Lt Col. Adcock
WRAMC - HPO

HARRY DIAMOND LABORATORIES

ADELPHI, MARYLAND

THE PLAN

FOR

THE DECOMMISSIONING OF THE DIAMOND ORDNANCE RADIATION FACILITY

AT

FOREST GLEN SECTION

WALTER REED ARMY MEDICAL CENTER

30 JUNE 1977

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I. Introduction

A. Background. The Harry Diamond Laboratories (HDL) operates the Diamond Ordnance Radiation Facility (DORF) which utilizes a research reactor with associated experimental equipment. The facility occupies a single remote building on 4.2 acres of the Forest Glen Annex of the Walter Reed Army Medical Center (WRAMC) near Silver Spring, Maryland. An intraservice agreement between the Commanding Officer, WRAMC and Commanding Officer, HDL establishes the WRAMC support services for DORF.

The reactor is the familiar General Atomic Company TRIGA Mark F, moderated by light water and mounted on a track support carriage assembly which can be moved through a 15,000 gallon capacity pool. The reactor core consists of 85 (maximum 87) fuel elements, four control rods, neutron source, and miscellaneous neutron detectors. The fuel elements are composed of zirconium hydride moderator homogeneously combined with 20% enriched uranium fuel. The control system consists of borated graphite safety, shim, regulating and pulse control rods, having either solid aluminum or fuel followers. Experiments are conducted in a 20 x 20 x 8 foot high fast neutron exposure room adjacent to pool, the pool itself, and within the core.

The facility was orginally developed in late 1959 and began operations in September 1961. Modifications applied since then include (1) replacement of the aluminum clad fuel elements with stainless steel clad elements (1964), (2) automatic SCRAM timing (1969), (3) replacement of the poison-followed transient rod with an aluminum follower (1964), (4) replacement of aluminum follower control rods with fuel-followed control rods (1971), and (5) replacement of reactor instrumentation with up-to-date instrumentation (1973).

The reactor has the capability of the following modes of operation:

1. Steady-state operation up to 250 kW.

2. Square-wave operation up to 250 kW.

3. Pulse operation resulting in up to a maximum peak power of 2000 MW with a pulse width of 9.5 ms at half maximum.

B. The decision to decommission the DORF reactor is the culmination of an Army reactor utilization study begun in mid-1975 to examine the requirement for the three Army research reactors. This study was done by HDL Nuclear Weapons Effects Program Office (NWEPO) which investigated the following alternatives:

1. Operation of the three reactors in their present locations.

2. The shutting down of one pulse reactor.

3. The consolidation of the Aberdeen Pulse Reactor with the Diamond Ordnance Radiation Facility (DORF) at either Forest Glen, Md. or Aberdeen Proving Ground, Md.

4. The consolidation of the DORF with the White Sands Missile Range (WSMR) pulse reactor with the closing of the Aberdeen Pulse Reactor.

5. Closing down of all Army effects research reactors with experiments conducted at either ERDA or other service reactor sites.

6. The closing of one pulse reactor and the DORF facility.

In addition, an Army Scientific Advisory Panel Ad Hoc Group on Pulse Reactors was asked to review the technical capabilities of the Army's two pulsed reactor facilities at WSMR and APRF and to review documentation including the Study for Requirements for Nuclear Weapons Effects Research Reactors prepared by NWEPO, HDL. The Ad Hoc Group also visited DORF and although they were not asked to address the DORF facility, their conclusion was that if any of the three reactors should be closed, DORF would be the logical candidate. This conclusion is based on the availability of TRIGA type reactors at the Armed Forces Radiological Research Institute and other facilities nationally. In January 1977 the HDL were directed by Headquarters, US Army Materiel Development and Readiness Command to decommission DORF with an initial closure date of 1 July 1977. The closure date was subsequently changed to 1 October 1977 so that planned experiments could be completed.

C. The primary objective of the decommissioning of the DORF is: (1) to remove the Special Nuclear Material (SNM), i.e., reactor fuel elements, and to return it to US Energy Research and Development Administration (ERDA) for disposal; (2) to remove all radioactive material from the facility and ship to a Nuclear Regulatory Commission (NRC) licensed burial site; and (3) to decontaminate and prepare the facility building for alternate use. Upon completion of decommissioning the facility building will be given to Walter Reed Army Medical Center (WRAMC) for their use. The purpose of this Decommissioning Plan is to outline the method of accomplishing this objective in a safe manner which will allow least exposure to personnel or contamination to the environment.

D. In decommissioning the DORF, the removal and shipment of the fuel elements is relatively straightforward but time consuming. There are approved TRIGA fuel shipping casks available, however they may not be available until early 1978 or later. Because of the Army's difficulty in obtaining indemnification, the shipment of the fuel will be performed by ERDA. The precise details of all the decommissioning and dismantling operations cannot be specified since tasks to be accomplished at a latter period are dependent upon results and judgement of earlier operations. For example, after the removal of the wood lining in the exposure room, the amount of concrete to be removed will depend upon its radioactivity. Therefore, this Plan will not give detailed procedures which may change at a latter date, but will give firm objectives which must be accomplished to complete the decommissioning operations. The detailed procedures and hazard analysis will be attached to the plan as appendices.

II ORGANIZATION AND RESPONSIBILITIES

A. The health and safety aspects of the Department of the Army nuclear reactor systems is a command responsibility. The specific command responsibilities are stated in AR 385-80, Chapter 2. The organization chart for the DORF Decommissioning is shown in Fig. 1.

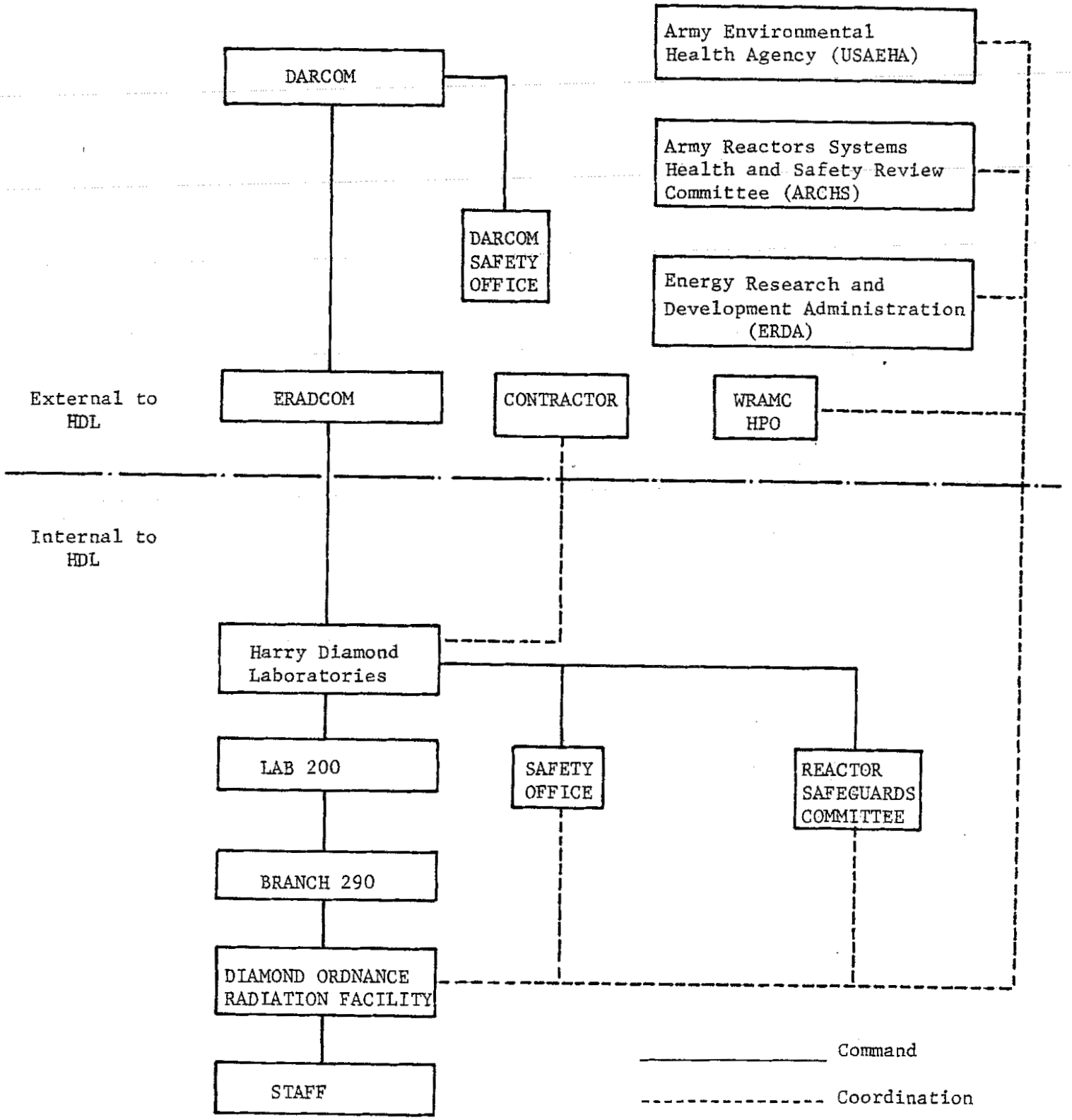
1. Responsible Commander. The Commanding Officer, Harry Diamond Laboratories is the responsible commander. He has ultimate responsibility for the operation of the reactor. He shall comply with the requirements of AR 385-80 and insure that all operations of the DORF reactor systems are conducted in a safe manner.

