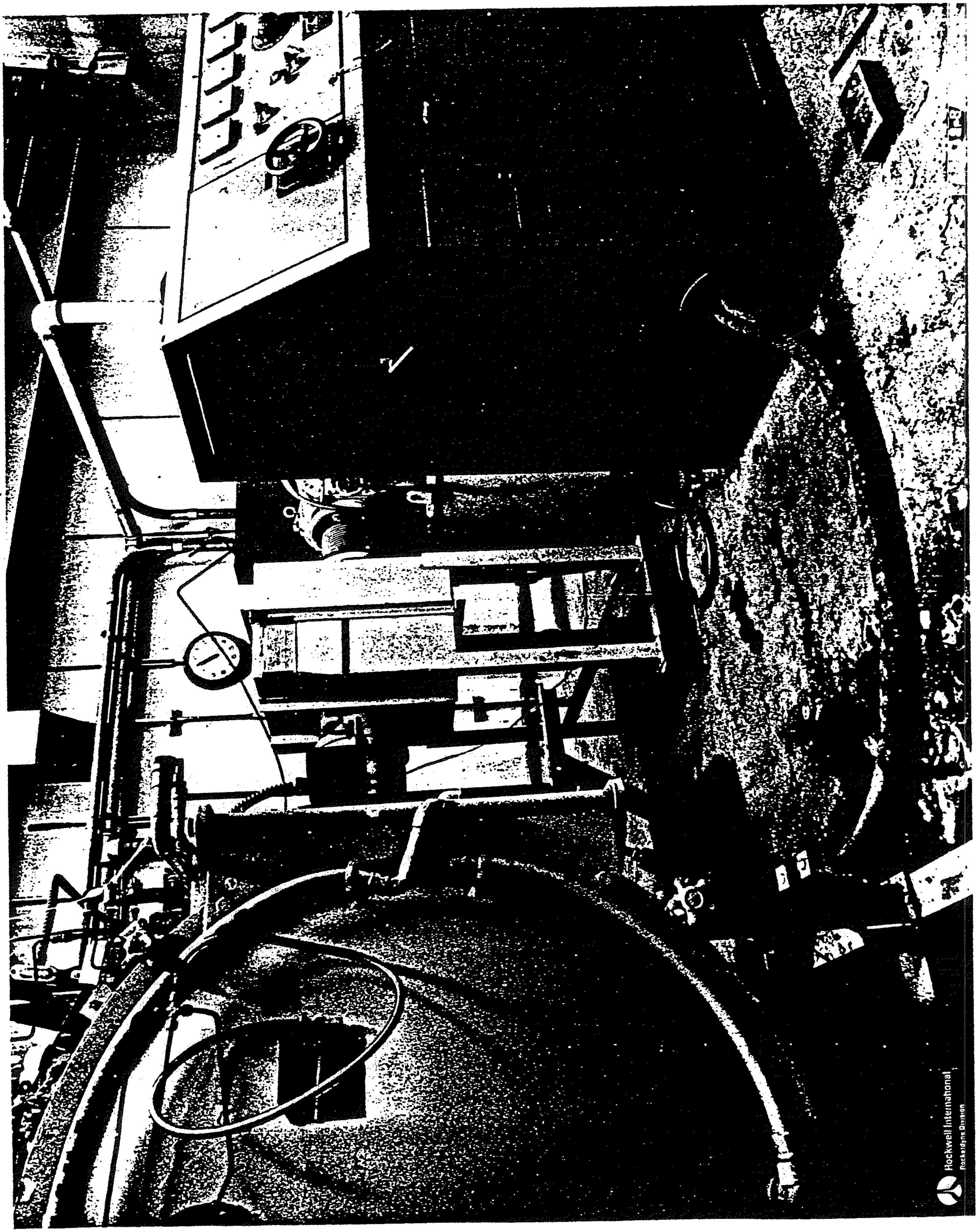
APPENDIX E



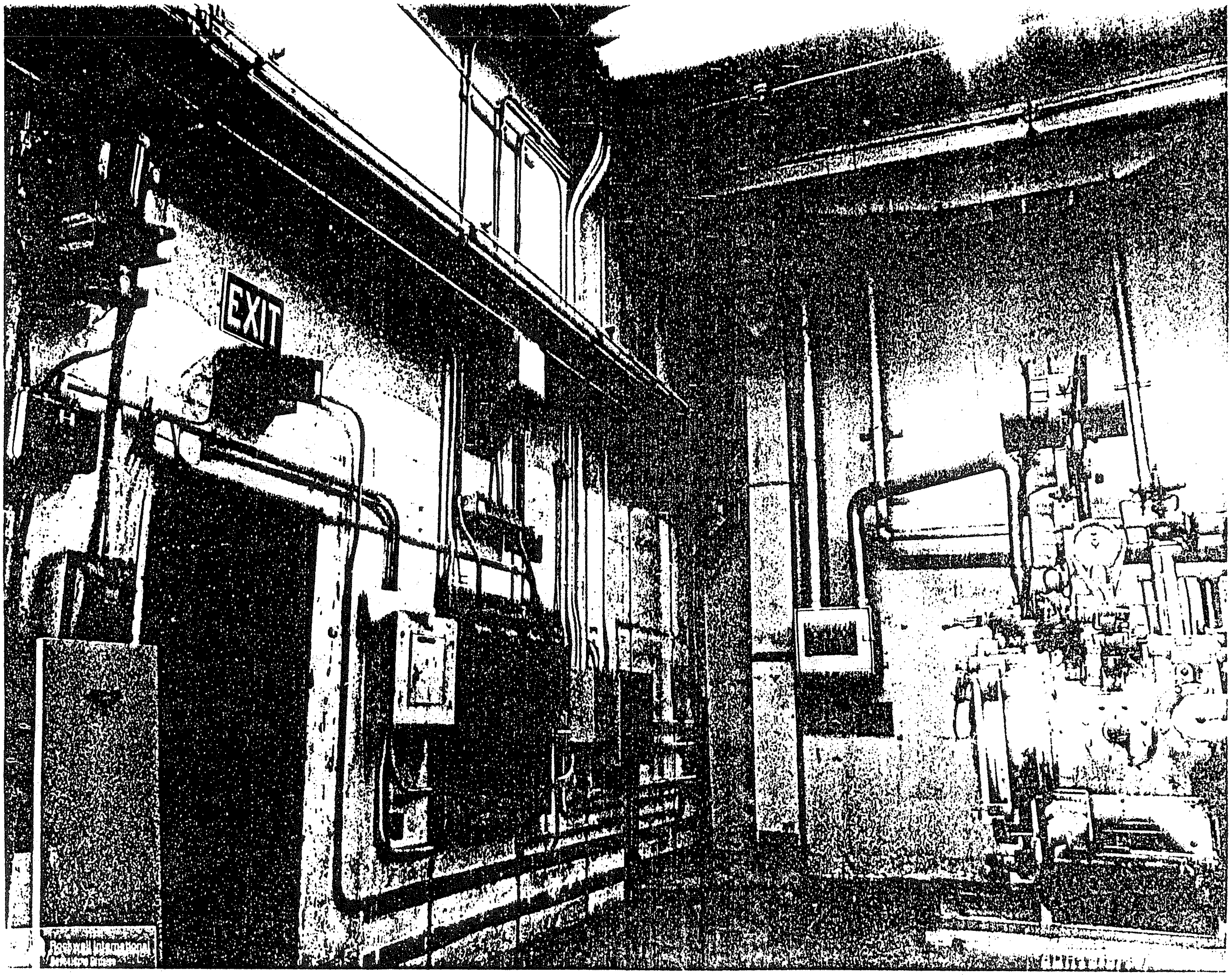
Room 101—Furnace and Vacuum Pump

Room 101—After Equipment Removal and Decontamination

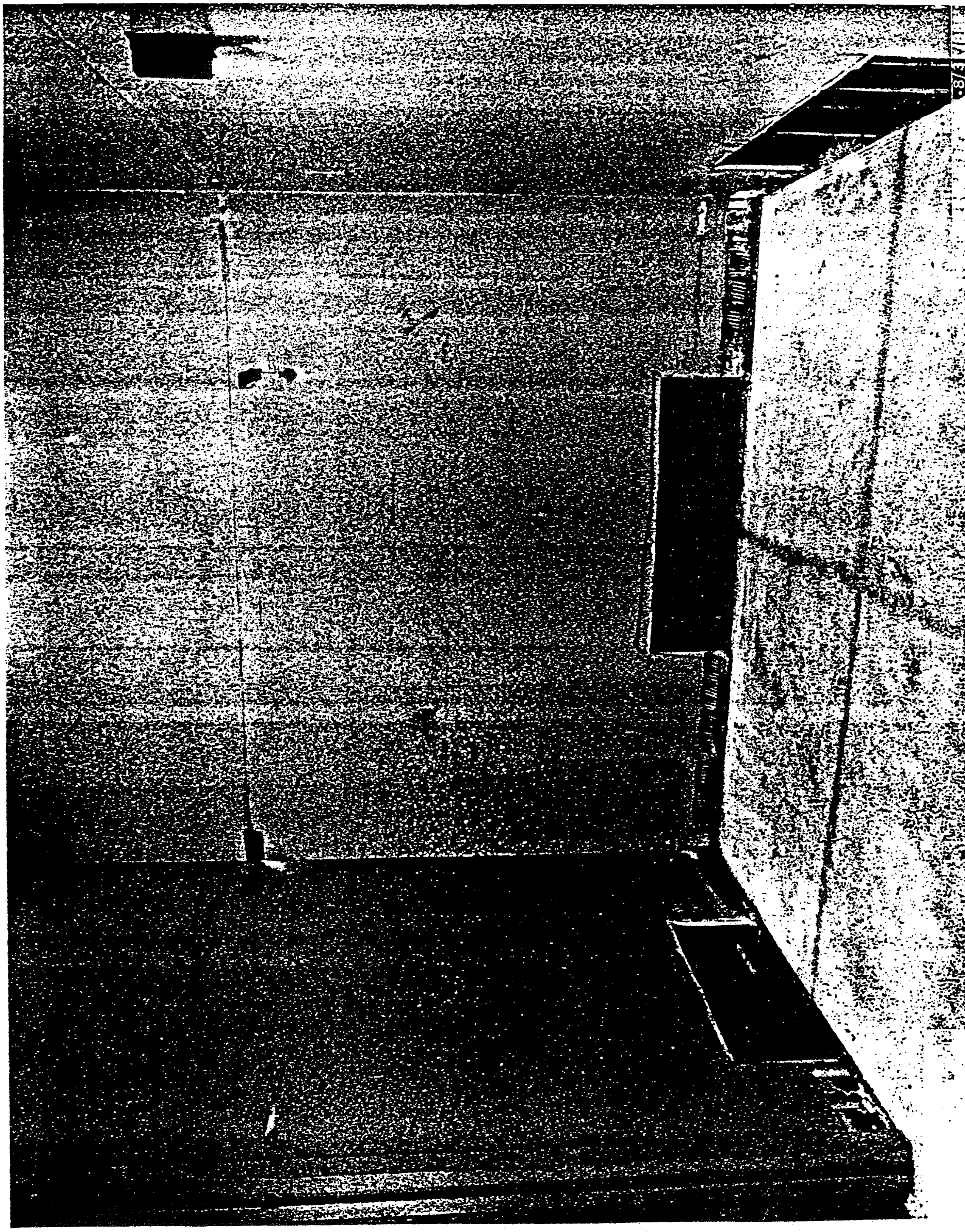
P[illegible] Dem[illegible]n V[illegible]f F[illegible]y



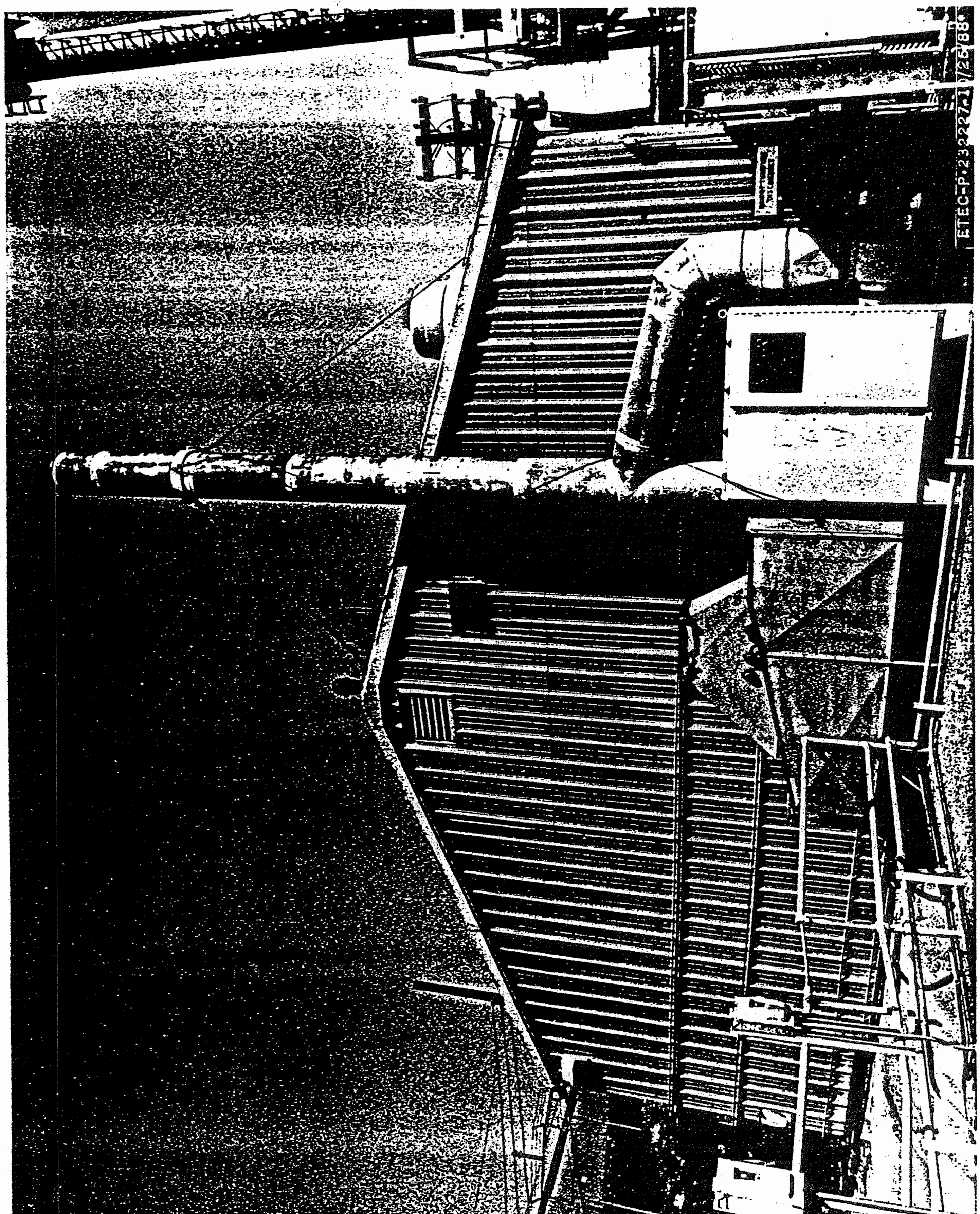
Room 101—Furnace, Filter, and Power Supply



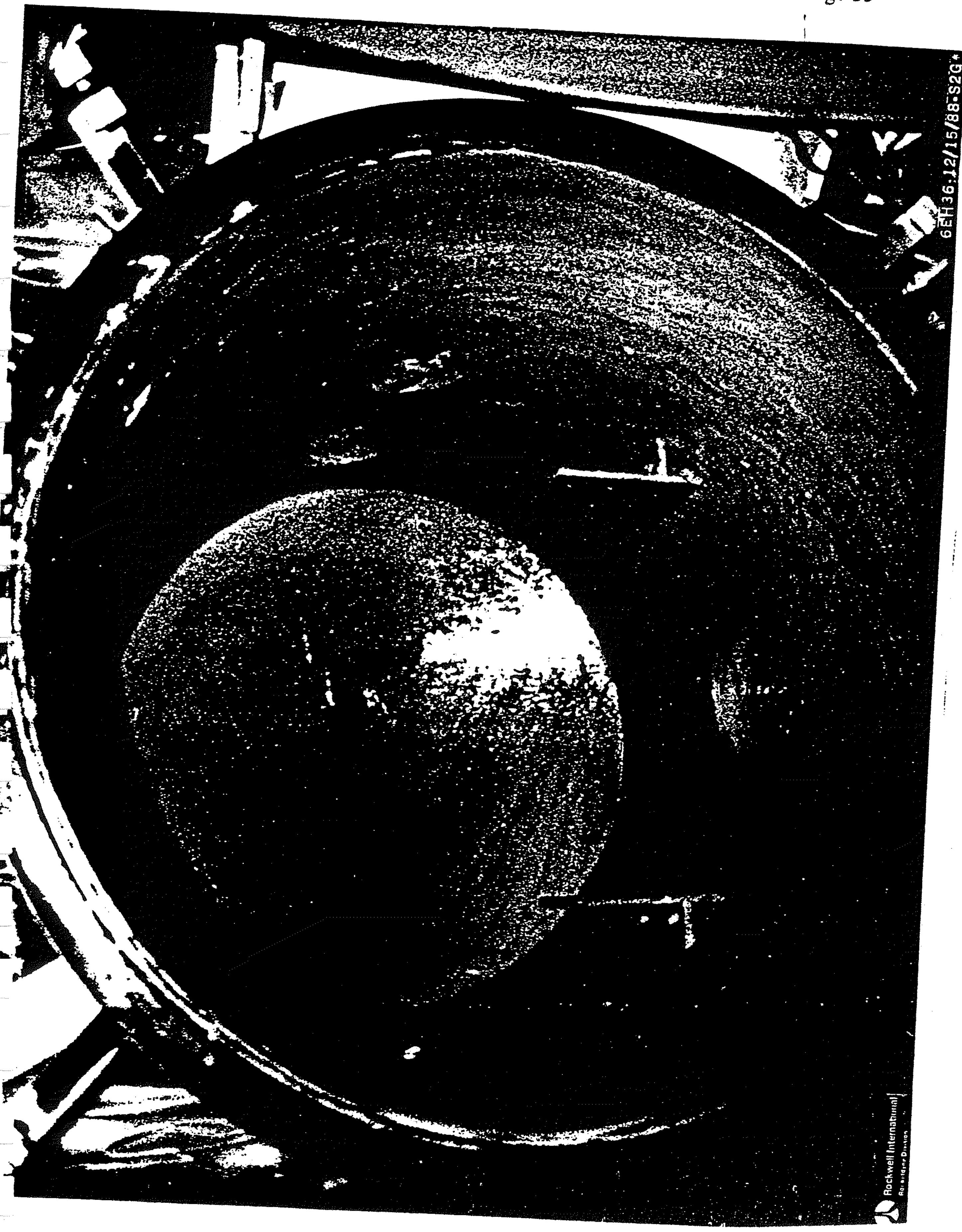
Room 101 – Furnace System and Radioactive Exhaust



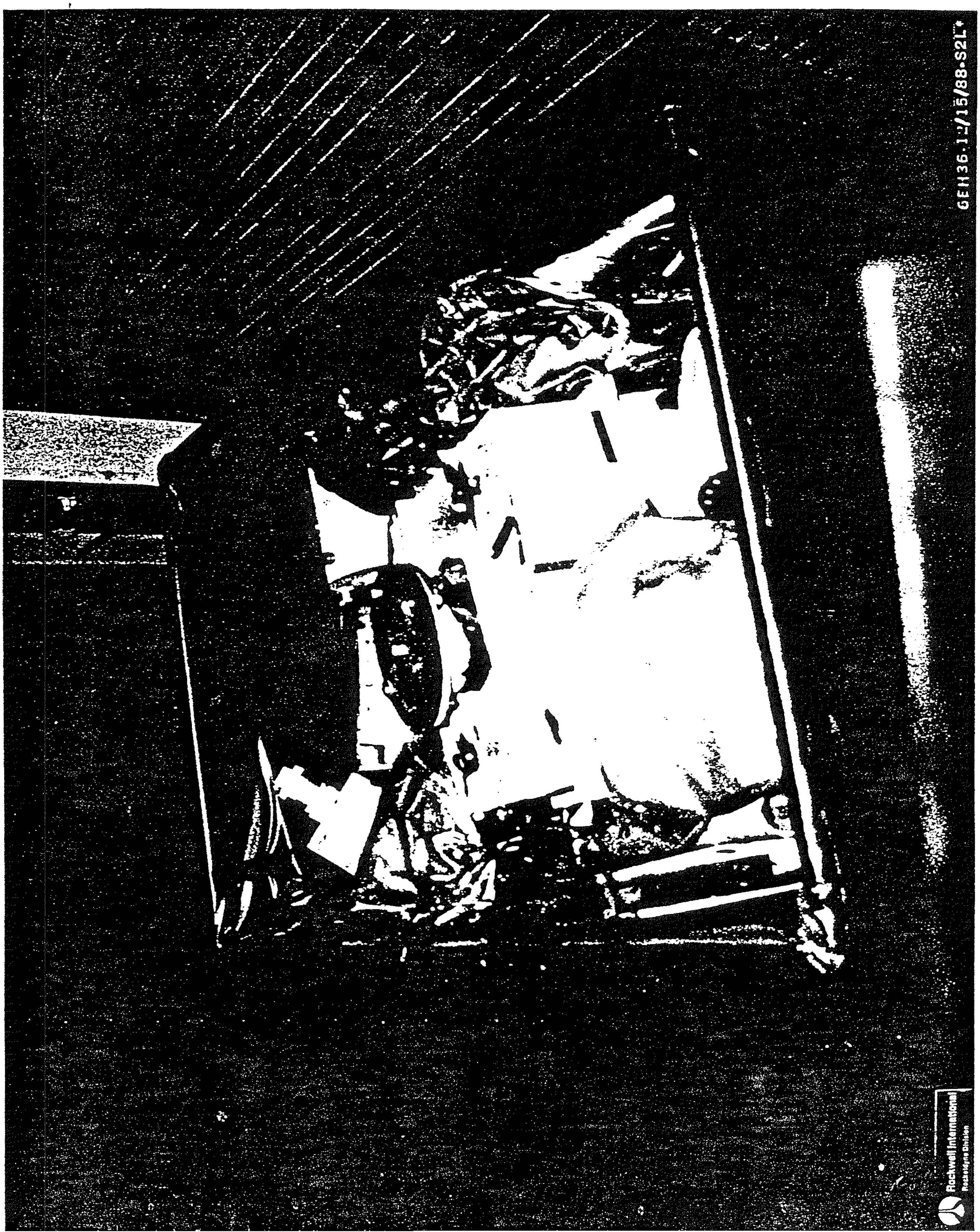
Room 102A—After Decontamination



Radioactive Exhaust System – Filter Housing and Blower



Inside of Furnace – Painted Surface



Packaged LSA Waste – Furnace System Components

Internal Letter

Rockwell International

Date May 10, 1976

No.

TO W. F. Heine
Address 713, 071-NB02

FROM L. Johnson
Address 779, 071-TI43

Phone 6503

Subject Final Radiation Survey - Building T-028

A final radiation survey has been conducted at the T-028 complex, to include all interior spaces and external areas (Figure 1), with the Technical Associates PUG-1 and Eberline E-510 with the 7 mg/cm^2 absorber probe. A final radiation survey summary is attached and the maximum radiation level detected with the 7 mg/cm^2 absorber probe was 0.08 mrad/hr at 1 cm. (General background was 0.02-0.04 mr/hr.) The maximum removable contamination level is 0 $dpm/100cm^2$ α and $<$ 60 dpm/100 cm^2 $\beta - \gamma$.

The T-028 complex is hereby certified to be clean of all contamination and activation as set forth by D&D Program Document SRR-704-900-001 of December 9, 1974, Revision B and released for unrestricted use.

RJ Tuttle for
L. Johnson
Radiation and Nuclear Safety

nht:1/2

Enclosures

cc w/o enclosures:

J. M. Harris T034
W. R. McCurnin T020

cc w/enclosures:

L. Johnson TI43
R. K. Owen TI43
R. J. Tuttle NB13
B. F. Ureda NB02

FORM 131-R REV. 2-73

T-028 STIR

FINAL RADIOLOGICAL SURVEY SUMMARY

	LOCATION	SURVEY TYPE	TOTAL SMEARS	MAXIMUM REMOVABLE CONTAMINATION LEVEL	MAXIMUM RADIATION LEVEL
1.	Office Area	A&B	250	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04 mrad/hr[a]
2.	Control Room	A&B	270	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04 mrad/hr[a]
3.	Change Room	A&B	160	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04 mrad/hr[a]
4.	Darkroom	A&B	120	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.03 mrad/hr[a]
5.	Laboratory	A&B	265	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04 mrad/hr[a]
6.	Reactor Room	A&B	280	0 dpm/100 cm^2 α <60 dpm/100 cm^2 β-γ	0.04 mrad/hr[a]
7.	Stairway & Tunnel	A&B	95	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04 mrad/hr[a]
8.	Test Vault	A&B	760	0 dpm/100 cm^2 α <50 dpm/100 cm^2 β-γ	0.07 mrad/hr[a]
9.	Exhaust System	A&B	100	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04 mrad/hr[a]
10.	Cooling System Area	B			0.04 mrad/hr[a]
11.	Blacktop Surfaces	B			0.04 mrad/hr[a]
12.	Stairway	B			0.08 mrad/hr[a]
13.	Reactor Cavity & Thermal Column	C	23.7 ± 2.6 pCi/gm Beta (Soil) 19.0 pCi/gm Beta (Concrete)		

A-Smear
B-Survey Meter (PUG-1)
C-Radiometric-BETA(LB)
a-Total radiation reading with E-510 and 7 mg/cm^2 absorber probe

NOTE: General background level - 0.02-0.04 mrad/hr

T-028 STIR

INTERNAL AND EXTERNAL SURVEY

LOCATIONS

TABLE 1

mrad/hr.

1.	0.03	21.	0.06	41.	0.04	61.	0.03
2.	0.03	22.	0.05	42.	0.04	62.	0.03
3.	0.03	23.	0.05	43.	0.04	63.	0.03
4.	0.02	24.	0.04	44.	0.04	64.	0.04
5.	0.02	25.	0.04	45.	0.04	65.	0.04
6.	0.02	26.	0.04	46.	0.03	66.	0.04
7.	0.02	27.	0.04	47.	0.03	67.	0.04
8.	0.02	28.	0.04	48.	0.03	68.	0.03
9.	0.02	29.	0.04	49.	0.03	69.	0.03
10.	0.02	30.	0.03	50.	0.04	70.	0.04
11.	0.03	31.	0.04	51.	0.04	71.	0.04
12.	0.03	32.	0.04	52.	0.04	72.	0.04
13.	0.03	33.	0.03	53.	0.03	73.	0.04
14.	0.02	34.	0.03	54.	0.03	74.	0.04
15.	0.02	35.	0.03	55.	0.04	75.	0.04
16.	0.03	36.	0.03	56.	0.04	76.	0.04
17.	0.03	37.	0.03	57.	0.04	77.	0.04
18.	0.04	38.	0.03	58.	0.04	78.	0.03
19.	0.04	39.	0.03	59.	0.03	79.	0.04
20.	0.08	40.	0.04	60.	0.03	80.	0.04

NOTE: Background - 0.02 - 0.04 mrad/hr.

TABLE 2

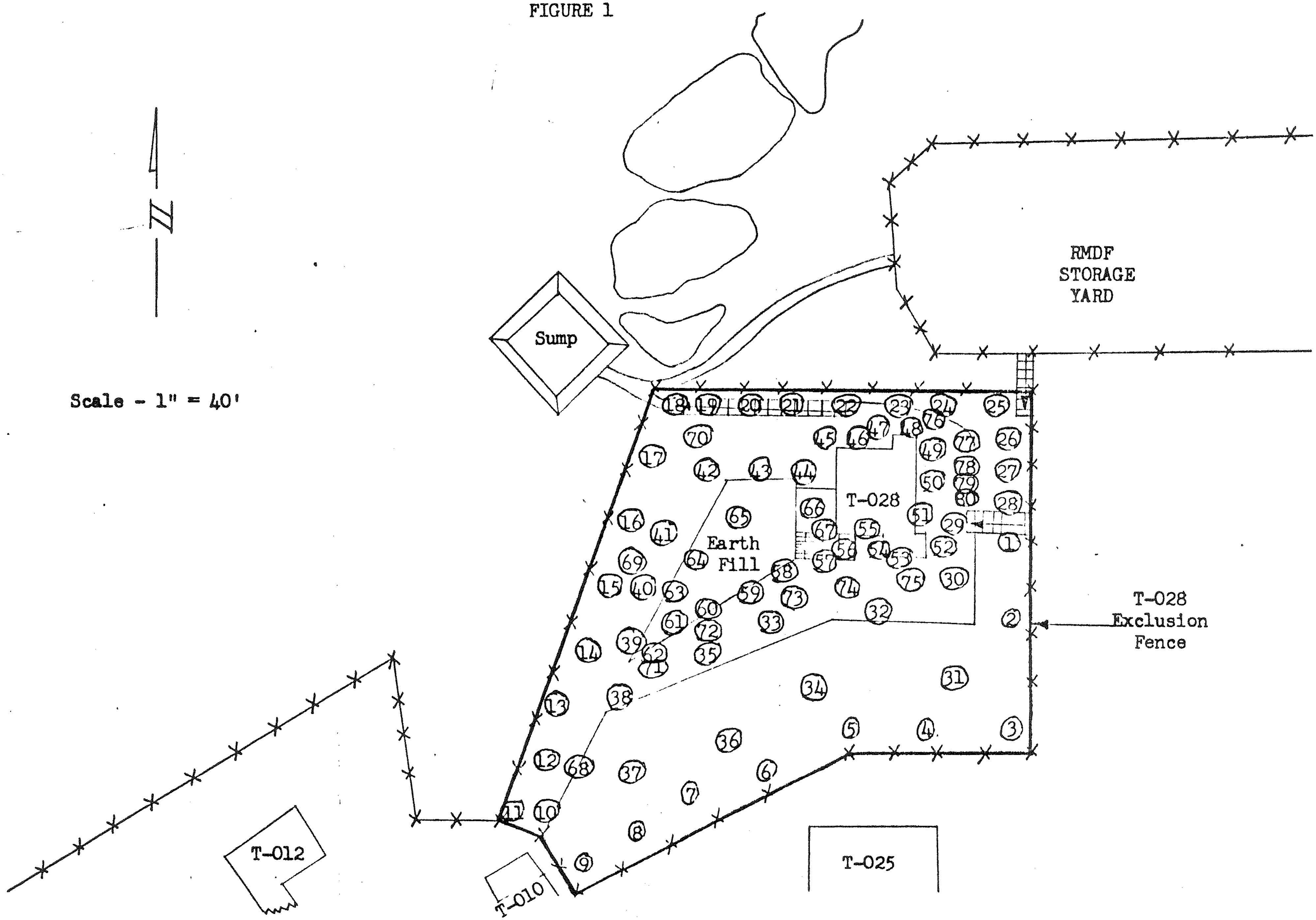
mrad/hr.

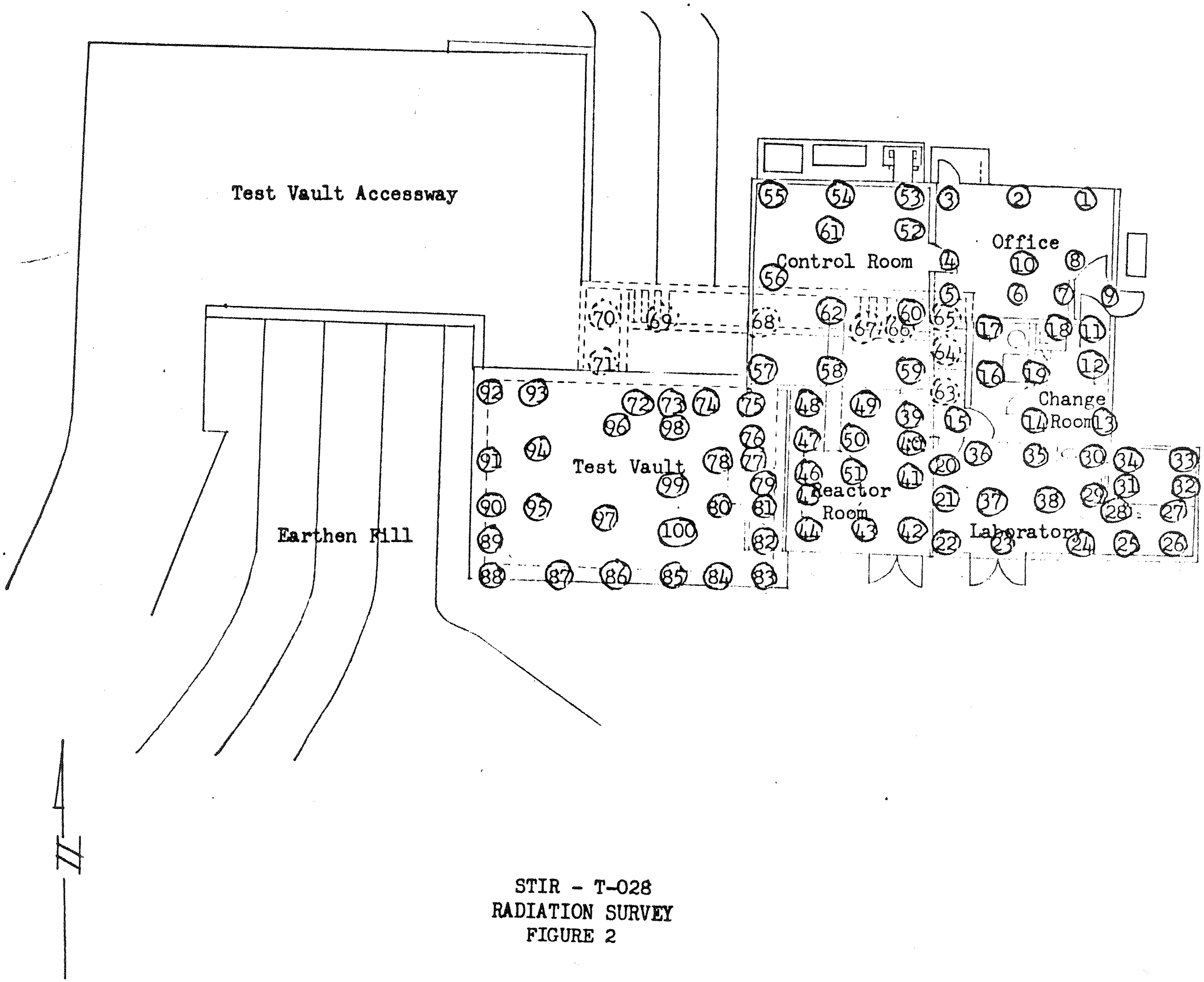
1.	0.03	17.	0.03	33.	0.03	49.	0.03	65.	0.03
2.	0.04	18.	0.04	34.	0.03	50.	0.03	66.	0.04
3.	0.03	19.	0.04	35.	0.03	51.	0.03	67.	0.04
4.	0.03	20.	0.04	36.	0.03	52.	0.04	68.	0.04
5.	0.04	21.	0.03	37.	0.03	53.	0.03	69.	0.03
6.	0.03	22.	0.03	38.	0.03	54.	0.03	70.	0.03
7.	0.04	23.	0.03	39.	0.03	55.	0.04	71.	0.03
8.	0.03	24.	0.04	40.	0.04	56.	0.04	72.	0.04
9.	0.03	25.	0.04	41.	0.03	57.	0.04	73.	0.04
10.	0.04	26.	0.04	42.	0.03	58.	0.03	75.	0.04
11.	0.03	27.	0.04	43.	0.03	59.	0.03	76.	0.04
12.	0.03	28.	0.03	44.	0.03	60.	0.03	77.	0.07
13.	0.04	29.	0.04	45.	0.04	61.	0.04	78.	0.05
14.	0.04	30.	0.04	46.	0.04	62.	0.04	79.	0.04
15.	0.04	31.	0.03	47.	0.03	63.	0.03	80.	0.05
16.	0.04	32.	0.03	48.	0.03	64.	0.03	81.	0.07

82.	0.04	92.	0.04
83.	0.04	93.	0.04
84.	0.04	94.	0.04
85.	0.04	95.	0.04
86.	0.04	96.	0.04
87.	0.04	97.	0.04
88.	0.04	98.	0.04
89.	0.04	99.	0.04
90.	0.04	100.	0.04
91.	0.04		

NOTE: Background - 0.03 - 0.04 mrad/hr.

STIR - T-028
RADIATION SURVEY
FIGURE 1





STIR - T-028
RADIATION SURVEY
FIGURE 2

R. K. Owen
T 143

AI-ERDA-13168

STIR FACILITY
DECONTAMINATION AND DISPOSITION
FINAL REPORT

ERDA Research and Development Report

Prepared for the United States
Energy Research and Development Administration,
Environmental Controls Technology Division,
under Contract Number AT(04-3)-701

Rockwell International

Atomics International Division
8900 DeSoto Avenue
Canoga Park, California 91304

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AI-ERDA-13168
WASTE MANAGEMENT
UC-70

STIR FACILITY
DECONTAMINATION AND DISPOSITION
FINAL REPORT

B. F. Ureda

Rockwell International

Atomics International Division
8900 DeSoto Avenue
Canoga Park, California 91304

CONTRACT: AT(04-3)-701
ISSUED: AUGUST 26, 1976

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CONTENTS

TABLES

TABLES

FIGURES

FIGURES

FIGURES

ABSTRACT

The decontamination and disposition (D&D) of Building 028, Shield Test Irradiation Reactor (STIR) facilities, are complete. The core tank, the activated concrete structures surrounding the core tanks, the thermal column, the reactor shield, the test vault carriage, the water cooling systems, and the water shield door were removed, and the facility exhaust system was partially dismantled. The facilities were decontaminated to levels which were as low as practicable, but in all cases to levels below the limits described as acceptable for future unrestricted use. The more significant D&D activities are summarized, and special techniques are noted. Results of the radiological monitoring in support of the D&D operations and of the final radiological survey are presented.

I. INTRODUCTION

The Shield Test and Irradiation Reactor (STIR) located at the AI Santa Susana field laboratories was a 1-Mwt pool-type reactor, used primarily to conduct basic shielding experiments. The reactor was operated with a 50-kwt capability between 1961 and 1964, and with a 1-Mwt capability between 1964 and 1972. The "Hazards Summary Report" and "Startup and Operation" reports[1,2] provide additional detail of the facility history. The fuel elements were removed and the pool water was drained in June 1973. The STIR facilities were declared excess, and the dismantling proceeded as described in the "Decontamination and Disposition (D&D) of Facilities Program Plan," PP-704-990-002. The actual dismantling of STIR began on September 24, 1975, and was completed March 26, 1976. Contaminated and irradiated components and structures associated with the reactor, water cooling system, thermal column, test carriage, and facility exhaust system were removed, packaged, and shipped to Beatty, Nevada for disposal by land burial. Nonradioactive peripheral equipment such as the cooling tower, shield door, and film conveyor were removed as salvage. Floor and wall openings resulting from the D&D operations were filled and covered with concrete as required to restore the facility to a safe condition.

The dismantling activities were conducted with a minimum of exposure to personnel, in keeping with "as low as practicable" (ALAP) principles. Upon completion of the facility decontamination and disposition, a radiological survey verified that the facility had been decontaminated to levels as low as practicable below the limits (Table 1) described as acceptable for future unrestricted use.

TABLE 1

CONTAMINATION LIMITS FOR DECONTAMINATION AND DISPOSAL OF BUILDING 028, STIR FACILITIES

	Total	Removable
Beta-Gamma Emitters	0.1 mrad/hr at 1 cm with 7 mg/cm^2 absorber	100 $dpm/100\ cm^2$
Alpha Emitters	100 $dpm/100\ cm^2$	20 $dpm/100\ cm^2$

This report summarizes the more pertinent decontamination and disposition activities, discusses special techniques used, and reviews major problems and their resolution.

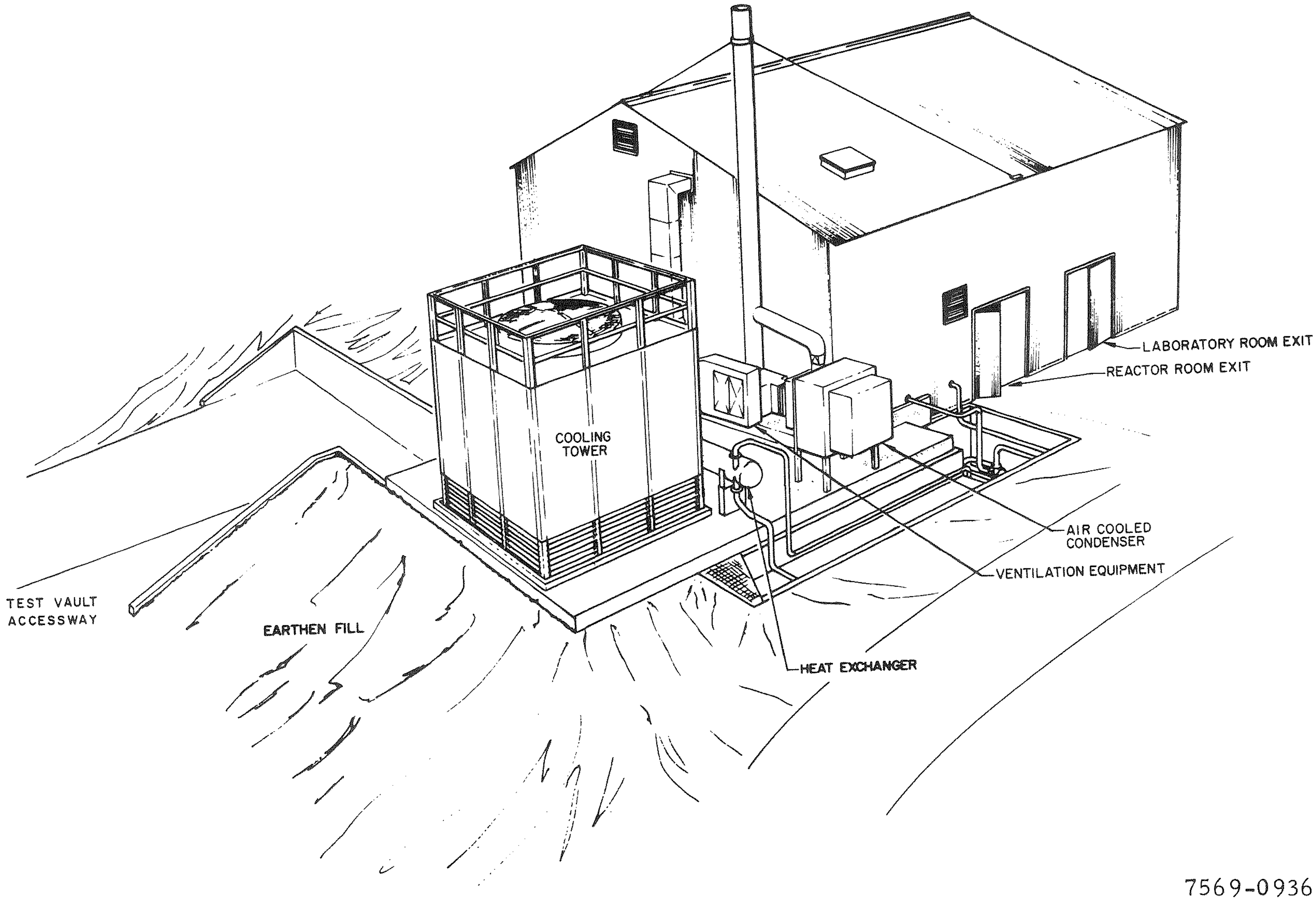

Figure 1. STIR Architectural Elevation

II. FACILITY DESCRIPTION

The STIR facility, shown in Figures 1 through 4, consisted essentially of a reactor core tank, control room, cooling system, test vault, graphite thermal column, fission plate, test carriage, and radiological shielding. The facility was deactivated in 1973, at which time the fuel and fission plate were removed and the reactor control room was dismantled.

A. COOLING SYSTEM

The reactor was cooled by natural convection flow of the pool water. There were two separate cooling systems for the pool water. The 50-kw auxiliary system consisted of a 15-ton capacity water refrigeration installation (Figure 5) and an airblast heat exchanger. For operations above 50 kw, a 1-Mw cooling system was used (Figure 6). The 1-Mw system consisted of a heat exchanger and a one-cell induced draft counterflow cooling tower. The water purification loop consisted of a particulate filter, a mixed-bed demineralizer, pumps, and control valves.

B. REACTOR

The reactor core was located at the bottom of a 5-ft diameter by 20-ft deep water-filled aluminum tank (Figure 4). Although the fuel elements had been removed in 1973, the grid plate and support structure remained in place. The tank sat in a concrete well, with a 6-in. annulus of pea gravel between the vessel and the concrete. The lower end of the tank mated with the thermal column which led to the test vault. A 7-in. lead shutdown shield filled with lead shot was located at the thermal column and tank interface. The center of the shield contained a 10 by 16 in. bismuth window. The thermal column was a 5-ft by 5-ft by 4-ft aluminum box, filled with 4-in. by 4-in. by 4-ft long graphite logs. Figure 7 shows the thermal column as viewed from the test vault. The wall immediately surrounding the thermal column was constructed of dense magnetite concrete.

C. TEST VAULT

The test vault contained a test carriage (Figure 8), upon which was mounted a concrete auxiliary shield also referred to as the "donut." A fission plate

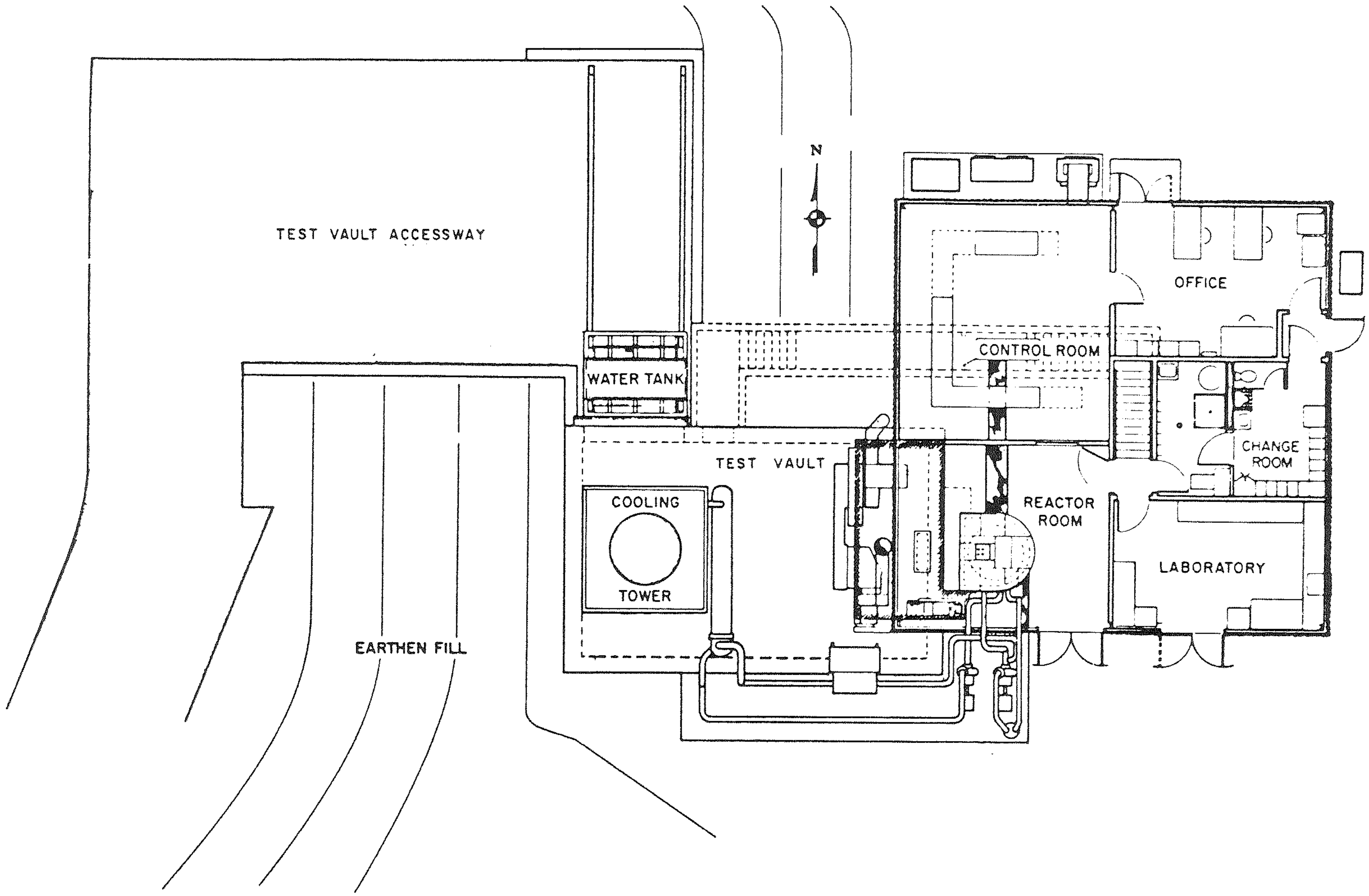

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Figure 2. STIR Architectural Floor Plan

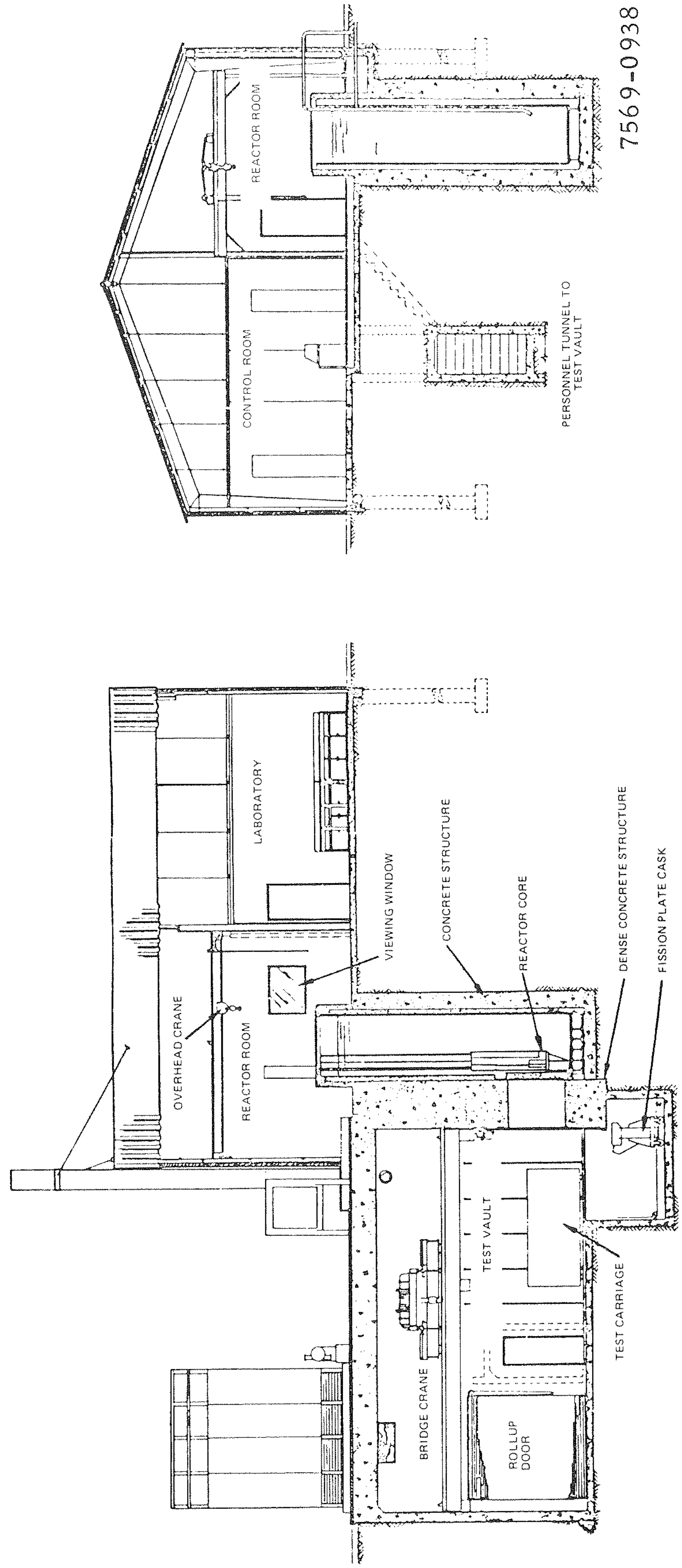

Figure 3. STIR Architectural Sections and Details

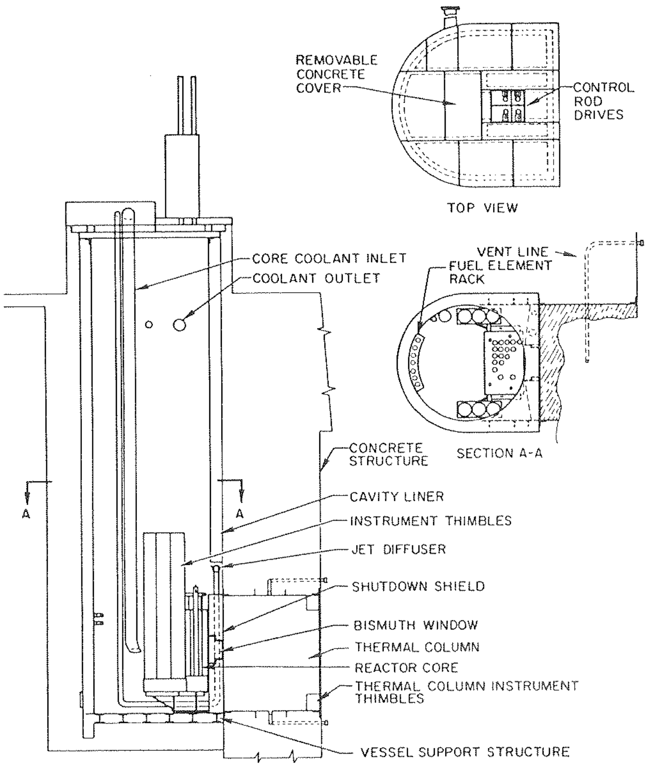

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Figure 4. Reactor Complex

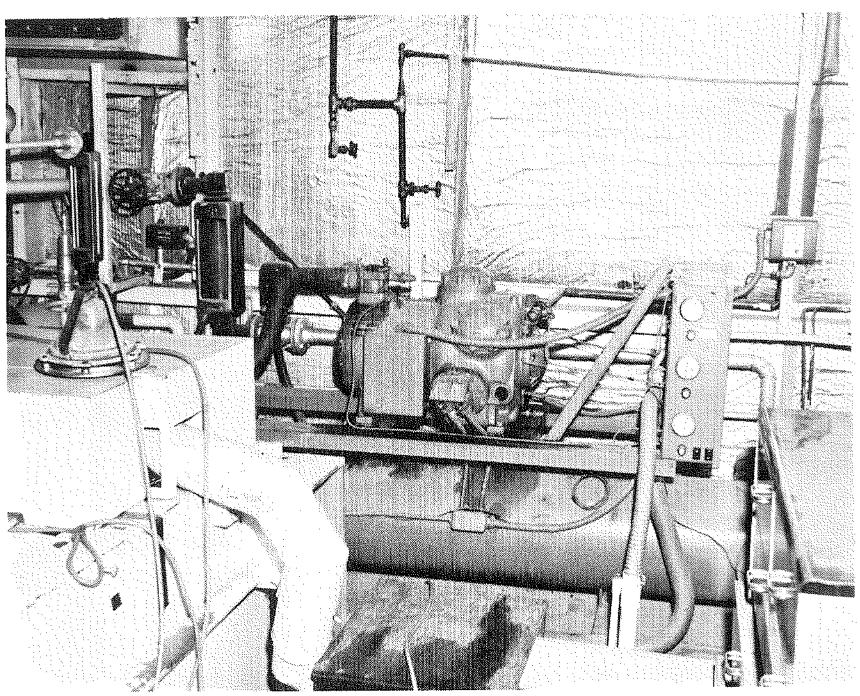

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Figure 5. 15-ton Refrigeration Water Cooling Unit

Figure 6. 1-Mw Cooling Tower and Heat Exchanger

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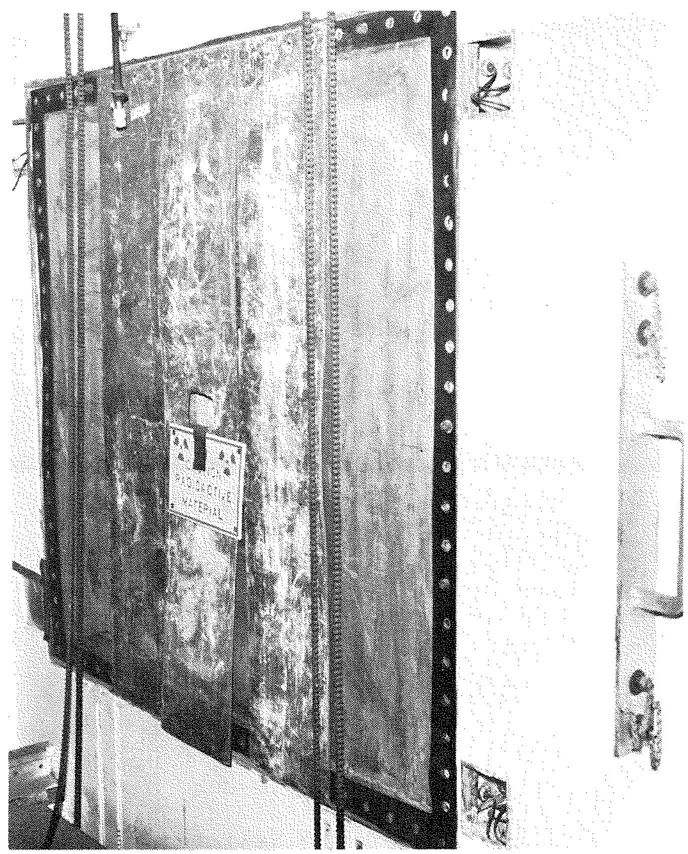

Figure 7. Thermal Column Face Viewed From Test Vault

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assembly, located in the pit directly below the thermal column area, provided a fission spectrum neutron flux source for shielding and irradiation experiments. A movable, water-filled steel tank (Figure 9) provided shielding at the overhead door of the test vault.

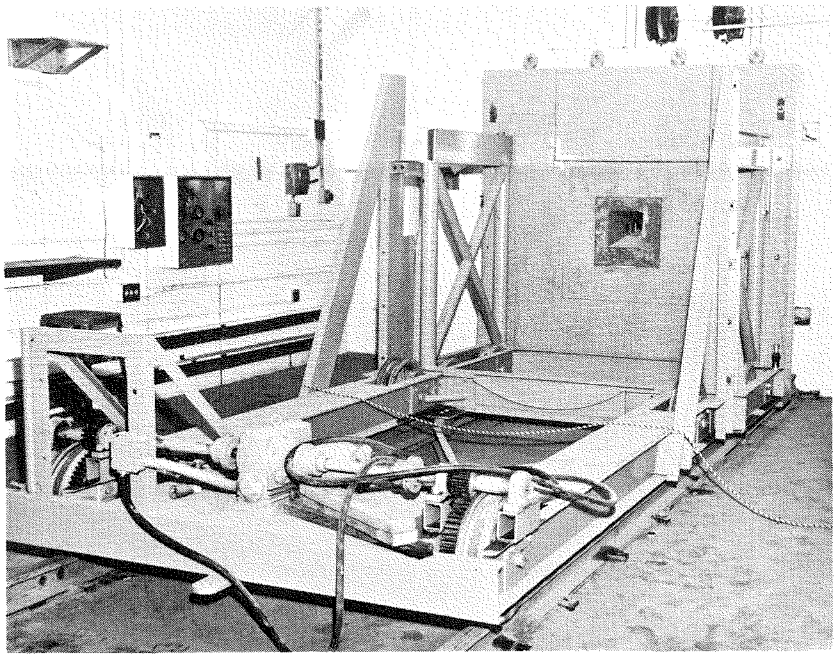

Figure 8. Test Carriage With "Donut" Positioned Against Thermal Column

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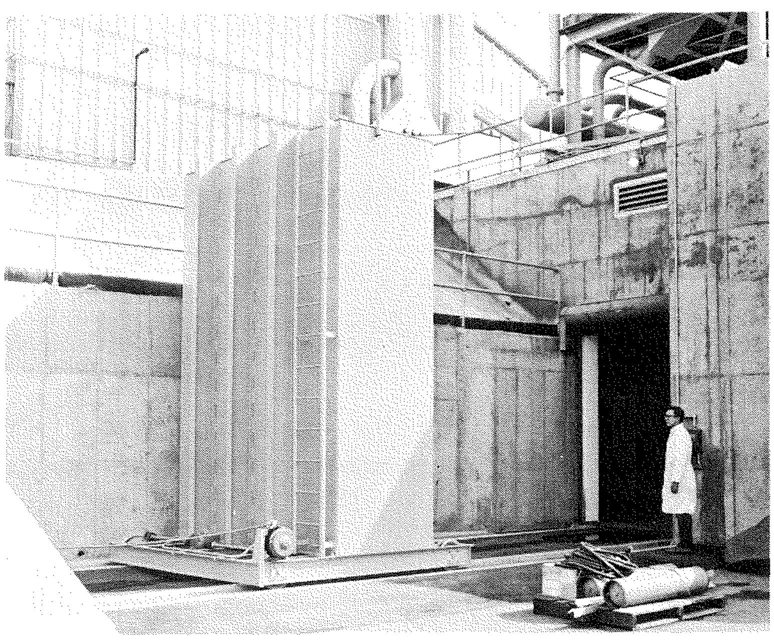

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Figure 9. Movable Water-Filled Shield Door

III. SUMMARY OF DISMANTLING ACTIVITIES

Physical dismantling of the STIR facility began on September 24, 1975, and was completed on March 26, 1976. The documentation prepared to support the dismantling activities was as follows:

1) Facilities Dismantling Plan for STIR, Building T028, FPD-704-990-004 (in Appendix)
2) Building T028 (STIR), Activity Requirement 001, Removal of Radioactive Components and Materials Excluding Concrete Structure, N704-ACR-900-001
3) Dismantling of Peripheral Equipment for STIR, Building T028, Detailed Working Procedure, N704-DWP-990-005
4) Radiological Survey Plan in Support of D&D Program Operations at Building T028 (STIR), N704-TP-990-004
5) Detailed Working Procedure for Decontamination and Dismantling for Shield Test Irradiation Reactor Building T028, Excluding Concrete Structures, N704-DWP-990-006
6) Building T028 (STIR) Activity Requirement 002, Removal of Activated Concrete, N704-ACR-900-002
7) Detailed Working Procedure for Removal of Activated Concrete and Associated Materials From Building T028 (STIR), N704-DWP-990-007.

The documents were reviewed and approved by Quality Assurance, Operating Groups, Health, Safety, and Radiation Services (HSRS), the D&D Program Office, and the Isotopes Committee of the AI Nuclear Safeguards Review Panel. The work was performed by the AI Remote Technology Unit, which includes personnel trained to handle radioactive materials. HSRS provided health physics and safety support. Industrial Engineering coordinated demolition and salvage contractors' work and arranged for plant maintenance assistance for utility disconnections. The demolition contractors' work included breakout and removal of the activated concrete, and backfilling and sealing the reactor enclosure. The salvage contractor removed peripheral systems. Health physics surveillance was continuous. All radioactive wastes were packaged and sent to the Radioactive Material Disposal Facility (RMDF), for shipment to Beatty, Nevada for burial.

TABLE 2

RADIOLOGICAL SURVEY OF BUILDING 028 FACILITIES BEFORE DISMANTLING

Facility	Sample Type	Maximum Specific Radiation
Floor Areas Reactor Control Room Office Area Change Room Laboratory	Smear (100 cm^2)	<5 dpm α, 50 dpm β-γ
Six Fuel Storage Cells	Smear (100 cm^2)	<20 dpm α, 60 dpm β-γ
Reactor Coolant in Pump Pit	Water	4.2×10^{-8} μCi/cc β
Leg of Cooling Tower	Filings	No activity detected
Cooling System Pipe	Filings	No activity detected
Test Carriage	Filings	25.0 pCi/gmβ
Tank-Concrete Enclosure Annulus	Pea Gravel (top surface)	24.1 pCi/gmβ
Core Tank Walls	Paint	289 pCi/gmβ
Manipulator	Smear (100 cm^2)	<50 dpm β-γ 1.0 mrad/hr
Reactor Grid Plate	Survey Meter (Jordan with Remote Detector)	1.2 rad/hr
Bismuth Window	Survey Meter (Juno)	430 mrad/hr

A. PREPARATIONS

The existing personnel change room in Building 028 was reactivated and resupplied. An HSRS work station was established in the office area, and equipped with radiation detection instrumentation. Personnel dosimeters, portable radiation survey instruments, respiratory protective devices, airborne particulate radioactivity samplers, and protective clothing were provided. A radiological survey of the facility was conducted before work was begun. The survey results (Table 2) show that the radiation sources were essentially confined to the reactor vessel internals and surrounding materials, thermal column, and test carriage.

Before beginning the dismantling, all personnel who were to be associated with the work were fully briefed by the unit manager on the scope of the work, the radiation hazards expected, and the precautions necessary to safely accomplish the dismantling tasks. A familiarization review of the Detailed Working Procedures, and the requirements for keeping the personnel radiation exposure levels as low as practicable, as defined in Reference 3, were also presented to the operating personnel by the unit manager.

A contract was issued to a salvage contractor for the removal of the peripheral nonradioactive equipment and materials which included the cooling tower, heat exchanger, water shield door, portions of the test carriage and associated piping.

B. PROCEDURES

The Detailed Working Procedures described the dismantling steps and delineated the activity sequence. When changes to the procedure were necessary, they were noted on the work copy of the procedure, and were instituted after review and approval. Separate procedures were prepared for removal of the peripheral equipment, the reactor systems, and the activated concrete.

A major activity in the STIR facility decontamination and disposition was the radiological monitoring and surveying of the total operation. Smear surveys, portable instrument surveys, air sampling, and radioanalyses of water, soil, and concrete were conducted.

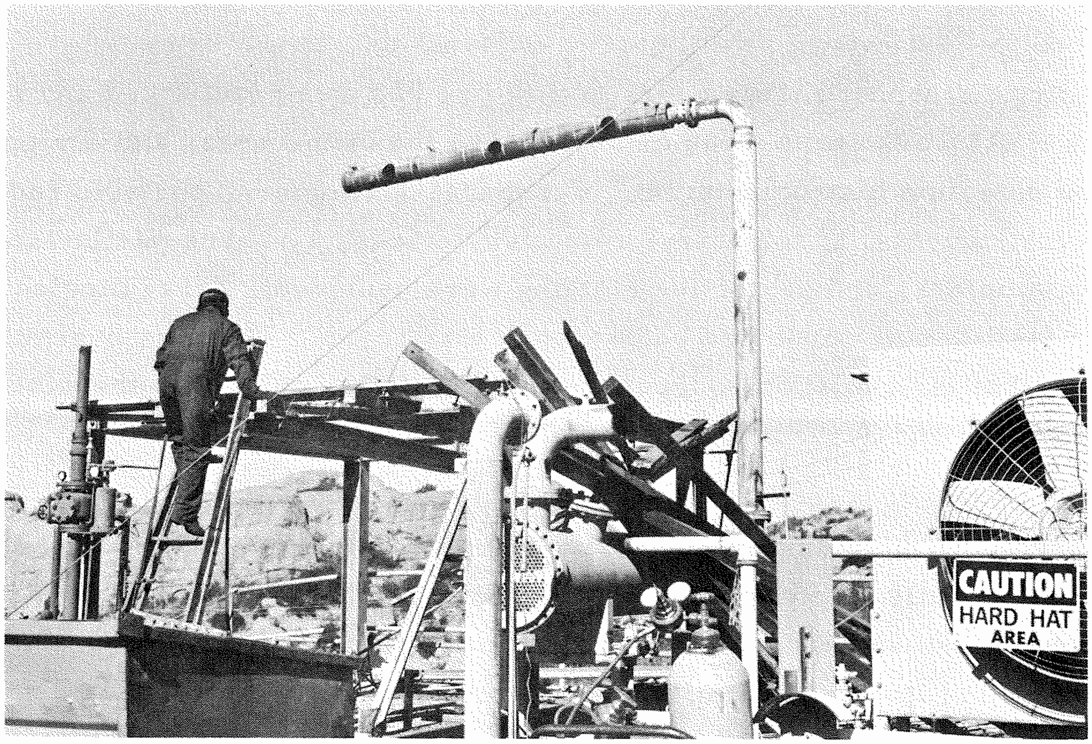

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Figure 10. 1-Mw Water-Cooling Tower and Heat Exchanger Dismantling

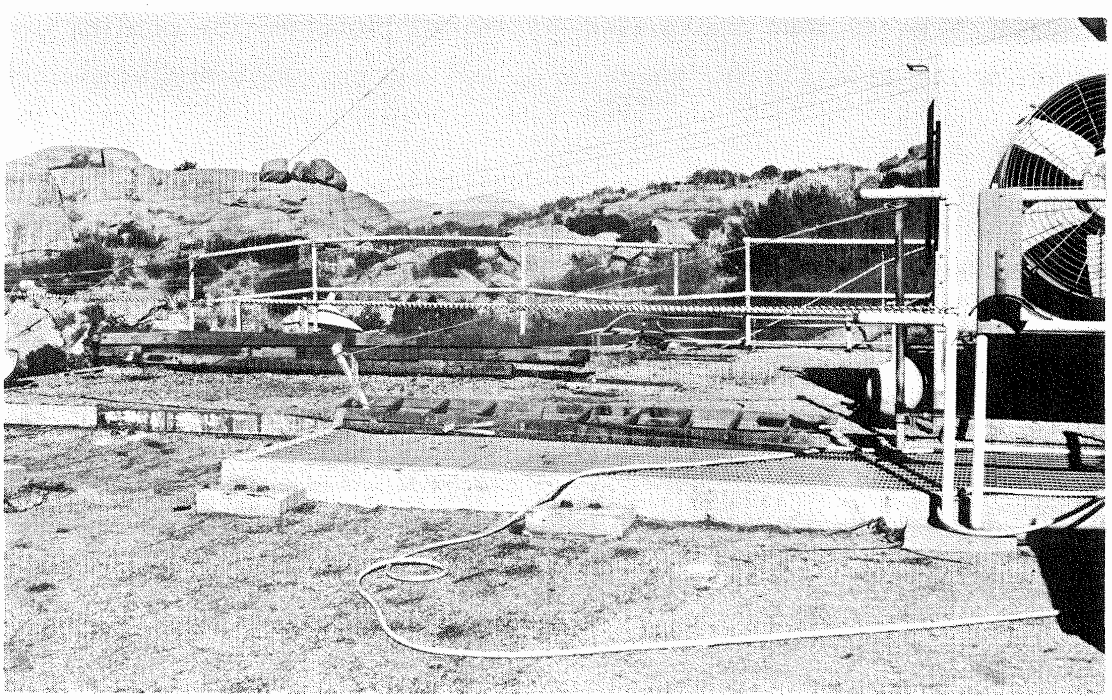

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Figure 11. Completion of Water Tower and Heat Exchanger Dismantling

1. Cooling System

The 50-kw reactor water cooling system and the primary (reactor side) of the 1-Mw reactor water cooling systems were dismantled. The secondary side of the 1-Mw system was removed by the salvage contractor. The piping, valves, and pumps were nonradioactive, and were disposed of as salvage. The 50-kw refrigeration water cooler was found to contain low levels of radioactivity in the water trapped inside the unit. The unit was removed from the facility and transferred to storage for possible reuse.

The water demineralizer and filters were cut out, packaged as radioactive waste, and sent to the RMDF for shipment to off-site burial. The nonradioactive 1-Mw water cooling tower and the 1-Mw heat exchanger and associated piping were dismantled and removed from the site by the salvage contractor. Figures 10 and 11 show stages of dismantling.

2. Reactor Vessel

The concrete shield blocks over the top of the reactor opening shown in Figure 12 were removed. These blocks were nonradioactive and were set aside for eventual burial in the reactor cavity.

Samples of the aluminum core tank walls were taken using a drill to produce metal shavings. Analysis of these samples revealed that the upper portion (11 ft) was not radioactive and that the radioactivity in the lower portion resulted mainly from neutron activation of the paint on the inside surface. Figure 13 shows the arrangement of the core tank internals: the grid plate at the right, six storage thimbles — three on each side, and coolant piping. The flexible duct at the right provided fresh air and air movement for personnel working in the tank.

The instrumentation thimbles, grid plate, and the core support structure were removed from the tank. These components were packaged as radioactive waste and sent to the RMDF for shipment to off-site burial.

Holes were sawed in the radiation shield, Figure 14, and the lead shot (about 3000 lb) was removed, placed in small drums, and sent to the RMDF for shipment to off-site burial. The annulus between the core tank and the concrete enclosure was opened and the pea gravel was removed by vacuuming and placing in 55-gal. drums. Figure 15 shows the removal technique. Radioactivity in the

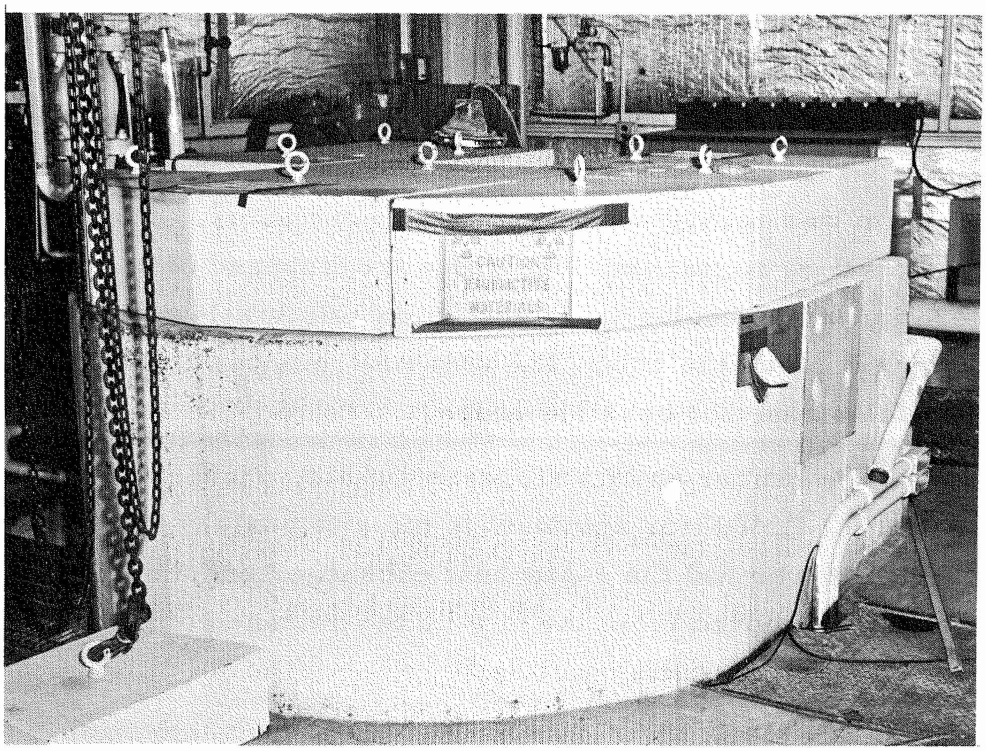

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Figure 12. Shield Blocks and Upper Reactor Opening

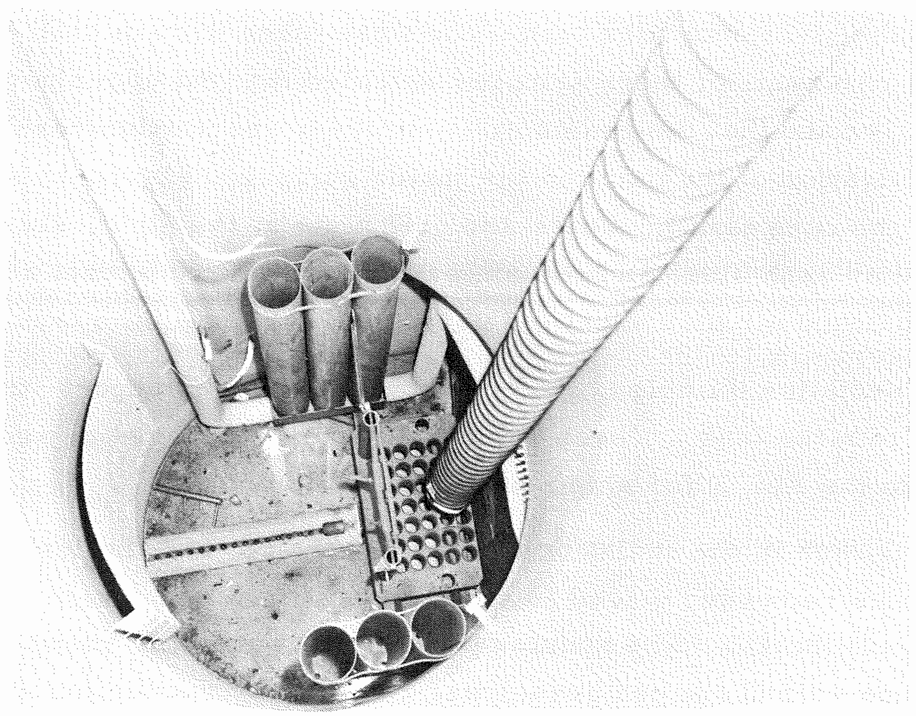

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Figure 13. Reactor Vessel Internals

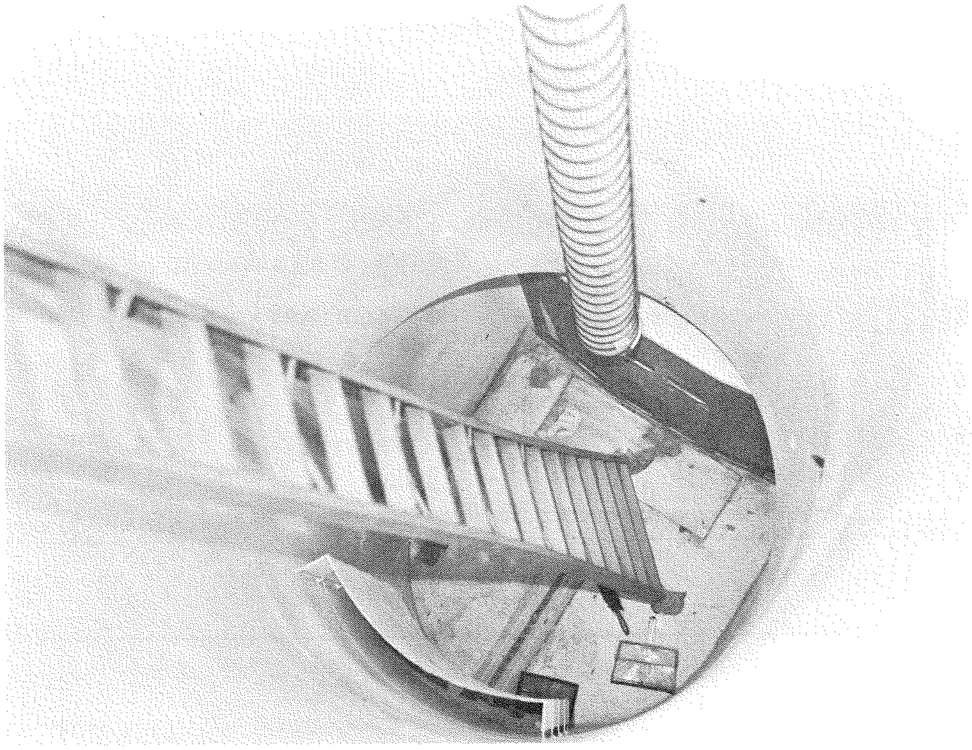

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Figure 14. Saw Holes in Shield for Lead Shot Removal

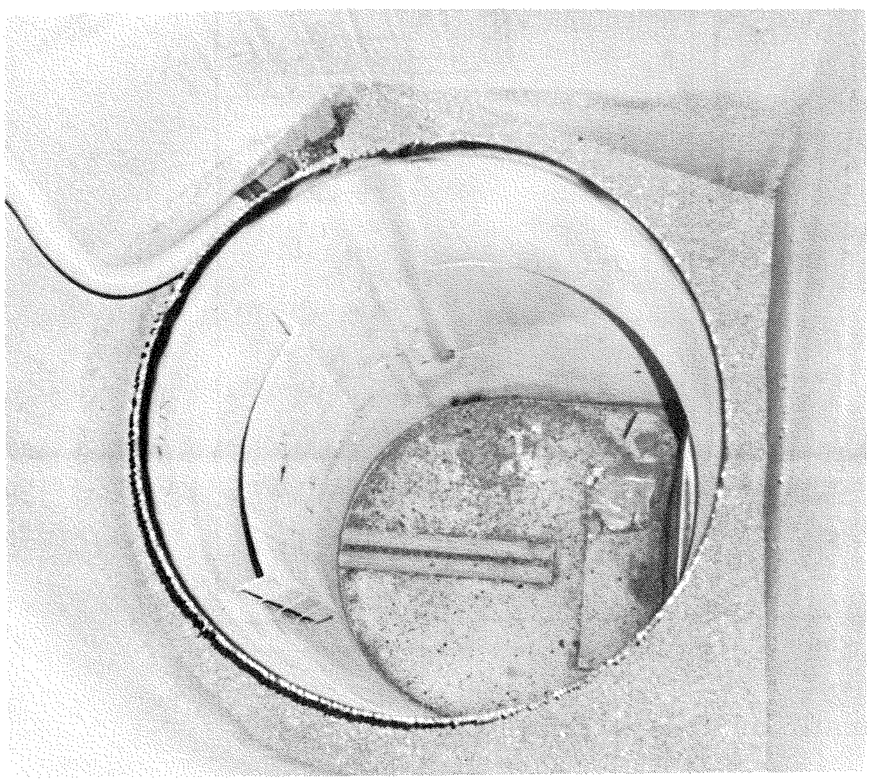

Figure 15.
Pea Gravel Removal
From Annulus

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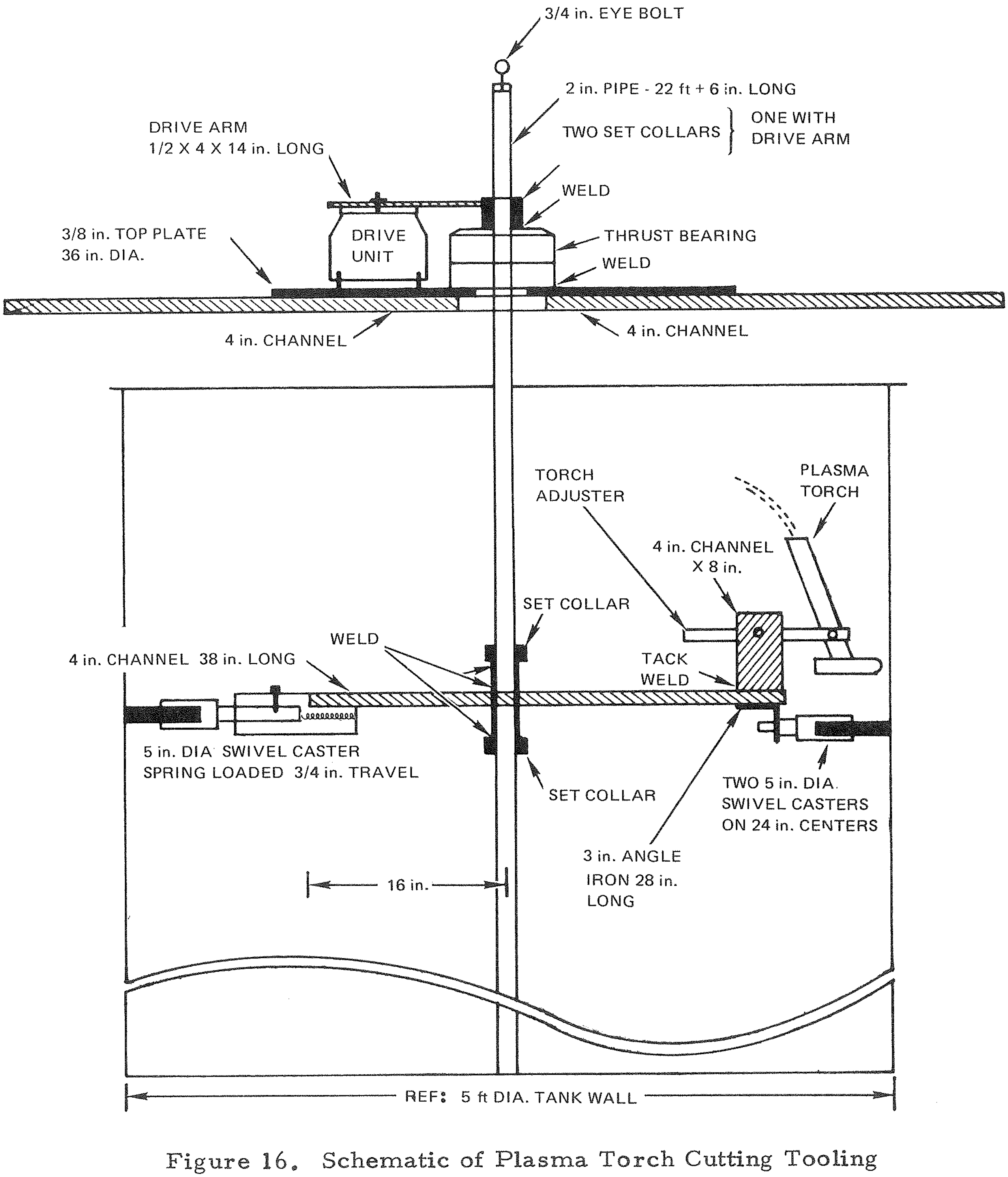

Figure 16. Schematic of Plasma Torch Cutting Tooling

pea gravel in the annulus for the first 10 ft below the tank top was limited to natural radioactivity: 24.1 pCi/gm. The gravel below 10 ft was determined to be neutron activated and was handled as radioactive waste. Radiation levels of up to 15 mrad/hr were measured at the surface of the drums of activated gravel.