2. Reactor Safeguards Committee. The Reactor Safeguards Committee (RSC) is appointed by and reports to the Commanding Officer, HDL. The committee consists of personnel who collectively provide a broad spectrum of experience in reactor technology. The RSC will review the plans and procedures of decommissioning.

3. Chief, Nuclear Radiation Effects Laboratory (Lab 200). The Chief, Lab 200 is the direct line of authority from the reactor commander to the HDL commanding officer.

4. Reactor Commander. The Chief, Nuclear Effects Simulation Technology Branch (Br. 290), is the reactor commander as defined in AR 385-80. He has the direct responsibility for the safe, competent and efficient operation and use of the facility. These responsibilities will continue through DORF decommissioning.

5. Physicist-in-Charge. The Physicist-in-Charge (PIC) is the Facility Supervisor. He has the direct and immediate responsibility for the facility. He is responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the DORF Technical Specifications. These responsibilities will continue until the DORF decommissioning is concluded.



ORGANIZATION FOR DECOMMISSIONING

Figure 1

B. Support from Other Commands and Agencies

1. Army Reactors Systems Health and Safety Review Committee (ARCHS)

The Safety Director, Office of Chief of Engineers, has been delegated the responsibility of surveillance and general guidance in all health and safety matters relating to Army nuclear reactor systems, which includes the decommissioning of DORF. The Army Nuclear Reactor Systems Health and Safety Review Committee (ARCHS) was established to assist in discharging this responsibility. The ARCHS will review and approve this decommissioning plan, safety analysis and other applicable documentation.

2. US Army Environmental Health Agency (USAEHA)

Following decommissioning and in-house radiological surveys, the USAEHA will be requested to conduct an independent radiological survey of the DORF site.

3. US Energy Research Development Administration (USERDA)

ERDA will be requested to supply the fuel transfer casks and the transportation of the fuel elements to the disposal site.

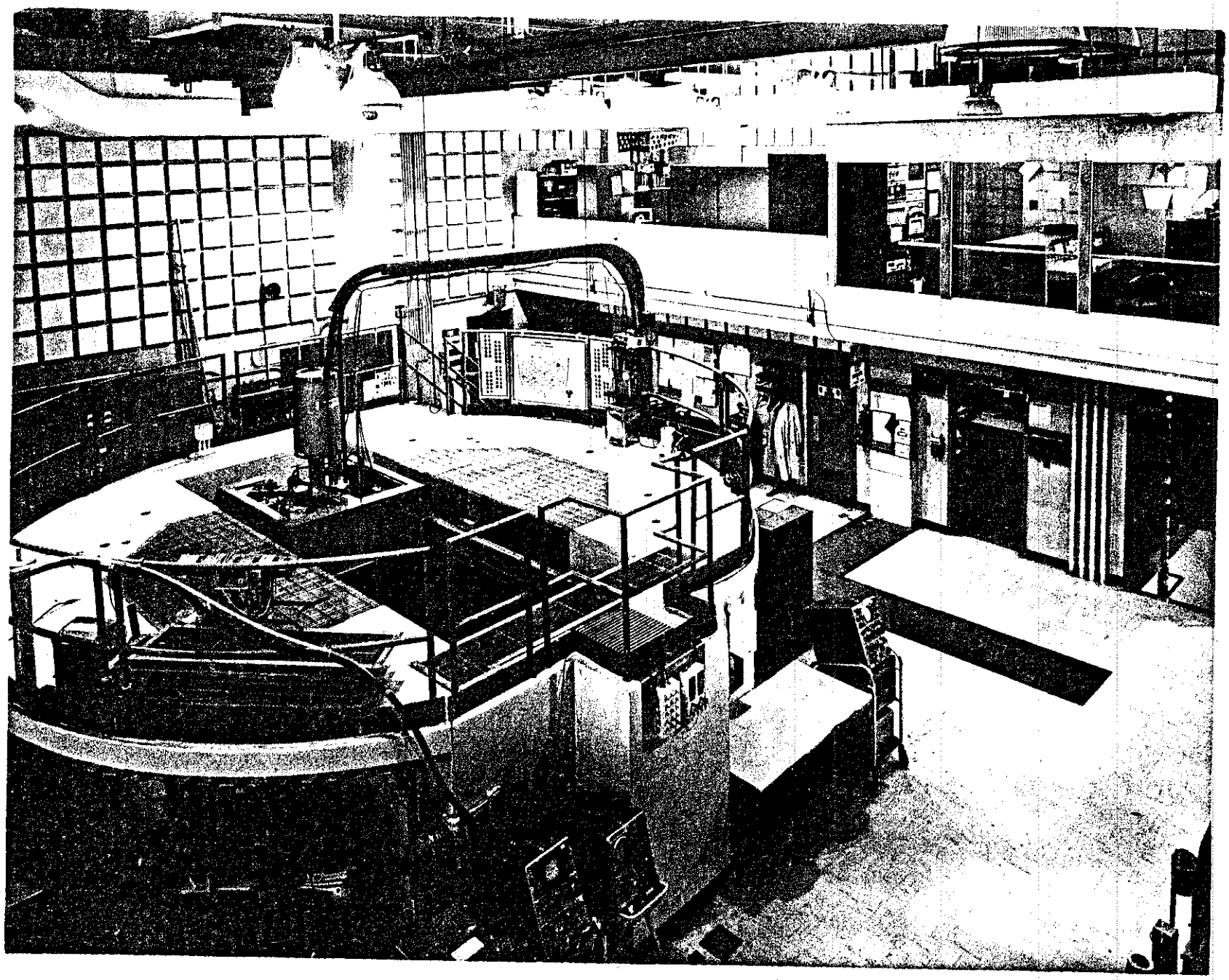
4. Civilian Contractor

A contract will be negotiated for the dismantlement and decontamination of the facility, the packing and shipment for disposal of radioactive waste, and the preparation of the building for alternate use.

III OBJECTIVES

A. Summary of Tasks

The facility decommissioning will begin immediately after the approval of this Decommissioning Plan. The decommissioning operation will be completed when all accessible areas of the facility are radiologically safe and unrestricted for personnel occupancy. The Energy Research and Development Administration ERDA has been asked to negotiate the fuel cask and shipment contract. This approach is used because ERDA is indemnified and the Army is not. The fuel elements will be sent to the chemical reprocessing plant in Idaho or to other TRIGA reactor facilities specified by the University Relation Division, ERDA. The dismantling, removal of radioactive equipment, and building decontamination will be done by contract. The Decommissioning Plan will be prepared by HDL personnel and approved by the Army Reactor Systems Health and Safety Review Committee (ARCHS) before implementation. The overall layout of the reactor is shown in Figures 2 through 5. Generally, the decommissioning operation will proceed as follows:



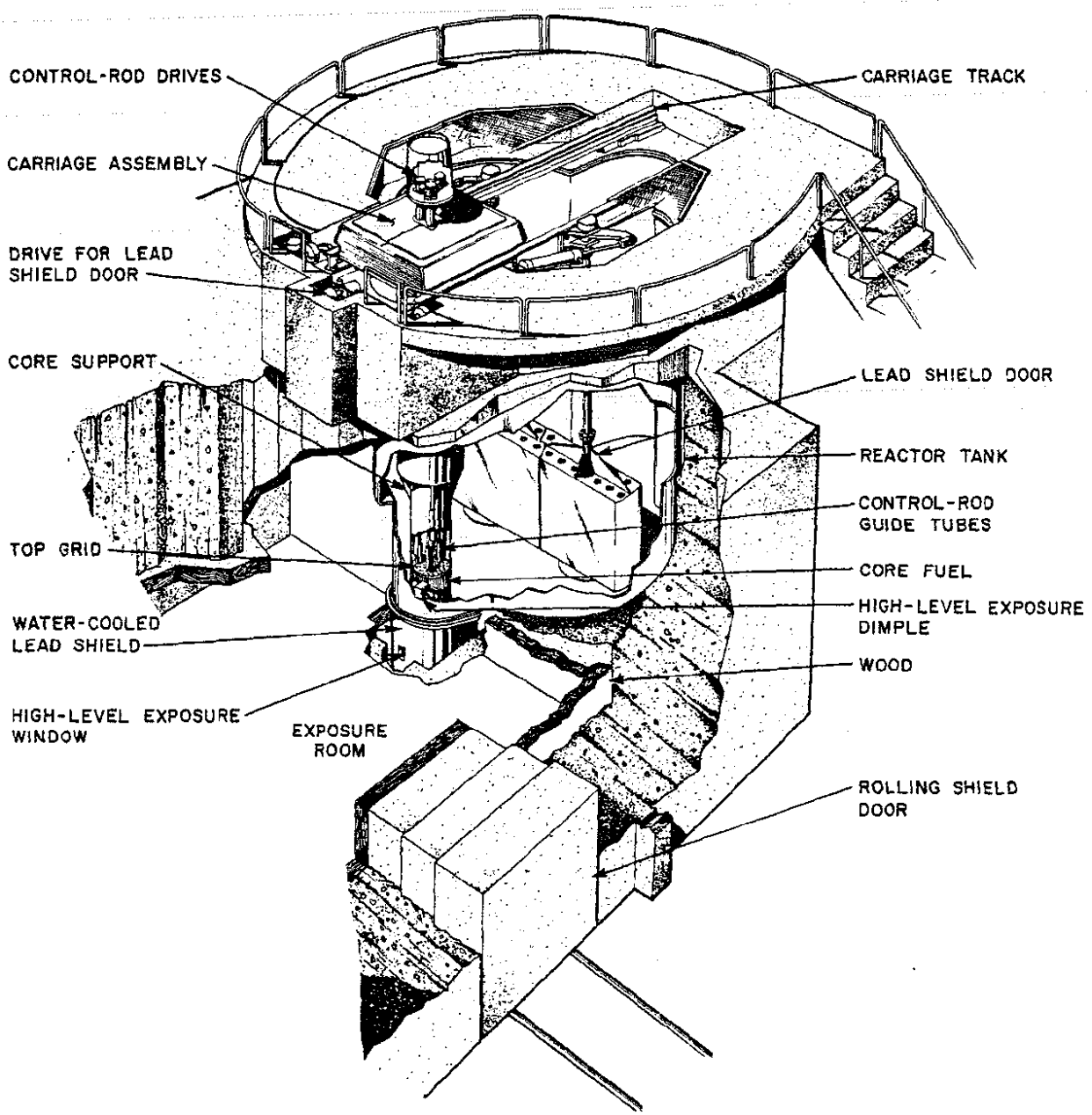


Fig. 5-1 -- Perspective view of DORF reactor

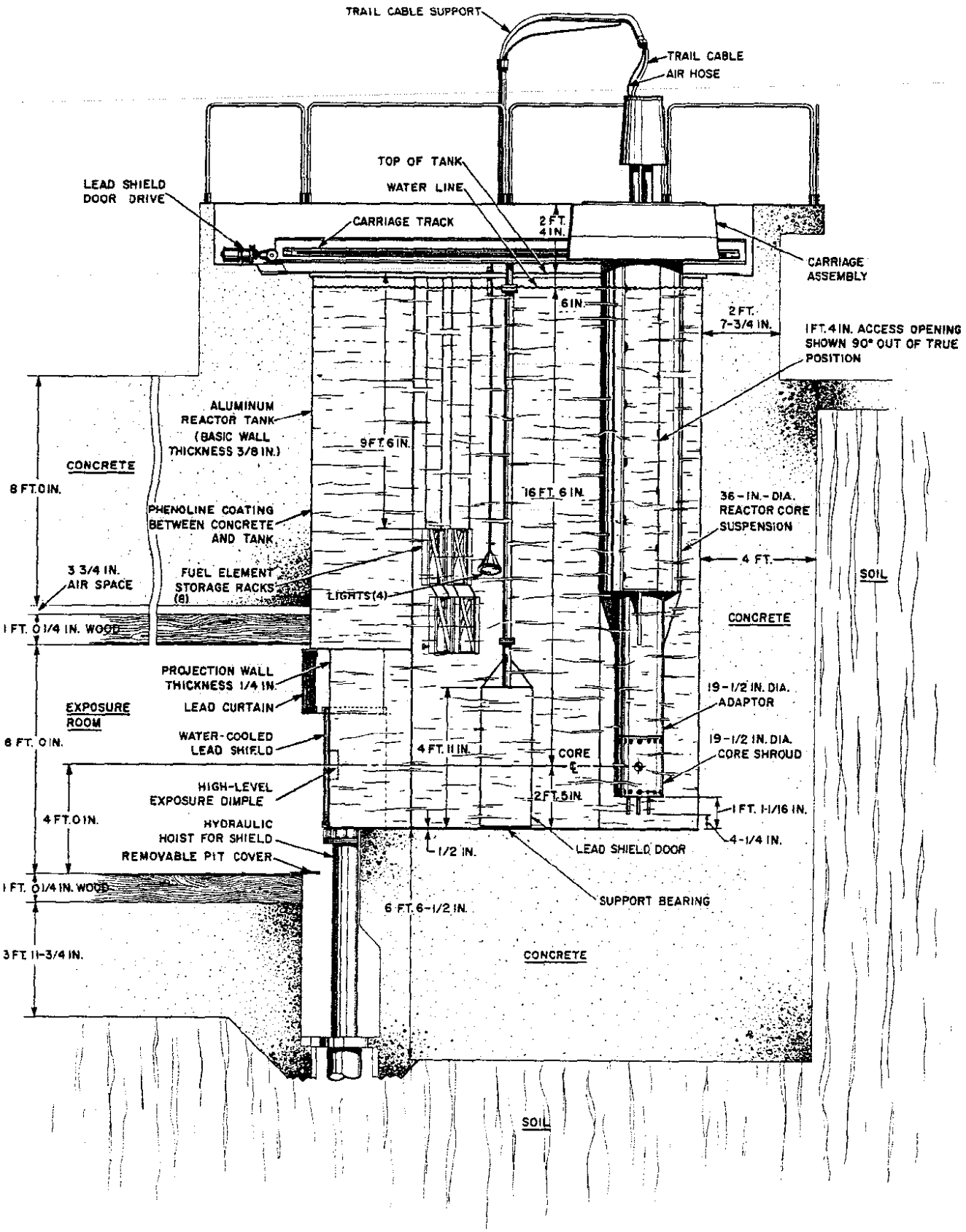


Fig. 5-3 -- Sectional elevation of DORF reactor

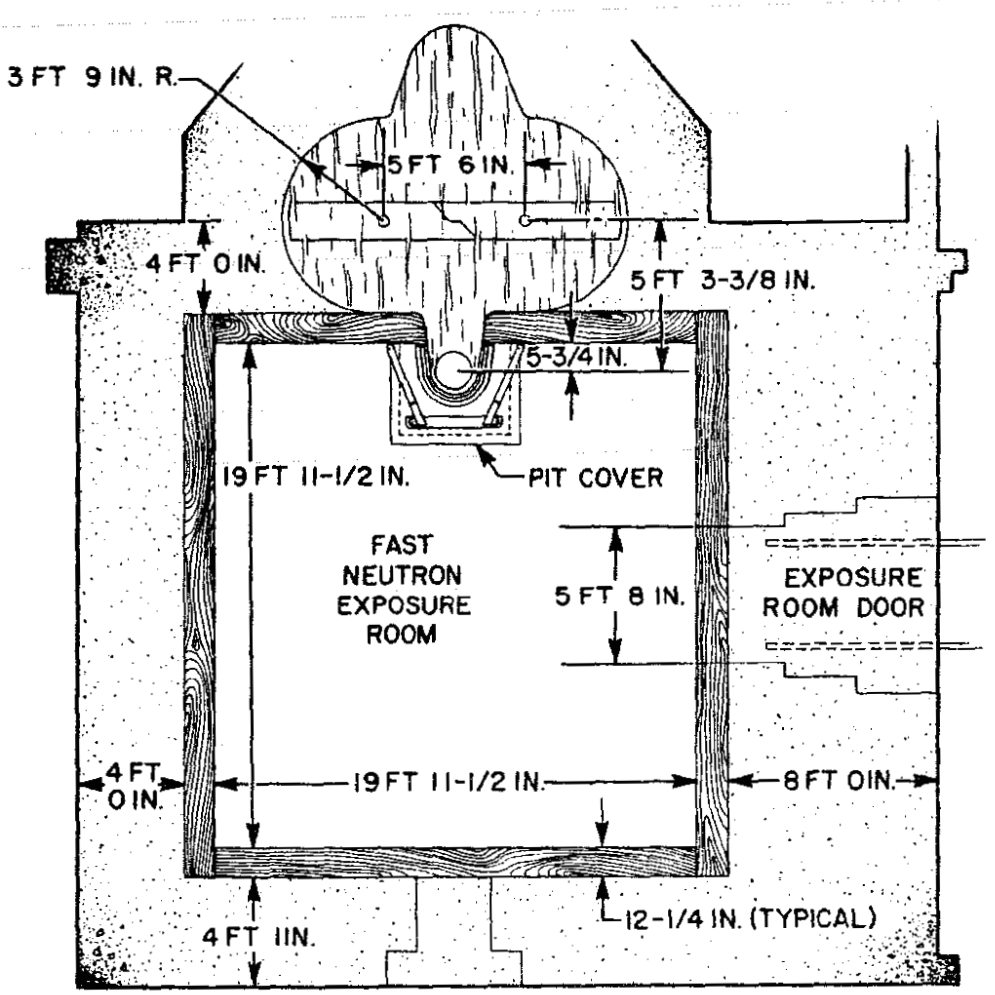


Fig. ~~7-2~~ 5 -- Plan view of DORF exposure room

1. The reator fuel will be transferred from the reactor core to the in-pool storage racks. (Note: This will be done immediately after operations cease on 1 Oct 1977. These are approved standard pool storage for the fuel elements and storage will assure reactor non-operability).