To minimize the technician's working time inside the core tank, where the radiation was highest, and to facilitate the required vessel cutting operations, a special plasma torch cutting fixture, Figures 16, 17, 18, and 19, was assembled to cut the upper nonactivated portion of the core tank. Figure 16 is a schematic showing the general arrangement and operation of the special fixture. Figure 17 shows the plasma torch mounted on the radial arm inside the core tank. Figure 18 shows the support and drive structure at the top of the tank. Figure 19 shows the plasma torch power supply. Once the torch was set up, the 1/2-in. thick aluminum tank wall was cut in approximately 15 min for each circumferential cut. Three cuts were made. The three vessel sections shown in Figure 20 were disposed of as nonradioactive scrap.

The lower 9-ft radioactive portion of the tank was cut into two sections. The bottom section is shown in Figure 21 resting on the top of the reactor enclosure where it was placed after hoisting from below. A longitudinal cut of the section was made to facilitate packaging. Figure 22 shows the sections in the shipping box. The shutdown shield with the bismuth windows at the center was an integral part of the lower section of the tank. The radiation level at the bismuth window was 600 mrad/hr. In packaging the lower section of the tank for shipment, special shielding was placed over the bismuth window. The thermal column — core tank interface plate was sawed, removed, and placed in a shipping box. The tank support structure at the bottom of the reactor enclosure and the remaining pea gravel were removed. Figure 23 shows the reactor enclosure after the tank support structure and thermal column interface plate were removed. Figure 24 shows the thermal column liner.

3. Test Vault Dismantling

Test vault dismantling began with the removal of the test carriage. The test carriage concrete "donut" was removed from the carriage and stored in the rear of the vault. The test carriage (Figure 8) was disassembled and removed

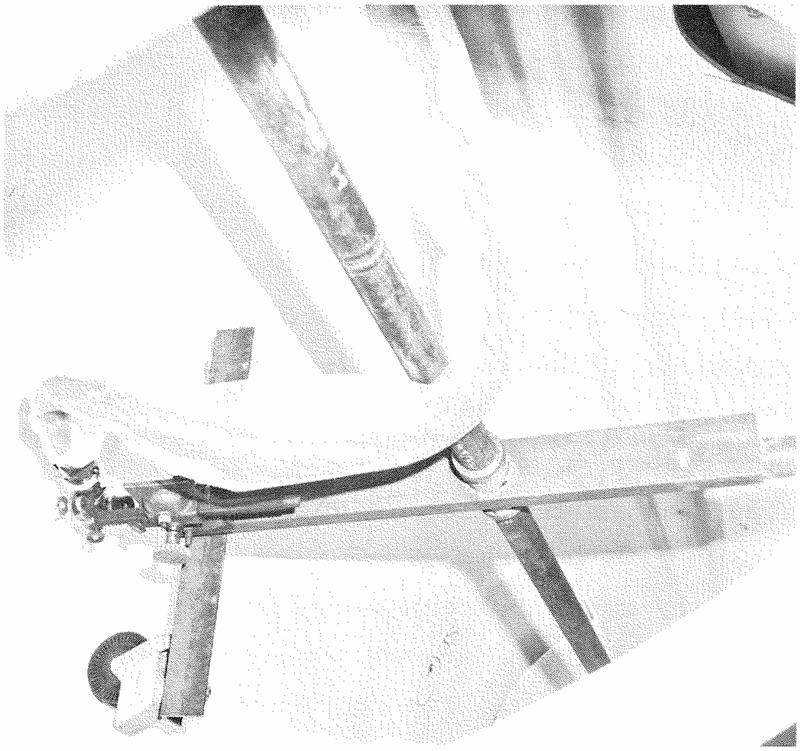

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Figure 17. Plasma Torch Cutting Fixture

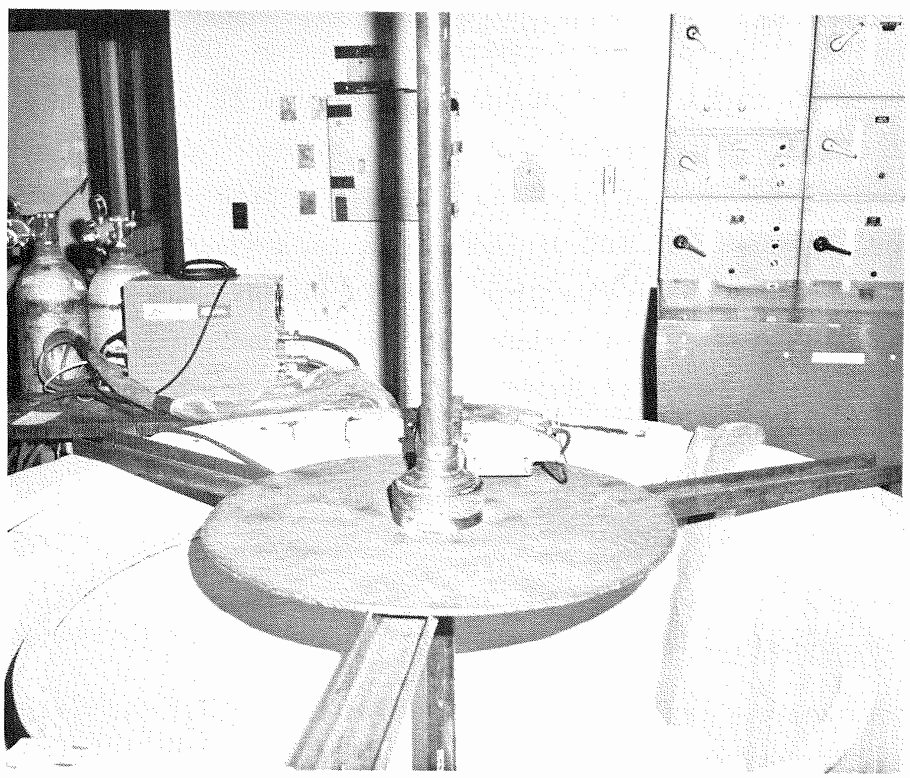

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Figure 18. Plasma Torch Support Fixture for Cutting Reactor Vessel

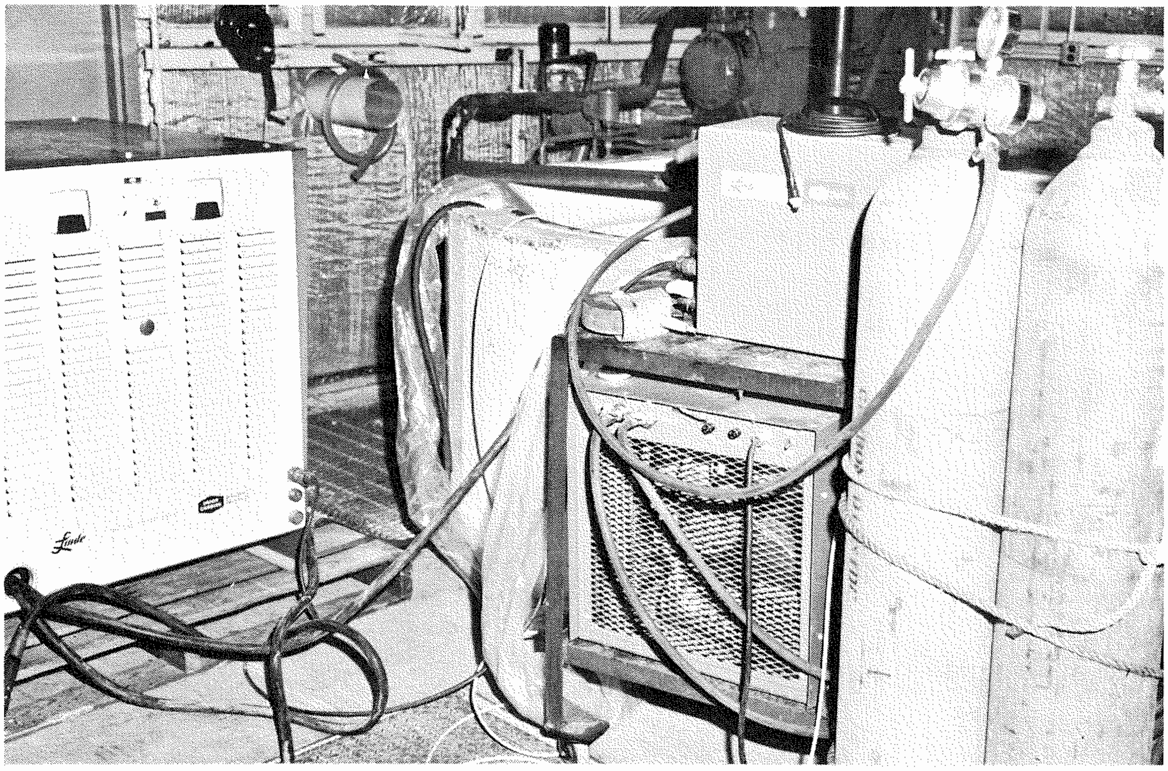

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Figure 19. Plasma Torch Power Supply and Gas Supply

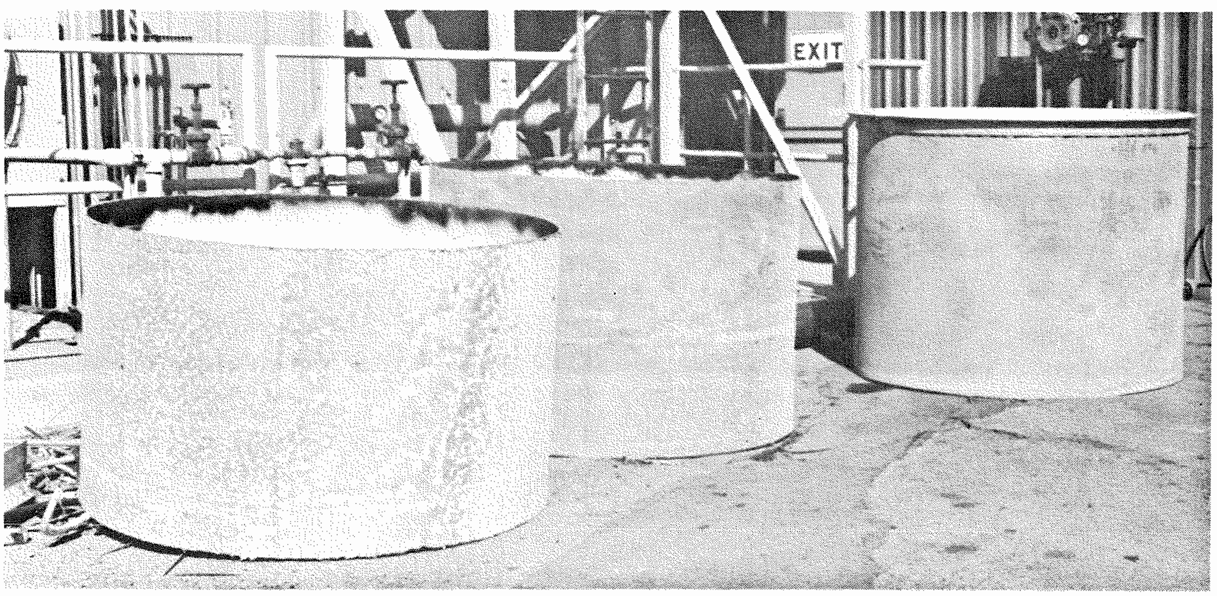

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Figure 20. STIR Reactor Vessel Sections Cut With Plasma Torch

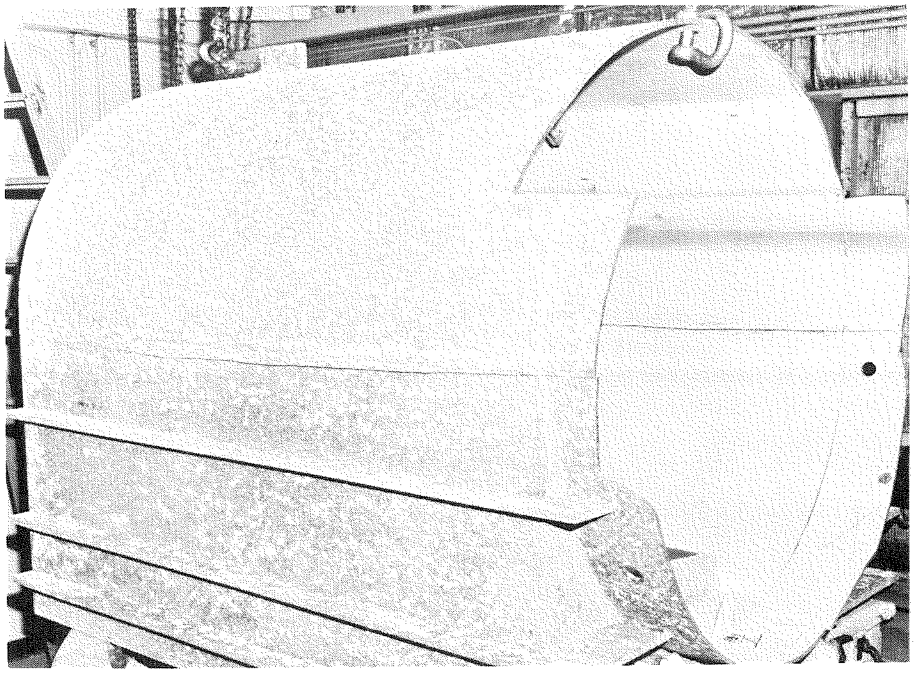

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Figure 21. STIR-Longitudinal Cut of Reactor Vessel After Removal From Pit

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Figure 22. STIR-Bottom Portion of Reactor Vessel Cut Longitudinally and Boxed for Shipment to Burial

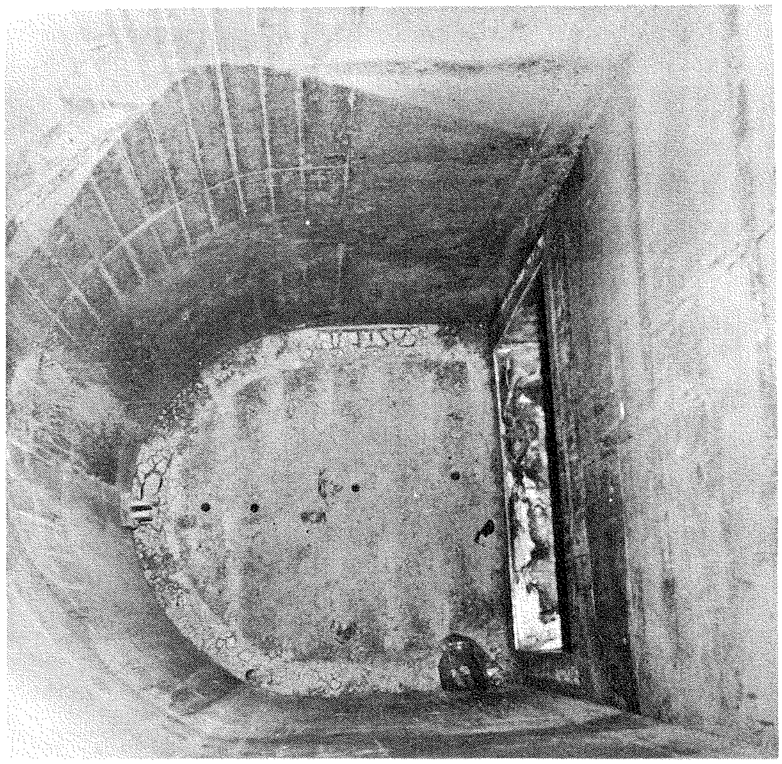

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Figure 23. STIR-Reactor Concrete Enclosure After Removal of Vessel, Shield and Bismuth Window

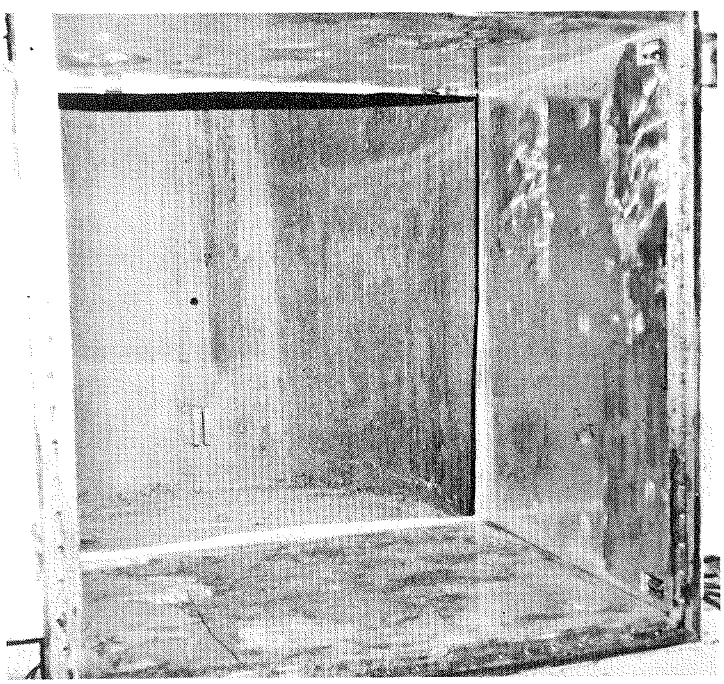

Figure 24.

Thermal Column Liner Looking Into Reactor Enclosure

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from the test vault. The salvage contractor cut up the noncontaminated portions of the test carriage, and removed them from the site. A radiological survey of the test carriage scrap material was made prior to release from the site. The forward end of the carriage that supported the "donut" was found to be neutron activated. This section was cut off, disassembled, and placed in boxes for shipment to the land burial site. The test carriage rails on the test vault were removed and disposed of as scrap.

The fission plate pit was opened, radiologically surveyed, and found to be free of radioactivity. The rails and structural support hardware were removed from the pit. The steel cover plates were disposed of as radioactive waste.

Plastic sheeting was spread over the floor area, directly in front of the thermal column, in preparation for disassembly of the thermal column. The lead shielding was removed from the thermal column front face (Figure 7). The boral sheet, which was nonradioactive, was then removed, exposing the graphite logs. Radiation levels associated with the graphite logs ranged from 15 mrad/hr at the ends exposed to the test vault to 50 mrad/hr at the ends nearest to the reactor. The graphite logs were removed, placed in shipping containers and sent to the RMDF for subsequent shipment to Beatty, Nevada for burial. Six thousand pounds of graphite logs were removed. The thermal column liner (Figure 25) was wiped down to remove loose contamination. The radiation level, after wiping, at the thermal column back wall was 500 mrad/hr in the center and 200 mrad/hr at the edges. The plastic sheeting on the floor was picked up and placed in the shipping boxes. The test vault area was then vacuumed.

a. Survey of Test Vault Before Activated Concrete Removal

A radiation survey of the test vault area, including the thermal column, reactor enclosure floor and walls, and the "donut" was performed prior to removal of the activated concrete. The survey was conducted using a Nuclear Chicago 2650 GM-type survey instrument with the beta shield open and readings taken at waist level. Radiation levels are shown in Tables 3 through 7. Figures 26 through 30 are schematics which show the locations in the STIR facility at which the radiation measurements were taken. All readings are total radiation readings including background radiation levels.

7704-62147

Figure 25. Thermal Column Liner After Removal of Graphite Logs

TABLE 3

RADIATION SURVEY OF TEST VAULT AREA
(Relating to Figure 26)

Survey Location	Radiation Level (mrad/hr)	Survey Location	Radiation Level (mrad/hr)
1	0.05	10	0.17
2	0.07	11	0.25
3	6.00	12	0.25
4	0.07	13	0.17
5	0.05	14	0.03
6	0.15	15	0.07
7	0.50	16	0.05
8	0.17	17	0.05
9	0.17	18	0.03

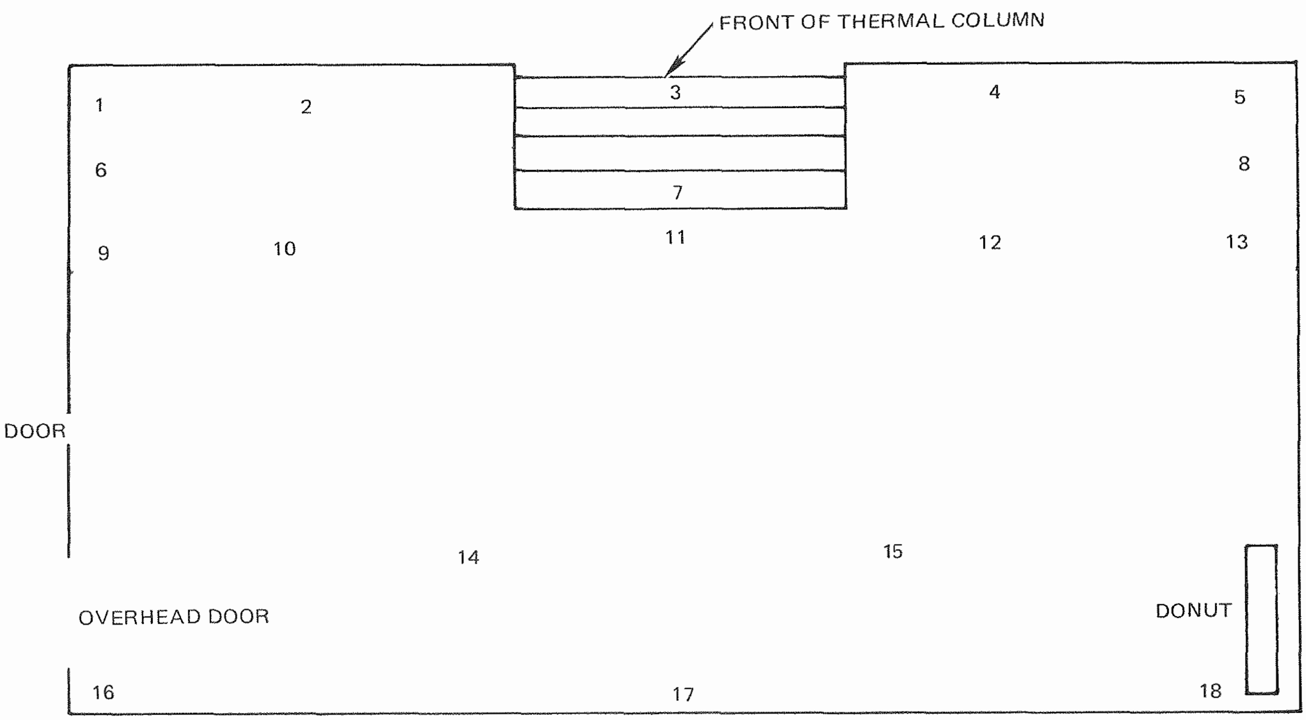


Figure 26. Radioactive Survey Locations in Test Vault Area
(Table 3)

TABLE 4

RADIATION SURVEY OF DONUT
(Relating to Figure 27)

Survey Location	Radiation Level (mrad/hr)
1	0.10
2	0.10
3	0.10
4	0.10
5	0.10
6	0.05
7	0.05
8	0.05
9	1.0

Note: Reading 1-8 taken 1/2 in. from surface. Reading 9 taken inside donut opening.

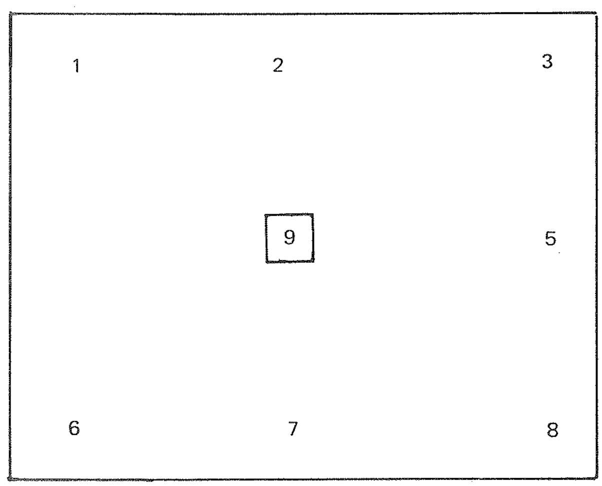


Figure 27. Donut Survey Locations (Table 4)

TABLE 5

RADIATION SURVEY OF THERMAL COLUMN WALLS
(Relating to Figure 28)

28a.		28b.		28c.	
Location	Radiation Level (mrad/hr)	Location	Radiation Level (mrad/hr)	Location	Radiation Level (mrad/hr)
1	7.0	1	15.0	1	15.0
2	10.0			2	10.0
3	15.0			3	5.0
4	7.0			4	32.0
5	15.0			5	12.0
6	33.0			6	5.0
7	4.0			7	15.0
8	8.0			8	8.0
9	12.0			9	3.0

Note: Readings in 28a. and 28c. taken 1/2 in. from surface. Reading in 28b. taken at center.

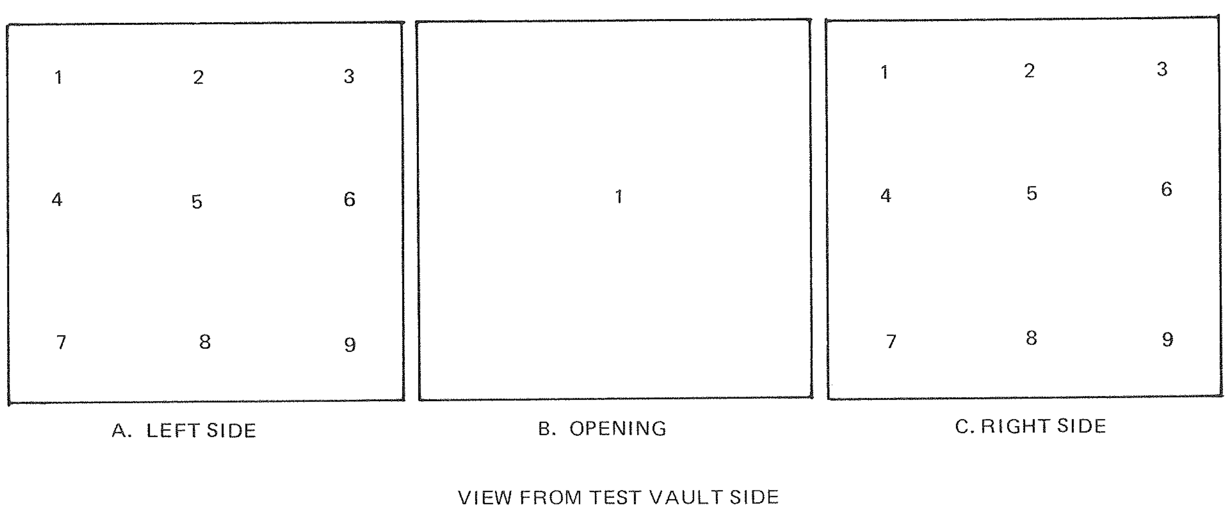


Figure 28. Reactor Thermal Column Survey Locations
(Table 5)

TABLE 6

REACTOR CAVITY FLOOR RADIATION SURVEY
(Relating to Figure 29)

Location	Radiation Level (mrad/hr)	Location	Radiation Level (mrad/hr)
1	10.0	8	10.0
2	32.0	9	5.0
3	55.0	10	3.0
4	60.0	11	3.0
5	35.0	12	3.0
6	10.0	13	3.0
7	5.0	14	3.0
		15	3.0

Note: Readings taken 1/2 in. from surface

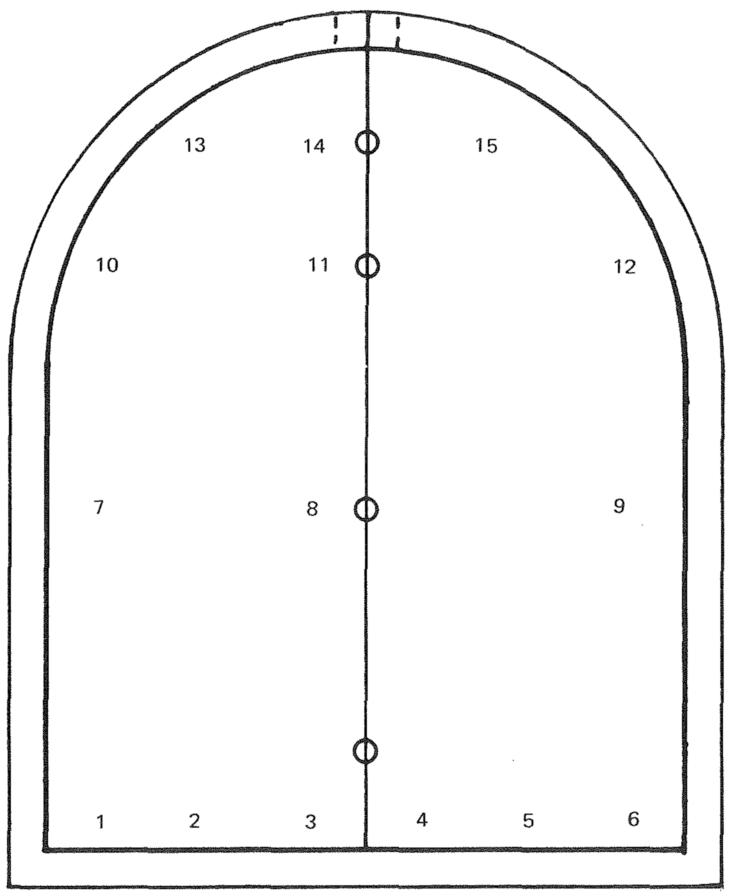


Figure 29. Reactor Cavity Floor Survey Locations (Table 6)

TABLE 7

LOWER REACTOR CONCRETE WALL
(Relating to Figure 30)

Location	Radiation Level (mrad/hr)
A	35.0
B	8.0
C	1.5
D	1.7
E	3.5
F	50.0
G	6.0
H	4.0
I	1.2
J	1.6
K	2.0
L	4.0

Note: Readings taken 1/2 in. from surface

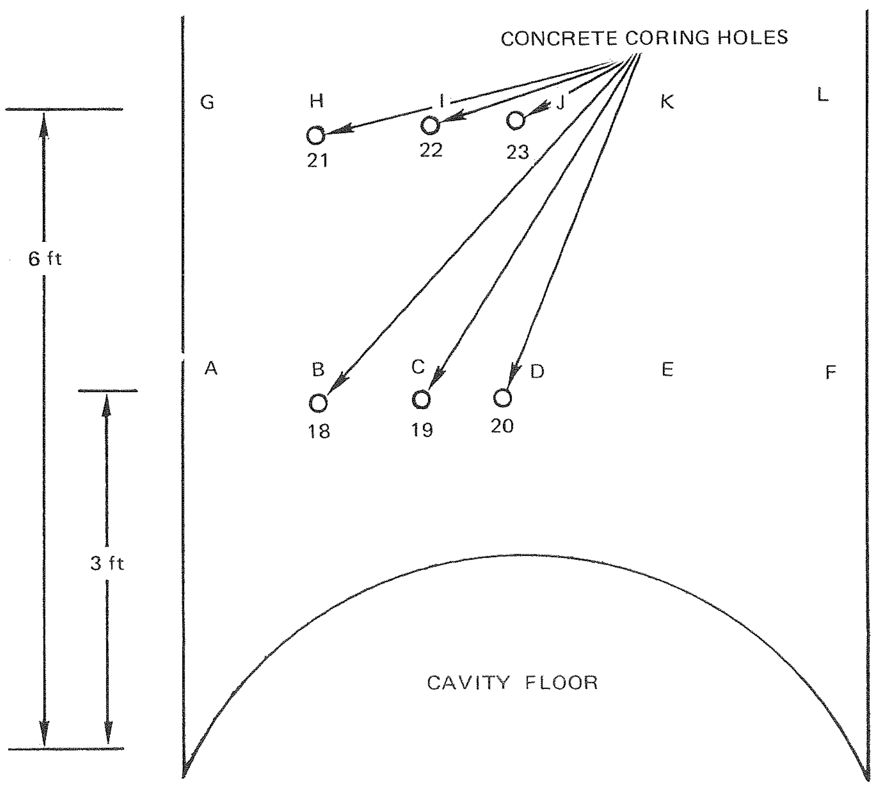


Figure 30. Lower Reactor Cavity Wall Survey Locations (Table 7)

b. Concrete Sampling

Removal of the concrete structures which were neutron irradiated during the reactor operations was a prime project requirement. Because of the accessibility of the activated concrete in the shield and the reactor enclosure structure, removal of all concrete containing statistically significant activity in excess of the natural radioactivity in the concrete was deemed practicable by the application of ALAP principles.

Before the extent of the concrete removal could be defined, it was necessary to determine the level of natural background radioactivity in the concrete structures. Nine concrete core samples (1 in. diameter by 18 in. long) were collected from the unirradiated concrete structures of the STIR facility for use as natural radioactivity standards. The 18-in. long cores were crushed and mixed, to make possible the collection of aliquots for radiometric analysis. Table 8 and the sample-identifying Figure 31 describe the results of this analysis. The mean concentration of the samples and the observed standard deviation of the data were calculated, to make possible an overall standard for the natural radioactivity in the concrete. Subsequent concrete samples were considered free of statistically significant activity, in excess of natural radioactivity, if they contained no radioactivity in excess of three times the standard deviation of the mean background radioactivity level, as established in the following listing.

mean	16.8 pCi/g
Standard deviation (σ)	1.4 pCi/g
3 (σ)	4.2 pCi/g
acceptable upper limit	21.0 pCi/g

Concrete core samples were taken from the irradiated concrete prior to initiating concrete demolition. Table 9 and the sample-identifying Figures 32, 33, and 34 describe the radiometric analyses of core samples taken from the irradiated concrete structures. Note that the samples were of various lengths, reflecting the thickness of the concrete at the sample location. Note also that the analyses were performed on segments of the samples, so that the depth of the irradiation could be assessed. The radioactivity level in the high-density (magnetite) concrete surround the thermal column (Sample 11) was lower than the level of the natural radioactivity standard for the ordinary concrete.