2. The fuel elements and fuel-followed control rods will be packaged in approved fuel shipping casks and shipped to the US Energy Research and Development Administration (ERDA), Idaho Reprocessing Plant. There is a possibility that some of the irradiated fuel elements will be shipped to other TRIGA reactor facilities. The loading will be performed by DORF personnel. The fuel shipment will be performed by ERDA on a cost recoverable basis. Unirradiated fuel elements will be transferred to the Armed Forces Radiological Research Institute (AFRRI).

3. Standard control rods and other radioactive reactor structures will be shipped from the facility for disposal at NRC licensed burial site. Selected reactor structures will be shipped to interested TRIGA reactor owners.

4. The reactor pool water will be discharged to the sanitary sewer provided the radioactivity is within the standards set by 10CFR20.

5. The reactor carriage, core support structure, and pool shield doors will be removed and activated sections will be shipped for disposal at a NRC licensed burial site.

6. The aluminum pool tank will be removed. Radioactive sections will be shipped from the facility site to an approved burial site.

7. Sufficient activated concrete beneath the aluminum liner will be removed to comply with acceptable surface contamination levels in NRC regulatory Guide 1.86, Termination of Operating Licenses for Nuclear reactors. (See Table 1)

8. Wood liner in the exposure room will be removed and shipped for disposal at a NRC licensed burial site.

9. Activated concrete in exposure room and exposure room plug door will be removed and shipped for disposal at a NRC licensed burial site. Sufficient concrete will be removed to comply with contamination levels in Table 1.

10. Exposure room lead shield and hoist and lead curtain will be removed and activated sections shipped for disposal at a NRC licensed burial site.

11. All activated structures will be decontaminated or removed for disposal.

TABLE I

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVERAGE[b c]	MAXIMUM[b d]	REMOVABLE[b e]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm β-γ/100 cm^2	15,000 dpm β-γ/100 cm^2	1000 dpm β-γ/100 cm^2

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

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

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

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

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

12. Reactor console and associated equipment will be removed by HDL personnel and shipped to AFRRI.

13. Building will be prepared for alternate use by removing parapet and restoring floor to bay level.

14. Remove three 5000 gallon waste water hold-up tanks and reconnect sewer system. Tanks to be shipped for disposal as non-radioactive waste.

15. A final site survey will be made by an independent agency to insure that none of the areas will exceed NRC guidelines for un-restricted use.

IV TRAINING

A. DORF Personnel. Fuel element handling will be done by DORF personnel who have the experience and training in this operation. Special training and briefings will be conducted prior to transfer of the fuel elements to shipping casks. Other special training will be conducted, as appropriate, for those tasks not performed during normal reactor operations. Removal of large items requiring rigging techniques will require personnel (contractor) who are familiar with these operations. Training will be on-site and will be the responsibility of the Physicist-in-Charge.

B. NON-DORF Personnel. All non-DORF personnel associated with the decommissioning will receive a briefing which will include reactor facility entrance and exit procedures, radiation areas and exclusion areas, the identification and meaning of various radiation signs, controlled areas, and use of radiation monitoring devices.

C. SUPPORT UNITS. Periodic training of support units such as the fire department personnel and WRAMC military police is their own responsibility. These units have participated in DORF quarterly emergency drills and will be apprised, and briefed when necessary, of decommissioning operations.

V. QUALITY ASSURANCE

A. The DARCOM requirements for quality assurance at reactor facilities is contained in DARCOM Supplement 1 to AR 385-80. The surveillance requirements and management procedures in the DORF Technical Specifications, and ENRADMON and Health Physics Plans are designed to assure that adequate control over activities affecting all aspects of the reactor system operations are maintained. Additional quality assurance during decommissioning will

be implemented, as necessary, in accordance with the above DARCOM Supplement and present DORF procedures and plans.

VI. RADIATION PROTECTION

A. Personnel Monitoring

1. Basic Radiation Protection Standards

a. The basic radiation protection standards are prescribed in Army Regulation 40-14, and Walter Reed Army Medical Center Regulation 40-10. Every effort will be made to maintain the radiation dose equivalent as far below the following Radiation Protection Standards as practicable. Positive efforts will be carried out to fulfill this objective; and, determination of necessity will be weighed against the benefits to be expected.

b. Basic Radiation Protection standards adopted for the control of occupational exposures to ionizing radiation include:

(1) The accumulated dose equivalent of radiation to the whole-body; head and truck; active blood-forming organs; gonads; or lens of the eye will not exceed:

(a) 1.25 rem in any calendar quarter, nor

(b) 5 rem in any 1 calendar year

(2) The accumulated dose equivalent of radiation to the skin of the whole-body (other than hands and forearms); cornea of the eye; and bone will not exceed:

(a) 7.50 rem in any calendar quarter, nor

(b) 30 rem in any 1 calendar year

(3) The accumulated dose of radiation to the hands and wrists or the feet and ankles will not exceed:

(a) 18.75 rem in any calendar quarter, nor

(b) 75 rem in any 1 calendar year

(4) The accumulated dose of radiation to the forearms will not exceed:

(a) 10 rem in any calendar quarter, nor

(b) 30 rem in any 1 calendar year

(5) The accumulated dose equivalent or radiation to the thyroid; other organs; tissues; and organ systems will not exceed:

(a) 5 rem in any calendar quarter, nor

(b) 15 rem in any 1 calendar year

(6) Individual(s) under 18 years of age, females known to be pregnant, and occassionally exposed individual(s) will not be exposed to a whole-body dose equivalent of more than:

(a) 2 millirem in any 1 hour, nor

(b) 100 millirem in any 7 consecutive days, nor

(c) 500 millirem

(d) more than 10 percent of the values in (2), (3), (4) and (5) above, for other areas of the body

(7) Individuals over 18 years of age but who have not yet reached their 19th birthday may be occupationally exposed to ionizing radiation provided that they do not exceed 1.5 rem dose equivalent to the whole-body in any calendar quarter, nor 3 rem in the 12 consecutive months prior to their 19th birthday.

2. Monitoring Devices

a. A film badge is the primary dosimeter device for monitoring personnel exposure. Personnel selected for personnel monitoring will include:

(1) Individuals who are likely to be exposed to sufficient radiation from all occupational exposures to receive an accumulated dose in excess of ten (10) percent of the applicable basic Radiation Protection Standard.

(2) Those other individuals selected by the Health Physicist on duty.

(3) Contractors will provide dosimeters for their own personnel.

b. The following types of personnel monitoring devices will be available:

(1) Whole Body badge: Sensitive to beta, x-ray and gamma radiation and worn to measure the exposure received by the whole body

(2) Wrist badge: Same as the whole body except that it is provided with a wrist band so that it can be used to measure the dose to the hands

(3) Pocket ionization chambers will be used to provide a means of obtaining rapid indication of accumulated dose over short periods of time. This chamber enables individuals to monitor their own accumulated dose.

3. Documentation

a. WRAMC Health Physics will supply the film badges. Records are kept on WRAMC Form 119 (Film Badge Application and Record of Occupational External Radiation Exposure). Health Physics will exchange film badges and transmit the film packets, along with photodosimetry reports, to Lexington-Bluegrass Army Depot for monthly development and exposure evaluation. Records of exposure will be maintained as follows:

(1) The Lexington-Bluegrass Army Depot maintains permanent records of all exposures and returns the Photodosimetry Report (DA 3484) to the WRAMC Health Physics Officer.

(2) The WRAMC Health Physics Officer maintains DD Form 1141 for all military and civilian personnel assigned or attached to WRAMC.

b. Cumulative daily Pocket ionization chamber dose readings will be recorded on WRAMC Form 705, "Pocket Dosimeter Log".

c. In general, contractors will maintain their own records of film badge exposures. However, they will be supplied pocket dosimeters at the direction of the Health Physicist and exposure records will be maintained on Form 705.

4. Use of Monitoring Devices

Use of personnel monitoring devices is contained in WRAMC Regulation 40-10, Annex E (PERSONNEL MONITORING). During the DORF decommissioning the following general guidelines will be followed:

a. Film badges: Each person who occupies a controlled area will wear a film badge, (beta, gamma) unless specifically exempted by the Health Physicist. These badges will be worn at all times within the controlled area and left at the facility when leaving. Neutron and wrist badges will be worn at the direction of the Health Physicist.

b. Pocket dosimeters; Each person who occupies a controlled area will wear a self-reading pocket dosimeter to provide an immediate indication of accumulated dose over short periods of time.

c. Contract specifications will require, as a minimum, that the contractor will supply the above (a and b) monitoring devices for their personnel.

5. Administrative Control

a. Staff Personnel: The fuel elements will be removed from the reactor pool to the shipping casks by DORF staff personnel who have had experience in handling TRIGA Fuel elements. All personnel will be briefed on tasks and hazards that exist. Rehearsals and "dry runs" will be made to reduce radiation exposure and to reduce hazards where applicable.

b. Contractors and Visitors:

(1) Contractor work will begin after the fuel elements have been removed and shipped from the facility. Therefore, the major source of radiation hazard will be gone when they begin the facility dismantling and decontamination. However, the contractors will be throughly briefed on the location of the remaining radioactive material and hazards involved, also use and control of monitoring devices and safety precautions to be utilized.

(2) Visitors will be accompained at all times when within the controlled area. The existing entrance sign-in-log and personnel dosimeter monitoring procedures will be followed.

B. Liquid Waste Processing

All liquid waste at the facility, except comode and urinals, drain into 15,000 gallon hold-up tanks. Before discharge to the sanitary sewer, the tank affluent is sampled and analyzed for radioactivity to insure that the maximum permissible concentration (MPC) listed in 10CFR20 is not exceeded.

C. Radiological Surveys

1. Plant surveys. The procedures for radiological surveys are contained in the DORF Health Physics Plan. These procedures insure that the facility is provided with adequate routine radiological daily, weekly and monthly checks. Particular attention will be made of radioactive dose rates during fuel transfer to shipping casks and during dismantlement of radioactive structures to insure minimum exposure to personnel. Additional surveys and activation analysis will be done during concrete removal to determine when sufficient material is removed so that the area can be made an unrestricted area. The Health Physicist will inform and advise persons of anticipated or existing radiation hazards and these hazards will be posted when required.

2. Contamination Control Plan. Control points will be established at the point of entry of any contaminated area. The procedures established in the Health Physics Plan will be followed. Materials and tools removed from the contaminated area will be surveyed. It is the goal of the decommissioning to remove sufficient activated materials so that the residual radiation levels do not exceed the values in Table 1.

3. Effluent Monitoring. Air borne effluents will continue to be monitored by the existing stack monitor as required by the DORF Technical Specifications. Local air borne activities will be monitored using a Staplex Air Sampler as directed by the HP. All building air is exhausted through the absolute filters and out the stack.

VII DOCUMENTATION AND RECORDS

A. Logs

1. Decommissioning log. A log will be kept of all events, as they occur, during decommissioning. The purpose of the log is to present as a record of all the events, drawings, sketches and other information that are required to perform decommissioning tasks. As a minimum, the following will be recorded:

a. Daily job status

b. Initiation and completion of each task

c. Personnel injuries

d. Accidents

e. Schedule delays

f. Facts bearing on any problem areas

g. Major shipments (incoming and outgoing)

h. Weekly summary: Work progress should be summarized in the log on a weekly basis.

2. Health Physics Log. A health physics log will be maintained recording the chronological decommissioning events. The health physicist on duty will be responsible for maintaining the log. As a minimum the following will be recorded:

a. Remarks pertaining to ~~daily~~ monitoring of decommissioning tasks to include any special monitoring performed.

b. The number and weight of radioactive waste containers filled

c. Status of radioactive waste in storage awaiting shipment

d. Amount (gallons and activity) of liquid waste discharge (to include hold-up tanks)

e. Amount and average concentration of airborne activity discharged.

f. Movement of all highly radioactive equipment, components of structures.

g. All incidents pertaining to radiological safety

h. Facts bearing on any problem areas

i. Issuance and termination of all radiation work permits (RWP)

j. Shipment of radioactive samples (quantity and description) and analysis results when received from Health Physics or a contractor.

3. Quality Control and Assurance. Quality control and quality assurance inspections and tests, where applicable, will be the responsibility of the Physcisit-in-Charge. Results of inspections, quality control and quality assurance observations will be kept in the decommissioning log.

4. Disposition. After completion of the decommissioning task, all logs will be forwarded to the HDL Technical & Administration Support Office for disposition.

B. Reports

1. Final Inspection Report. The reactor Commander (Chief Branch 290) shall be responsible for preparing a report on the final inspection of the decommissioned DORF.

2. Final Decommissioning Report. The Reactor Commander (Chief Branch 290) will be responsible for the preparation of the final decommissioning report. This report will contain all the aspects of the decommissioning tasks and contain the safety analysis of the facility in its ultimate status. The report will be forwarded through Commander, DARCOM (ATTN: DRCSF-P) to Chief of Engineers (ATTN: DAEN-SON).

3. Monthly Decommissioning Progress Report. A monthly decommissioning progress report will be prepared by the Physicist-in-Charge and forwarded to the Commander DARCOM (ATTN: CRCSF-P) by the 15th day of the month immediately following the reporting period.

C. Records

1. Fuel Accountability. The DORF fuel elements (Special Nuclear Material) accountability will be transferred from the US Army, Harry Diamond Laboratories to ERDA. The fuel elements will be sent to the Idaho Reprocessing Plant or to other TRIGA reactor plants as determined by the University Relations Division, ERDA.

2. Files and Records. After completion of decommissioning, all DORF operations logs and records will be transferred to HDL Technical and Administration Support Office for retention or final disposition in accordance with AMC Supplement to AR 385-80.

VIII EMERGENCY PLANS

A. DORF Emergency Plan

The Emergency Plan required by the Technical Specifications will be in effect through the Decommissioning operations. The purpose of the DORF Emergency Plan is to establish the procedures to be followed in the event of an accident or emergency at the reactor facility. Accordingly, these procedures are to insure the protection of the health and safety of personnel under accident or emergency conditions. The objective of the Emergency Plan is to provide complete and coherent procedures for dealing with emergency and accident situations. The potential emergency conditions which may arise at the DORF are classified into two categories:

(1) The Design Basis Accident and, (2) varying degrees of lesser accidents. After the SNM fuel has left the facility site, the Design Basis Accident considerations would no longer be in effect.

B. OTHER PLANS AND SOPS

The following plans and standard operating procedures will be in effect during decommissioning where applicable:

1. Technical Specifications for the DORF, 13 Aug 1973
2. Health Physics Plan for the DORF Reactor, 18 March 1975
3. DORF ENRADMON PLAN, 1 Feb 1977
4. Operations Procedures for DORF, HDL Pamphlet 70-5
5. DORF Emergency Plan 1973
6. Physical Security Plan, DORF, 25 Sept 1975

7. SOP No. 4 Area Radiation Monitoring System

8. SOP No. 6 Procedures for Measuring Fuel Elements and Fuel Following Control Rods

9. SOP No. 8 Criteria for Replacement of Pre and Absolute Filters in Exhaust System

10. SOP No. 12 Procedures for Special Nuclear Material (SNM) Inventory

11. SOP No. 13 Personnel Access Control

12. WRAMC Health Physics Regulations, 40-10

IX CONTRACTING

A. Nuclear Fuel Shipment

The shipment of the DORF Special Nuclear Material (SNM) is governed by AR 55-5, Department of Transportation Regulation and Nuclear Regulatory Commission Regulation 10CFR71. U.S. Energy Research and Development Administration has been asked to contract for the shipping cask and for the shipment of the fuel elements to the reprocessing plant in Idaho. A licensed Battelle, Columbus BM-1 shipping cask for the standard TRIGA fuel elements is available in November 1977, or if this date cannot be met early 1978. A different cask will be used for the fuel followed control rods and instrumented fuel elements because of their longer length. Some of the fuel elements may be shipped to other TRIGA reactor facilities.

B. Solid Radioactive Waste Shipments

1. Control Rods. The control rods will be shipped with the solid radioactive waste. The contact dose rates from these rods is not expected to exceed 100 mr/hr.

2. Solid Radioactive Waste. The shipment of solid radioactive waste will be contracted to a commercial carrier. Disposal will be in accordance with AR 755-15 at an NRC licensed burial site. It is expected that solid radioactive waste will be low level.

X. DECOMMISSIONING TASKS

A listing of the decommissioning tasks is tabulated in Appendix 2.

APPENDIX 1

NRC REGULATORY GUIDE 1.86

TERMINATION OF OPERATING LICENSES
FOR NUCLEAR REACTORS

June 1974

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

NRC REGULATORY GUIDE 1.86

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

A. INTRODUCTION

Section 50.51, "Duration of license, renewal," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each license to operate a production and utilization facility be issued for a specified duration. Upon expiration of the specified period, the license may be either renewed or terminated by the Commission. Section 50.82, "Applications for termination of licenses," specifies the requirements that must be satisfied to terminate an operating license, including the requirement that the dismantlement of the facility and disposal of the component parts not be inimical to the common defense and security or to the health and safety of the public. This guide describes methods and procedures considered acceptable by the Regulatory staff for the termination of operating licenses for nuclear reactors. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

When a licensee decides to terminate his nuclear reactor operating license, he may, as a first step in the process, request that his operating license be amended to restrict him to possess but not operate the facility. The advantage to the licensee of converting to such a possession-only license is reduced surveillance requirements in that periodic surveillance of equipment important to the safety of reactor operation is no longer required. Once this possession-only license is issued, reactor operation is not permitted. Other activities related to cessation of operations such as unloading fuel from the reactor and placing it in storage (either onsite of offsite) may be continued.