TABLE 8

STIR REACTOR CONCRETE ANALYSIS DATA
(Related to Figure 31)

Core Sample No.	Total Core Length (in.)	Sampled Core Segment Depth (in.)	Analysis (pCi/g β)
1	18	Composited	17.2
2	18	Composited	18.3
3	18	Composited	18.6
4	18	Composited	16.0
5	18	Composited	16.1
6	18	Composited	16.8
7	18	Composited	16.8
8	18	Composited	14.0
9	18	Composited	17.4

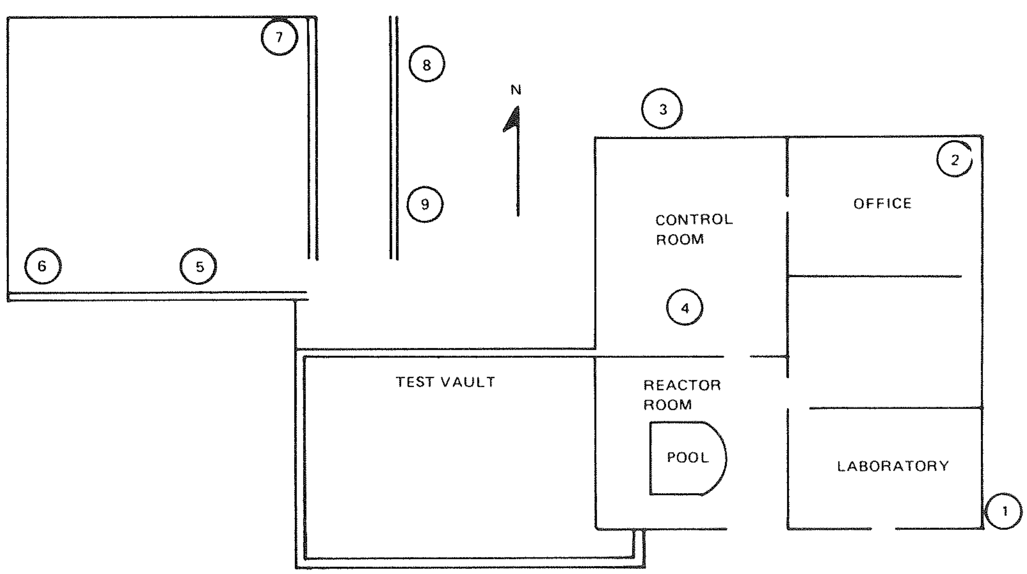


Figure 31. STIR Reactor Site Map Showing Core Sample Locations (Table 8)

TABLE 9

STIR IRRADIATED CONCRETE ANALYSIS DATA
(Related to Figures 32, 33, 34)
(Sheet 1 of 3)

Core Sample No.	Total Core Length (in.)	Sampled Core Segment Depth (in.)	Analysis (pCi/g β)
10-A	13	0-1	19.1
10-B		3-4	16.8
10-C		6-7	16.9
10-D		9-10	18.7
10-E		12-13	11.6
11-A	52	0-1	15.5
11-B		12-13	2.9
11-C		24-25	2.9
11-D		36-37	3.2
11-E		48-49	2.3
12-A	50	0-1	15.2
12-B		12-13	16.8
12-C		24-25	16.6
12-D		36-37	14.9
12-E		48-49	13.7
13-A	39	0-1	20.1
13-B		11-12	14.4
13-C		22-23	16.8
13-D		30-31	14.6
13-E		37-38	16.3
14-A	51	0-1	4904.1
14-B		12-13	21.4
14-C		24-25	25.3
14-D		36-37	17.8
14-E		50-51	16.5
15-A	36	0-1	308.8
15-B		10-11	30.0
15-C		16-17	19.4
15-D		24-25	14.3
15-E		35-36	16.6
16-A	28	0.1	24.6
16-B		6.7	18.7
16-C		12-13	13.0
16-D		18-19	10.5
16-E		27-28	16.0

TABLE 9

STIR IRRADIATED CONCRETE ANALYSIS DATA
(Related to Figures 32, 33, 34)
(Sheet 2 of 3)

Core Sample No.	Total Core Length (in.)	Sampled Core Segment Depth (in.)	Analysis (pCi/g β)
17-A	18	0-1	15.5
17-B		4-5	17.5
17-C		8-9	15.2
17-D		12-13	17.4
17-E		17-18	14.6
18-A	18	0-1	173.4
18-B		4-5	96.5
18-C		8-9	43.4
18-D		12-13	35.2
18-E		17-18	20.0
19-A	18	0-1	19.3
19-B		4-5	19.4
19-C		8-9	20.8
19-D		12-13	15.5
19-E		17-18	14.9
20-A	18	0.1	25.3
20-B		4-5	14.7
20-C		8-9	17.1
20-D		12-13	16.8
20-E		17-18	18.3
21-A	18	0-1	51.7
21-B		4-5	23.7
21-C		8-9	14.8
21-D		12-13	18.6
21-E		17-18	16.2
22-A		0-1	13.8
22-B		4-5	25.9
22-B	Duplicate Aliquot		28.4
22-C		8-9	20.2
22-D		12-13	18.0
22-E		17-18	16.4

TABLE 9

STIR IRRADIATED CONCRETE ANALYSIS DATA
(Related to Figures 32, 33, 34)
(Sheet 3 of 3)

Core Sample No.	Total Core Length (in.)	Sampled Core Segment Depth (in.)	Analysis (pCi/g β)
23-A	18	0-1	22.7
23-A	Duplicate Aliquot		18.4
23-B		4-5	12.2
23-C		8-9	24.3
23-D		12-13	17.1
23-E		17-18	14.3
24-A	18	0-1	14.2
24-B		4-5	14.8
24-C		8-9	16.3
24-D		12-13	17.1
24-E		17-18	16.0
25-A	18	0.1	16.2
25-B		4-5	18.9
25-C		8-9	14.7
25-D		12-13	15.2
25-E		17-18	13.3
26-A	18	0-1	12.1
26-B		4-5	15.5
26-C		8-9	18.6
26-D		12-13	16.0
26-E		17-18	28.3
26-E	Duplicate Aliquot		17.5

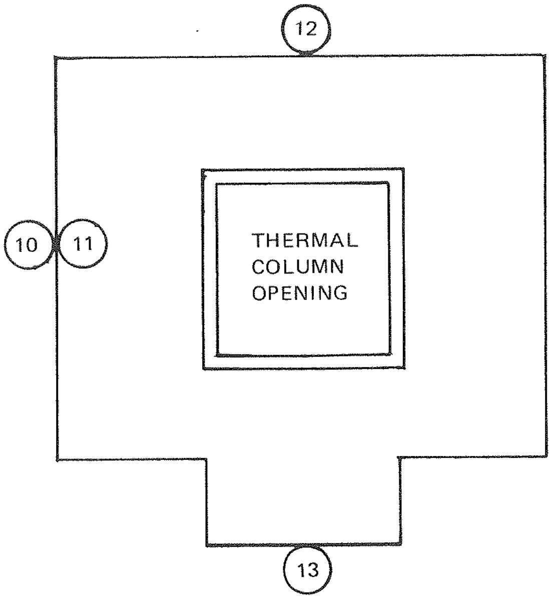
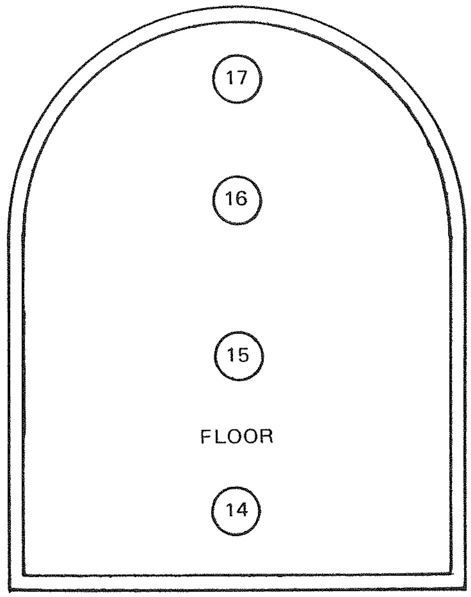


Figure 32. Thermal Column Core Sample Locations (Table 9)

Figure 33. Core Sample Locations Floor of Reactor Enclosure (Table 9)



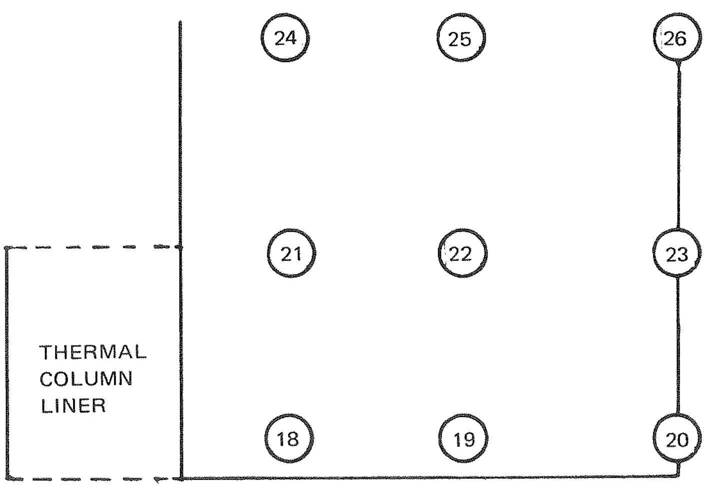


Figure 34. Cavity Wall Concrete Sample Locations (Table 9)

c. Removal of Activated Concrete

A specification defining the extent of the required activated concrete removal was prepared. Bids from demolition contractors were obtained and the contract was awarded to the lowest bidder. The activated concrete was broken out using an air driven, hydraulically positioned Hoe-Ram. The Hoe-Ram is a large jack hammer with a 4-in. diameter bit. Figure 35 shows the Hoe-Ram in action. Water was sprayed on the rubble to decrease the amount of airborne dust. Figure 36 highlights the personnel protective clothing and equipment required during the concrete removal. The concrete rubble was placed in boxes and sent to the RMDF for shipment to off-site burial. Sealed boxes of radioactive concrete rubble are shown in Figure 37. These boxes were later steel-banded prior to shipment. Figures 38, 39, 40, 41, and 42 are closeup views of the activated concrete excavation.

After removal of the thermal column liner, which was embedded 4 to 6 in. in the magnetite concrete, the side walls of the reactor enclosure were broken out. A wall area of 7 ft high and 3 ft wide was removed from each side. A radiological survey of the remaining exposed concrete and rebar revealed radiation levels in excess of 0.1 mrad/hr. Radiometric analysis of concrete samples from the remaining concrete indicated specific activities which were greater than the established limits. On the basis of the survey and sample analyses, the area of concrete excavation was widened an additional 2 ft, leaving a concrete wall 3 ft wide at the rear of the enclosure. The activity in this wall was below the established limits. The entire floor area of the reactor enclosure and the concrete pad directly below the floor area were removed. In addition, the concrete structure which supported the thermal column shielding and extended under the floor area was removed to a depth of 1.5 ft. Excavation of the floor area extended to a depth of 3 ft below the original floor level at the rear of the reactor cavity and 4.5 ft at the front. Radioanalysis of concrete samples taken from the concrete remaining in the wall and below the floor indicated a maximum specific activity of 19.0 pCi/g.

Removal of the concrete walls and floor exposed the surrounding fill soil. Results of the analysis of soil samples taken from this area are reported in

Figure 35. STIR-Excavation of Activated Concrete Near Thermal Liner and Reactor Enclosure

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Figure 36. STIR-Hoe-Ram Crew Suited Up for Removal of Activated Concrete

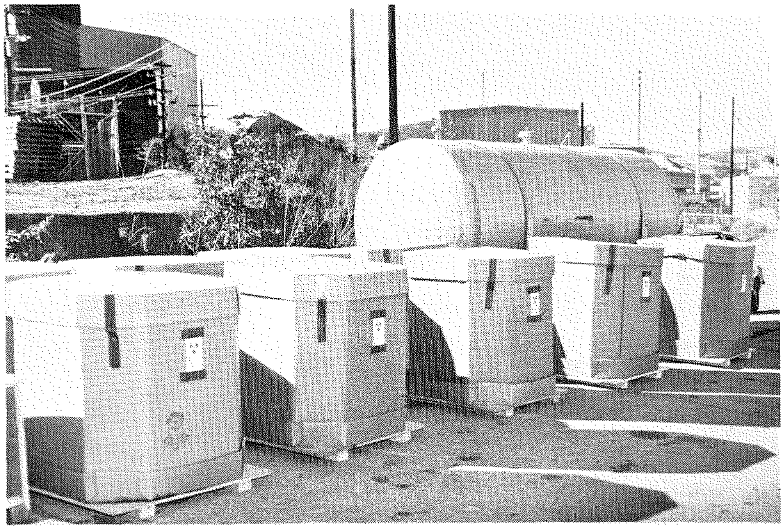

Figure 37.
Rubble Containers for Activated Concrete

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Figure 38.
Activated Concrete Removal

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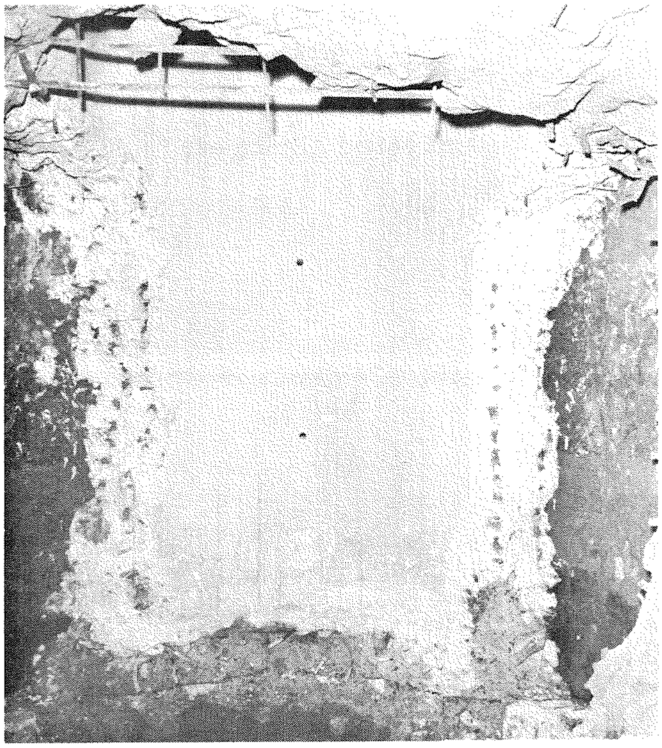

Figure 39. Excavation Showing Removal of Side Walls and Floor

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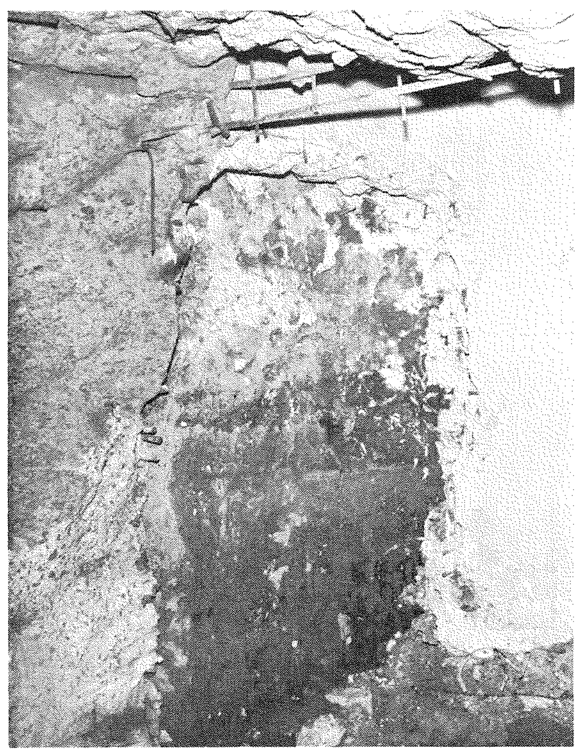

Figure 40. Excavation of Activated Concrete North Side of Enclosure

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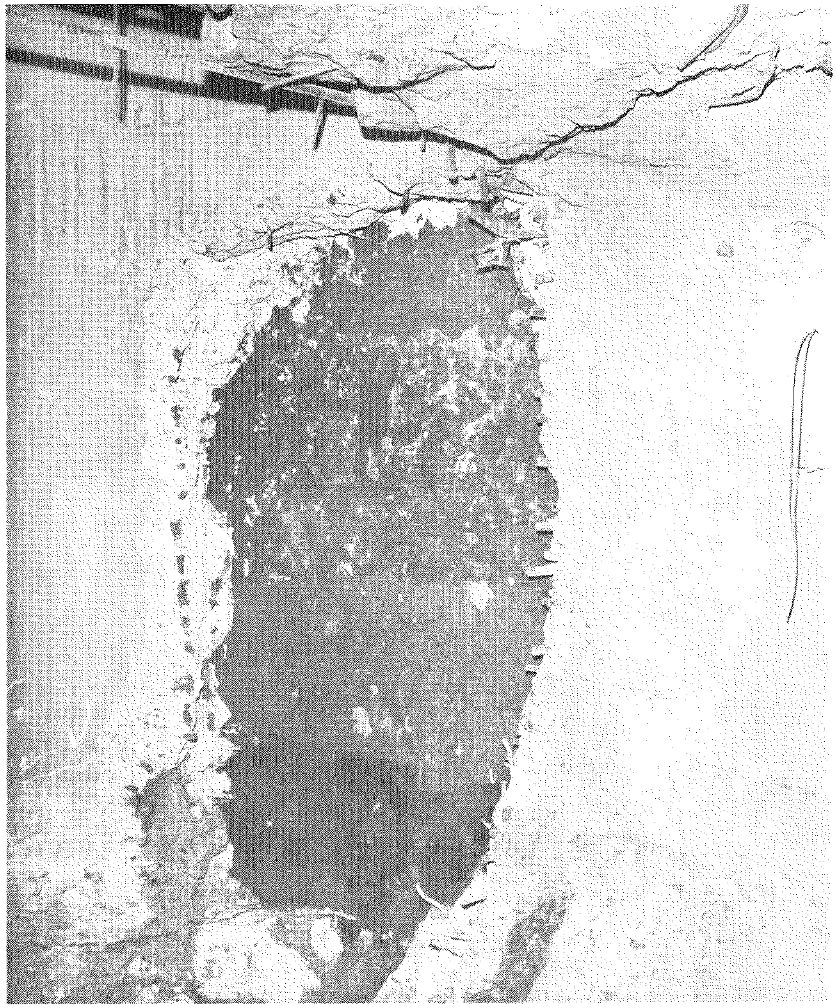
Figure 41. Excavation of Activated Concrete South Side of Enclosure



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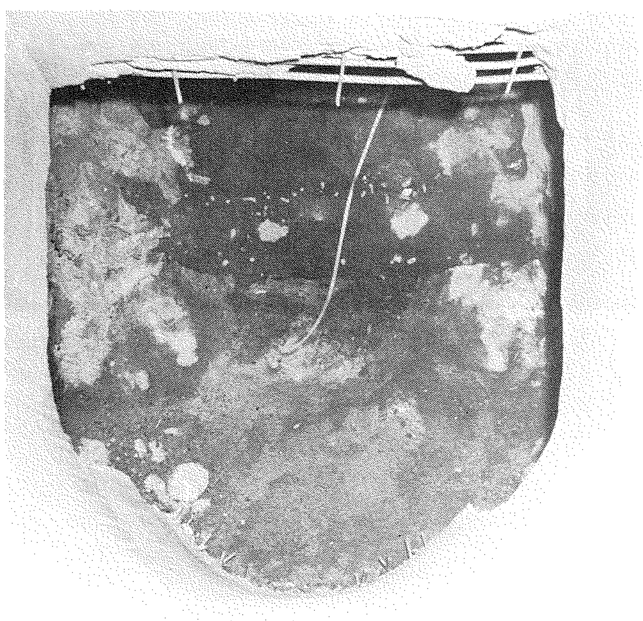

Figure 42. Excavation of Activated Concrete at Lower End of Enclosure

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TABLE 10

RADIATION LEVELS OF SOIL SURROUNDING REACTOR CAVITY

Sample Number	Description and Location		Sample Weight (g)	Analysis (pCi/g β)
1	Soil, North Wall Reactor Cavity	West End	2.0	22.6
2	Soil, North Wall Reactor Cavity	West End	2.0	24.1
3	Soil, North Wall Reactor Cavity	West End	2.0	30.8
4	Soil, North Wall Reactor Cavity	West End	2.0	23.1
5	Soil, North Wall Reactor Cavity	Center	2.0	22.8
6	Soil, North Wall Reactor Cavity	Center	2.0	22.4
7	Soil, North Wall Reactor Cavity	Center	2.0	25.5
8	Soil, North Wall Reactor Cavity	Center	2.0	26.7
9	Soil, North Wall Reactor Cavity	East End	2.0	22.0
10	Soil, North Wall Reactor Cavity	East End	2.0	23.2
11	Soil, North Wall Reactor Cavity	East End	2.0	24.4
12	Soil, North Wall Reactor Cavity	East End	2.0	22.9
13	Soil, Floor of Reactor Cavity	North Side	2.0	19.5
14	Soil, Floor of Reactor Cavity	North Center	2.0	20.7
15	Soil, Floor of Reactor Cavity	Center	2.0	18.5
16	Soil, Floor of Reactor Cavity	South Center	2.0	14.4
17	Soil, Floor or Reactor Cavity	South Side	2.0	15.2
18	Soil, South Wall Reactor Cavity	West	2.0	23.8
19	Soil, South Wall Reactor Cavity	West	2.0	29.1
20	Soil, South Wall Reactor Cavity	West	2.0	27.9
21	Soil, South Wall Reactor Cavity	Center	2.0	20.8
22	Soil, South Wall Reactor Cavity	Center	2.0	21.7
23	Soil, South Wall Reactor Cavity	Center	2.0	25.0
24	Soil, South Wall Reactor Cavity	East	2.0	20.8
25	Soil, South Wall Reactor Cavity	East	2.0	26.1

Table 10. The natural background radioactivity levels of soil in the general environs of the AI Santa Susana site have historically measured from 20 to 30 pCi/g β. The data in this table show that the soil surrounding the concrete is at these background radioactivity levels. Surface radiation levels associated with the rebar projecting from the remaining concrete in all cases were below the 0.1 mrad/hr established limit.

Table 11 represents the results of radiological survey of the test vault area upon completion of the activated concrete removal.

d. Air Sampling

Continuous air sampling was conducted by HSRS whenever the potential for airborne radioactivity existed, e.g., when using the Hoe-Ram for removing the activated concrete, which generated considerable dust. Control of the dust was effected by use of a water spray and by sealing the test vault area with plastic sheeting, taped at all openings, i.e., the stairway opening, the upper end of the reactor enclosure, and the roll-type door. Two air samplers were operated continuously during these operations and no significant airborne contamination was found. The data obtained from these samplers are reported in Table 12.

e. Contractor's Equipment

A contamination survey of the contractor's equipment following decontamination revealed that the equipment was not contaminated and could be released. Removable contamination levels on all equipment released were <30 dpm β-γ/100 cm^2.

4. Facility Exhaust System

Upon completion of the concrete removal, the facility exhaust system was radiologically surveyed. Only in one location, the grille opening directly over the thermal column area in the test vault, was measurable radioactivity detected. The exhaust system ducts directly associated with the grille were removed and sent to the RMDF. Radiological surveys of the entire remaining exhaust system were performed, and no radioactivity levels above the established limits were found. Table 13 presents the survey data for the exhaust system. The filters in the exhaust system were removed and packaged for disposal as radioactive waste.

TABLE 11

SMEAR SURVEY OF TEST VAULT
AFTER CONCRETE REMOVAL
(Sheet 1 of 3)

Sample Number	Description and Location	Analysis (dpm α/100 cm^2)	Analysis (dpm β-γ/100 cm^2)
1	Floor Area — T-028 Test Vault	0	30
2	Floor Area — T-028 Test Vault	0	30
3	Floor Area — T-028 Test Vault	0	30
4	Floor Area — T-028 Test Vault	0	30
5	Floor Area — T-028 Test Vault	0	30
6	Floor Area — T-028 Test Vault	0	30
7	Floor Area — T-028 Test Vault	0	30
8	Floor Area — T-028 Test Vault	0	30
9	Floor Area — T-028 Test Vault	0	30
10	Floor Area — T-028 Test Vault	0	30
11	Floor Area — T-028 Test Vault	0	30
12	Floor Area — T-028 Test Vault	0	30
13	Floor Area — T-028 Test Vault	0	30
14	Floor Area — T-028 Test Vault	0	30
15	Floor Area — T-028 Test Vault	0	30
16	Floor Area — T-028 Test Vault	0	30
17	Floor Area — T-028 Test Vault	0	30
18	Floor Area — T-028 Test Vault	0	30
19	Floor Area — T-028 Test Vault	0	30
20	Floor Area — T-028 Test Vault	0	30
21	Floor Area — T-028 Test Vault	0	30
22	Floor Area — T-028 Test Vault	0	30
23	Floor Area — T-028 Test Vault	0	30
24	Floor Area — T-028 Test Vault	0	30
25	Floor Area — T-028 Test Vault	0	30
26	Floor Area — T-028 Test Vault	0	30
27	Floor Area — T-028 Test Vault	0	30
28	Floor Area — T-028 Test Vault	0	30
29	Floor Area — T-028 Test Vault (Change Area Temporary)	0	30

TABLE 11

SMEAR SURVEY OF TEST VAULT
AFTER CONCRETE REMOVAL
(Sheet 2 of 3)

Sample Number	Description and Location	Analysis (dpm α/100 cm^2)	Analysis (dpm β-γ/100 cm^2)
30	Floor Area — T-028 Test Vault (Change Area Temporary)	0	30
31	Stair Well to T-028 Test Vault	0	30
32	Stair Well to T-028 Test Vault	0	30
33	Stair Well to T-028 Test Vault	0	30
34	Stair Well to T-028 Test Vault	0	30
35	T-028 Test Vault Walls - South Wall - East Corner	0	30
36	T-028 Test Vault Walls - South Wall - East Corner	0	30
37	T-028 Test Vault Walls - South Wall - East Corner	0	30
38	T-028 Test Vault Walls - South Wall - East Corner	0	30
39	T-028 Test Vault Walls - South Wall - East Corner	0	30
40	T-028 Test Vault Walls - South Wall - East Corner	0	30
41	T-028 Test Vault Walls - South Wall	0	30
42	T-028 Test Vault Walls - South Wall	0	30
43	T-028 Test Vault Walls - South Wall	0	30
44	T-028 Test Vault Walls - South Wall	0	30
45	T-028 Test Vault Walls - South Wall	0	30
46	T-028 Test Vault Walls - West Wall	0	30
47	T-028 Test Vault Walls - West Wall	0	30
48	T-028 Test Vault Walls - West Wall	0	30

TABLE 11

SMEAR SURVEY OF TEST VAULT
AFTER CONCRETE REMOVAL
(Sheet 3 of 3)

Sample Number	Description and Location	Analysis (dpm α/100 cm^2)	Analysis (dpm β-γ/100 cm^2)
49	T-028 Test Vault Walls - West Wall	0	30
50	T-028 Test Vault Walls - West Wall	0	30
51	T-028 Test Vault Walls - West Wall	0	30
52	T-028 Test Vault Walls - West Wall	0	30
53	Roll-Up Door - North Wall	0	30
54	Roll-Up Door - North Wall	0	30
55	North Wall	0	30
56	North Wall	0	30
57	North Wall	0	30
58	North Wall	0	30
59	North Wall	0	30
60	North Wall	0	30
61	T-028 Test Vault Walls - North Wall	0	30
62	T-028 Test Vault East Wall	0	30
63	T-028 Test Vault East Wall	0	30
64	T-028 Test Vault East Wall	0	30
65	T-028 Test Vault East Wall	0	30
66	Weather Proof Work Lights (North Wall)	0	30
67	Weather Proof Work Lights (North Wall)	0	30

TABLE 12

AIR SAMPLING DURING CONCRETE REMOVAL

Sample No.	Sampler Location No.*	Date of Sample	Immediate Count ($\mu Ci/cm^3\ \beta$)	Delay Count Date	Delay Count ($\mu Ci/cm^3\ \beta$)
1 (Background)	1	1-14-76	2.03×10^{-12}	1-15-76	8.0×10^{-13}
2 (Background)	2	1-14-76	2.42×10^{-12}	1-15-76	2.6×10^{-13}
3 (Max for Date)	1	1-15-76	4.67×10^{-11}	1-16-76	2.56×10^{-12}
4 (Max for Date)	2	1-15-76	5.18×10^{-11}	1-16-76	7.04×10^{-12}
5 (Max for Date)	1	1-16-76	8.26×10^{-12}	1-19-76	2.93×10^{-12}
6 (Max for Date)	2	1-16-76	9.07×10^{-12}	1-19-76	1.73×10^{-12}
7 (Max for Date)	1	1-19-76	4.13×10^{-12}	1-20-76	3.73×10^{-12}
8 (Max for Date)	2	1-19-76	8.00×10^{-12}	1-20-76	5.60×10^{-12}
9 (Max for Date)	1	1-20-76	3.48×10^{-11}	1-21-76	3.73×10^{-13}
10 (Max for Date)	2	1-20-76	4.18×10^{-11}	1-21-76	2.67×10^{-13}
11 (Max for Date)	1	1-21-76	1.29×10^{-11}	1-22-76	1.85×10^{-13}
12 (Max for Date)	2	1-21-76	1.69×10^{-11}	1-22-76	2.62×10^{-12}
13 (Max for Date)	1	1-22-76	3.22×10^{-11}	1-23-76	1.33×10^{-12}
14 (Max for Date)	2	1-22-76	2.88×10^{-11}	1-23-76	2.67×10^{-13}
15 (Max for Date)	1	1-23-76	1.78×10^{-11}	1-26-76	5.66×10^{-12}
16 (Max for Date)	2	1-23-76	1.78×10^{-11}	1-26-76	3.94×10^{-12}
17 (Max for Date)	1	1-26-76	3.13×10^{-12}	1-27-76	3.80×10^{-13}
18 (Max for Date)	2	1-26-76	2.66×10^{-12}	1-27-76	3.60×10^{-13}
19 (Max for Date)	1	1-27-76	1.35×10^{-12}		
20 (Max for Date)	2	1-27-76	5.37×10^{-12}		

*Location 1 - Test Vault Exit Door
Location 2 - Near Thermal Column Opening

TABLE 13

EXHAUST SYSTEM RADIOLOGICAL SURVEY REPORT

Sample Number	Description and Location	Analysis (dpm β-γ/100 cm^2)	Analysis (dpm α/100 cm^2)
1	Back Side of Fume Hood Inside Panel (R) Side	<50	<5
2	Back Side of Fume Hood Inside Panel (R) Side	<50	<5
3	Back Side of Fume Hood Inside Panel (R) Side	<50	<5
4	Back Side of Fume Hood Inside Panel (L) Side	<50	<5
5	Back Side of Fume Hood Inside Panel (L) Side	<50	<5
6	Back Side of Fume Hood Inside Panel (L) Side	<50	<5
7	Back Side of Fume Hood Inside Panel Top (Exhaust Opening)	<50	<5
8	Back Side of Fume Hood Inside Panel Top (Exhaust Opening)	<50	<5
9	Back Side of Fume Hood Inside Panel Top (Exhaust Opening)	<50	<5
10	Back Side of Fume Hood Inside Panel Top (R) Side	<50	<5
11	Back Side of Fume Hood Inside Panel Top (L) Side	<50	<5
12	Test Vault (L) Wall Exhaust Opening	<50	<5
13	Test Vault (L) Wall Exhaust Opening	<50	<5
14	Test Vault (L) Wall Exhaust Opening	<50	<5
15	Duct/Exhaust Reactor Room	<50	<5
16	Duct/Exhaust Reactor Room	<50	<5
17	Duct/Exhaust Reactor Room	<50	<5
18	Duct/Exhaust Reactor Room	<50	<5
19	Facility Exhaust Stack (Top End of Stack)	<50	<5
20	Facility Exhaust Stack (Top End of Stack)	<50	<5

5. Facility Repairs

The demolition contractor filled the reactor cavity with fill dirt and non-radioactive rubble. The opening in the test vault was sealed with a 6-in. thick concrete-steel reinforced wall. The reactor cavity opening in the reactor room was paved with concrete. Other pits and trenches deemed unsafe were also filled and paved. Included were the storage pit in the laboratory room, the shield door rail excavations, and the pipe pits near the reactor cavity. Figure 43 shows the concrete forming for the test vault wall repair. Figure 44 shows the completed wall. Figure 45 shows the reactor room floor after paving.

6. Disposal of Radioactive Waste

All radioactive waste generated from the STIR D&D activities was sent to the RMDF. Contaminated water from the concrete coring and Hoe-Ram operations was evaporated. Solid waste was packaged in containers and shipped in three shipments to Beatty, Nevada for land burial. A total of 1500 ft^3 of waste was shipped.

7. Personnel Dosimetry

Monitoring of internal and external radiation exposure to personnel, as prescribed in the Operational Safety Plan, was conducted throughout the STIR dismantling operations.

Personnel were periodically evaluated, by urinalysis, for internal exposure to mixed fission products, activation products, and nonspecific gross alpha emitters. All results were at or below the appropriate minimum detection limits for the analysis performed.

The external radiation exposure of the nine persons directly associated with the dismantling operations, during the period of September 24, 1975 through January 31, 1976, when the reactor internals, reactor vessel, and reactor shielding were removed, averaged 193 mrem, with a maximum individual exposure of 420 mrem. The entire operation was performed with a total radiation exposure of 1.7 man-rem.

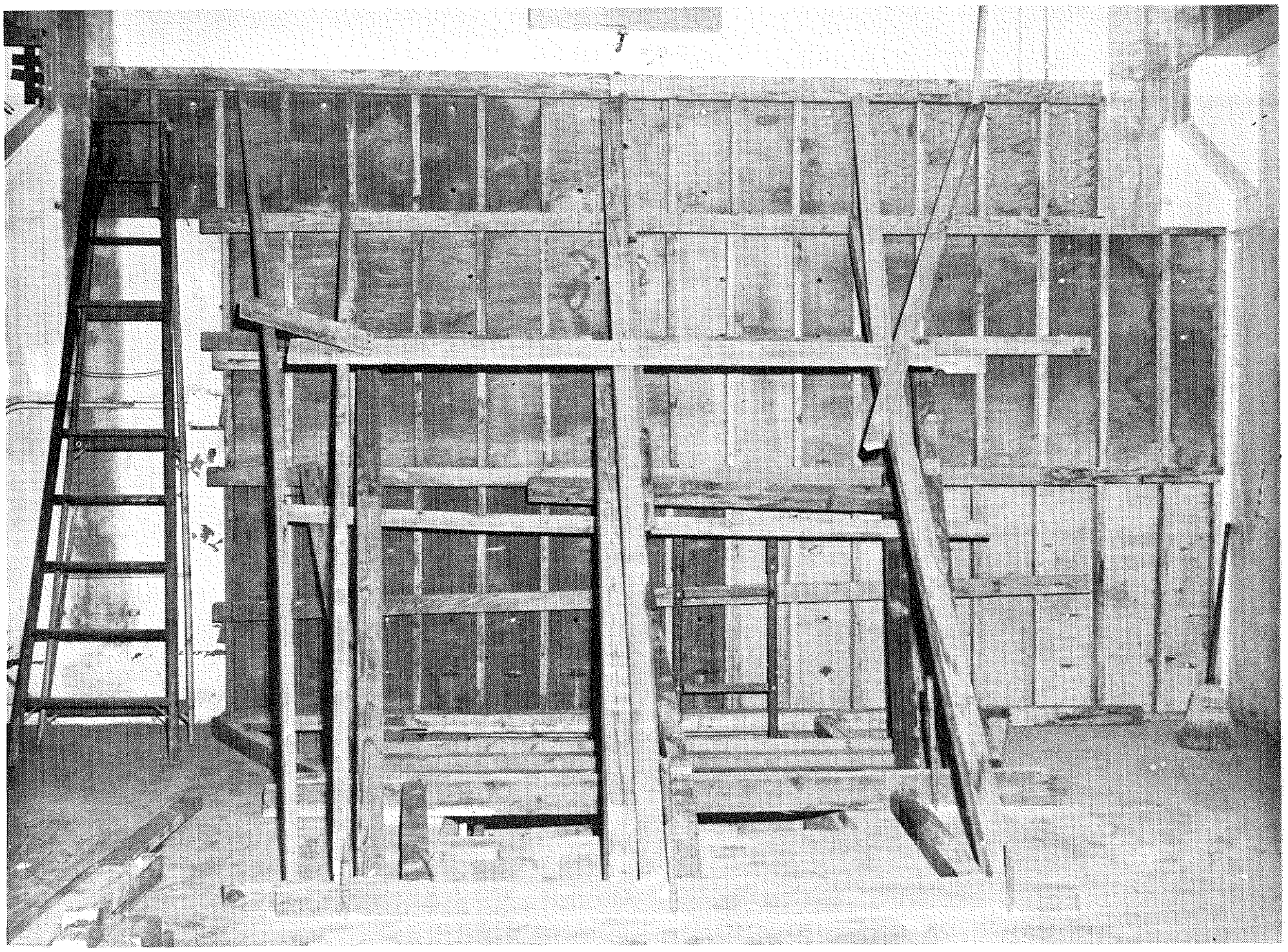

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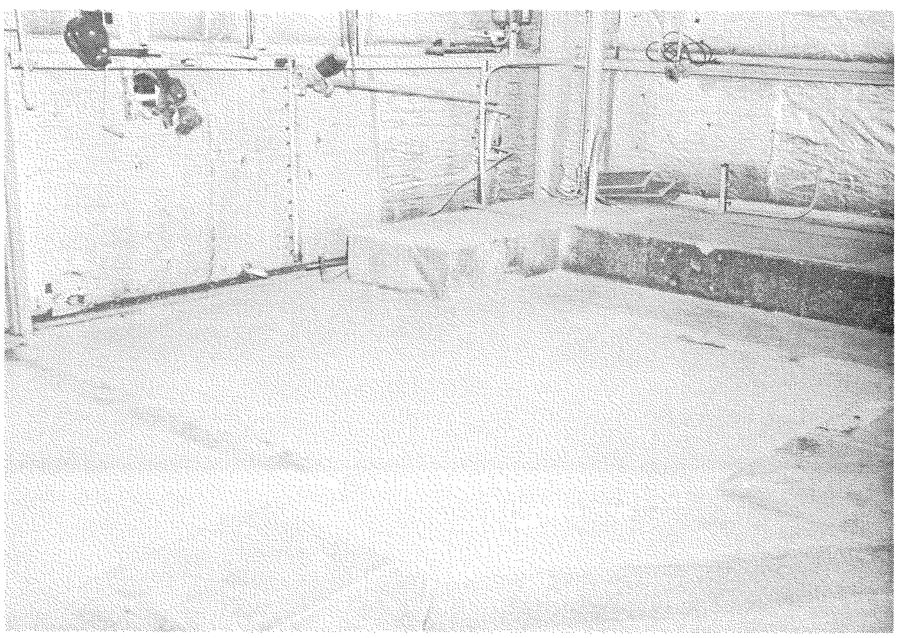
Figure 43. STIR-Concrete Forming in Repair of Excavation in Test Vault

7704-62442CN

Figure 44. Repaired Wall in Test Vault



7704-62470CN

Figure 45. Repaired Floor in Reactor Room

8. Final Survey of the STIR Facility

A final survey of the total facility was conducted to verify that the radiation levels in the facility have been reduced to <0.1 mrad/hr. The radiation survey was conducted in the interior spaces of Building T028, with a Technical Associates PUG-1 thin-window GM survey instrument and an Eberline E-510 GM survey instrument equipped with a 7 mg/cm^2 absorber over the detector window. The radiation levels measured throughout the building with the 7 mg/cm^2 absorber detector ranged from 0.02 to 0.05 mrad/hr above background. The maximum level measured with the 7 mg/cm^2 absorber detector was 0.07 mrad/hr at the west end of the thermal column in the test vault. The radiation levels on the reactor cavity excavation ranged from 0.02 to 0.04 mrad/hr. The radiation levels in the fission plate storage pit directly below the thermal column ranged from 0.02 to 0.05 mrad/hr above background. The surveys were conducted throughout the interior of Building 028 and throughout the fenced-in area surrounding the building.

Tables 14 and 15 describe the final radiation survey meter measurements at specific interior and exterior locations shown in Figures 46 and 47 respectively. Table 16 summarizes the final radiological survey, including radiation and removable contamination measurements.

TABLE 14

T028 STIR INTERIOR FACILITY SURVEY
(Refer to Figure 46)

(mrad/hr)							
1.	0.03	26.	0.04	51.	0.03	76.	0.04
2.	0.04	27.	0.04	52.	0.04	77.	0.07
3.	0.03	28.	0.03	53.	0.03	78.	0.05
4.	0.03	29.	0.04	54.	0.03	79.	0.04
5.	0.04	30.	0.04	55.	0.04	80.	0.05
6.	0.03	31.	0.03	56.	0.04	81.	0.07
7.	0.04	32.	0.03	57.	0.04	82.	0.04
8.	0.03	33.	0.03	58.	0.03	83.	0.04
9.	0.03	34.	0.03	59.	0.03	84.	0.04
10.	0.04	35.	0.03	60.	0.03	85.	0.04
11.	0.03	36.	0.03	61.	0.04	86.	0.04
12.	0.03	37.	0.03	62.	0.04	87.	0.04
13.	0.04	38.	0.03	63.	0.03	88.	0.04
14.	0.04	39.	0.03	64.	0.03	89.	0.04
15.	0.04	40.	0.04	65.	0.03	90.	0.04
16.	0.04	41.	0.03	66.	0.04	91.	0.04
17.	0.03	42.	0.03	67.	0.04	92.	0.04
18.	0.04	43.	0.03	68.	0.04	93.	0.04
19.	0.04	44.	0.03	69.	0.03	94.	0.04
20.	0.04	45.	0.04	70.	0.03	95.	0.04
21.	0.03	46.	0.04	71.	0.03	96.	0.04
22.	0.03	47.	0.03	72.	0.04	97.	0.04
23.	0.03	48.	0.03	73.	0.04	98.	0.04
24.	0.04	49.	0.03	74.	0.04	99.	0.04
25.	0.04	50.	0.03	75.	0.04	100.	0.04

NOTE: Background of 0.03 - 0.04 mrad/hr included in radiation measurements.