A licensee having a possession-only license must retain, with the Part 50 license, authorization for special nuclear material (10 CFR Part 70, "Special Nuclear Material"), byproduct material (10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Material"), and source material (10 CFR Part 40, "Licensing of Source Material"), until the fuel, radioactive components, and sources are removed from the facility. Appropriate administrative controls and facility requirements are imposed by the Part 50 license and the technical specifications to assure that proper surveillance is performed and that the reactor facility is maintained in a safe condition and not operated.

A possession-only license permits various options and procedures for decommissioning, such as mothballing, entombment, or dismantling. The requirements imposed depend on the option selected.

Section 50.82 provides that the licensee may dismantle and dispose of the component parts of a nuclear reactor in accordance with existing regulations. For research reactors and critical facilities, this has usually meant the disassembly of a reactor and its shipment offsite, sometimes to another appropriately licensed organization for further use. The site from which a reactor has been removed must be decontaminated, as necessary, and inspected by the Commission to determine whether unrestricted access can be approved. In the case of nuclear power reactors, dismantling has usually been accomplished by shipping fuel offsite, making the reactor inoperable, and disposing of some of the radioactive components.

Radioactive components may be either shipped offsite for burial at an authorized burial ground or secured

on the site. Those radioactive materials remaining on the site must be isolated from the public by physical barriers or other means to prevent public access to hazardous levels of radiation. Surveillance is necessary to assure the long term integrity of the barriers. The amount of surveillance required depends upon (1) the potential hazard to the health and safety of the public from radioactive material remaining on the site and (2) the integrity of the physical barriers. Before areas may be released for unrestricted use, they must have been decontaminated or the radioactivity must have decayed to less than prescribed limits (Table I).

The hazard associated with the retired facility is evaluated by considering the amount and type of remaining contamination, the degree of confinement of the remaining radioactive materials, the physical security provided by the confinement, the susceptibility to release of radiation as a result of natural phenomena, and the duration of required surveillance.

C. REGULATORY POSITION

1. APPLICATION FOR A LICENSE TO POSSESS BUT NOT OPERATE (POSSESSION-ONLY LICENSE)

A request to amend an operating license to a possession-only license should be made to the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545. The request should include the following information:

a. A description of the current status of the facility.

b. A description of measures that will be taken to prevent criticality or reactivity changes and to minimize releases of radioactivity from the facility.

c. Any proposed changes to the technical specifications that reflect the possession-only facility status and the necessary disassembly/retirement activities to be performed.

d. A safety analysis of both the activities to be accomplished and the proposed changes to the technical specifications.

e. An inventory of activated materials and their location in the facility.

2. ALTERNATIVES FOR REACTOR RETIREMENT

Four alternatives for retirement of nuclear reactor facilities are considered acceptable by the Regulatory staff. These are:

a. **Mothballing.** Mothballing of a nuclear reactor facility consists of putting the facility in a state of protective storage. In general, the facility may be left intact except that all fuel assemblies and the radioactive fluids and waste should be removed from the site. Adequate radiation monitoring, environmental surveillance, and appropriate security procedures should be established under a possession-only license to ensure that the health and safety of the public is not endangered.

b. **In-Place Entombment.** In-place entombment consists of sealing all the remaining highly radioactive or contaminated components (e.g., the pressure vessel and reactor internals) within a structure integral with the biological shield after having all fuel assemblies, radioactive fluids and wastes, and certain selected components shipped offsite. The structure should provide integrity over the period of time in which significant quantities (greater than Table I levels) of radioactivity remain with the material in the entombment. An appropriate and continuing surveillance program should be established under a possession-only license.

c. **Removal of Radioactive Components and Dismantling.** All fuel assemblies, radioactive fluids and waste, and other materials having activities above accepted unrestricted activity levels (Table I) should be removed from the site. The facility owner may then have unrestricted use of the site with no requirement for a license. If the facility owner so desires, the remainder of the reactor facility may be dismantled and all vestiges removed and disposed of.

d. **Conversion to a New Nuclear System or a Fossil Fuel System.** This alternative, which applies only to nuclear power plants, utilizes the existing turbine system with a new steam supply system. The original nuclear steam supply system should be separated from the electric generating system and disposed of in accordance with one of the previous three retirement alternatives.

3. SURVEILLANCE AND SECURITY FOR THE RETIREMENT ALTERNATIVES WHOSE FINAL STATUS REQUIRES A POSSESSION-ONLY LICENSE

A facility which has been licensed under a possession-only license may contain a significant amount of radioactivity in the form of activated and contaminated hardware and structural materials. Surveillance and commensurate security should be provided to assure that the public health and safety are not endangered.

a. Physical security to prevent inadvertent exposure of personnel should be provided by multiple locked barriers. The presence of these barriers should make it extremely difficult for an unauthorized person to gain access to areas where radiation or contamination levels exceed those specified in Regulatory Position C.4. To prevent inadvertent exposure, radiation areas above 5 mR/hr, such as near the activated primary system of a power plant, should be appropriately marked and should not be accessible except by cutting of welded closures or the disassembly and removal of substantial structures

and/or shielding material. Means such as a remote-readout intrusion alarm system should be provided to indicate to designated personnel when a physical barrier is penetrated. Security personnel that provide access control to the facility may be used instead of the physical barriers and the intrusion alarm systems.

b. The physical barriers to unauthorized entrance into the facility, e.g., fences, buildings, welded doors, and access openings, should be inspected at least quarterly to assure that these barriers have not deteriorated and that locks and locking apparatus are intact.

c. A facility radiation survey should be performed at least quarterly to verify that no radioactive material is escaping or being transported through the containment barriers in the facility. Sampling should be done along the most probable path by which radioactive material such as that stored in the inner containment regions could be transported to the outer regions of the facility and ultimately to the environs.

d. An environmental radiation survey should be performed at least semiannually to verify that no significant amounts of radiation have been released to the environment from the facility. Samples such as soil, vegetation, and water should be taken at locations for which statistical data has been established during reactor operations.

e. A site representative should be designated to be responsible for controlling authorized access into and movement within the facility.

f. Administrative procedures should be established for the notification and reporting of abnormal occurrences such as (1) the entrance of an unauthorized person or persons into the facility and (2) a significant change in the radiation or contamination levels in the facility or the offsite environment.

g. The following reports should be made:

(1) An annual report to the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, describing the results of the environmental and facility radiation surveys, the status of the facility, and an evaluation of the performance of security and surveillance measures.

(2) An abnormal occurrence report to the Regulatory Operations Regional Office by telephone within 24 hours of discovery of an abnormal occurrence. The abnormal occurrence will also be reported in the annual report described in the preceding item.

h. Records or logs relative to the following items should be kept and retained until the license is terminated, after which they may be stored with other plant records:

(1) Environmental surveys,

(2) Facility radiation surveys,

(3) Inspections of the physical barriers, and

(4) Abnormal occurrences.

4. DECONTAMINATION FOR RELEASE FOR UNRESTRICTED USE

If it is desired to terminate a license and to eliminate any further surveillance requirements, the facility should be sufficiently decontaminated to prevent risk to the public health and safety. After the decontamination is satisfactorily accomplished and the site inspected by the Commission, the Commission may authorize the license to be terminated and the facility abandoned or released for unrestricted use. The licensee should perform the decontamination using the following guidelines:

a. The licensee should make a reasonable effort to eliminate residual contamination.

b. No covering should be applied to radioactive surfaces of equipment or structures by paint, plating, or other covering material until it is known that contamination levels (determined by a survey and documented) are below the limits specified in Table I. In addition, a reasonable effort should be made (and documented) to further minimize contamination prior to any such covering.

c. The radioactivity of the interior surfaces of pipes, drain lines, or ductwork should be determined by making measurements at all traps and other appropriate access points, provided contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement should be assumed to be contaminated in excess of the permissable radiation limits.

d. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated in excess of the limits specified. This may include, but is not limited to, special circumstances such as the transfer of premises to another licensed organization that will continue to work with radioactive materials. Requests for such authorization should provide:

(1) Detailed, specific information describing the premises, equipment, scrap, and radioactive contaminants and the nature, extent, and degree of residual surface contamination.

(2) A detailed health and safety analysis indicating that the residual amounts of materials on surface areas, together with other considerations such as the prospective use of the premises, equipment, or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

e. Prior to release of the premises for unrestricted use, the licensee should make a comprehensive radiation survey establishing that contamination is within the limits specified in Table I. A survey report should be filed with the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, with a copy to the Director of the Regulatory Operations Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report should:

(1) Identify the premises;

(2) Show that reasonable effort has been made to reduce residual contamination to as low as practicable levels;

(3) Describe the scope of the survey and the general procedures followed; and

(4) State the finding of the survey in units specified in Table 1.