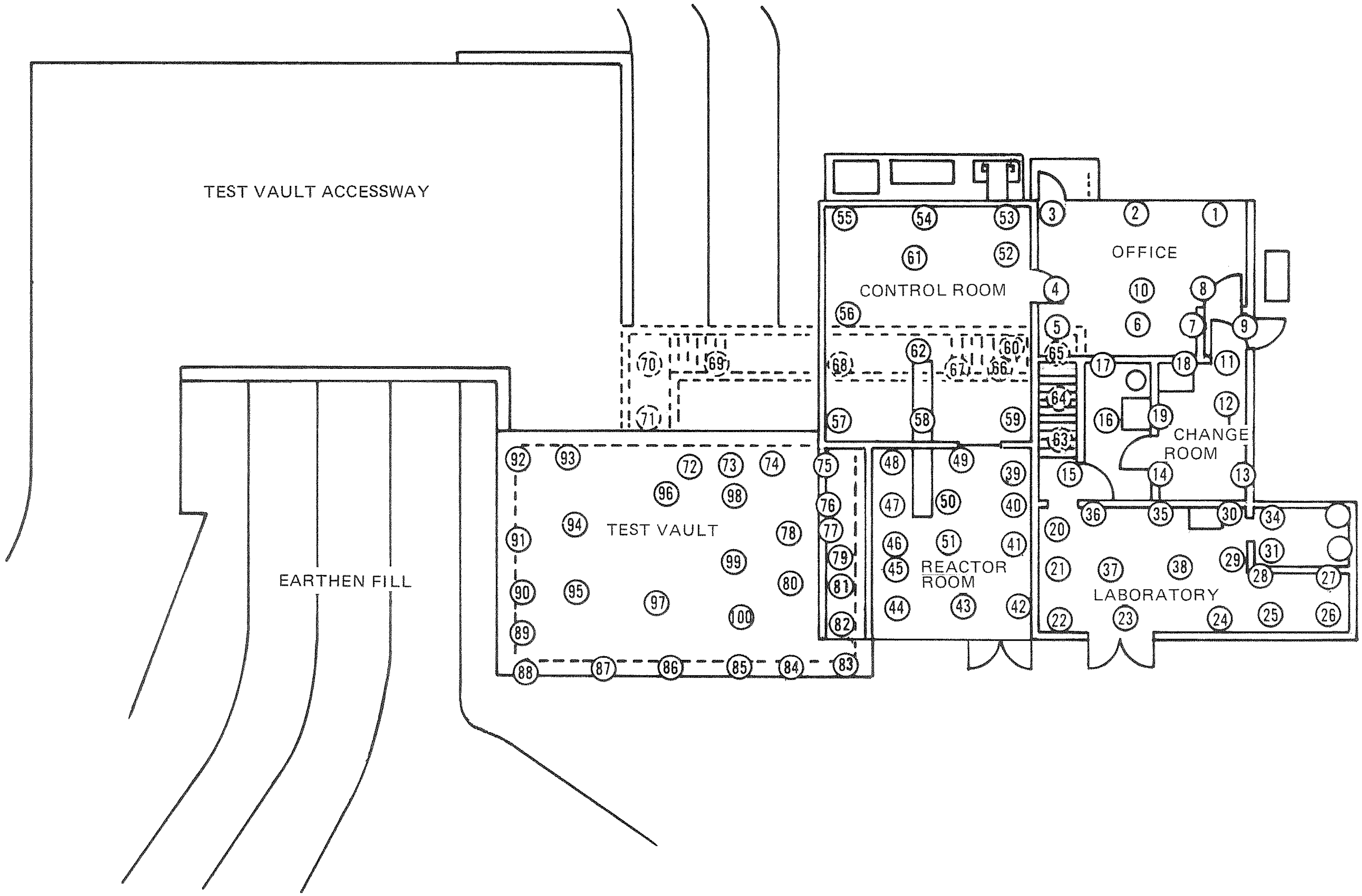


Figure 46. STIR — T028 Interior Radiation Survey

TABLE 15

T028 STIR EXTERIOR FACILITY SURVEY
(Refer to Figure 47)

(mrad/hr)							
1.	0.03	21.	0.06	41.	0.04	61.	0.03
2.	0.03	22.	0.05	42.	0.04	62.	0.03
3.	0.03	23.	0.05	43.	0.04	63.	0.03
4.	0.02	24.	0.04	44.	0.04	64.	0.04
5.	0.02	25.	0.04	45.	0.04	65.	0.04
6.	0.02	26.	0.04	46.	0.03	66.	0.04
7.	0.02	27.	0.04	47.	0.03	67.	0.04
8.	0.02	28.	0.04	48.	0.03	68.	0.03
9.	0.02	29.	0.04	49.	0.03	69.	0.03
10.	0.02	30.	0.03	50.	0.04	70.	0.04
11.	0.03	31.	0.04	51.	0.04	71.	0.04
12.	0.03	32.	0.04	52.	0.04	72.	0.04
13.	0.03	33.	0.03	53.	0.03	73.	0.04
14.	0.02	34.	0.03	54.	0.03	74.	0.04
15.	0.02	35.	0.03	55.	0.04	75.	0.04
16.	0.03	36.	0.03	56.	0.04	76.	0.04
17.	0.03	37.	0.03	57.	0.04	77.	0.04
18.	0.04	38.	0.03	58.	0.04	78.	0.03
19.	0.04	39.	0.03	59.	0.03	79.	0.04
20.	0.08	40.	0.04	60.	0.03	80.	0.04

NOTE: Background of 0.02 - 0.04 mrad/hr included in radiation measurements

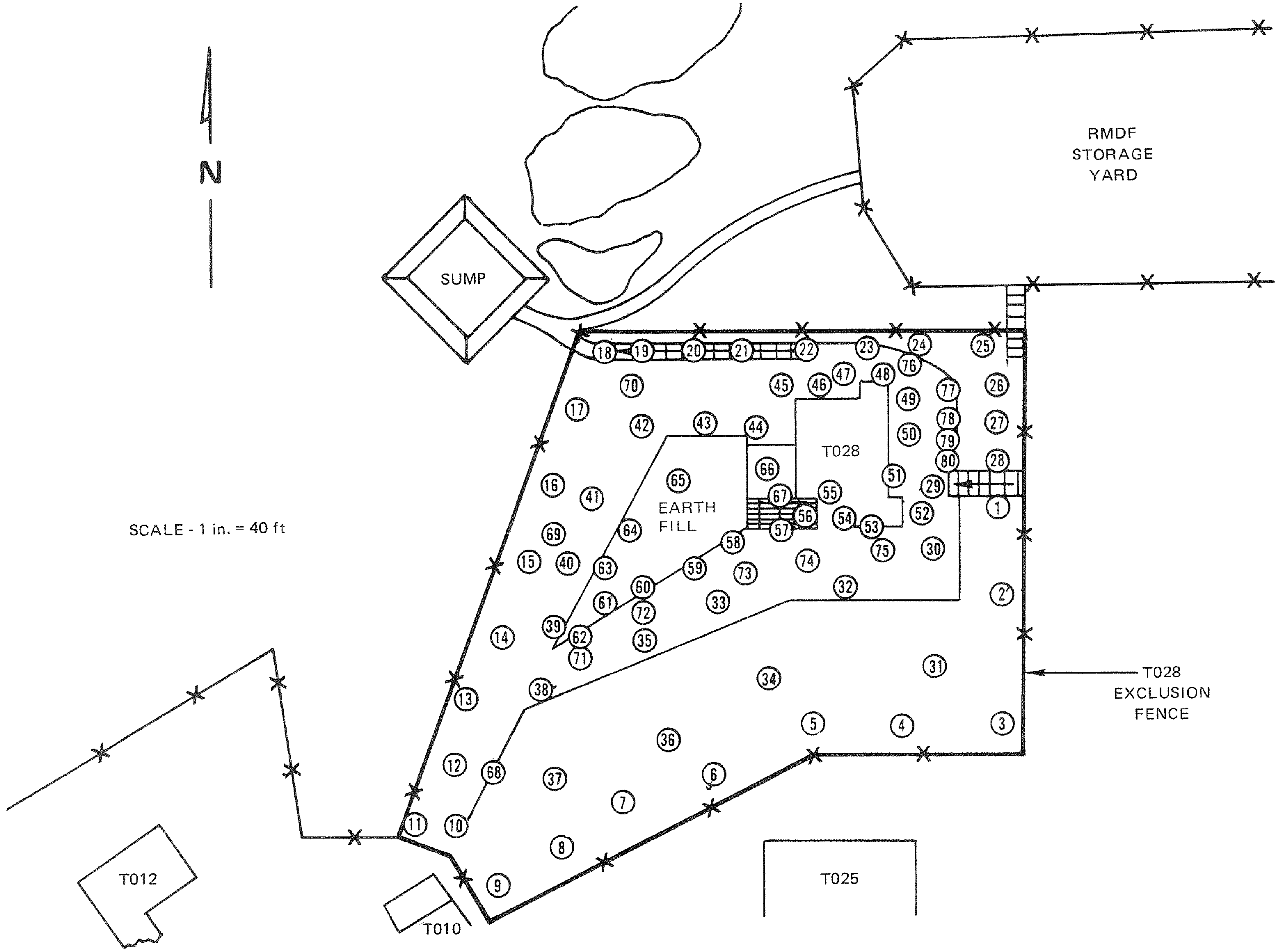


Figure 47. STIR – T028 Exterior Radiation Survey

TABLE 16

T028 STIR FINAL RADIOLOGICAL SURVEY SUMMARY

Location	Survey Type	Total Smears	Maximum Removable Contamination Level	Maximum Radiation Level* (mrad/hr)
1. Office Area	A&B	250	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04
2. Control Room	A&B	270	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04
3. Change Room	A&B	160	0 dpm/100 cm^2 <30 dpm/100 cm^2 β-γ	0.04
4. Darkroom	A&B	120	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.03
5. Laboratory	A&B	265	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04
6. Reactor Room	A&B	280	0 dpm/100 cm^2 α <60 dpm/100 cm^2 β-γ	0.04
7. Stairway and Tunnel	A&B	95	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04
8. Test Vault	A&B	760	0 dpm/100 cm^2 α <50 dpm/100 cm^2 β-γ	0.07
9. Exhaust System	A&B	100	0 dpm/100 cm^2 α <30 dpm/100 cm^2 β-γ	0.04
10. Cooling System Area	B			0.04
11. Blacktop Surfaces	B			0.04
12. North Perimeter Stairway	B			0.08
13. Reactor Cavity and Thermal Column	C		23.7 ± 2.6 pCi/g β(Soil) 19.0 pCi/g β(Concrete)	

A — Smear

B — Survey Meter (PUG-1)

C — Radiometric

* — Total radiation reading with E-510 and 7 mg/cm^2 absorber detector

NOTE: General background level of 0.02 to 0.04 mrad/hr included in radiation measurements.

IV. STIR FACILITY D&D COSTS

The total costs for the STIR D&D are presented in Table 17. The major cost is represented by AI labor.

Nuclear Engineering Company was the contractor for burial of the radioactive waste material. Lester Cushing Company was employed as the demolition contractor and United Scrap Metals as the salvage contractor.

TABLE 17

STIR FACILITY D&D COSTS

Total Labor Costs	
AI	$ 88,442
Rocketdyne	710
Subcontracted Costs	
Nuclear Engineering Corp.	$ 6,908
Lester Cushing	18,370
Other Costs	
Materials	$ 4,104
Miscellaneous	
a. G&A	7,749
b. Fee	8,639
Total D&D Costs	$134,922

REFERENCES

1. "Hazards Summary Report, STIR Modifications for 1-Megawatt Operation, NAA-SR-MEMO-9129 (December 15, 1963)

2. "Startup and Operation of the 1 Megawatt STIR," NAA-SR-11175 (March 25, 1966)

3. Operational Safety Plan for the AI Decontamination and Disposition of Facilities Program, SRR-704-990-001, Rev. B (October 21, 1975)

APPENDIX

Atomics International Division
Rockwell International

SUPPORTING DOCUMENT

NUMBER	FDP-704-990-004
REV LTR/CHG NO.	SEE SUMMARY OF CHG
PROGRAM TITLE	Decontamination and Disposition of Facilities Program
DOCUMENT TYPE	Facilities Dismantling Plan
KEY NOUNS	Dismantling Plan
DOCUMENT TITLE	Facilities Dismantling Plan for STIR, Building 028
ORIGINAL ISSUE DATE	5-28-75
GO NO.	09070
S/A NO.	15100
PAGE 1 OF TOTAL PAGES	14
REL. DATE	5-29-75 nk

PREPARED BY/DATE	DEPT	MAIL ADDR
V. A. Swanson	713-540	T093

IR&D PROGRAM? YES ☐ NO ☒ IF YES, ENTER TPA NO. ______

APPROVALS DATE

W. Heine W Heine M. Remley M Remley 28 May

P. Higgins P. Higgins 5-28-75 B. Ureda B. Ureda 5-28-75

A. Graves A. Graves, 5-28-75

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ABSTRACT

The Shield Test Irradiation Reactor (STIR) will be dismantled, and materials and components disposed of to the extent necessary to allow unrestricted use of the remaining facilities. All contaminated or radioactive materials, equipment, and facility structures will be decontaminated or removed, packaged and shipped for burial. Utilities, ventilation systems, hoist, and other items which would have future general use will not be removed. Items that will be removed include: the reactor tank, thermal column, activated concrete, cooling systems, water purification system, water door, and test carriages. The reactor tank cavity will be filled with sand and topped with concrete flush with the reactor floor.

FORM 734-C REV. 2-74

I. OBJECTIVE

The Shield Test and Irradiation Reactor (STIR) facility is shown in Figures 1-3. The reactor was operated with a 50 kwt capability between 1961 and 1964 and with a 1 Mwt capability between 1964 and 1972. The MTR type fuel elements were removed and the pool water drained in June 1973. The maximum radiation level observed in the facility in a February 1975 survey was ~800 mR/hr on the core grid plate next to the lead gamma shield.

A. DESCRIPTION OF THE STIR FACILITY

1. Reactor

The reactor core was located at the bottom of a 5 ft diameter x 20 ft deep, water-filled aluminum tank. The fuel elements have been removed but the grid plate and support structure are still in place. The tank sits in a concrete well with a 6-inch annulus of pea gravel between the concrete and the tank. The west side of the tank near the bottom was modified to mate to the thermal column leading to the test vault and to provide a lead and bismuth gamma shield between the core and the thermal column. The control rods and drives, and the exposure thimbles and neutron detectors have already been removed. A 2000-lb capacity, manually operated chain hoist is provided in the reactor room.

2. Thermal Column and Test Vault

The 5 ft x 5 ft x 4 ft thermal column interfaces with the reactor tank on the east side and with the test vault on the west side. It consists of an aluminum box filled with graphite logs of 4-in. x 4-in. cross section. The wall immediately around the thermal column is dense concrete.

The test vault is 20 ft x 33 ft x 17 ft - 8 in. high. A 7.5 ton bridge crane with a remotely operated manipulator attached to it services the area. Access to the test vault is through a 9 ft x 10 ft freight door or through a stairwell leading to the main floor of the building. An electrically driven, 5 ft thick, water-filled tank can be moved into a position just outside the freight door for radiation shielding. An electrically driven test carriage runs on rails in an east-west direction inside the vault. A 5 ft x 10-1/2 ft x 6 ft - 10 in. deep pit in the floor of the vault served to hold a fission plate and its shield cask. The plate and cask have been removed.

3. Cooling System

Cooling for the reactor was provided by two systems; a 50 kw refrigeration unit and a 1 Mw cooling tower. The refrigeration system consists of a freon-to-water heat exchanger in the reactor room, an airblast heat exchanger outside the reactor room, and the associated pump and plumbing. The 1 Mw cooling system consists of a cooling tower, on the secondary side, and a 4-pass, tube and shell type heat exchanger located on the roof of the test vault. Two pumps are used to circulate water through the cooling tower and a single pump, located in a trench outside the reactor room on the south side, is used to circulate water through the primary side. The water purification system, valves and piping are also located in the trench outside the reactor room. A 1000 gal distilled water make-up tank is located just south of the building.

4. Support Facilities

Located on the same level as the reactor room are the control room, office area, change room and laboratory as shown in Figure 2. The laboratory has been extended 12 ft to the south since the figures were drawn. A fume hood is provided in the laboratory area.

The ventilation system maintains the reactor room and test vault at a negative pressure relative to surrounding areas. Exhausted air passes through a particulate air filter bank before being released through the building ventilation stack.

B. DISMANTLING AND DISPOSITION

All contaminated or radioactive materials, equipment, and facility structures will be decontaminated or removed, packaged and shipped for burial. All areas of the facility and all material and equipment released for unrestricted use will be decontaminated to levels which are as low as practicable but in all cases to levels below those in Table 1. Acceptable specific activity levels for the concrete biological shielding remaining in place following completion of the dismantling operations will be developed in the Activity Requirements for the concrete removal.

TABLE 1

Contamination Limits for Decontamination and Disposition of the STIR Facility

	Total	Removable
Beta-Gamma Emitters	0.1 mrad at 1 cm with 7 mg/cm^2 absorber	100 dpm/100 cm^2
Alpha Emitters	100 dpm/100 cm^2	20 dpm/100 cm^2

The facility will not be completely dismantled. Utilities, ventilation system, hoists, and other items that might be of general use to some future project will not be removed. Items that will be removed include the reactor tank, the thermal column, the two cooling systems, the water purification system, the water shield for the test vault freight door, the test carriage and miscellaneous items which are not generally useful. The control room instrumentation and equipment, most of the laboratory equipment, and miscellaneous hardware were removed in June 1973.

II. SCOPE OF PLAN

The Dismantling Plan delineates the activities necessary to realize the objectives stated above. These activities have been categorized as follows:

1. Planning, monitoring, and control
2. Radiological survey
3. Tooling and support equipment procurement
4. Dismantling and disposal
5. Documentation

III. PLANNING, MONITORING, AND CONTROL

A schedule listing the detailed tasks and the sequence of performance has been prepared (see Figure 4). The level of manpower requirements for these activities are also shown in Figure 4.

Specific tasks will be initiated and monitored by the Program Office. The work authorizations, work releases, and progress report issuance will generally follow the format and guidelines set out in the Decontamination and Disposition of Facilities Program Plan. Quality Assurance and Health Safety and Radiation Services actions will be governed by the Quality Assurance Plan and the Operational Safety Plan, respectively. The schedule and manpower loading charts and the cost records will serve as the overall criteria to measure progress and accumulated costs.

IV. RADIOLOGICAL SURVEY

An initial radiological survey will be made to determine the extent of radioactivity present in the facility. An assessment of the probable levels of radioactivity are as follows:

A. REACTOR TANK

The highest observed radiation level as of February 1975 was 800 mR/hr measured at the top of the core grid plate next to the lead gamma shield. Most of this radiation is due to activated impurities in the 6061 T6 aluminum structure but some is probably due to Po-210 generated in the bismuth shield and activation of the gravel and concrete around the pool tank.

B. THERMAL COLUMN

The maximum radiation level at the test vault side of the thermal column is about 3 mR/hr. This is probably due to a combination of activated structural material and activated samarium oxide contamination in the graphite.

C. TEST VAULT

The concrete around the thermal column is probably activated. The rest of the vault structure indicates acceptable radiation levels. Parts of the test carriage structure and the shield mounted on it indicate radiation levels as high as 1 mR/hr.

D. COOLING SYSTEM

No radiation was detected external to the cooling system piping, heat exchangers, pumps, etc. This was true of the water purification system also. There may be some low level internal contamination.

V. TOOLING AND SUPPORT EQUIPMENT PROCUREMENT

No special tooling requirements are anticipated. Handling equipment, containers and packaging materials required for radioactive waste will be procured from the Radioactive Materials Disposal Facility (RMDF) at AI. Cranes and rigging needed for lifting and moving heavy equipment will be provided by AI Maintenance or an outside contractor.

VI. DISMANTLING AND DISPOSAL

Activity Requirements and detailed Working Procedures will be written to guide the dismantling and disposal operations. A brief description of the principal tasks are as follows.

A. PREPARATION FOR DISMANTLING AND DISPOSITION

A change area and a radiological survey station will be set up. Health and Safety equipment, instrumentation, and materials will be made available. A radiological survey will be made of all areas.

B. PERIPHERAL SYSTEMS REMOVAL

A Salvage Contractor will be used to remove non-radioactive equipment. Items such as the water door, cooling tower, and associated piping will be removed by the contractor. To facilitate his removal of the non-radioactive equipment, possibly contaminated equipment physically near will be removed early in the STIR dismantling.

and disposed of accordingly. The 50 Kw cooling system will likewise be checked for contamination and removed and disposed of accordingly. All signal cables will be removed and all electrical wiring will be removed back to the circuit breakers.

C. DISMANTLING OF TEST VAULT AREA

The water tank shield outside the freight door and the rails on which it runs will be removed. The channels provided for the rails in the concrete will be filled with concrete.

The concrete shield will be removed from the test carriage and broken up into pieces of manageable size for disposal. The test carriage and rails will be dismantled and disposed of as necessary. The drive mechanism and coolant hoses for the fission plate will be removed. The conveyor system in the stairwell and vault will be dismantled and removed. Miscellaneous hardware and equipment will be disposed of. The rails in the fission plate pit will be removed, but cleanup of the pit will be deferred until after the thermal column and pool tank have been removed.

D. REACTOR TANK - THERMAL COLUMN DISMANTLING

The grid plate, detector thimbles and internal piping will be removed from the reactor tank. The gravel in the annulus between the tank and the concrete liner will be taken out. The lead shot and bismuth "window" in the gamma shield will be removed. The aluminum tank will be cut into small sections and removed. The I-beam supports for the tank will be removed.

The cover plate on the test vault side of the thermal column will be taken off and the graphite logs removed. The aluminum liner will be removed.

The concrete around the reactor tank and around the thermal column will be checked for radioactivity and will be jackhammered or blasted out where necessary and disposed of. The concrete tank liner extending above floor level in the reactor room will be removed down to floor level. The storage wells in the reactor room floor will be decontaminated or removed for disposal. The gamma counter pit will be surveyed and decontaminated if radioactive.

E. FINAL CLEANUP

All debris from the dismantling work will be cleaned up and disposed of. A radiological survey will be made of all areas and a final cleanup will be done in those areas which are above permissible levels.

The filters in the building exhaust system will be removed and disposed of and ducting and stack checked for contamination. Any part of the exhaust system which is contaminated will either be cleaned or disposed of.

The thermal column will be plugged with concrete on the test vault side flush with the east wall of the test vault. The reactor tank cavity will be filled with sand and the top capped with concrete, flush with the reactor room floor.

VII. DOCUMENTATION

A. PROCEDURES

As indicated above, Activity Requirements and Detailed Working Procedures will be written to guide the decontamination and dismantling operations. Specific radiological and industrial safety hazards and the means for working with and elminating these hazards will be identified. The procedures will be consistent with the requirements of the Operational Safety Plan, and compliance with these requirements will be monitored by Quality Assurance and Health,Safety and Radiation Services. Detailed procedures will be released and controlled by the AI Engineering Data Release System.

B. REPORTING

Progress on the STIR D&D activities will be reported to ERDA in the Decontamination and Disposition of Facilities Program Monthly Report.

C. RECORD INFORMATION

The results of radiological surveys of the areas, materials, and equipment will be recorded. A complete accounting of all radioactivity disposed of by RMDF will be maintained. Photographic coverage of the more significant phases of dismantling will be obtained in still photos.

D. FINAL REPORT

The final report will describe the dismantling and decontamination activities. Problem areas and the subsequent solutions will be highlighted. Shipping records, showing quantities of material and the level of associated radioactivity, will be included. The report will contain the QA and HSRS records certifying the reported status of the STIR area upon completion.

REFERENCES

1) PP-704-990-002, Decontamination & Disposition of Facilities Program Plan, January 23, 1975

2) PP-704-990-001, Quality Assurance Program Plan for the Decontamination and Disposition of Facilities, Revision A, January 16, 1975

3) SRR-704-990-001, Operational Safety Plan for the AI Decontamination and Disposal of Facilities Program, Revision A, February 17, 1975

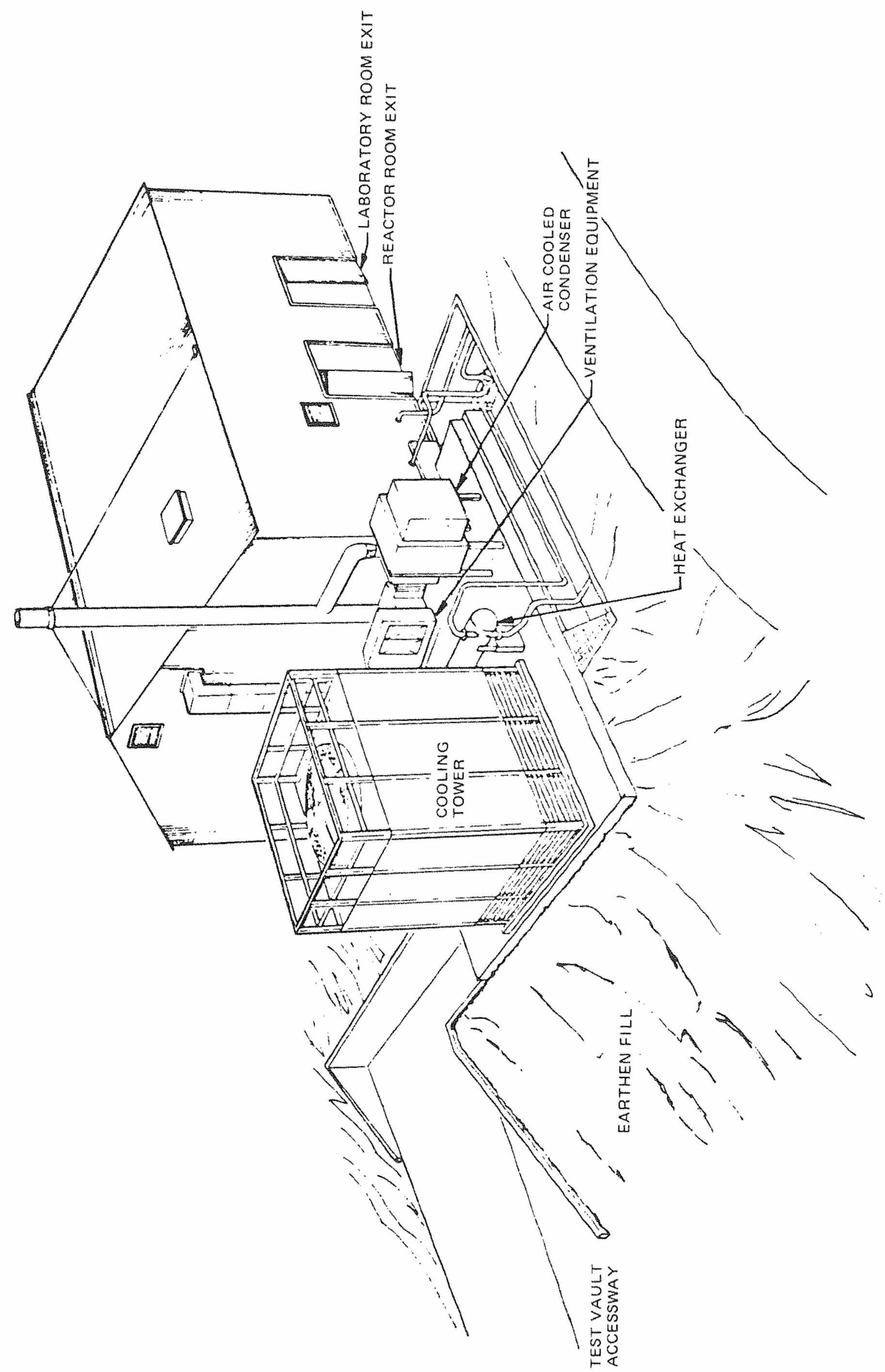

Figure 1. STIR Architectural Elevation

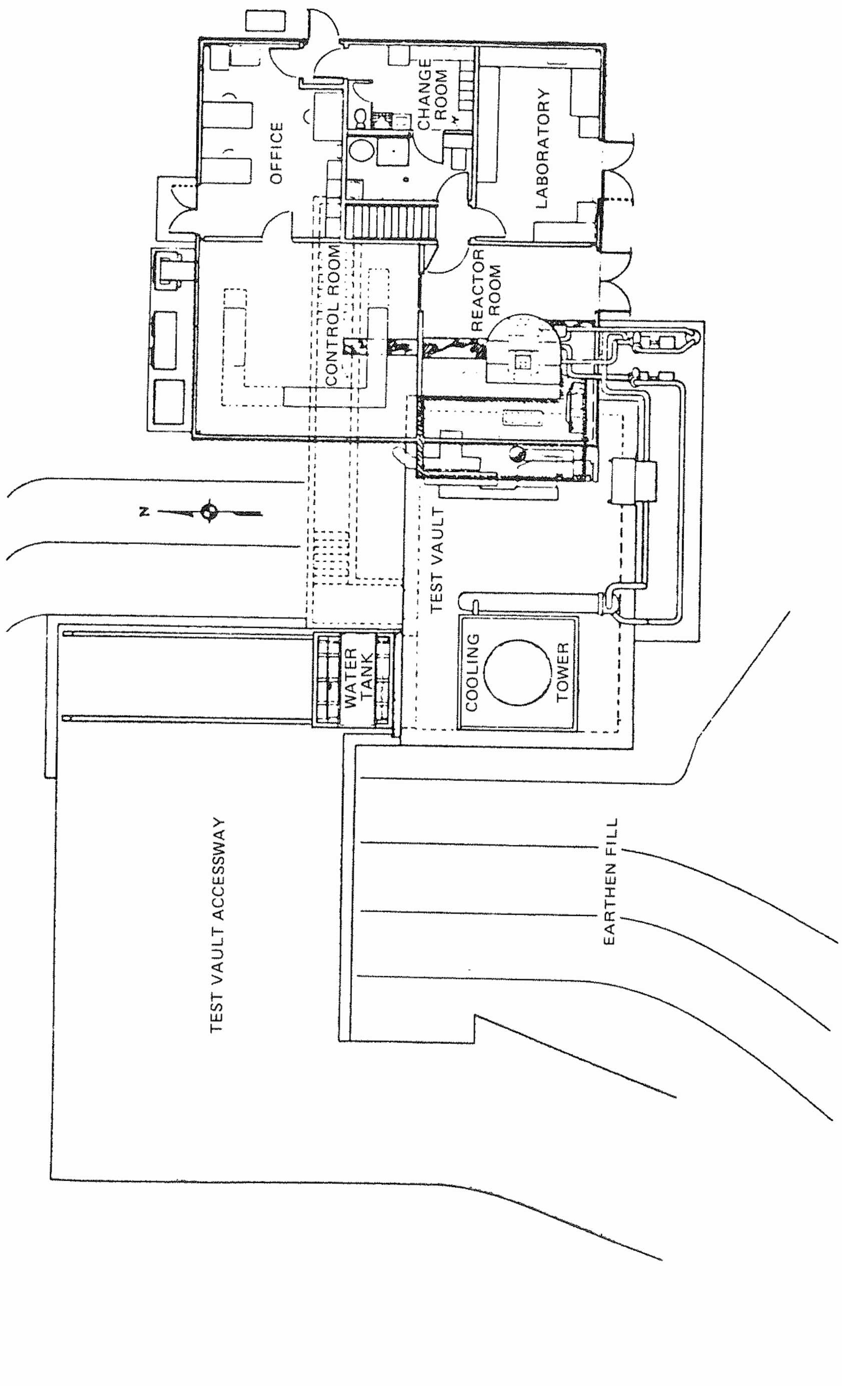

Figure 2. STIR Architectural Floor Plan

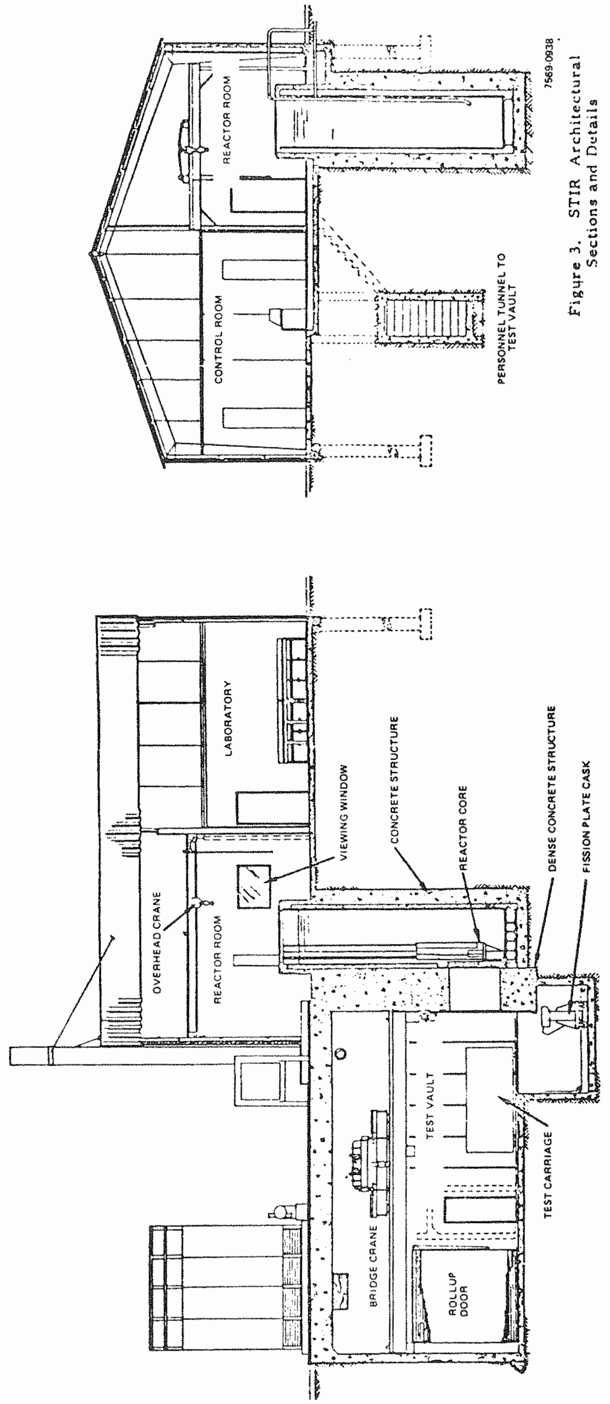

Figure 3. STIR Architectural Sections and Details

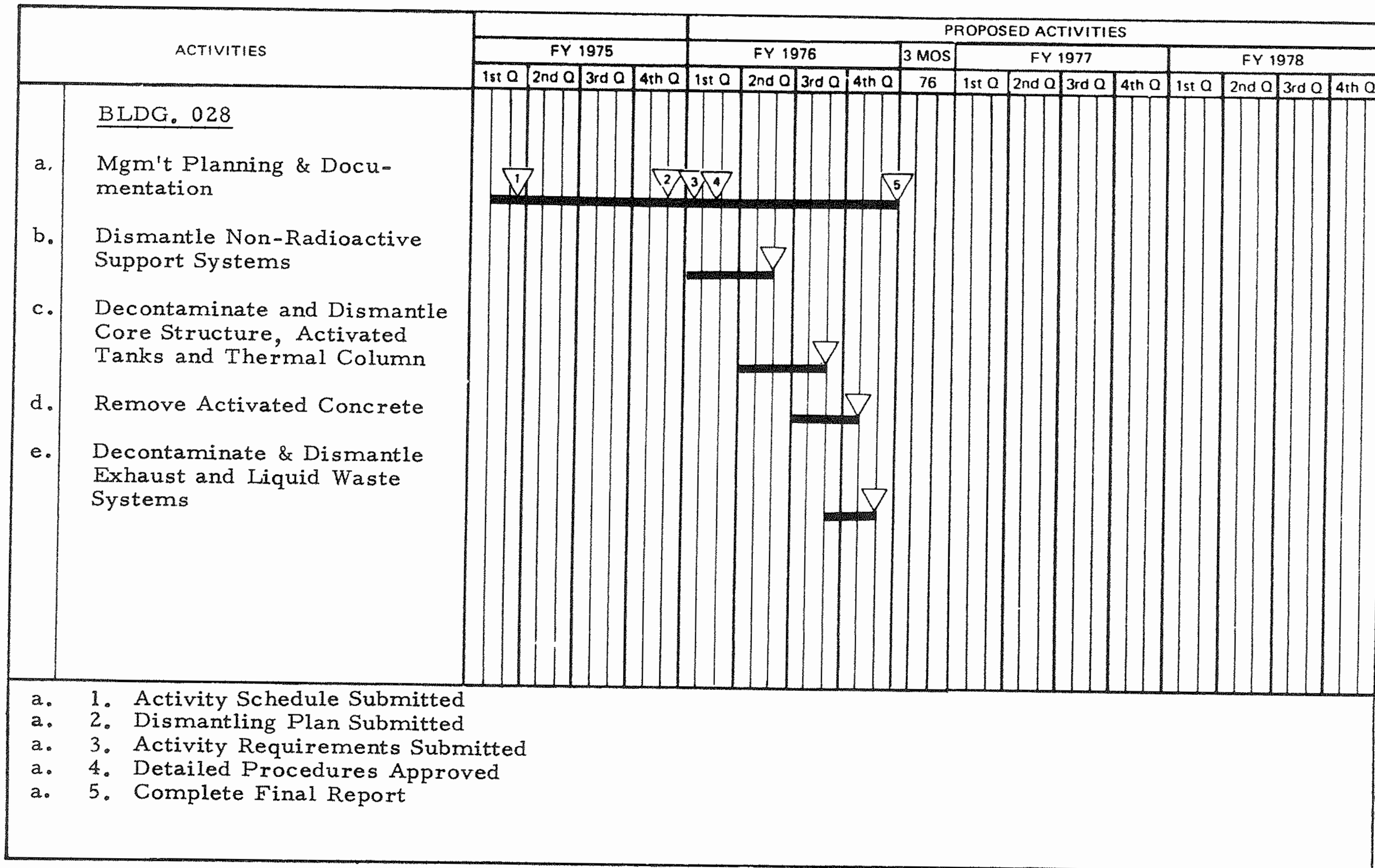

Figure 4. Decontamination and Disposition of Facilities Program, Program Activity Network, STIR Facilities

EXHIBIT V

FINAL DECONTAMINATION AND RADIOLOGICAL SURVEY OF BUILDING 028

Rockwell International
Rocketdyne Division

SUPPORTING DOCUMENT

GO NO.	S/A NO.	PAGE 1 OF	TOTAL PAGES	REV LTR/CHG NO. SEE SUMMARY OF CHG	NUMBER
96110		57	57	New	N704SRR990033

PROGRAM TITLE

Building T028 Decontamination

DOCUMENT TITLE

Final Decontamination and Radiological Survey of Building T028

DOCUMENT TYPE

Safety Review Report

KEY NOUNS

Decontamination, survey, release

ORIGINAL ISSUE DATE

REL. DATE

2-27-91 cw

PREPARED BY/DATE | DEPT | MAIL ADDR

B. M. Oliver | 635 | NA02

B.M. Oliver 2/2/91

IR&D PROGRAM? YES ☐ NO ☒

IF YES, ENTER TPA NO.

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ABSTRACT

SSFL Building T028 originally housed the STIR test reactor. The reactor was removed, and the facility was decontaminated and decommissioned for release for unrestricted use in 1976. Subsequent uranium-oxide melting experiments conducted in the facility resulted in some recontamination of equipment and localized areas. Following completion of the uranium experiments, the facility underwent additional decontamination and decommissioning operations in 1988, and the above-grade structures were demolished and removed in 1989. As part of this effort, a final radiological survey was conducted (prior to demolition), which included both indication-only surveys, and detailed grid surveys of total and removable alpha/beta activity, and ambient gamma exposure rate.

The results and statistical analysis of the final radiological survey data are presented in this report. The data show very small levels of residual radioactive contamination (just above background) in isolated areas of the facility, but at levels that are far below any regulatory limits. Ambient gamma exposure rates were observed to be slightly higher in the basement area, but still at levels below allowable limits. Some evidence of residual activation in the concrete wall adjacent to the removed STIR reactor was noted, but also at levels below the allowable limits. Based on the present results, it is concluded that residual radioactivity in all areas of Building T028 is below applicable limits. Therefore, these areas meet the requirements of DOE Order 5400.5, "Radiation Protection of the Public and Environment" (February 1990), and are suitable for release without radiological restrictions.

RESERVED FOR PROPRIETARY/LEGAL NOTICES

CONTENTS

CONTENTS

TABLES

FIGURES

FIGURES

1.0 INTRODUCTION

Decontamination and decommissioning (D&D) of a number of formerly used nuclear facilities and sites is underway at Rockwell International's Santa Susana Field Laboratories (SSFL). During D&D of these facilities, reasonable efforts are being made to eliminate radioactive contamination or to reduce residual contamination to levels that are as low as reasonably achievable (ALARA). Upon completion of D&D, radiological surveys are performed, using formal procedures, to determine that any remaining radioactivity does not exceed applicable regulatory limits. The scope of these surveys includes both known and suspected areas of contamination.

To promote efficient use of the facilities at SSFL, buildings are often decontaminated and decommissioned following one use with radioactive materials and then reused in new projects that may or may not involve radioactive materials. Building T028 has been recycled in this manner, starting as a research reactor facility which was decommissioned, and then reused for research on simulated accident conditions involving molten uranium oxide. Following completion of this latter project, it was determined that there would be no future need for the building and so the subsequent decontamination was followed by demolition of the above-grade structures.

Prior to demolition, the building was completely surveyed for detectable radioactive contamination. Small areas that indicated some contamination were completely decontaminated before releasing the above-grade structure to contractors for demolition. All contaminated material was sent to the RMDF for eventual disposal at an authorized site. After the sections of the building that were to remain in place were decontaminated, a final radiological survey was performed. The results of the final radiological survey are described in this report.

The findings presented in this SRR include a statistical treatment of measured gamma radiation exposure rates and surface contamination from sections of the above-grade structure prior to demolition, and the present below-grade portion of Building T028. The gamma exposure rates and the surface contamination are compared with regulatory acceptance limits. These comparisons show that residual radioactivity is well below acceptable levels and that the remaining structure is suitable for release without radiological restrictions.

This report is organized as follows: A background on Building T028 that includes its location and operating history is provided in the next section (Section 2). The scope of the survey and applicable regulatory limits are given in Section 3. Section 4 summarizes the statistical techniques used to interpret the survey data. Section 5 summarizes the survey methods and procedures. Results are provided and discussed in Section 6, and Section 7 states the conclusions drawn from the review.

Additional data and information pertaining to Building T028 are provided in Appendices A through E. Appendix A describes the method used to determine the applicable regulatory limit for ambient gamma exposure rate above background in a concrete vault area; Appendices B, C, and D list the various radiological data obtained during the final release survey; and Appendix E provides a list of items collected during the decontamination and decommissioning operation which are archived at Rockwell.

2.0 BACKGROUND

2.1 LOCATION

Building T028 is located within Rockwell International's Santa Susana Field Laboratory (SSFL) in the Simi Hills of southeastern Ventura County, California, adjacent to the Los Angeles County Line and approximately 29 miles northwest of downtown Los Angeles. Location of the SSFL relative to Los Angeles and vicinity is shown in Figure 1. An enlarged map of neighboring SSFL communities is shown in Figure 2. Figure 3 is a plot plan of the western portion of SSFL, known as area IV where Building T028 is located. A drawing (plan view) of Building T028, as it existed prior to above-grade demolition, is shown in Figure 4.

Figure 5 shows the relevant portion of a 1967 edition of the U.S. Geological Survey (USGS) topographic map of the Calabasas Quadrangle where the SSFL is located. Using USGS terminology, the description for Building T028 is: Section 25 of Township T2N: Range R18W: Calabasas Quadrangle.

2.2 AREA CHARACTERISTICS

Figures 6 and 7 are photographs of Building T028 taken from the west end of the facility. Figure 6 shows the remaining slab floor after demolition and removal of the above-grade structures. Figure 7 shows the remaining below-grade structure, consisting of the original test vault area.

The terrain throughout most of the SSFL areas is uneven due to rock outcroppings. Rock outcroppings exist upslope from the facility to the north, and to the south and west. Water runoff is primarily to the west at the western end of the facility. Surrounding the facility in all directions is asphalt paving. The minimum distance to the SSFL boundary is approximately 300 ft. This boundary lies in a northeasterly direction (Simi Valley direction). Grade floor elevation is approximately 1,800 ft above sea level.

2.3 OPERATING HISTORY

Building T028 was originally constructed to perform tests of space reactor shields using a fission plate driven by neutrons from the thermal column of a 50-kW swimming pool-type reactor. This reactor was designated the Shield Test Reactor (STR) and operated from 1961 to 1964, when it was modified to operate at 1 MW. This latter configuration was renamed the Shield Test and Irradiation Reactor (STIR) and operated through 1972. Following shutdown of the test program and removal of the reactor, the facility was decommissioned and made available for alternate use in March 1976 (Ref. 1).

In 1977, operations were started to investigate the behavior of molten UO_2 relative to simulated reactor accidents; in particular, its reaction with floor and structural

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Figure 1. Map of Los Angeles Area

Figure 2. Map of Neighboring SSFL Communities

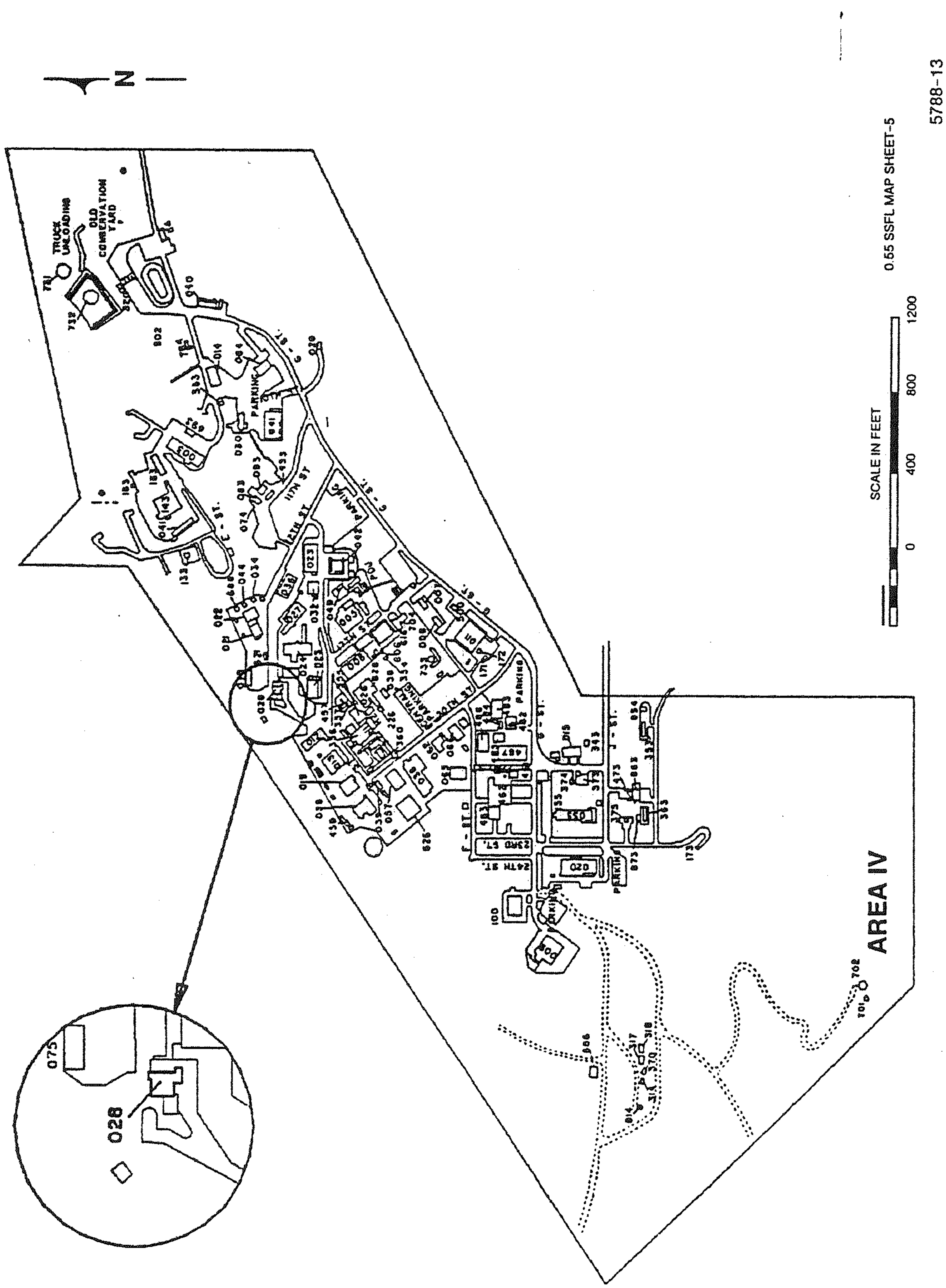


Figure 3. SSFL Layout Showing Location of Building T028

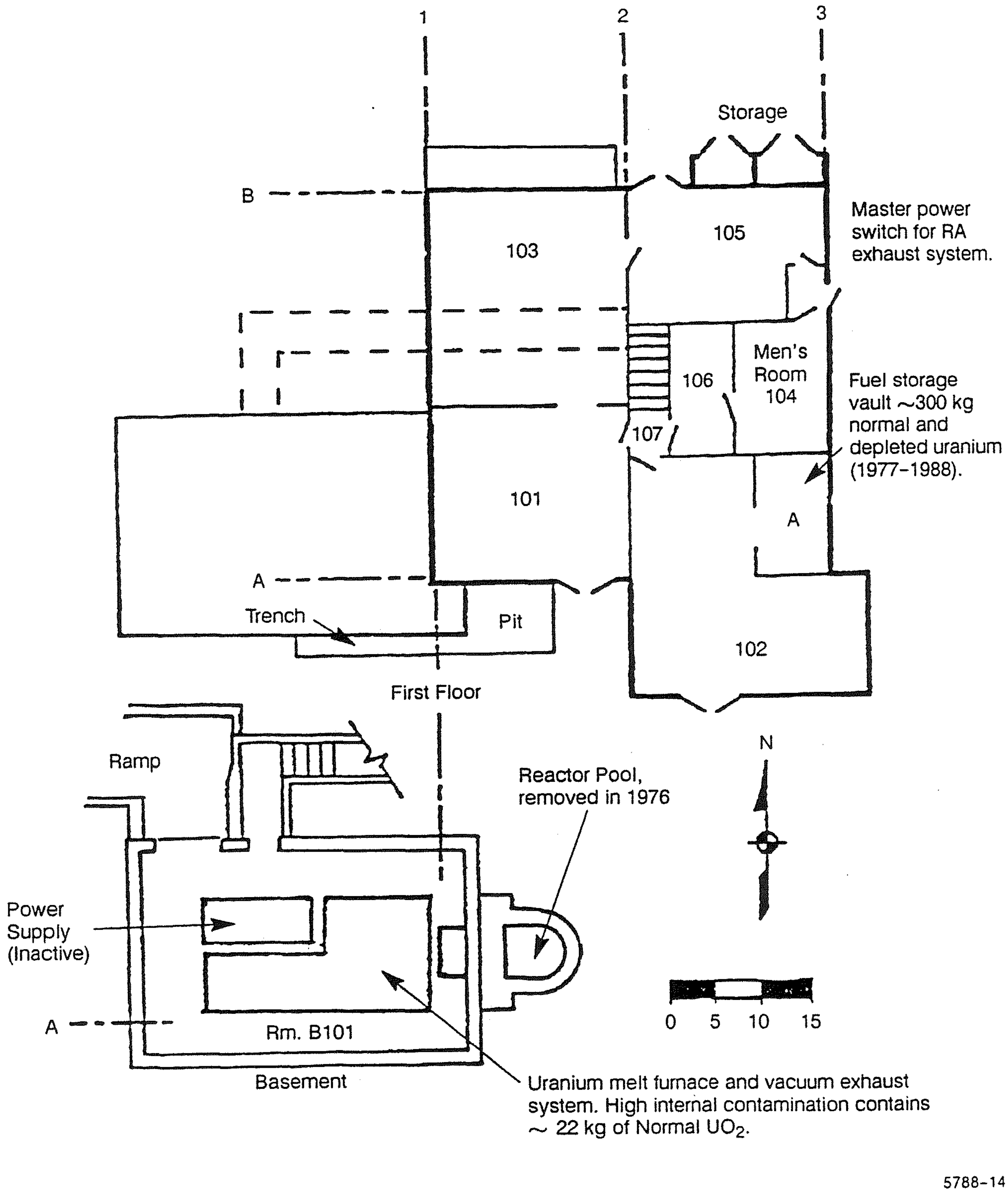
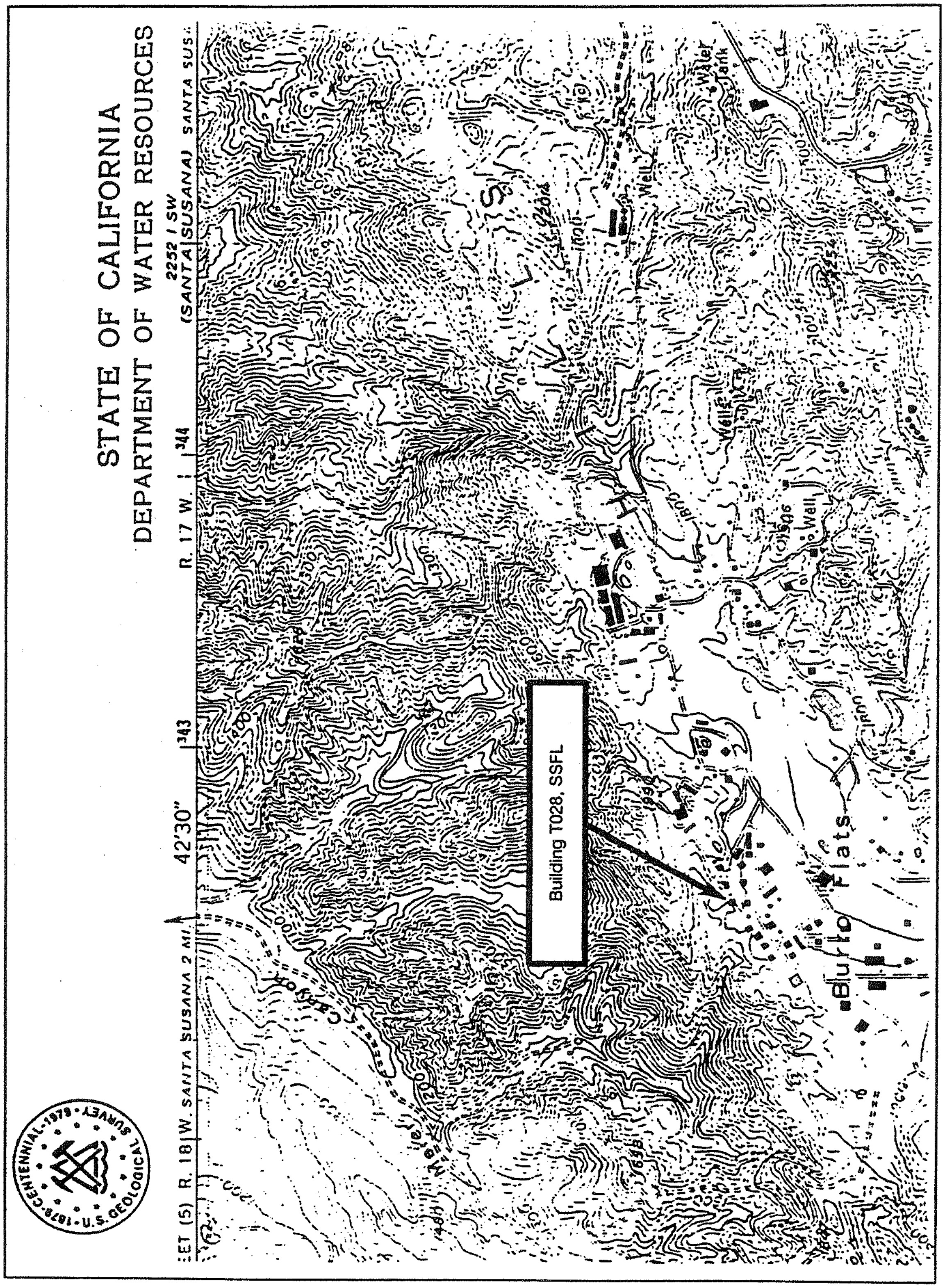


Figure 4. Plan View of Building T028 Prior to Decontamination and Above-Grade Demolition (1977–1988)



5788-15

Figure 5. USGS Topographic Map of Portion of Calabasas Quadrangle (Bottom-Left Area Corresponds to the SSFL)

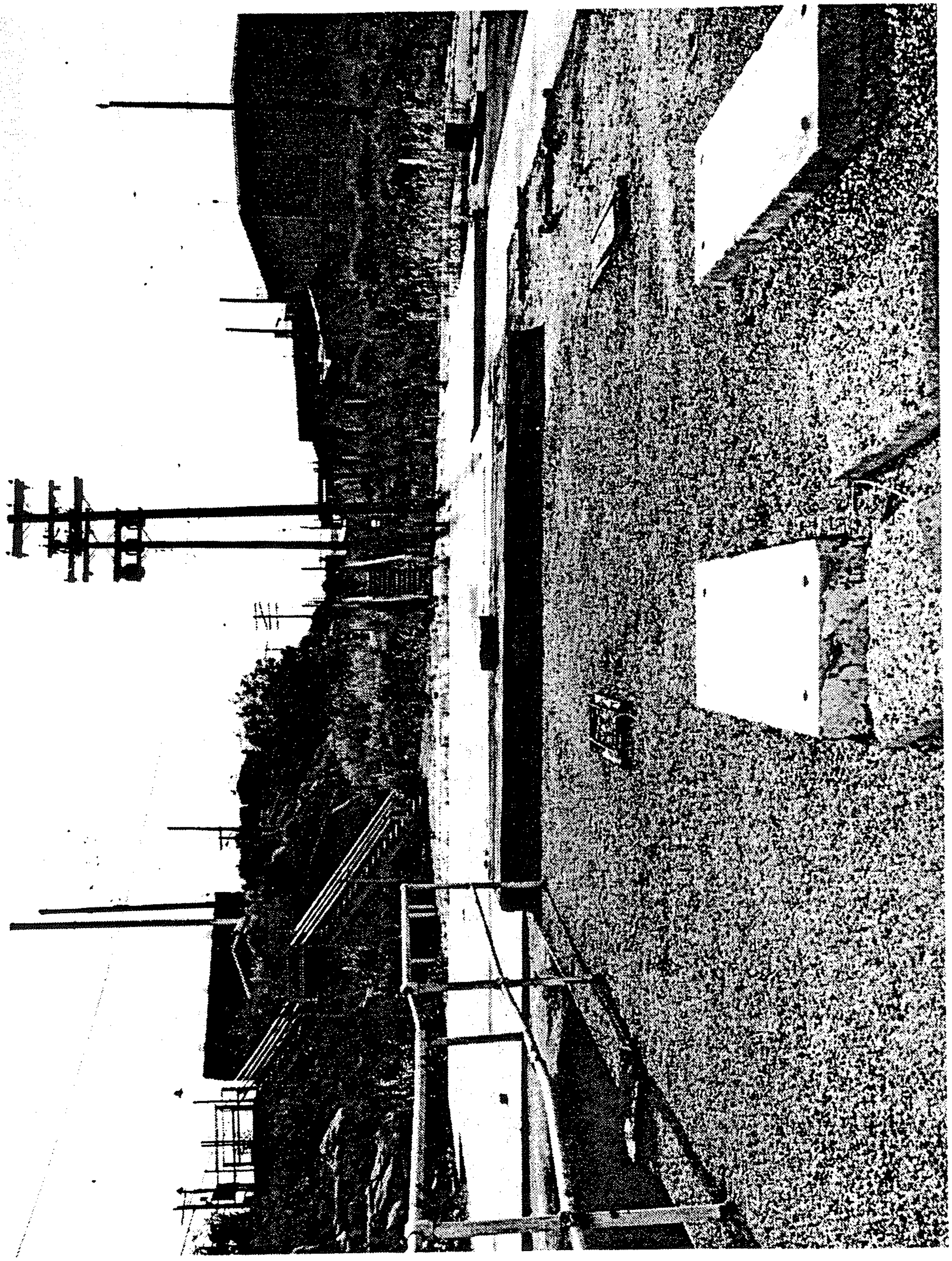


Figure 6. Above-Grade Portion of Building T028 After Demolition

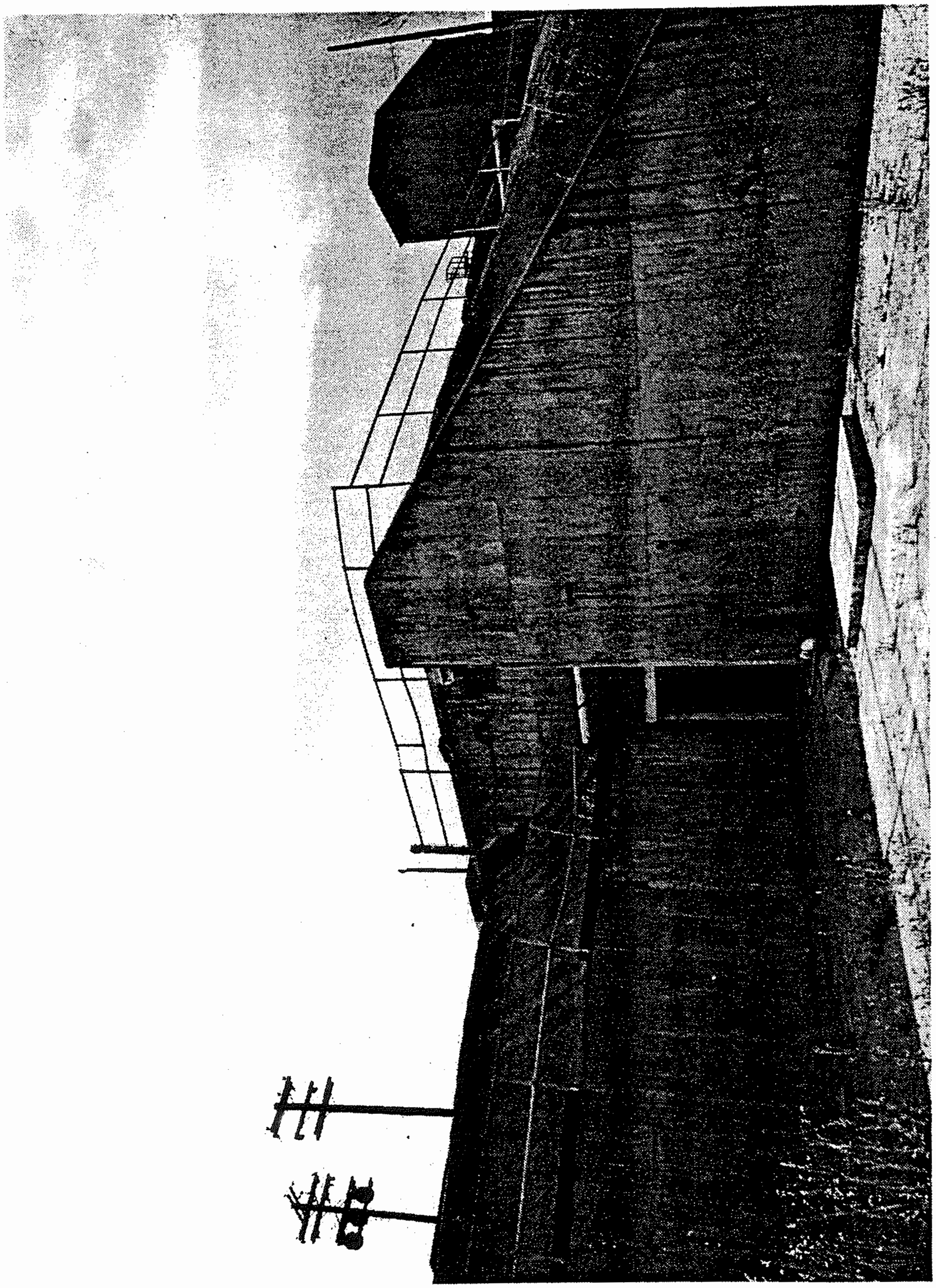


Figure 7. Existing Below-Grade Portion of Building T028

materials . These experiments resulted in some recontamination of various parts of the building that were used for the preparation and the melting of the UO_2. Tests continued intermittently through 1981. Some facility modifications were done after that, and a decision to terminate operations was made in 1984. The building remained inactive, under periodic surveillance, until 1988 when cleanout and decontamination began.

In April 1989 it was determined that there was no remaining radioactive contamination in the above–grade portion of the building and that part of the structure was demolished. Only the concrete floor and the below–grade test vault and stairway currently remain.

2.4 SUMMARY OF DECONTAMINATION ACTIVITIES

Decontamination and decommissioning (D&D) of the Building T028 facility occurred from about July through December 1988. All work was done following approved written procedures. Details of the work are discussed in Ref. 2.

Briefly, the D&D steps involved were (1) removal of surplus normal and depleted uranium oxide; (2) decontamination and removal of equipment and electrical components, including the furnace system used for the uranium–oxide experiments; (3) removal of the R/A ducting system; (4) building surfaces decontamination, including scabbling of Room 101A concrete floor; (5) final miscellaneous cleanup operations; and (6) final radiological survey of the T028 building facility (above–grade and basement).

Following qualitative analysis of the final radiological survey data, which showed no residual radionuclide contamination above acceptable levels (Ref. 3), the building was released to Taylor Wrecking Co. for demolition and removal of the above–grade structures. The structure demolition and removal work was completed in July 1989.

All radioactive waste from the facility D&D was sent to the RMDF for packaging and shipment to Hanford, Washington. A total of about 1,200 ft^3 of waste was shipped to Hanford.

3.0 SCOPE OF SURVEY

The scope of the decontamination and decommissioning (D&D) survey included radiological inspections of the interior above-grade building areas and the basement area which had been used previously for uranium-oxide melting experiments. With the exception of Rooms 102 and 102A, all above-grade interior rooms were inspected by indication-only surveys with a μR survey meter and with a thin-window pancake GM survey meter. No activity above background levels was observed in these areas.

Rooms 102, 102A, and the basement (Room B101), which had potential for uranium and activation product contamination, were quantitatively characterized by measuring total and removable alpha/beta activity on surfaces, and ambient gamma exposure rates 1 m above the floor. Total and removable alpha/beta activity was measured in 67, 1 m^2 wall, floor, and ceiling locations, and in 30 selected areas covering several structural features remaining in B101. Gamma exposure rates were measured in 29 floor grids, ranging in size from ~1 m to 2.4 m in size.

Total and removable alpha/beta surface contamination is reported in disintegrations per minute per 100 cm^2 area (dpm/100 cm^2). Indication-only alpha/beta measurements are reported as No Detectable Activity (NDA), or less than 20 or 50 dpm/100 cm^2, respectively. Ambient gamma exposure rates are reported in micro-roentgens per hour (μR/h). All quantitative data were statistically analyzed against appropriate residual contamination acceptance limits.

3.1 UNRESTRICTED-USE ACCEPTANCE LIMITS

Comparison of the survey data with unrestricted-use acceptance limits was performed using a statistical sampling inspection by variables. This approach is discussed further in Section 4. Acceptance limits for contamination and gamma exposure rates are those prescribed in DOE guidelines (Ref. 4), Regulatory Guide 1.86, NRC license SNM-21, and other references.

Typically, the lowest (most conservative) limits are chosen. Table 1 shows the composite of conservative limits derived from the aforementioned references and adopted by Rocketdyne with respect to Building T028.

Two limits are indicated for ambient gamma exposure rate. The first, 5μR/h above background, applies to large open areas outside buildings or to the interior of buildings with standard concrete slab floor and standard above-grade construction. For areas either largely or totally surrounded by concrete, such as shielded basement rooms, a somewhat modified approach can be taken, and this is discussed below.

3.1.1 Radiation Exposure Limit in Concrete Vaults

The State of California Radiologic Health Branch has recognized the difficulty in determining the exposure rate corresponding to "natural" background in a concrete vault. An approximate estimate has been developed by the State, as described in Appendix A, that permits estimation of the interior background as being 3 μR/h greater than the outside background. This discussion also presents the consideration that because of the all-encompassing nature of a concrete vault, 10 μR/h at one meter from a surface closely corresponds to 5 μR/h from a single plane area, and states that a limit of 10 μR/h may be applied in such circumstances.

Table 1. Maximum Acceptable Contamination Limits

Parameter	Limit
Total surface alpha/beta activity	5,000 dpm/100 cm^2
Removable surface alpha/beta activity	1,000 dpm/100 cm^2
Ambient gamma exposure rate (at 1 m)	5 μR/h above background[a]
	10 μR/h above background[b]

[a] Limit applicable for outside areas or building interiors with standard slab floor and standard above-grade construction. The average background gamma exposure rate at the SSFL has a value of about 15 μR/h with a range (maximum–minimum) of about 4 μR/h. Although DOE guidelines (Ref. 5) recommend a value of 20 μR/h above background for gamma exposure rate, the NRC Dismantling Order for the L-85 reactor decommissioning (Ref. 6) required 5 μR/h above background. For conservatism, 5 μR/h above background is used at Rocketdyne to compare survey results.

[b] Limit applicable for areas such as concrete shielded vaults. For this case, the ambient gamma rate background to be applied is the average outside ambient gamma rate increased by 3 μR/h (see text and Appendix A).

3.2 ACTION LEVELS

Three specific action levels were established for the survey. If the surveyor detected radiation, action was initiated according to the following criteria:

1. **Characterization Level.** That level of exposure rate which is less than 50% of the maximum acceptable limit. This level encompasses the range of natural background levels at the SSFL and requires no further action.

2. **Reinspection Level.** That level of exposure rate which is between 50% and 90% of the maximum acceptable limit. A general survey of the area and a resampling of the area are required in this case.

3. **Investigation Level**. That level of exposure rate which exceeds 90% of the maximum acceptable limit. Specific investigation of the occurrence is required in this case.

4.0 STATISTICAL ANALYSIS

A statistical procedure is required to validate the applicability of radiological survey data collected at selected locations to an entire area or region. A statistical method known as "sampling inspection by variables" (Ref. 7) was used to analyze the data from the present survey. This method has been widely applied in industry and the military and it is both effective and efficient for cases where destructive tests must be performed (e.g., in quality control) or for cases where the lot size is impractically large. A detailed description of this method is given in Ref. 8. For completeness, however, the technique is summarized below.

In sampling inspections by variables, the number of data points on which measurements are obtained is first chosen to be sufficiently large (greater than ~30) so that the distribution of the data should be normal (i.e., Gaussian). The mean of the distribution, x_m, and its standard deviation, s, then determine a "test statistic," TS, as follows:

$$TS = x_m + ks.$$

TS and x_m are compared with an acceptance limit, U, to determine acceptance or other plans of action, including rejection of the area. In the above expression k is known as the tolerance factor. The value of k is determined from the sample size and two other statistical sampling coefficients that are related to the "consumer's risk" of accepting a lot, given that a fraction of the lot has rejectable items in it. The values chosen for these coefficients for the survey correspond to assuring with 90% confidence, that 90% of the area has residual contamination below 100% of the applicable limit (a 90/90/100 test). The choice of values for the two coefficients is consistent with industrial sampling practices and State of California guidelines (Ref. 9).

Data from the present survey are treated using this statistical approach. The reduced data are plotted against the cumulative Gaussian probability on a probability–grade scale. Display of data in this manner permits clear identification of values with significantly greater exposure rates than expected for the lot, based on a Gaussian distribution.

5.0 SURVEY METHODS AND PROCEDURES

A detailed working procedure was developed and used for the final radiological survey of Building T028 (Ref. 10). Relevant details from this procedure are repeated below.

5.1 SAMPLING PLAN

For the final radiological survey, Building T028 was divided into two general survey areas. These areas were those determined to have potential for radionuclide contamination based on the previous use history of the building. The selected areas were the above–grade rooms 102 and 102A, and the basement test vault, Room B101. Details on the sampling plan for both areas are given below.

For purposes of the present final radiological survey report, total and removable alpha/beta contamination measurements from both areas were treated together as two individual lots for comparison with acceptance limits. Because of expected increased gamma exposure rate in a concrete vault (see Table 1), however, the gamma exposure rate measurements for the above–grade sections (Rooms 102 and 102A) were treated separately from the basement measurements (Room B101).

All other areas of the facility were given an indication–only survey using a µR survey meter and a thin–window pancake GM survey meter. No above normal indications were seen in the indication–only survey.

5.1.1 Walls, Floors, and Ceilings

A minimum of 11% of the total surface area of all walls, floors, and ceilings was surveyed. Measurements for total and removable alpha/beta contamination, and ambient gamma exposure rates, were made in the three sample lots. The sampling inspection plan used was based on a uniform 3–m square grid (9 m^2) superimposed on a uniform inspection area. A 3–m square grid has been adopted to be consistent with NRC and State of California guidance for releasing a facility for unrestricted use. Within each 3–m x 3–m grid, one 1–m x 1–m area was selected for survey. This area was randomly selected, except, that where possible, it was biased toward that area which was expected to have the highest contamination level. To obtain a sufficient number of data points for later statistical analysis, a higher density of sampling grids, within each 3–m x 3–m area, was used for the ambient gamma exposure rate measurements.

A grid was superimposed on walls, floors, and ceilings. Each survey area was designated by its location and number (e.g., F–1 indicates the number 1 floor grid). A drawing was made of each area to clearly show the location of each survey grid. This gridding arrangement resulted in obtaining 67 total and removable alpha/beta measurements and 29 ambient gamma exposure–rate measurements for the T028 facility. Figure 8 shows the basement vault Room B101, with some of the 1–m square survey grids indicated.

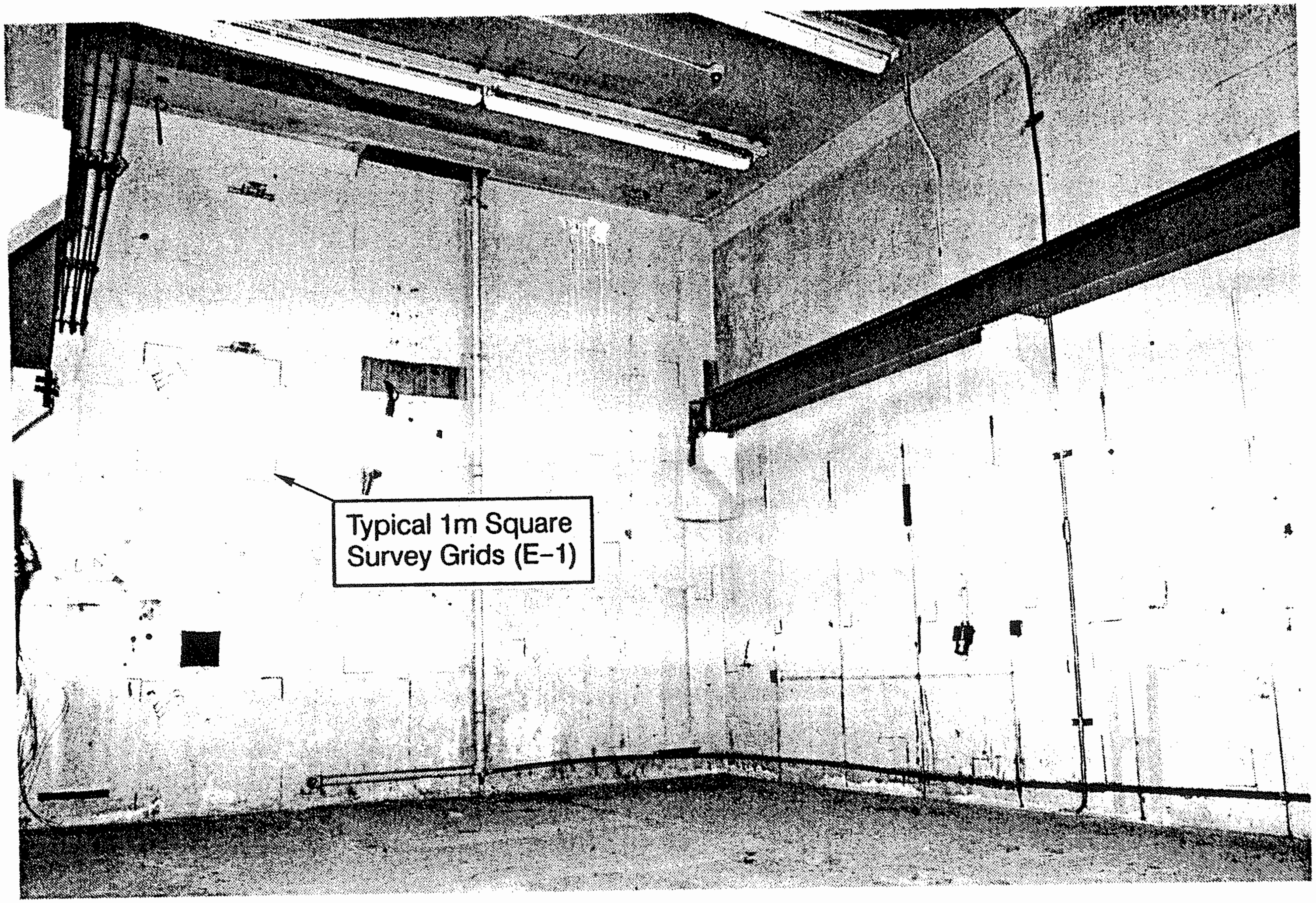


Figure 8. Building T028 Test Vault (Room B101) After Decontamination. Evident in the photograph are some of the 1 m square grids used in the final radiological survey.

5.1.2 Structural Surfaces

Structural surfaces consisted of beams, pipes, conduits, and other surfaces that were not amenable to large surface measurements. Except as otherwise noted, for these surfaces, 20% of the surface area was surveyed. Structural surfaces surveyed included the overhead bridge crane and rails, and light fixtures, all in Room B101.

5.2 DATA ACQUISITION

In each selected survey area, total and removable alpha/beta contamination and ambient gamma exposure rates were measured. The exact location within the survey area where the measurements were made was left to the surveyor's judgment; it was to be the area that was most likely to have retained the greatest amount of contamination. This decision was based on surface discoloration, stains, or chemical residues, debris, and crevices or cracks in tile and concrete. This procedure provides a uniform survey biased toward the high end of the distribution. Locations of noticeably greater radioactivity were to be noted. Upon any indication, surrounding locations were to be surveyed.

5.3 DATA REDUCTION

Each radiological measurement data value was input into a spreadsheet code developed at Rocketdyne. This code allows multiple computations to be performed on raw data values. Columns were established to calculate the total, maximum, and removable alpha/beta contamination per 1 m^2 in dpm/100 cm^2 and surface ambient gamma exposure rate in μR/h. The standard deviation of each measurement was also calculated. Software was developed in Microsoft QuickBASIC® to read data from the spreadsheet file and then plot the radiological measurements against the Gaussian cumulative distribution function. For convenience, the distribution function, G(x), is plotted as the abscissa (probability grades), and x, the measurement value, is plotted as the ordinate (linear grades).

Input for this data reduction was:

1. Room number
2. Grid location; e.g., W-1 west wall, grid 1
3. Alpha total activity, averaged over 1 m^2 (counts in 5 min)
4. Alpha maximum activity for hot spot, if present (counts in 5 min)
5. Alpha removable activity from 100 cm^2 smear (counts in 5 min)
6. Beta total activity, averaged over 1 m^2 (counts in 5 min)
7. Beta maximum activity for hot spot, if present (counts in 5 min)
8. Beta removable activity from 100 cm^2 smear (counts in 5 min)

9. Alpha survey instrument background (5 min), efficiency factor (dpm/cpm), and area factor
10. Alpha gas-proportional detector background (5 min) and efficiency factor (dpm/cpm)
11. Beta survey instrument background (5 min), efficiency factor (dpm/cpm), and area factor
12. Beta gas-proportional detector background (5 min) and efficiency factor (dpm/cpm)
13. Ambient gamma exposure rate (counts in 5 min, cpm)
14. Gamma survey instrument background (5 min)
15. Gamma survey instrument efficiency factor (μR/h/cpm)

Output for the Gaussian plots was:

1. Alpha total activity averaged over 1 m^2 and standard deviation (dpm/100 cm^2)
2. Alpha maximum activity and standard deviation (dpm/100 cm^2), only if observed
3. Alpha removable activity and standard deviation (dpm/100 cm^2)
4. Beta total activity averaged over 1 m^2 and standard deviation (dpm/100 cm^2)
5. Beta maximum activity and standard deviation (dpm/100 cm^2), only if observed
6. Beta removable activity and standard deviation (dpm/100 cm^2)
7. Ambient gamma exposure rate and standard deviation (μR/h)

5.4 DATA ANALYSIS

An arithmetic mean and standard deviation of the radiological measurement values is calculated for each data set. The test statistic, x_m + ks, is also calculated for each distribution. The acceptance criteria presented in Section 3.2 is applied to each sampling distribution using the acceptance limits given in Table 1.

From the plot of measurement values vs cumulative probability, and assuming a Gaussian distribution of data, the mean radiological value of the lot is the point on the ordinate axis where the distribution intersects the 50% cumulative probability. When an acceptance limit is applied to a test case, a horizontal line is displayed on the graph at the acceptance limit for comparison with the calculated test statistic, TS.

5.5 DIRECT ALPHA/BETA CONTAMINATION MEASUREMENTS

Direct alpha/beta contamination measurements were made using Ludlum model 2220–ESG portable scalers to detect pulses from a Ludlum 43–1 alpha scintillation probe and a Ludlum 44–9 thin-window pancake GM beta probe, respectively.

5.5.1 Instrument Calibration

Each detector was calibrated two or three times daily by the operator (see Ref. 10). The alpha detector was calibrated with ^{230}Th; the beta detector with ^{99}Tc. Background levels were determined by 5–min measurements on a representative area outside the facility survey plan.

5.5.2 Data Acquisition and Reduction

Each location where a measurement was made was identified on a map and in matrix notation. The gross number of alpha and beta counts recorded in 5 min along with the matrix notation location was input into the spreadsheet code. Columns were established to calculate total–average alpha and beta surface activity and the standard deviation (in dpm/100 cm^2) according to Eq. 5–1 and 5–2. Conversion from gross counts observed to dpm/100 cm^2 is given by:

$$SA = \frac{(C - B) \cdot EF \cdot 100}{5 \cdot A} \qquad (5\text{–}1)$$

where

SA = surface activity
C = total counts in 5 min
5 = count time, min
B = background count in 5 min (generally 0–5 for alpha and about 440–460 for beta)
EF = Efficiency factor, dpm/cpm (averages about 4.8 for alpha and about 3.5 for beta)
100 = 100 cm^2 standard area
A = probe sensitive area (71 cm^2 for Ludlum model 43–1 circular alpha scintillator; 20 cm^2 for Ludlum model 44–9 pancake GM).

Note that the analysis is done using counts rather than count rates. The standard deviation of the measurement in dpm/100 cm^2 is given by:

$$S = \frac{\sqrt{C + B} \cdot EF \cdot 100}{5 \cdot A} \qquad (5\text{–}2)$$

5.5.3 Data Analysis

Total–average alpha/beta radioactivity in dpm/100 cm^2 per square meter were plotted, in order of magnitude from left to right, against the cumulative probability. The test statistic, $x_m + ks$, was also calculated for the lot, and compared against the acceptance limits in Table 1. Criteria for accepting the area as uncontaminated are presented in Section 3.2.

If the measurements taken are represented by a Gaussian distribution, the data will fall along a straight line. Large breaks or changes in slope in the distribution will indicate some specific areas are contaminated to differing levels.

5.6 REMOVABLE ALPHA/BETA CONTAMINATION MEASUREMENTS

A 100 cm^2 area of each square meter surveyed for fixed alpha/beta contamination was sampled for removable alpha/beta contamination. Each smear sample was placed in a gas–flow proportional counter for analysis.

5.6.1 Instrument Calibration

The Canberra Model 2201 gas–flow proportional counter was calibrated twice daily by the operator (Ref. 10). Alpha efficiencies were determined by using a ^{230}Th calibration source. Beta efficiencies were determined by using a ^{99}Tc calibration source. A "clean" smear–paper was used to determine background radiation levels.

5.6.2 Data Acquisition and Reduction

Gross alpha and beta counts for each sample location were entered into the spreadsheet code. Columns were established for input of instrument efficiency and background. Removable surface activity is converted to dpm/100 cm^2 by the expression:

$$SA = \frac{(C - B) \cdot EF}{5} \qquad (5\text{–}3)$$

where the appropriate alpha and beta backgrounds and efficiency factors were used. Backgrounds (B) are typically 0–2 counts for alpha and 40–50 counts for beta in a 5–min time period. Efficiency factors (EF) are about 3.5 for alpha and 3.9 for beta.