After review of the report, the Commission may inspect the facilities to confirm the survey prior to granting approval for abandonment.

5. REACTOR RETIREMENT PROCEDURES

As indicated in Regulatory Position C.2, several alternatives are acceptable for reactor facility retirement. If minor disassembly or "mothballing" is planned, this could be done by the existing operating and maintenance procedures under the license in effect. Any planned actions involving an unreviewed safety question or a change in the technical specifications should be reviewed and approved in accordance with the requirements of 10 CFR §50.59.

If major structural changes to radioactive components of the facility are planned, such as removal of the pressure vessel or major components of the primary system, a dismantlement plan including the information required by §50.82 should be submitted to the Commission. A dismantlement plan should be submitted for all the alternatives of Regulatory Position C.2 except mothballing. However, minor disassembly activities may still be performed in the absence of such a plan, provided they are permitted by existing operating and maintenance procedures. A dismantlement plan should include the following:

a. A description of the ultimate status of the facility

b. A description of the dismantling activities and the precautions to be taken.

c. A safety analysis of the dismantling activities including any effluents which may be released.

d. A safety analysis of the facility in its ultimate status.

Upon satisfactory review and approval of the dismantling plan, a dismantling order is issued by the Commission in accordance with §50.82. When dismantling is completed and the Commission has been notified by letter, the appropriate Regulatory Operations Regional Office inspects the facility and verifies completion in accordance with the dismantlement plan. If residual radiation levels do not exceed the values in Table I, the Commission may terminate the license. If these levels are exceeded, the licensee retains the possession-only license under which the dismantling activities have been conducted or, as an alternative, may make application to the State (if an Agreement State) for a byproduct materials license.

TABLE I

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVERAGE[b c]	MAXIMUM[b d]	REMOVABLE[b e]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm β-γ/100 cm^2	15,000 dpm β-γ/100 cm^2	1000 dpm β-γ/100 cm^2

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

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

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

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

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

Appendix 2

Decommissioning Tasks

The DORF decommissioning tasks are divided into three phases: (1) Reactor fuel removal and shipment, (2) Facility decommissioning, and (3) Preparation of building for alternate use and Post-decommissioning tasks.

Phase I Tasks: Reactor Fuel Removal and Shipment

1. Prepare decommissioning Plan.
2. Obtain ERDA agreement to ship and dispose reactor fuel.
3. Prepare fuel handling Safety Analysis Report (FHSAR).
4. Obtain ARCHS approval of FHSAR.
5. Negotiate rental of licensed shipping casks (either HDL or ERDA).
6. Prepare Appendix A data, "Description of Specification Material and Designation of Processing Batch Size" and "Idaho Chemical Processing Plant Fuel Receipt Criteria", and send to ERDA, Idaho.
7. Remove fuel elements, prepare for shipment.
8. Transportation of fuel elements to ERDA, Idaho or to TRIGA reactor facilities.

Phase II Tasks: Facility Decommissioning

1. Prepare dismantlement plan.
2. Obtain ARCHS approval of dismantlement plan.
3. Establish acceptable surface contamination levels for release of premises for unrestricted use.
4. Prepare specifications for dismantlement contract.
5. Award dismantlement contract.
6. Radioactive equipment and structure removal:

    a. Removal of core support structure
    b. Removal of pool lead shield doors
    c. Pool water discharge to sanitary sewer. (water analysis to assure within 10CFR20 limits)
    d. Removal of aluminum tank liner and activated concrete under the liner
    e. Removal of wood timber lining and concrete in exposure room and plug-door
    f. Removal of lead shield hoist and curtain in exposure room
    g. Activation analysis/dose rate measurements at bottom of pool and in the exposure room to assure acceptable residual contamination levels.
    h. Packaging of radioactive material for shipment to disposal site
    i. Removal of water treatment system and activated sections of exposure room air-conditioning ducts.

7. Transfer reactor console instrumentation and other ancillary equipment to the Armed Forces Radiological Research Institute (AFRRI).

8. Remove jib crane hoist.

9. Remove machine shop equipment and 100 kW emergency generator (to be surplused or transferred to other users).

10. Prepare procedures or discriptions of method for accomplishing the dismantling tasks where applicable.

11. Description of the biological shield and estimates of the content and extent of induced radioactivity.

12. Contamination control:

(1) Dust collection and absolute filtering procedures.
(2) Respirator and protective clothing requirements.
(3) Air Sampling Procedures.
(4) Other Health Physics requirements or procedures.

13. Removal of three 5000 gallon water hold-up tanks.

Phase III Preparation of Building for Alternate Use

1. Post decommissioning radioactivity survey prior to covering/ resurfacing former activated areas.
2. Preparation of building for alternate use:

a. Dismantle concrete parapet to floor level.
b. Fill-in pool hole and resurface to bay floor level.
c. Fill-in lead-shield hoist hole in exposure room and restore floor to level of entrance.
d. Remodel and reactivate air-conditioning system.
e. Remodel electrical distribution and sewer system where applicable.
f. Remove office and laboratory trailer.
g. Remove all other HDL equipment and furniture.
h. Reconnection of sewer lines at hold-up tank area, fill hole and relandscape.

3. Post decommissioning inspection by Independent Agency.

4. Post decommissioning I.G. Inspection and final decommissioning report.

Appendix III

TRIGA Fuel Shipping Cask Analysis

The shipping casks to be used to ship the DORF fuel are the Battelle Memorial Institute BMI-1 cask and another cask not yet determined. The BMI-1 cask will be used to ship the standard length DORF fuel elements. Another cask must be used for the fuel follower control rods (FFCR) and the thermocouple instrumented fuel elements (TCFE) because of the longer overall length.

This appendix contains the BMI-1 certificate of compliance and the cask safety analysis as follows:

1. Certificate of Compliance No. 5957, Revision 1, of the BMI-1 shipping package.

2. Safety Analysis Report for shipment of the TRIGA fuel by the University of Arizona.

3. Supplement No. 1 to request for license to transport irradiated TRIGA fuel in BMI-1 Shipping Cask, June 1972.

4. University of Arizona TRIGA loading-to-critical experiment.

5. Summary of initial Criticality Experiment for Torrey Pines TRIGA Mark III Reactor Startup.

6. Supplement No.1 to request for license to transport irradiated TRIGA fuel in BMI-1 Shipping Cask, June 1973 (replacement pages 12 through 20 to item 3).

7. Analysis of Shells of Revolution Subjected to Symmetrical and Nonsymmetrical Loads.

8. DORF reactor start-up.

APPENDIX IV

ACTIVATION ANALYSIS OF RADIOACTIVE MATERIAL IN THE DORF STRUCTURE BEFORE DECOMMISSIONING

INTRODUCTION

This report documents information of the amount and type of radioactive material that will be present in the structure and building of the Diamond Ordnance Radiation Facility after removal of the reactor fuel in the spring of 1978. Such information is required for decommissioning plans and must be supplied to the Army Reactor Committee for Health and Safety (ARCHS) prior to their approval of such plans. The information is also needed by the waste-disposal area directorate who must budget for specific volumes and radioactive levels. Finally, the isotopic composition of the radioactive waste is necessary for labeling containers at the time of shipment.

The first section of this report is a summary for those who need only the final results on type, location and amount of residual radioactivity. Section two describes the investigative procedures, discusses the possible sources of radioactivity and the properties of the radioactive isotopes found. Graphs of isotopic analyses and calculations, which convert detector response to specific activities, are included in this section. The second section also provides the detailed calculations of volumes, weights and total radioactivity in the various sections of DORF. The final section contains recommendations based on things discovered during this study.

SUMMARY

The radioactivity that will remain at DORF after the fuel removal in the spring of 1978 has been carefully estimated based on criteria, measurements and necessary assumptions documented in this report. A concise summary of that radioactivity is given in Table I. The most predominant radioactive isotopes in the concrete are cobalt-60 and europium-152 and -154. The most predominant isotopes in lead are antimony-124 and silver-110^{m}. The wood and steel (mainly in the lead-shield hoist) are not very radioactive and easy to dispose of. The aluminum itself is almost non-radioactive but there is a radioactive Phenoline

liner which tends to stick to the aluminum. Its radioactivity comes from cobalt-60 and zinc-65. All of these readioactive isotopes have half-lives in excess of 60 days.

TABLE I. Summary[1/] of total radioactivity to be expected from materials in the DORF structure after core removal.

Material	Mass (lbs)	Volume (ft^3)	Radioactivity (millicuries)
CONCRETE	82,170	412	36.24
(If whole plug door included).	(170,050)	(850)	(36.24)
LEAD	55,753	112	13.34
ALUMINUM	2,288	15	75.71
WOOD	34,944	1344	0.33
STEEL	2,662	5.5	0.03
GRAND TOTAL	177,817 lbs (89 tons)	1889 ft^3	0.126 Curies

[1/] This represents a summation of the values given in Table X.