The standard deviation, s, of this measurement (in dpm/100 cm^2) is:

$$s = \frac{\sqrt{(C + B)} \cdot EF}{5} \qquad (5\text{–}4)$$

5.6.3 Data Analysis

Removable alpha/beta radioactivity in dpm/100 cm^2 per square meter were plotted, in order of magnitude from left to right, against the cumulative probability. The same analytical criteria apply here as those presented in Section 5.5.3.

5.7 AMBIENT GAMMA EXPOSURE RATE

Measurements of ambient gamma exposure rate were made by using a 1 in. by 1 in. NaI scintillation crystal coupled to a Ludlum Model 2220–ESG portable scaler. This device was mounted on a tripod so that the sensitive crystal was 1 m from the floor. The detector is nearly equally sensitive in all directions, i.e., 4π geometry, and can detect variations in exposure rate down to about 0.5 μR/h, using the digital scaler for a 1–min count time. Because of the natural variability of ambient radiation (particularly outdoors), a 3 to 5 μR/h exposure rate above "background" is considered the practical instrument sensitivity in terms of identifying increased exposure values. At this level, a surveyor would decide to collect additional measurements.

5.7.1 Instrument Calibration

The gamma detection system is calibrated quarterly using ^{137}Cs as the calibration source. A voltage plateau is plotted and the voltage is set at a nominal 800 V. The detector is placed on a calibration range and readings taken at 5, 2, 1, 0.9, 0.5, 0.4, 0.3, and 0.2 μR/h. A detector efficiency plot as a function of exposure rate is then generated.

Because of an exposure rate–dependent effect and because the calibration range does not read less than 200 μR/h, this instrument was cross–calibrated against a Reuter Stokes High Pressure Ion Chamber (HPIC). Count rates were converted to exposure rates using the relationship that 215 cpm = 1 μR/h, at background exposure rates.

Instrument response was checked three times a day using a ^{226}Ra source. The source was placed 1 ft from the detector and counted for 5 min. If the scaler reading fell within $\pm 5\%$ of the nominal value, then the instrument was qualified as operable for the day, under the calibration conditions previously described. Recalibration because of "instrument out of tolerance" was not necessary during the period this survey took place.

5.7.2 Data Acquisition and Reduction

Each location where a gamma measurement was made was identified on a map and in matrix notation. The gross number of counts recorded in 5 min along with the matrix location was put into the spreadsheet code. Columns were established to calculate the total exposure rate (μR/h) and its standard deviation according to Eq. 5–5 and 5–6. Gamma scintillations produced by a NaI detector were converted from gross counts to exposure rate R (μR/h) by:

$$R = \frac{C \cdot EF}{5} \tag{5-5}$$

where

C = gross counts in 5 min (cpm)
EF = efficiency factor (0.00465 μR/h/cpm) based on cross calibration with the HPIC.

The standard deviation, s, of a single measurement then becomes Eq. 5-6:

$$s = \frac{\sqrt{C} \cdot EF}{5} \tag{5-6}$$

5.7.3 Data Analysis

Analysis and interpretation of gamma exposure rate data is a five-step process:

1. Plot, in order of magnitude from left to right, total-gross exposure rates in μR/h against cumulative probability for at least three independent areas considered to be "natural background" at SSFL. These survey locations should be from areas where no radioactive material has ever been used, handled, stored, or disposed. If available, these areas should be of similar geologic characteristics to those of the inspected areas. Calculate the average, standard deviation, and range for each distribution. These distributions give the baseline for "natural" variability of exposure rate as a function of SSFL terrain.

2. Plot total-gross exposure rates in μR/h against the cumulative probability for each subject sampling lot. Calculate the average, standard deviation, and range for each distribution. Compare these statistics and probability distributions against "natural background" distributions.

3. Determine if there are any trends indicated by the probability plots of each subject sampling lot which show a potentially contaminated area. If necessary, investigate elevated measurements and/or trends in the distribution.

4. Determine whether the "natural background" distributions adequately represent "ambient background" for the tested areas. Determine if any nuclear-related operations in the local area are influencing "ambient background" in the test area. If so, make corrections.

5. Subtract the estimated "natural background" from each test-area measurement and compare the results against the acceptance criteria listed in Table 1. Use inspection by variables techniques to test for acceptance. Calculate the average, standard deviation, and test statistic, $x_m + ks$, for each test-area distribution. If "ambient background" in the test areas differs from "natural background," correct the data accordingly and retest. Often, "ambient background" is less than "natural background." When this is the case, a better

estimate of "ambient background" is the median gross-total exposure rate value from the same uncorrected data set. The median is an unbiased estimator of "ambient background."

The most critical step in the analysis of gamma exposure rate measurements is assessing what true "ambient background" radiation is for a test area. "Ambient background" accounts for three effects which result in the production of an electronic pulse from the gamma instrument (a count), which under ideal measurement conditions would not occur:

1. "Natural background" radiation from outer space, primordial radionuclides, and global fallout
2. Secondary influence of gamma exposure rate due to nearby facilities which handle radioactive materials or radiation producing machines
3. Instrument noise.

These individual contributions to "ambient background" complicate data interpretation against acceptable limits because both the NRC and DOE criteria for acceptance for unrestricted use are given in μR/h above background. In natural-terrain areas, significant deviations in "natural background" radiation occur as a function of landscape geometry. For example, when the detector is placed near a large sandstone outcropping, the exposure rate may increase by almost 4 μR/h. This increase is due to naturally occurring radionuclides in the sandstone, and a change in source geometry, from a planar 2π-steradian surface to a rocky 3π-steradian surface. "Natural background" is also more variable when measurements are made over, at, or near large metal pieces, scrap components, and other objects. "Natural background" is also different indoors and varies with construction materials, particularly concrete, and typically is higher in concrete-lined rooms.

Once all the best corrections for "ambient background" have been made, resulting distributions are compared against the appropriate acceptance limit. The test statistic, $x_m + ks$, is calculated for each distribution. Statistical acceptance criteria presented in Section 3.2 then apply.

6.0 SURVEY RESULTS AND DISCUSSION

A radiological survey of Building T028 was performed using the survey plan described in Ref. 10, and outlined in Section 5. Three sample lots were established for analyzing and interpreting radiological data: (1) total and removable alpha/beta activity measurements for the whole facility, (2) ambient gamma exposure rate measurements for the above-grade section of T028, and (3) ambient gamma exposure rate measurements for the basement section of T028.

Analytical interpretation using Gaussian statistics of gamma exposure rate measurements and total and removable alpha/beta contamination measurements show slight contamination in some areas, but at levels far below acceptance limits. Further investigation is not required in any location.

6.1 INDICATION–ONLY SURVEYS

As part of the final release survey, indication–only surveys were first conducted using a μR survey meter and a pancake GM survey meter to search for contamination in those areas not specifically selected for grid measurements in the survey plan. No detectable activity (NDA) was observed in any of these areas.

6.2 GRID MEASUREMENTS

Total and removable alpha/beta activity was measured in 67 floor, wall, and ceiling locations in Rooms 102, 102A, and B101. Ambient gamma exposure rates were measured in 29 floor locations in the same areas. The results of all these measurements are listed in Appendixes B and C and summarized in Table 2. The table shows four parameters for each of the three data sets: average value, maximum value, standard deviation of the distribution, and the test statistic TS (TS = $x_m + ks$).

6.2.1 Total Alpha/Beta Activity

Total alpha/beta measurements were made in all of the 67 survey grid locations, including 14 locations in Room 102, 16 locations in Room 102A, and 37 locations in the basement room B101. These data are shown plotted vs the cumulative probability in Figures 9 and 10 (negative values occur when the observed count is less than the value adopted for background). As is evident in Figure 9, there is some deviation from a Gaussian distribution, with a few possible outliers. No outliers are evident in the total beta data set in Figure 10.

These same two data sets are shown on reduced scales in Figures 11 and 12 to more clearly display the survey results relative to the acceptance limits. Here, the appropriate acceptance limit of 5,000 dpm/100 cm^2 is shown as the top limit of each graph. As is clear, both data sets show TS values which are well below the acceptance limit, and

Table 2. Summary of Survey Results for Building T028

Data Set	Data Points	Average Value	Maximum Value	Standard Deviation	Test Statistic	Acceptance Limit
Total alpha (grids)[a]	67	12.3	72.8	15.9	36.2	5,000
Total beta (grids)[a]	67	523	1,303	413	1,148	5,000
Removable alpha (grids)[a]	67	5.1	109	18.7	33.3	1,000
Removable beta (grids)[a]	67	14.7	307	49.7	89.9	1,000
Ambient gamma (102, 102A)[b]	16	0.2	1.0	0.6	1.3	5
Ambient gamma (B101)[c]	13	–0.7	2.3	2.2	3.4	10
Removable alpha (structures)[d]	30	1.3	14.7	3.0	6.2	1,000
Removable beta (structures)[d]	30	11.8	50.4	12.2	31.8	1,000

[a] Total and removable alpha/beta measurements on 67 grid locations in Rooms 102, 102A, and B101.

[b] Background subtracted ambient gamma exposure rates in the above-grade Rooms 102 and 102A. The ambient background gamma rate subtraction was 11.2 μR/h, which is the median of the data set (see text).

[c] Background subtracted ambient gamma exposure rate in basement room B101. The ambient background gamma rate subtraction was 17.9 μR/h (see text).

[d] Removable alpha/beta measurements on various structures remaining at the facility.

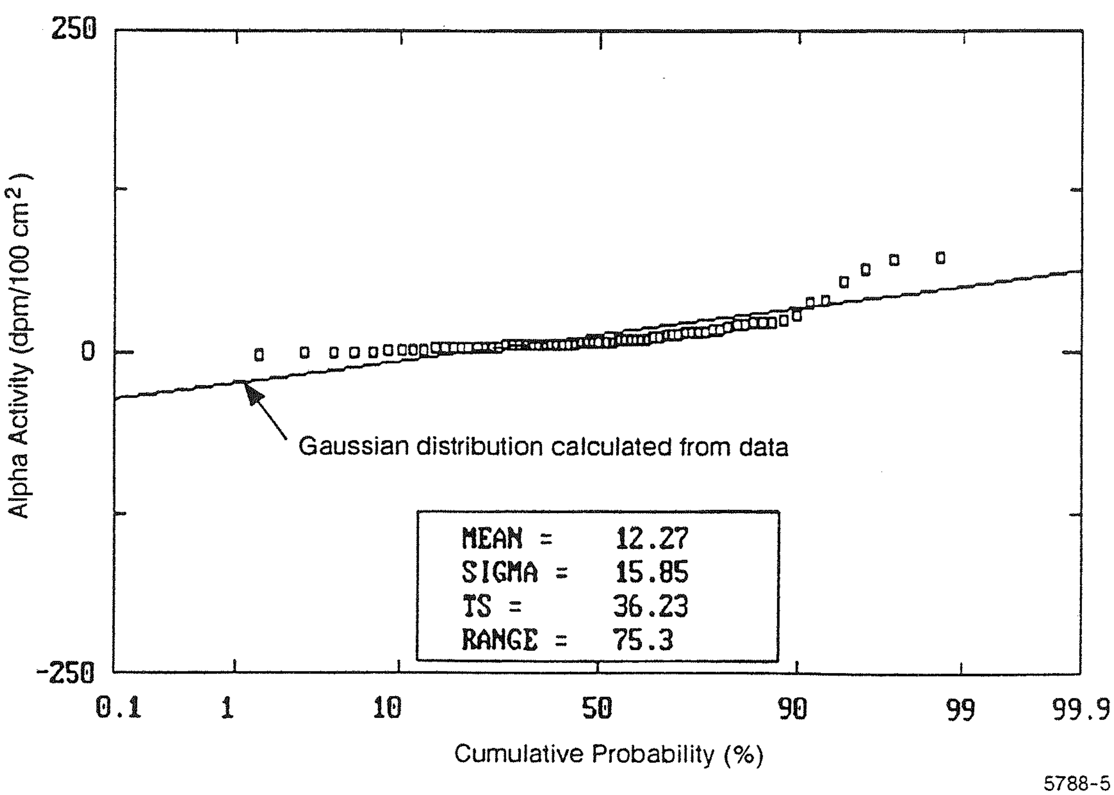


Figure 9. Total Alpha Activity in T028 Survey Grids

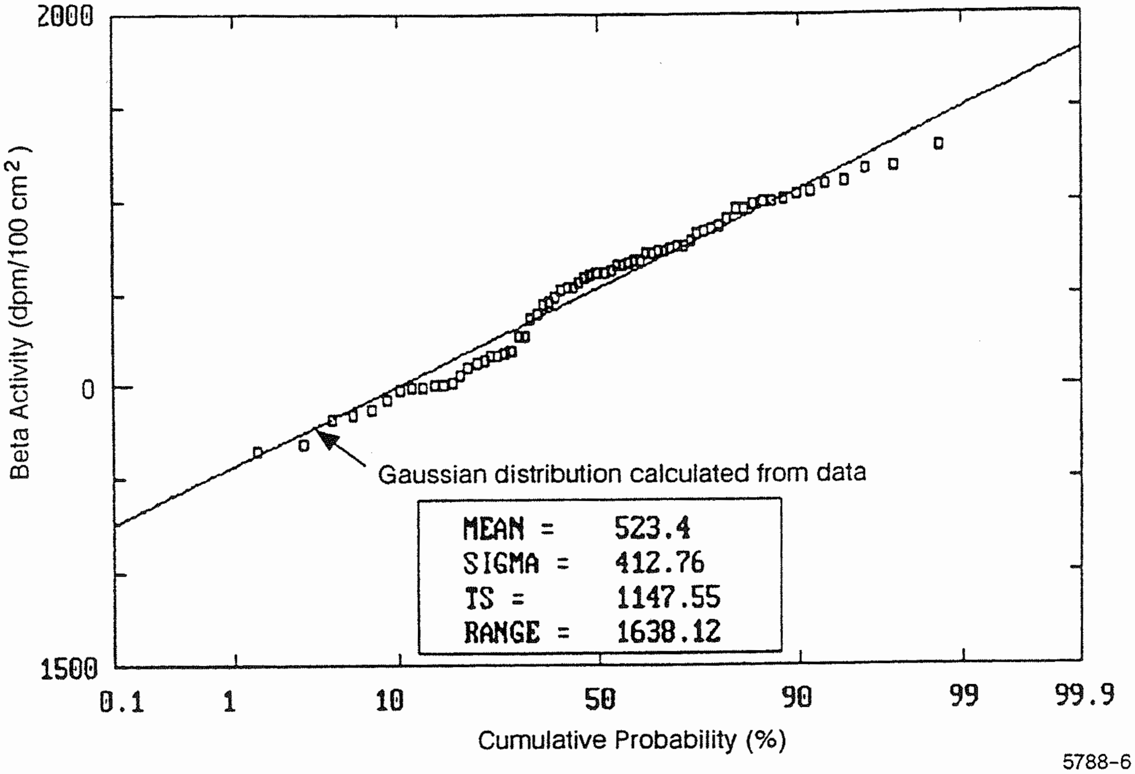


Figure 10. Total Beta Activity in T028 Survey Grids

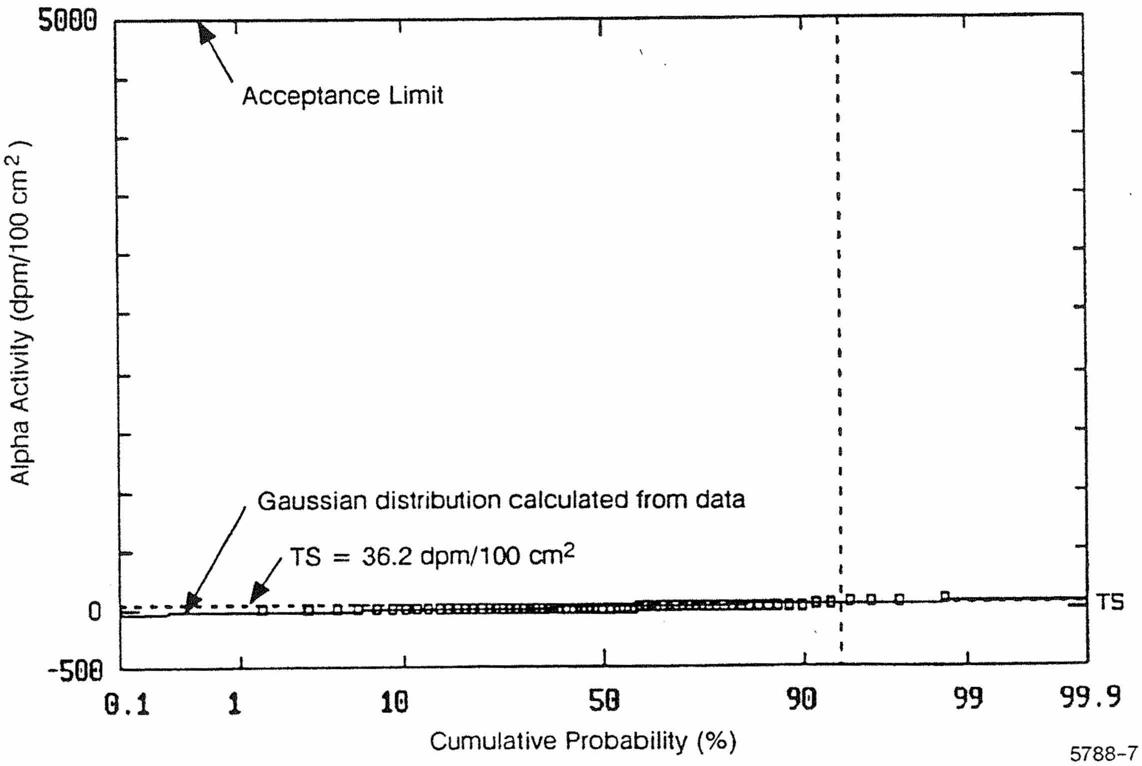


Figure 11. Total Alpha Activity in T028 Survey Grids–Reduced Scale

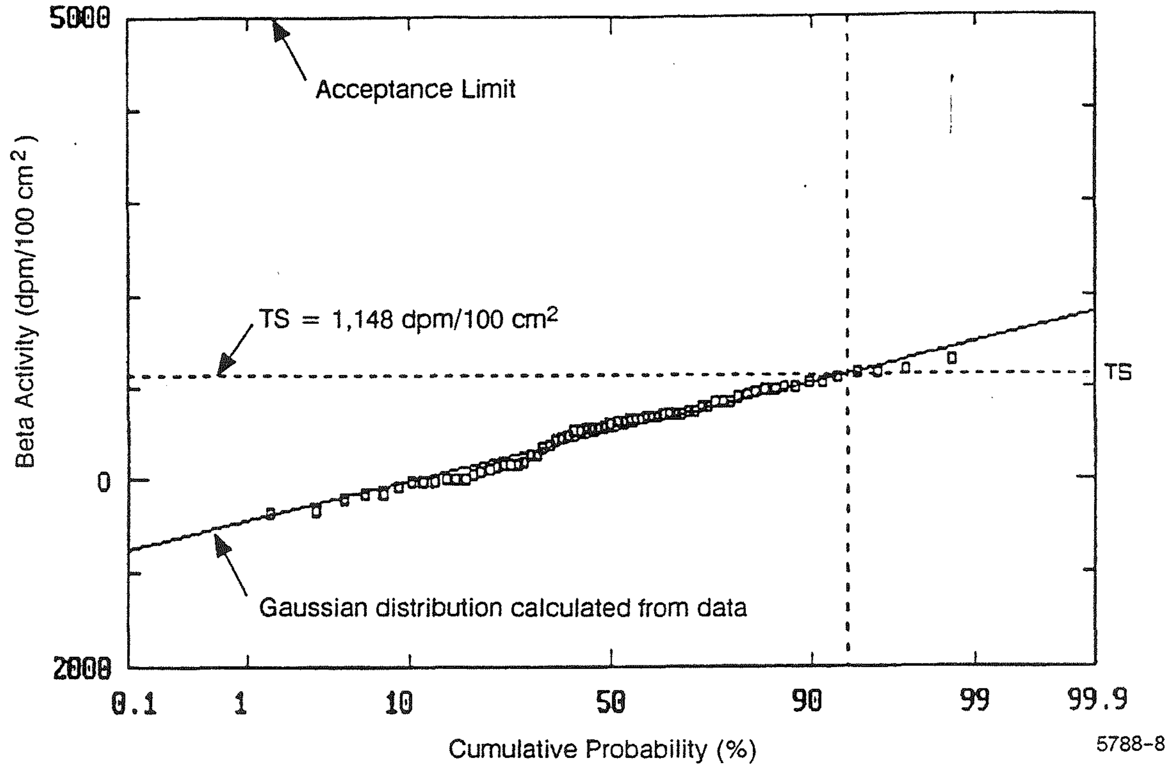


Figure 12. Total Beta Activity in T028 Survey Grids–Reduced Scale

indicate no need for further action. The reduced-scale plot of the data in Figure 11 clearly show that the few potential alpha outliers indicated in Figure 9 are of no regulatory concern.

6.2.2 Removable Alpha/Beta Activity

Removable (smear) alpha/beta measurements were also conducted in all 67 survey grid locations. These data are shown plotted vs the cumulative probability in Figures 13 and 14. As was the case in Figure 9, there is some deviation from a Gaussian distribution in both data sets, with several outliers.

The removable alpha and beta data sets are shown on reduced scales in Figures 15 and 16. Here, the appropriate acceptance limit of 1,000 dpm/100 cm^2 is shown as the top limit of each graph. Again, both data sets show TS values well below the acceptance limit, and indicate no need for further action. The few potential outliers, indicated in Figures 13 and 14, are far below levels that are of regulatory concern.

6.3 AMBIENT GAMMA EXPOSURE MEASUREMENTS

Ambient gamma exposure rate measurements were made in 29 grid locations, including 16 locations in Rooms 102 and 102A, and 13 locations in Room B101. Because of expected differences in natural background gamma exposure rates in the above-grade vs

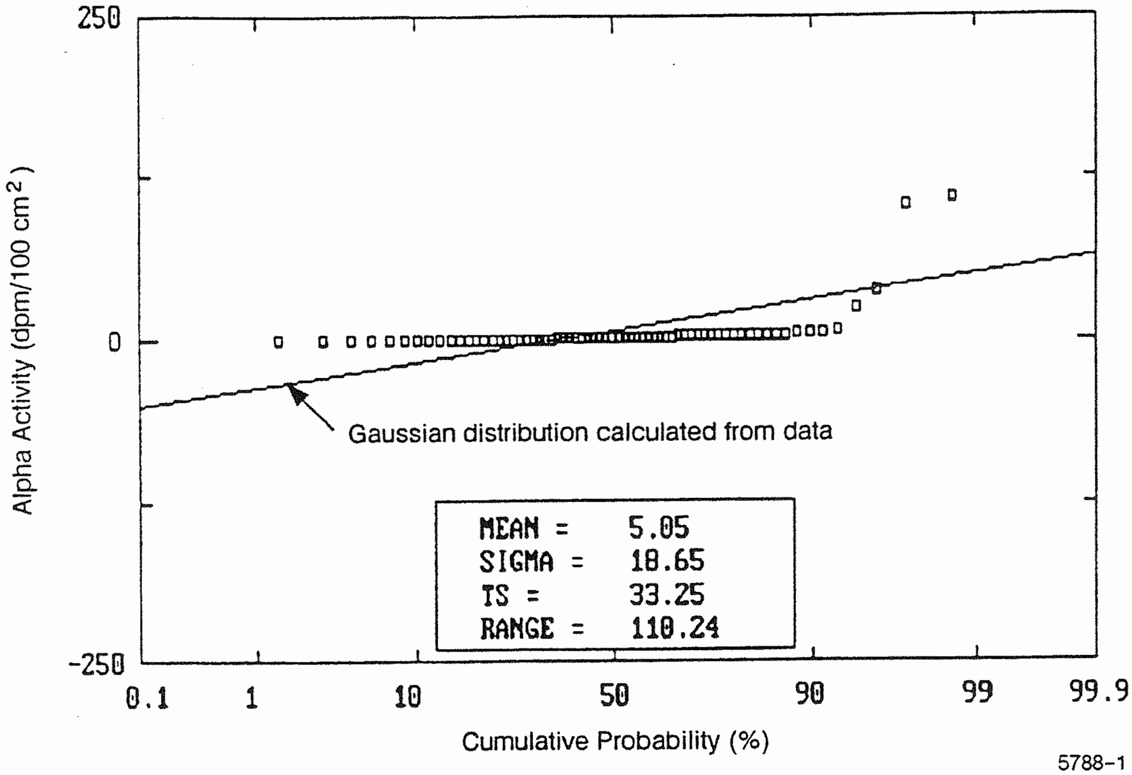


Figure 13. Removable Alpha Activity for T028 Survey Grids

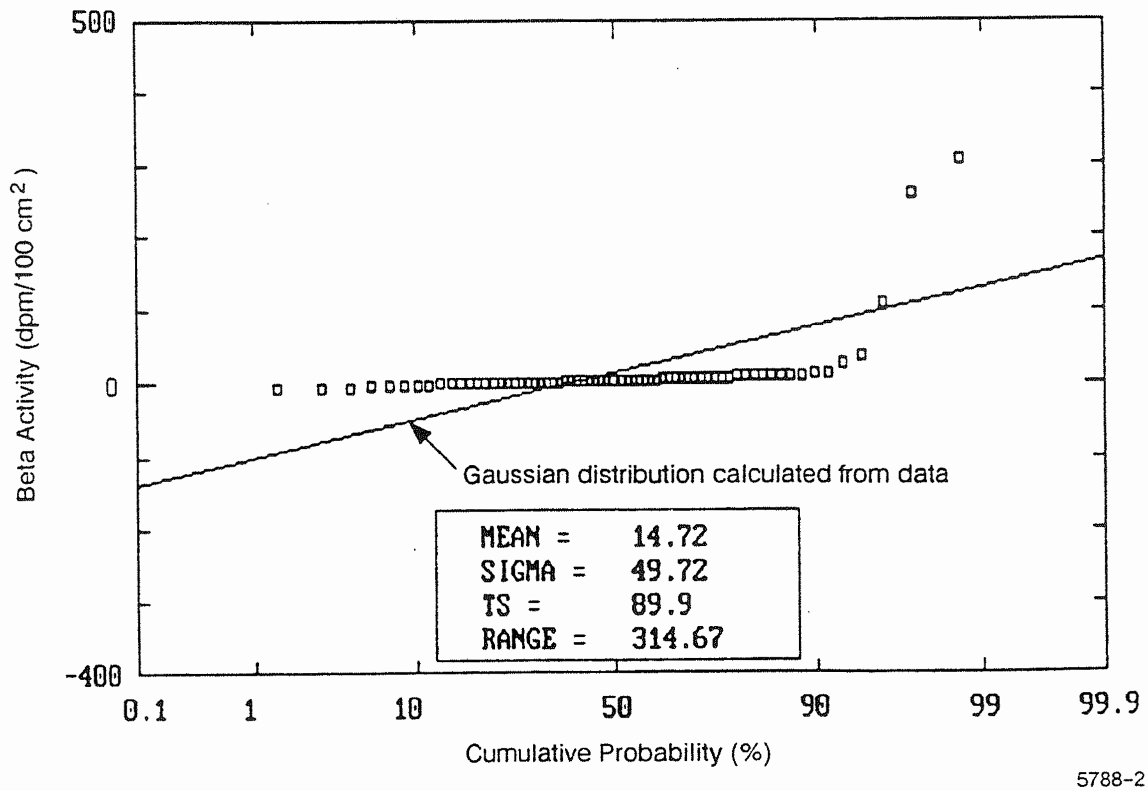


Figure 14. Removal Beta Activity for T028 Survey Grids

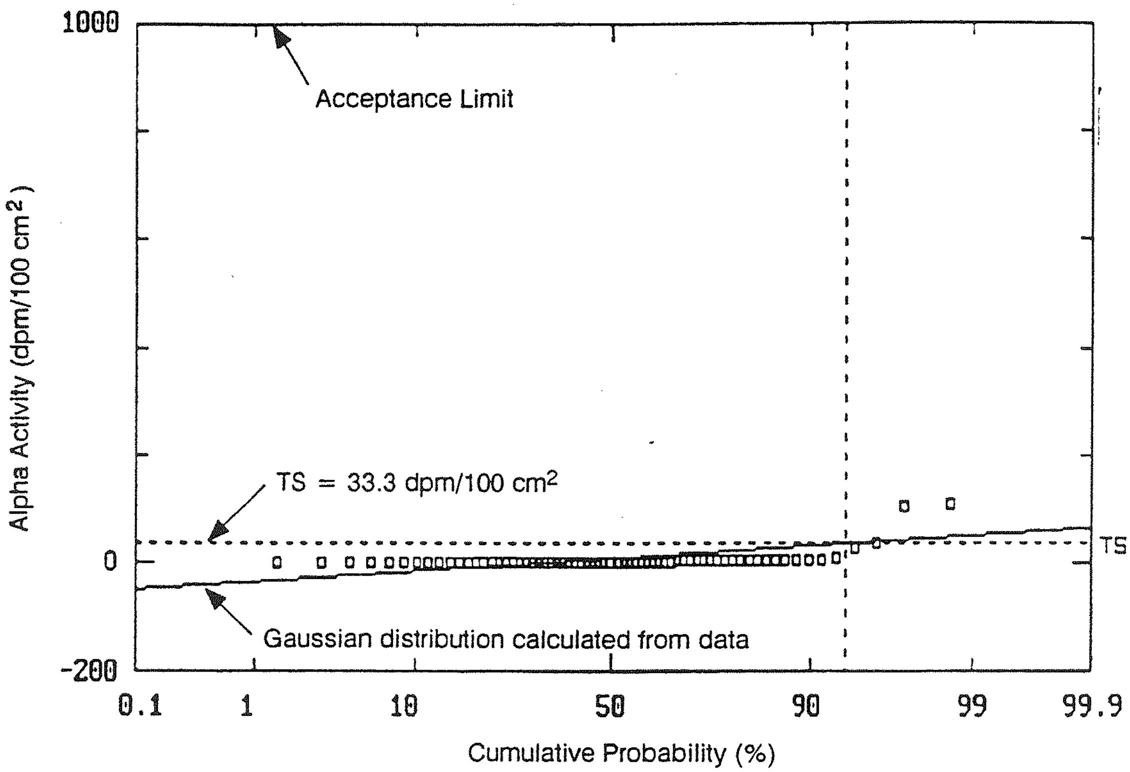


Figure 15. Removable Alpha Activity in T028 Survey Grids – Reduced Scale

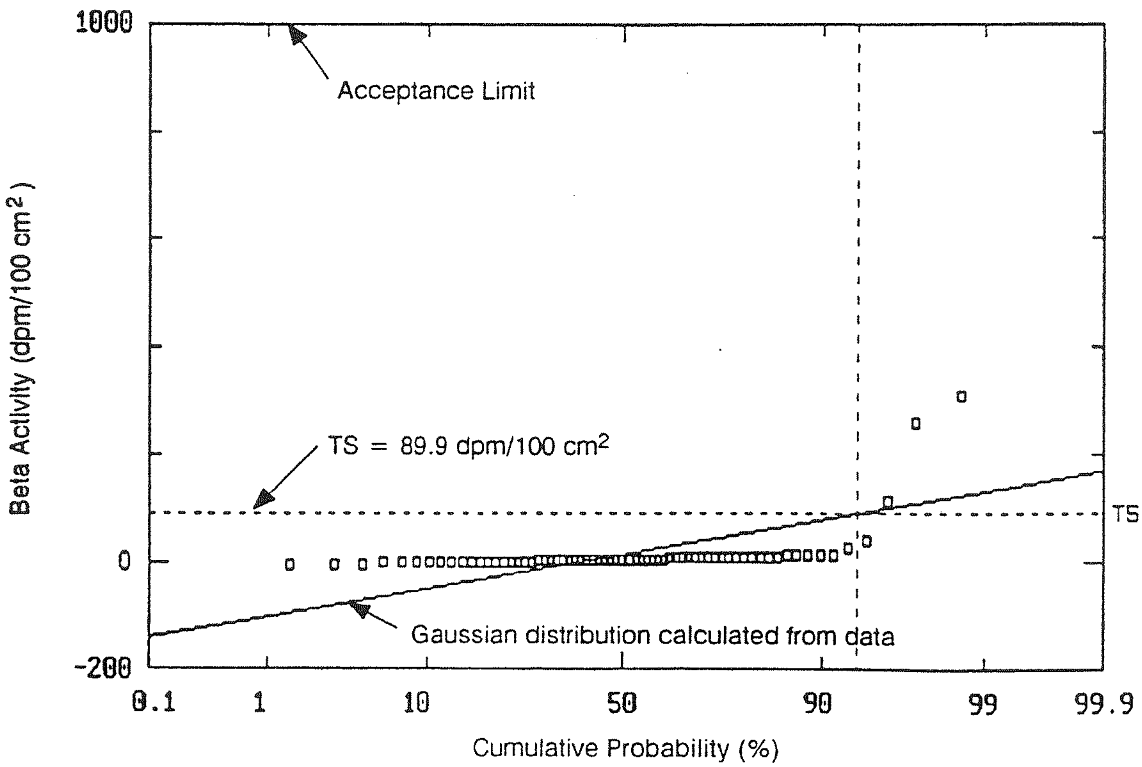


Figure 16. Removable Beta Activity in T028 Survey Grids – Reduced Scale

the basement area, the two areas were treated as separate data sets for statistical analysis. Treatment of the data and determination of gamma background levels appropriate to each data set are discussed below.

6.3.1 Background Gamma Exposure Rate at the SSFL

Because the variability in the background gamma exposure rate at the SSFL approaches 3 to 4 μR/h, the choice of a suitable value to use for the background exposure rate is critically important. Ideally, the best approach is to choose an area whose characteristics (geographic, location, etc.) are identical to the area under study. For the present survey, where the two areas of interest included a bare concrete slab floor and a concrete vault, no genuinely suitable "background" area was readily available. Therefore, the approach that was taken here was to average the available background gamma exposure rate data from a variety of areas at the SSFL. The five areas considered and summary data for each are listed in Table 3. All five area data sets have been used in previous radiological surveys at the SSFL and are outdoor areas where no radioactive materials have ever been used, stored, or disposed of.

The first three data sets were from areas specifically chosen based on their known history of use at the SSFL, which effectively precluded the possibility of there ever having been radioactive materials present at the sites. The latter two data sets, on the other hand, were established and used separately during the final radiological surveys of the Old Conservation Yard (a portion of the old Rocketdyne Barrel Storage Yard) (Ref. 11) and the Building T064 Sideyard (Ref. 12). Each of these latter data sets were subsets of gamma survey data taken in 1988 in the immediate vicinity of these two SSFL sites. The data points included in the subsets were taken from a single contiguous area within each

Table 3. Background Gamma Exposure Rates (μR/h) at the SSFL

Location	Data Points	Average Value	Range	Standard Deviation (1σ)
Bldg 309 Area	36	15.6	3.4	0.8
Well No. 13 Road	43	16.2	2.2	0.5
Incinerator Road	35	14.0	1.4	0.4
Old Conservation Yard	75	13.1	4.2	0.8
T064 Side Yard	24	15.5	3.0	0.8
	Average:	14.9	2.8	0.7
	$\pm 1\sigma$	1.3	1.1	0.2

D635-0172

of the larger data sets, where it could be reasonably ascertained that no previous use of radioactive materials had ever taken place.

The combined data in Table 3 give an average outdoor gamma exposure rate background at the SSFL of 14.9 ± 1.3 μR/h (1σ). The range (maximum minus minimum) of measured data from the five data sets varied from 1.4 for the Incinerator Road, to 4.2 for the area east and adjacent to Building T064.

6.3.2 Gamma Exposure Rates for Rooms 102 and 102A

Ambient gamma exposure rates measured in the 16 floor grid locations in Rooms 102 and 102A are presented in Figure 17. The data are plotted vs the cumulative Gaussian probability. The mean measured gamma exposure rate is 11.4 μR/h, with a range of 2.7 μR/h. Comparison with the data in Table 3 shows that this average value is significantly lower than generally observed at the SSFL, although individual data values in the 9 to 10 μR/h range have been observed.

Following the procedure guidelines given in Section 5.7.3, therefore, the median of the data set (11.2 μR/h) was used for a representative (unbiased) background estimate. The resulting background-subtracted gamma exposure rates above-grade in T028 are shown in Figure 18, compared against the appropriate acceptance limit of 5 μR/h shown

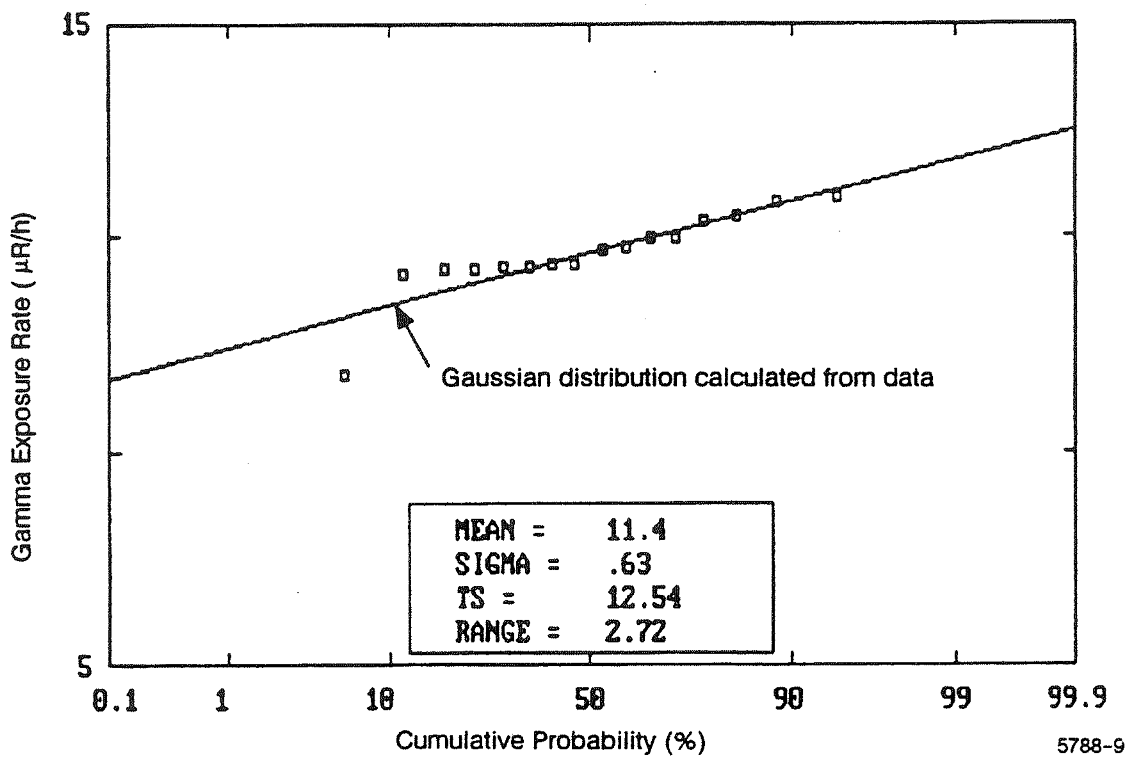


Figure 17. Gamma Exposure Rates for T028 Survey Grids (Rooms 102 and 102A)

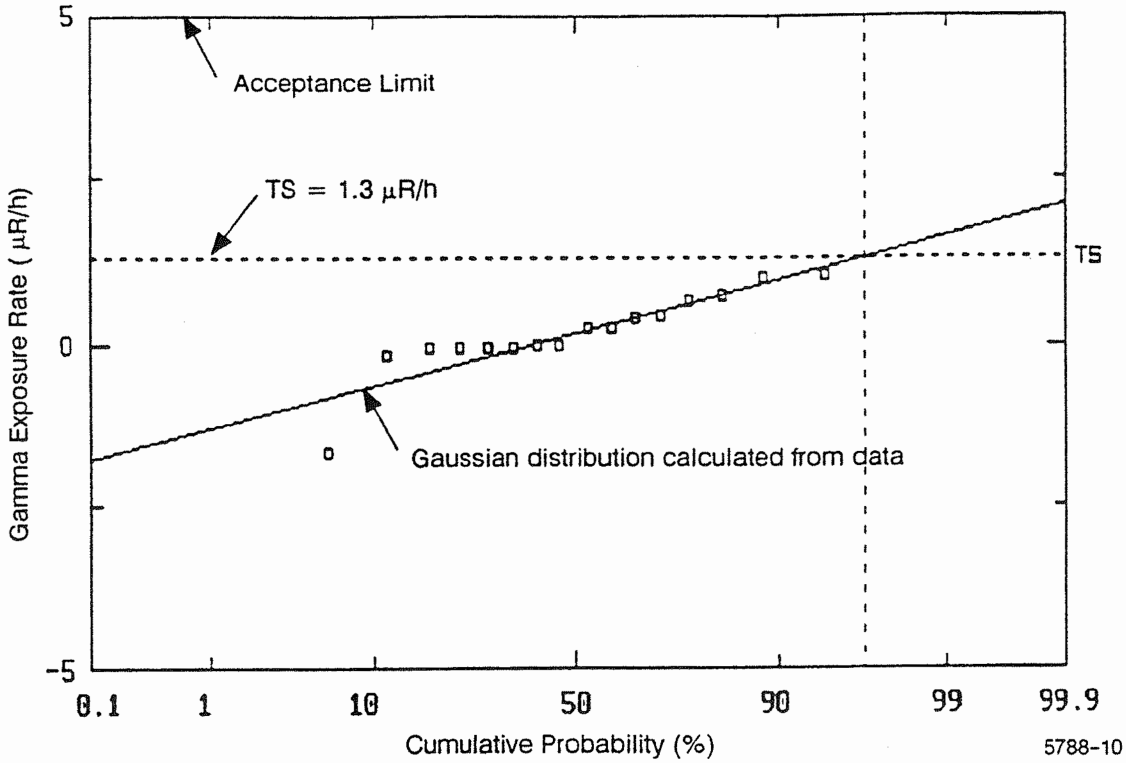


Figure 18. Gamma Exposure Rates for T028 Survey Grids (Rooms 102 and 102A)–Reduced Scale

at the top of the graph. The test statistic for the distribution is 1.3 μR/h, well below the acceptance limit.

6.3.3 Gamma Exposure Rates for Room B101

The second set of gamma exposure rates measured, that for the basement of Building T028, is shown in Figure 19. As expected for a concrete vault, the data set shows values higher than observed above grade. For this distribution, it was appropriate to use an adjusted background level following the method described earlier in Section 3.1.1. This method specifies adding 3 μR/h to the natural ambient gamma exposure rate for the SSFL of 14.9 μR/h, resulting in a background gamma exposure rate of 17.9 μR/h. The method also specifies using the higher acceptance limit of 10 μR/h given in Table 1. The resulting background–subtracted Room B101 data is shown in Figure 20. The test statistic for the distribution is 3.4 μR/h, which is well below the applicable 10 μR/h acceptance limit. Correlation of the measured gamma data with corresponding survey location, however, does indicate a gradient across the room of ~4.7 μR/h. This result is also below the 10 μR/h limit, but, as expected, does indicate some remaining low–level residual activation in the concrete wall section that was adjacent to the previously decommissioned STIR reactor.

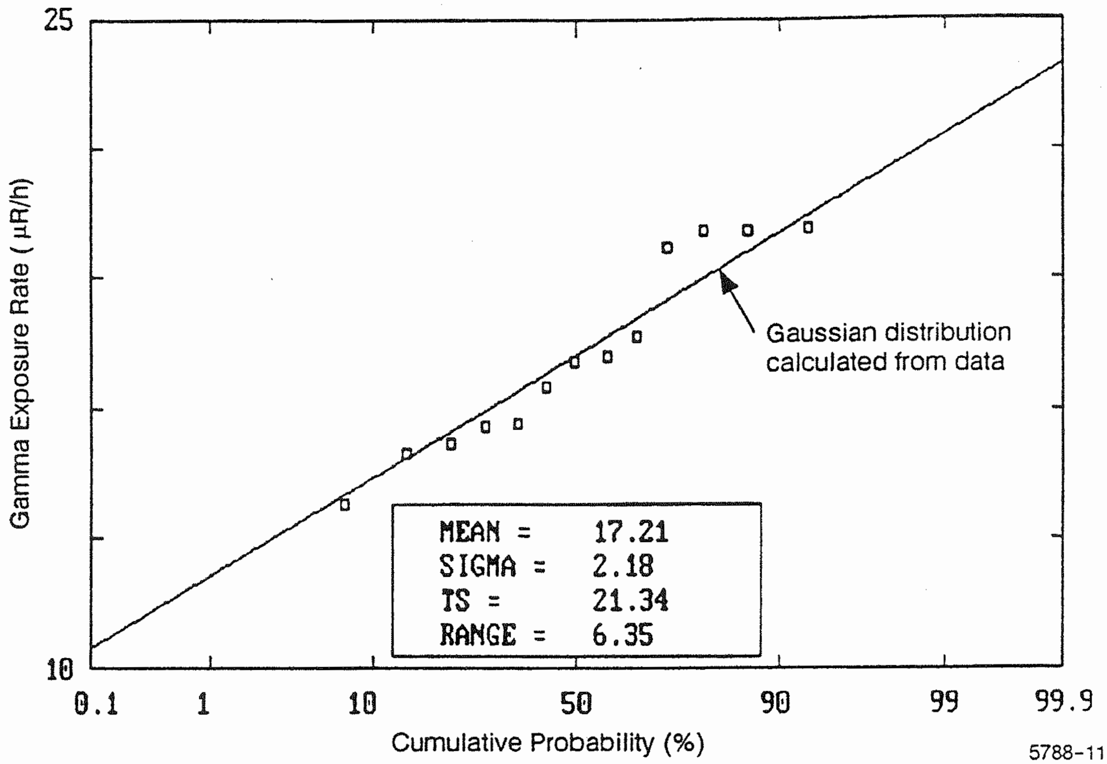


Figure 19. Gamma Exposure Rates for T028 Survey Grids (Room B101)

6.4 ADDITIONAL SURVEYS

Several additional nongridded areas of T028 were surveyed as part of the final radiological release survey. These data are discussed below.

6.4.1 Room B101

Removable alpha/beta surveys were conducted on several special structural surfaces in Room B101. These included the 7–1/2 ton bridge crane and rails (20 smears, 90% survey), and the ceiling light fixtures (10 locations). Results of the smear surveys are shown in Figures 21 and 22 plotted vs the cumulative Gaussian probability and are listed in Appendix C. Figures 23 and 24 show the same data plotted on a reduced scale for comparison with the acceptance limit of 1,000 dpm/100 cm^2. Test statistic values for both data sets are well below the acceptance limit.

6.4.2 HEPA Filter Plenum Foundation

Ground–level surveys were conducted on the concrete foundation beneath the R/A exhaust HEPA filter plenum after removal of the plenum. The survey encompassed 17 grid locations. Survey results were reported as NDA for total alpha/beta, and as <20 and <50 dpm/100 cm^2 for removable alpha/beta, indicating no observable residual radionuclide activity. These results are well below the acceptance limits of 5,000 and 1,000

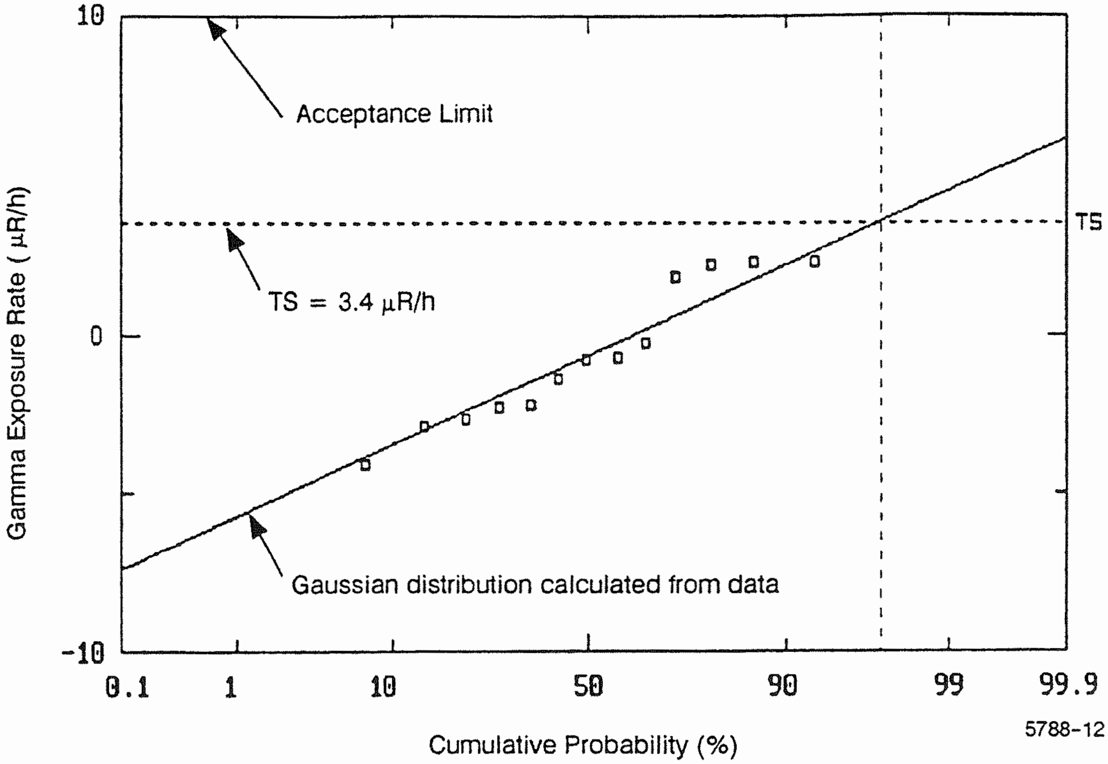


Figure 20. Gamma Exposure Rates for T028 Survey Grids (Room B101)–Reduced Scale

dpm/100 cm^2, respectively, and therefore, no further statistical analysis of these data was performed.

6.5 STATUS OF BUILDING

The above–grade concrete slab floor and the test vault (Room B101) are the only remaining features of T028 still intact. The basement room is currently inactive.

A decommissioning file for Building T028 has been established and is currently archived at Rockwell's SSFL Building T100. Appendix D contains a list of items archived in this file.

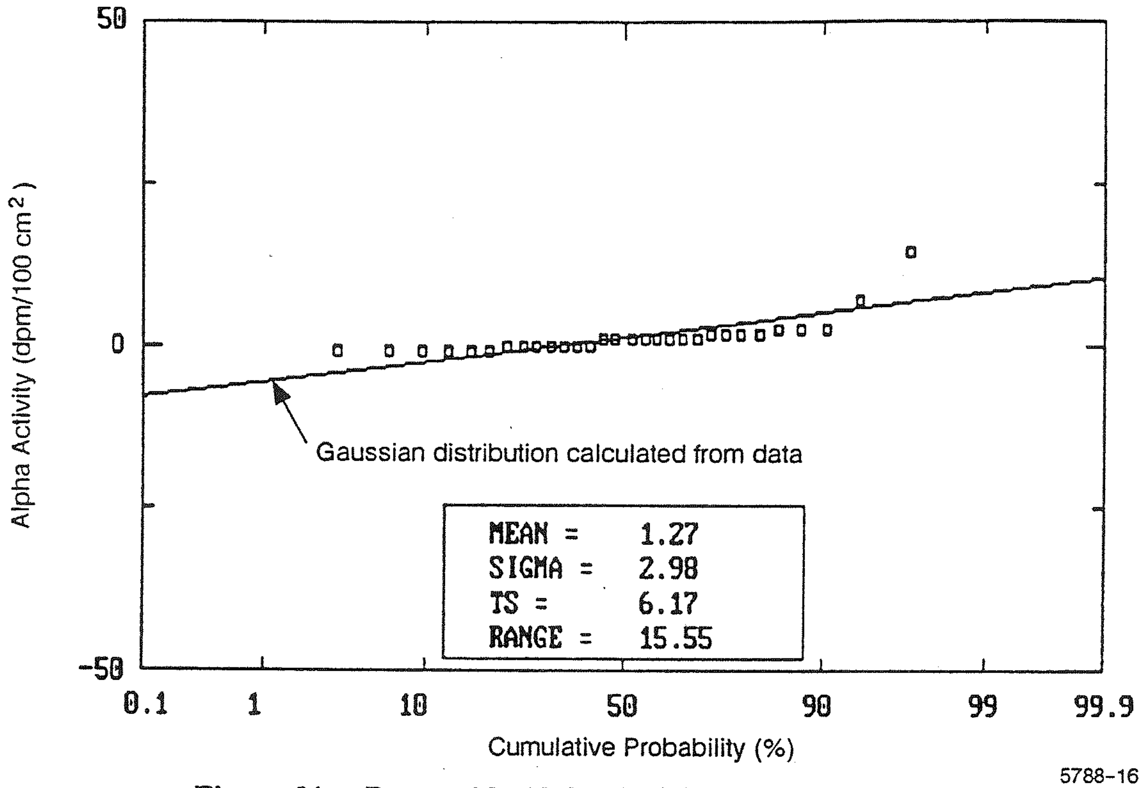


Figure 21. **Removable Alpha Activity for B101 Structures**

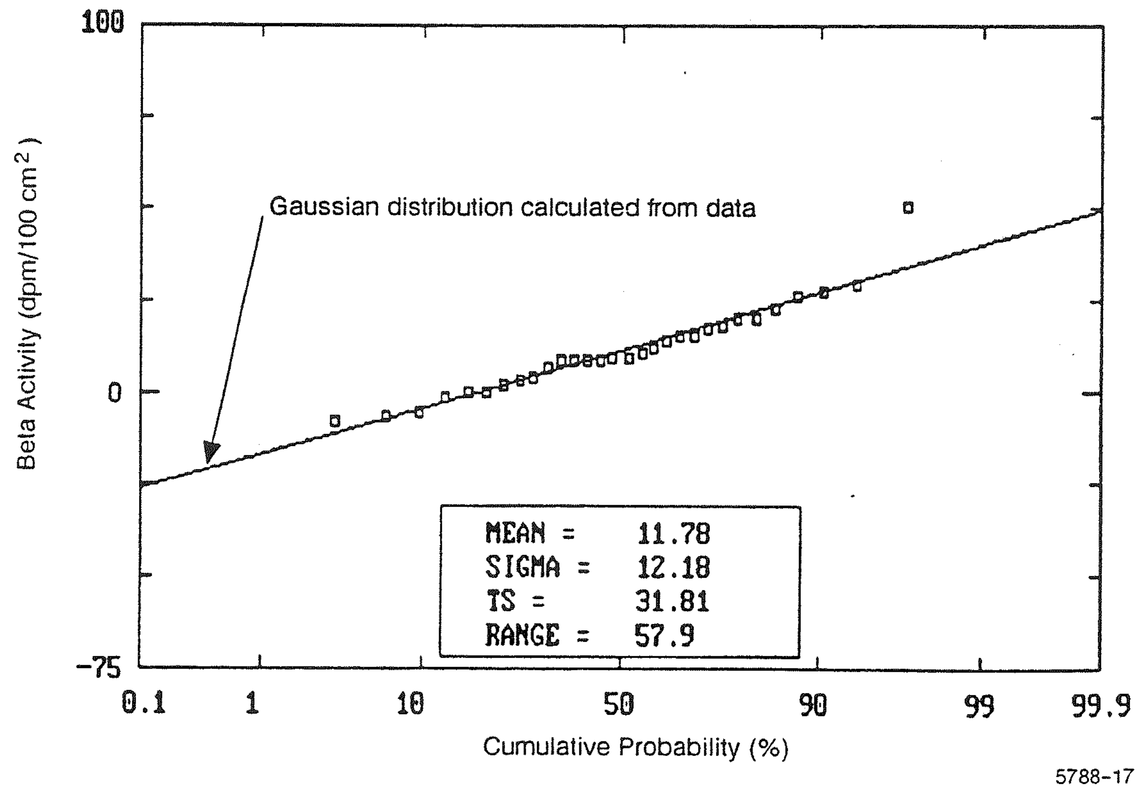


Figure 22. **Removable Beta Activity for B101 Structures**

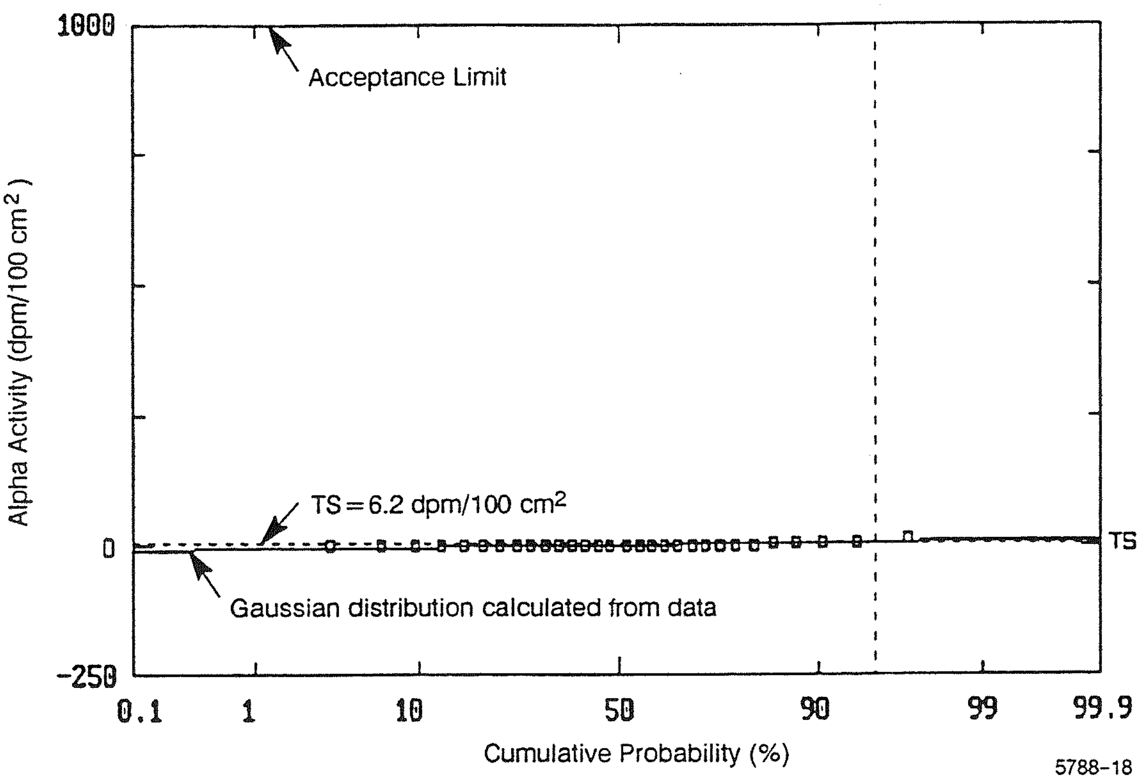


Figure 23. **Removable Alpha Activity for B101 Structures–Reduced Scale**

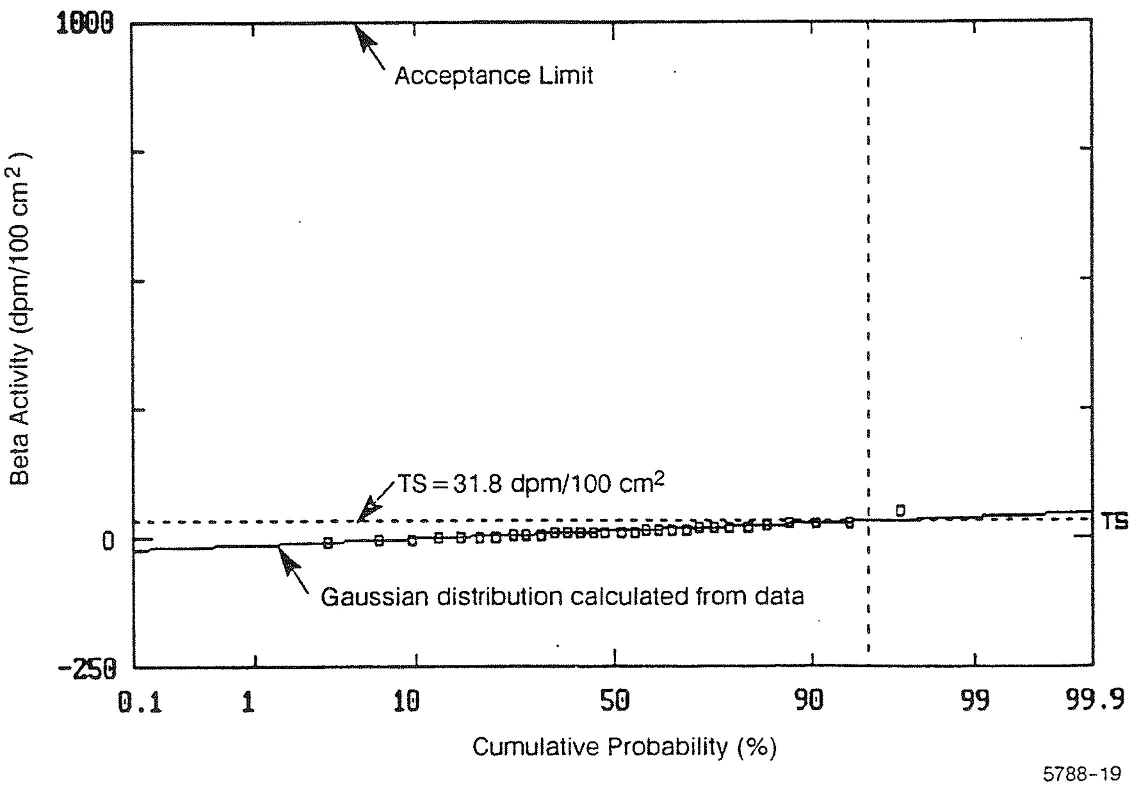


Figure 24. **Removable Beta Activity for B101 Structures–Reduced Scale**

7.0 CONCLUSIONS

Specific and overall conclusions relating to the current radiological status of T028 are given below.

7.1 SPECIFIC CONCLUSIONS

1. Indication–only radiological survey data on nongridded areas of T028 indicated no detectable residual radioactivity.

2. Total alpha/beta measurements made in 67 grid locations in T028 Rooms 102, 102A, and B101 showed test statistic (TS) values of 36.2 and 1,148 dpm/100 cm^2, respectively. Both values are well below the acceptance limit for surface contamination of 5,000 dpm/100 cm^2.

3. Removable alpha/beta measurements made in the same 67 grid locations showed TS values of 33.3 and 89.9 dpm/100 cm^2, respectively. Removable alpha/beta measurements on various remaining structures located at the facility showed TS values of 6.2 and 31.8 dpm/100 cm^2. All values are well below the acceptance limit for removable contamination of 1,000 dpm/100 cm^2.

4. Background–subtracted ambient gamma exposure rate measurements made in 16 grid locations in Rooms 102 and 102A, and 13 grid locations in B101, showed TS values of 1.3 and 3.4 μR/h, both below the applicable acceptance limits of 5 and 10 μR/h, respectively. Some slight residual gamma activity, less than the 10 μR/h limit, was observed near the north wall of B101, presumably from residual activation from the previously removed STIR reactor.

7.2 OVERALL CONCLUSIONS

1. Based on the results of the final radiological survey reported here, the remaining structures at SSFL Building T028 may be released for use without radiological restrictions.

8.0 REFERENCES

1. B. F. Ureda, "STIR Facility, Decontamination and Disposition, Final Report," Rockwell International Report AI–ERDA 13168, August 26, 1976.

2. A. Klein, "Building T028 Decontamination and Demolition Final Report," Rockwell International Supporting Document N001TI000322, June 6, 1990.

3. R. J. Tuttle, "Radiological Status of T028," Rockwell International Internal Letter 495 Il.rjt, April 17, 1989.

4. "Radiation Protection for Occupational Workers," DOE Order 5480.11, Attachment 2, July 20, 1989.

5. "Guidelines for Residual Radioactivity at FUSRAP and Remote SFMP Sites," U.S. Department of Energy Report, February 1985.

6. "Order Authorizing Dismantling of Facility and Disposition of Component Parts," U.S. Nuclear Regulatory Commission, Docket 50–375, February 22, 1983.

7. "Sampling Procedures and Tables for Inspection by Variables for Percent Defect," MIL–STD–414, June 11, 1957.

8. V. A. Swanson, "Building T020 Radiological Survey Procedure," Rockwell International Supporting Document 173DWP000021, May 24, 1988.

9. "State of California Guidelines for Decontaminating Facilities and Equipment Prior to Release for Unrestricted Use," DECON–1, June 1977.

10. V. A. Swanson, "Building T028 Radiological Survey Plan," Rockwell International Supporting Document N704DWP990095, February 1991.

11. G. Subbaraman and B. M. Oliver, "Final Decontamination and Radiological Survey of the Old Conservation Yard," Rockwell International Supporting Document N704SRR990030, August 16, 1990.

12. G. Subbaraman and B. M. Oliver, "Final Decontamination and Radiological Survey of the Building T064 Side Yard," Rockwell International Supporting Document N704SRR990031, October 30, 1990.

APPENDIX A

STATE OF CALIFORNIA—HEALTH AND WELFARE AGENCY GEORGE DEUKMEJIAN, Governor

DEPARTMENT OF HEALTH SERVICES
714 P Street
Sacramento, CA 95814
(916) 445-0931

February 7, 1986

RE: 1. Radioactive Materials License No.

2. "Initial Radiological Survey and Analysis - Facility", dated September 12, 1985.

3. "Final Radiological Survey, ", dated December 6, 1985.

Dear :

Upon review of the above referenced information, the Department is unable to concur with the conclusions made in the "Final Radiological Survey, ".

Your Plan describes your: (a) removal of interior activated concrete to a level of approximately 133 pCi/gm and your intent to (b) rubblize this 133 pCi/gm material with other concrete to reduce the concrete average concentration to 13.3 pCi/gm, and then (c) bury this aggregate mixture at 1 to 1.5 meters below the surface at a sanitary landfill.

The staff believes an appropriate method to evaluate the gamma emitting activation products in concrete and rebar for unrestricted release is specified in NUREG-2082 and NUREG-0613c (copy enclosed). Your facility presently exceeds the recommended criteria of 5 uR/hr at one meter from any surface. Because of your facility's interior configuration we estimate that 10 uR/hr at one meter above background inside the structure is a reasonable equivalent to the slab criteria.

Our information shows that interior background levels within concrete normally exceed comparative unconfined background levels by approximately 1-4 uR/hr. The problem of determining background levels may be accomplished by monitoring a similar confinement made of similar material with similar mass.

If no similar structure and material mass can be found, then nearby outside environmentally representative measurement should be taken and modified to reflect the interior concrete mass contribution such that: Measured exterior background + corrective number (~3 uR/hr) = estimated interior background.

APPENDIX B
TOTAL AND REMOVABLE ALPHA/BETA MEASUREMENTS IN T028 GRIDS

RADIOLOGICAL SURVEY DATA

T028 total and removable alpha/beta measurements

SAMPLE NAME	GRID NAME	ALPHA (DPM/100CM2) TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV	BETA (DPM/100CM2) TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV	GAMMA (uR/h) TOTAL	STD DEV
102	C-1	15.42	5.78			0.38	0.93	-131	93			9.38	5.61		
102	C-2	4.20	4.20			-0.38	0.53	0	95			0.00	4.66		
102	E-1	5.61	4.43			1.89	1.41	-317	89			-1.04	4.54		
102	E-2	4.20	4.20			-0.38	0.53	-154	92			-2.08	4.42		
102	F-1	8.41	4.86			0.38	0.93	732	107			-2.08	4.42		
102	F-2	8.41	4.86			1.13	1.20	540	104			-2.08	4.42		
102	F-3	8.41	4.86			0.38	0.93	356	101			1.04	4.78		
102	F-4	5.61	4.43			1.89	1.41	59	96			8.34	5.51		
102	N-1	14.02	5.61			0.38	0.93	-347	89			7.29	5.41		
102	N-2	2.80	3.96			-0.38	0.53	-7	95			4.17	5.10		
102	S-1	1.40	3.71			-0.38	0.53	20	95			7.29	5.41		
102	S-2	-2.80	2.80			2.65	1.60	-183	92			6.25	5.31		
102	W-1	0.00	3.43			-0.38	0.53	-72	94			0.00	4.66		
102	W-2	4.20	4.20			0.38	0.93	-26	95			5.21	5.21		
102A	C-1	8.16	3.84			1.13	1.20	123	99			1.04	4.78		
102A	E-1	1.36	2.35			0.38	0.93	133	99			3.13	5.00		
102A	E-2	2.72	2.72			-0.38	0.53	7	97			-7.29	3.76		
102A	F-1	6.80	3.60			0.38	0.93	564	106			7.29	5.41		
102A	F-2	10.87	4.30			-0.38	0.53	728	109			7.29	5.41		
102A	F-3	4.08	3.04			-0.38	0.53	1183	116			4.17	5.10		
102A	F-4	6.80	3.60			-0.38	0.53	980	113			3.13	5.00		
102A	F-5	9.51	4.08			-0.38	0.53	1169	116			0.00	4.66		
102A	F-6	12.23	4.51			0.38	0.93	1012	113			0.00	4.66		
102A	F-7	13.59	4.71			-0.38	0.53	721	109			9.38	5.61		
102A	N-1	-1.36	1.36			-0.38	0.53	161	99			3.13	5.00		
102A	N-2	5.44	3.33			0.38	0.93	179	100			-6.25	3.90		

102A	S-1	-1.36	1.36			-0.38	0.53	-14	96			1.04	4.78		
102A	S-2	2.72	2.72			1.13	1.20	98	98			2.08	4.89		
102A	W-1	2.72	2.72			1.13	1.20	189	100			3.13	5.00		
102A	W-2	2.72	2.72			3.40	1.77	161	99			3.13	5.00		
B-101	C-1	2.63	3.22			1.06	1.84	720	119			0.94	4.31		
B-101	C-2	11.83	4.74			2.12	2.12	1002	123			9.40	5.15		
B-101	C-3	2.63	3.22			1.06	1.84	748	119			-0.94	4.10		
B-101	C-4	6.57	3.94			2.12	2.12	952	122			0.00	4.20		
B-101	C-5	6.57	3.94			0.00	1.50	870	121			2.82	4.51		
B-101	C-6	3.94	3.48			3.18	2.37	820	121			10.34	5.23		
B-101	E-1	21.03	5.88			37.10	6.45	601	117			110.92	11.04		
B-101	E-2	3.94	3.48			3.18	2.37	584	117			6.58	4.88		
B-101	E-3	62.31	9.69			109.18	10.86	653	114			257.56	16.12		
B-101	E-4	38.77	7.83			0.00	1.50	755	115			27.26	6.58		
B-101	E-5	72.50	9.90			102.82	10.55	784	117			307.38	17.51		
B-101	F-1	54.53	8.65			2.12	2.12	998	121			1.88	4.41		
B-101	F-10	27.59	6.35			2.12	2.12	525	114			7.52	4.97		
B-101	F-2	69.93	9.73			4.24	2.60	1292	125			11.28	5.32		
B-101	F-3	23.74	5.95			4.24	2.60	1103	122			8.46	5.06		
B-101	F-4	36.57	7.20			4.24	2.60	676	116			5.64	4.79		
B-101	F-5	19.89	5.52			0.00	1.50	900	119			7.52	4.97		
B-101	F-6	21.17	5.67			1.06	1.84	952	120			9.40	5.15		
B-101	F-7	17.32	5.21			0.00	1.50	1043	121			4.70	4.70		
B-101	F-8	14.76	4.89			0.00	1.50	676	116			7.52	4.97		
B-101	F-9	22.45	5.81			0.00	1.50	837	118			5.64	4.79		
B-101	N-1	13.85	5.18			1.06	1.84	265	107			13.16	5.48		
B-101	N-2	4.15	3.66			2.12	2.12	671	114			15.04	5.64		
B-101	N-3	16.62	5.54			24.38	5.30	854	117			38.54	7.34		
B-101	S-1	5.54	3.92			6.36	3.00	755	115			12.22	5.40		
B-101	S-2	2.77	3.39			1.06	1.84	530	112			1.88	4.41		
B-101	S-3	4.15	3.66			3.18	2.37	438	110			4.70	4.70		
B-101	W-1	13.20	4.94			0.00	1.50	388	114			1.88	4.41		
B-101	W-2	7.92	4.17			0.00	1.50	459	115			11.28	5.32		
B-101	W-3	2.64	3.23			1.06	1.84	477	115			-4.70	3.64		
B-101	W-4	7.92	4.17			1.06	1.84	607	117			4.70	4.70		

B-101PIT	E-2	7.13	4.28			0.00	1.50	265	113			-2.82	3.88		
B-101PIT	F-5	19.98	6.05			2.12	2.12	1085	125			1.88	4.41		
B-101PIT	F-6	1.43	3.19			2.12	2.12	649	119			1.88	4.41		
B-101PIT	N-1	1.43	3.19			0.00	1.50	609	118			4.70	4.70		
B-101PIT	S-3	7.13	4.28			-1.06	1.06	629	119			-0.94	4.10		
B-101PIT	W-4	-1.43	2.47			2.12	2.12	1037	124			4.70	4.70		
Maximum:		72.50				109.18		1291.50				307.38			
Minimum:		-2.80				-1.06		-346.62				-7.29			
Average:		12.27				5.05		523.40				14.72			
SD:		15.85				18.65		412.76				49.72			

APPENDIX C
AMBIENT GAMMA EXPOSURE RATE MEASUREMENTS IN T028 GRIDS

RADIOLOGICAL SURVEY DATA

T028 gamma survey data

SAMPLE NAME	GRID NAME	TOTAL	STD DEV	ALPHA (DPM/100CM2) MAX	STD DEV	REM	STD DEV	TOTAL	STD DEV	BETA (DPM/100CM2) MAX	STD DEV	REM	STD DEV	GAMMA (uR/h) TOTAL	STD DEV
102	F-1													9.55	0.09
102	F-2													11.23	0.10
102	F-3													11.17	0.10
102	F-4													11.08	0.10
102	F-5													11.66	0.10
102	F-6													11.97	0.11
102	F-7													11.16	0.10
102	F-8													11.63	0.10
102	F-9													11.48	0.10
102A	F-1													11.18	0.10
102A	F-2													11.90	0.11
102A	F-3													12.22	0.11
102A	F-4													11.17	0.10
102A	F-5													11.46	0.10
102A	F-6													11.23	0.10
102A	F-7													12.27	0.11
B101	F-1													15.63	0.12
B101	F-10													20.05	0.14
B101	F-11													20.09	0.14
B101	F-12													19.67	0.14
B101	F-13													20.15	0.14
B101	F-2													14.97	0.12
B101	F-3													15.23	0.12
B101	F-4													15.68	0.12
B101	F-5													13.79	0.11
B101	F-6													16.52	0.12

B101	F-7													17.67	0.13
B101	F-8													17.10	0.13
B101	F-9													17.18	0.13
Maximum:														20.15	
Minimum:														9.55	
Average:														14.00	
SD:														3.30	

APPENDIX D
REMOVABLE ALPHA/BETA MEASUREMENTS ON T028 STRUCTURAL COMPONENTS

RADIOLOGICAL SURVEY DATA

T028 structure surveys (removable alpha/beta)

SAMPLE NAME	GRID NAME	ALPHA (DPM/100CM2) TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV	BETA (DPM/100CM2) TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV	GAMMA (uR/h) - TOTAL	STD DEV
B101CRANE	G-1					1.73	1.73					-7.52	11.40		
B101CRANE	G-2					0.00	1.22					-6.02	11.45		
B101CRANE	G-3					0.00	1.22					6.77	11.87		
B101CRANE	G-4					0.00	1.22					-5.26	11.48		
B101CRANE	G-5					6.91	2.73					9.02	11.94		
B101CRANE	G-6					-0.86	0.86					26.32	12.47		
B101CRANE	G-7					0.86	1.50					2.26	11.72		
B101CRANE	G-8					-0.86	0.86					10.53	11.98		
B101CRANE	G-9					-0.86	0.86					15.79	12.15		
B101LITFX	LF-1					1.73	1.73					50.38	13.18		
B101LITFX	LF-10					2.59	1.93					22.56	12.36		
B101LITFX	LF-2					1.73	1.73					3.76	11.77		
B101LITFX	LF-3					-0.86	0.86					14.29	12.10		
B101LITFX	LF-4					0.00	1.22					9.78	11.96		
B101LITFX	LF-5					0.86	1.50					0.00	11.65		
B101LITFX	LF-6					-0.86	0.86					15.79	12.15		
B101LITFX	LF-7					0.00	1.22					18.05	12.22		
B101LITFX	LF-8					-0.86	0.86					9.02	11.94		
B101LITFX	LF-9					14.69	3.77					0.00	11.65		
B101RAILS	R-1					2.59	1.93					20.30	12.29		
B101RAILS	R-10					0.00	1.22					-0.75	11.63		
B101RAILS	R-11					0.86	1.50					20.30	12.29		
B101RAILS	R-2					0.86	1.50					29.33	12.56		
B101RAILS	R-3					0.86	1.50					4.51	11.79		
B101RAILS	R-4					2.59	1.93					9.02	11.94		
B101RAILS	R-5					1.73	1.73					9.02	11.94		

B101RAILS	R-6					0.00	1.22					17.30	12.20		
B101RAILS	R-7					0.86	1.50					27.07	12.49		
B101RAILS	R-8					0.86	1.50					12.03	12.03		
B101RAILS	R-9					0.86	1.50					9.78	11.96		
.															
Maximum:						14.69						50.38			
Minimum:						-0.86						-7.52			
Average:						1.27						11.78			
SD:						2.98						12.18			

APPENDIX E

LIST OF ITEMS IN THE BUILDING T028 DECOMMISSIONING FILE

The following is a list of the documents on the decommissioning of SSFL Building T028. The documents are archived in SSFL Building T100.

1. V. A. Swanson, "Building T028 Radiological Survey Plan," Rockwell International Supporting Document N704DWP990095, to be released in 1991.
2. A. Klein, "Building T028 Decontamination and Demolition Final Report," Rockwell International Supporting Document N001TI000322, June 6, 1990.
3. Building T028 radiological survey and other supporting data, including Instrument Qualification Reports, survey location maps and diagrams, and Health and Safety Analysis Reports.
4. Spreadsheet data on measurements of total and removable alpha/beta activity, and ambient gamma exposure rate data, measured at T028 as part of the final radiological survey.
5. B. M. Oliver, "Final Decontamination and Radiological Survey of Building T028," Rockwell International Supporting Document N704SRR990033, February 1990.

EXHIBIT VI

NATIONAL ENVIRONMENTAL POLICY ACT
DOCUMENTATION FOR DECONTAMINATION AND
DECOMMISSIONING OF BUILDING 028 AT ENERGY
TECHNOLOGY ENGINEERING CENTER

ERWM
LIDDLE
4/15/92

ERWM
CULLEN
4/15/92

AMEMS
DAVIS
4/16/92

OCC
BRECHBILL
4/22/92

DAMA
LAMBERT
4/24/92

DM
VAETH
4/27/92

AMC
DAVIS
92
4-29-92

APR 29 1992

DOE San Francisco Field Office (ERWM)

Categorical Exclusion (CX) Determination for Environmental Remediation of Buildings and Work Areas by Decontamination and Removal and Disposal of Hazardous and Radioactive Waste

Susan Brechbill, Acting AMEMS

In accordance with DOE NEPA Guidelines, Section D, and SEN-15-90, I have determined that the subject project satisfies the requirements for exclusion from further NEPA review based on the following:

CX DETERMINATION

NEPA Document Number: ET-EM-92-12

Proposed Action: Environmental Remediation of Buildings and Work Areas by Decontamination and Removal and Disposal of Hazardous and Radioactive Waste

Location: Energy Technology Engineering Center (ETEC), Santa Susana Field Laboratory, Ventura County, CA

Description: Remove stored equipment, decontaminate facilities and adjacent grounds to remove low level radioactivity contamination, and restore them to conditions suitable for use without radiological restrictions. Also, excavate, as needed, adjacent grounds to remove hazardous and radioactively contaminated soil and debris. Package the hazardous and radioactively contaminated fixtures, surplus equipment and debris, and ship it to an approved radioactive waste disposal facility.

Buildings and Work Areas to be Remediated

Radioactive Materials Disposal Facility (ADS 4005-AC):
- Building 022, RA Materials Storage Vault
- Building 021, Decontamination and Packaging
- Building 034, Offices
- Building 044, Health-Physics Services
- Four peripheral storage structures & the storage yard

Building 023, Liquid Metals Chemistry Laboratory (ADS 5002-AC)

Buildings and Work Areas to be Remediated (Continued)

SSFL Work Areas Decontamination (ADS 4006-WC):
Sodium Reactor Experiment (SRE) Moderator Shipping Cask stored in:
Building 012, SNAP Critical Facility
Building 100 Area, Construction Work Trenches
Old Conservation Yard Packaged Waste Disposal

CX To Be Applied (from Section D, DOE NEPA Guidelines):

CX as identified in Federal Register Volume 55, Number 174, dated September 7, 1990, for "1. The removal actions and other actions described below, if it is determined that such an action would not threaten a violation of applicable statutory, regulatory or permit requirements, including requirements of DOE Orders; would not require siting and construction or major expansion of waste disposal, recovery, or treatment facilities (including incinerators and facilities for treating waste water, surface water, or ground water); and would not adversely affect environmentally sensitive areas.... c. Removal actions under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) (including those taken as final response actions and those taken before remedial action) and actions similar in scope under the Resource Conservation and Recovery Act (RCRA) and other authorities (including the Atomic Energy Act, as amended) and those taken as partial closure actions and those taken before corrective action.... (12) Use of chemicals and other materials to retard the spread of the release or to mitigate its effects, where the use of such chemicals would reduce the spread of, or direct contact with, the contamination; {and}.... (16) Treatment (including incineration), recovery, storage or disposal of wastes at existing facilities permitted for the type of waste resulting from the removal action, where needed, to reduce the likelihood of human, animal, or food chain exposure."

The project will not affect historic, archaeological, or architecturally significant properties; will not impact environmentally sensitive areas or critical habitats; is not located in a floodplain, wetland, or prime agricultural land; and will not utilize special sources of water, sole source aquifers, well heads, or other resources vital to the region.

I have determined that the proposed action meets the requirements for the CX referenced above. Therefore, I have determined that the proposed action may be categorically excluded from further NEPA review and documentation.

/s/
James T. Davis
Acting Manager

cc: D. Williams, EM-443
A. Kluk, EM-443
C. Borgstrom, EH-25