IDENTIFICATION OF THE RADIOACTIVITY

Isotope Identification:

The principal method of identification was gamma-radiation spectroscopy with a germanium lithium-drifted detector, or Ge(Li) crystal. The crystal is housed inside a very low-activity-lead cave lined with wood. Numerous background analyses confirm that for photons with energies greater than 140 keV, samples with low activities (two to three times background) can be successfully analyzed for specific photon energies. A plot of a multichannel analyzer spectrum of the background is given in Fig. 1. The principal higher energy peaks in the background spectrum are the 511-keV gammas associated with annihilation radiation and the 1461-keV peak from ^{40}K, a radioactive isotope which is found naturally in almost all "non-radioactive" materials.

The method of analyses provides for very good resolution of the photon energies in the range 140 keV to 2500 keV at approximately 9.7 keV per channel of 256 total channels. The electronic equipment is sufficiently stable over counting periods of 50,000 seconds to permit energy assignment within two percent. Graphs of the gamma spectra of the various materials investigated are shown. (See Fig. 2 through 5).

The method does not provide for the ultimate in accuracy for determining specific activity. The crystal efficiency (disintegrations per count as a function of energy) can only be accurately assigned for a well-defined geometry. The samples in the present situation varied in size and shape. Therefore, they were suspended above the crystal so that their centers of mass were approximately three centimeters from the active volume of the

detector and efficiencies were determined with calibrated point sources. The error associated with this procedure is estimated to be no greater than 50%, based on a volume integration of point-source response at points in space representative of the sample size. For the task at hand such accuracy is sufficient.

Rational of Sample Selection

The job was to identify the radioactive content and quantity of materials that will have to be removed from the DORF site so that it can be certified, by post-decommissioning radioactive survey, as an unrestricted area for possible public use. This survey, to be conducted by the Army Environmental Health Agency (AEHA), must be accomplished prior to any filling, sealing or burying activities. This presented two problems. How can we identify the radioactivity in presently inaccessible areas, such as below the reactor pool, before the reactor fuel and higher-level radioactive structures have been removed? What amount of material will have to be removed from walls and floors to reach an acceptable AEHA level?

The first problem was attacked as follows. Representative samples of all the material types are accessible in the exposure-room area. Because of the significantly larger thermal-neutron cross sections of materials and the fact that the DORF-TRIGA reactor is zirconium-hydride moderated and water-cooled reactor, the thermal component of the spectrum is the dominate source of induced radioactivity. As will be discussed later, the predominance of radioactive europium confirms this. Therefore, isotopic analyzes of exposure room samples are representative of those in presently inaccessible

areas. Furthermore, with facility dosimetry data for the various locations, we can estimate the residual radioactivity in remote locations with significantly different flux exposure levels.

The second problem of how much material to remove is more complex because we do not have good guidance on the amount of radioactivity in volume that can remain. NRC Regulation 1.86, the current guide, clearly specifies levels for removal surface contamination but is, at best, vague on volume activity and how to detect it. The criteria set for the analysis in this report are as follows:

(1) Once the reactor support structure has been removed there will be no high-level radioactive waste remaining in the DORF structure. Our analyses confirm this.

(2) Based on existing allowable concentrations of radioactive materials in water and a specific activity proportional to material density, we can set an allowable specific activity of 2×10^{-5} microcuries per gram as the maximum permissible concentration of radionuclides in water when it is known that Sr 90, I 129, (I 124, I 126, I 131, Table II only), Pb 210, Ra 226, Ra 228, Cm 248, and Cf 254 are not present.* Since the density of water is one g/cm^3 and there are 28317 cm^3/ft^3, 2×10^{-5} $\mu Ci/g$ corresponds to 0.57 $\mu Ci/ft^3$ of water.

(3) It is assumed that the radioactivity is distributed in the material to be removed in proportion to the incident thermal fluence (flux-time product) and attenuated exponentially according to thermal-neutron relaxation lengths, (i.e., the inverse of microscopic removal cross sections for broad beams). Half-life decay is taken into consideration for the period until spring 1978. Therefore, the depth of material to be removed, D in centimeters, is determined

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* 10 CFR20, note to Appendix B

by relative fluence level at the surface, ϕ/ϕ_o, and

$$D = L \ln \frac{\phi/\phi_o \times A}{0.57 \mu Ci/ft^3} \qquad (1)$$

where A is the activity in $\mu Ci/ft^3$ estimated from this study. The values of relaxation length are given in Table II.

TABLE II. Material densities and relaxation lengths, L

Material	Density	Relaxation Length
Concrete	2.35 g/cm^3	1.6 cm
Lead	11.0 g/cm^3	4.2 cm
Wood	0.42 g/cm^3	2.9 cm

Measured Radioactivity

Samples taken from the DORF exposure room were concrete, wood, aluminum, lead and a tar-paper-like liner installed between the aluminum pool tank and the concrete pool base. Although the aluminum itself has very little residual radioactivity (less than 8×10^{-6} $\mu Ci/gm$ for the sections counted), the Phenoline paper (i.e., the tar-paper liner) has the highest specific activity of all the materials examined. Since this liner tends to stick to the aluminum, for all practical purposes the aluminum tank exhibits this activity.

Tables IV through IX give a breakdown of the isotopic composition of the radioactivity in the various samples. Tables IV and V are composed of several additional pages that serve as detailed examples of the methods of analyses and are self explanatory when reference is made to the graphs of the multichannel-analyzer output. Figs. 1 through 5 are the multichannel-analyzer gamma spectra for the various types of samples. The energy of the photopeaks is related to

TABLE III. GAMMA* SPECIFIC ACTIVITY AND THE NUMBER OF MICROCURIES PER UNIT OF MATERIAL FOR VARIOUS RADIOACTIVE MATERIALS FROM THE DORF EXPOSURE ROOM

	Type of Material	Location in Exposure Room	Specific Activity (d/s•g)	Activity per unit of material
1.	PHENOLINE PAPER	On aluminum tank near exposure room end of pool	636	3.6 μCi/sq ft
2.	CONCRETE	From front part of room about 4 feet from reactor	78	140 $\mu Ci/ft^3$
3.	CONCRETE	Very near reactor at exposure room end of tank	141	252 $\mu Ci/ft^3$
4.	LEAD	From curtain above the movable lead shield	72	0.62 μCi/lb
5.	LEAD	From brick in middle of the movable lead shield	205	1.30 μCi/lb
6.	WOOD	From very near reactor and concrete sample #3, above.	1.3	0.40 $\mu Ci/ft^3$

*From gross beta plus gamma analyses, the beta-to-gamma activity of all these different materials is approximately 1.8.

spectra for the various types of samples. The energy of the photopeaks is related to start the channel number (abcissa) by the following equation:

$$E(keV) = (\text{channel} + 2.5) \times 9.69 \pm 2\% \quad (2)$$

For clarification, the gamma-ray peaks are indentified by isotope and their energies in keV (and in parenthsis) are given for most of the peaks.

The specific activity (d/s·g) for each measured sample is compared in Table III. This table also provides the number of microcuries per unit most practical for that type of material. This latter information is used in Table X to determine the total radioactivity in the volumes of radioactive materials at DORF.

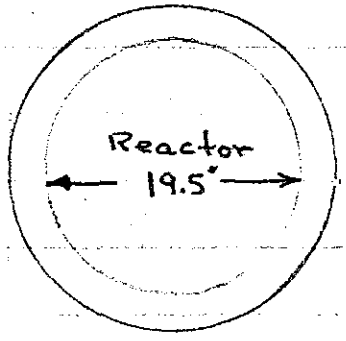
CONCRETE EXCAVATION

The maximum observed concrete-volume activity was 252 μCi/ft³. If we assume that the maximum fluence at any location is 100 times larger than at this monitored exposure room location, Eq. (1) ... the excavation-depth formula ... requires

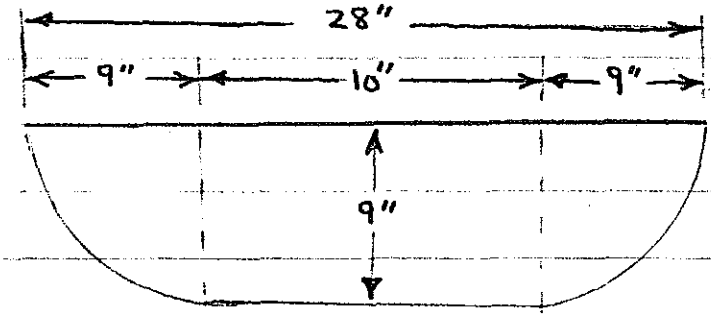
$$D = 1.6 \ln\left(\frac{100 \times 252}{0.57}\right) = 17.1 \text{ cm or } 6.7 \text{ inches}$$

of course, fast neutrons will also penetrate and thermalize but the factor of 100 is already very conservative so excavation depths of 9 to 10 inches in the immediate vicinity of the reactor and at the end positions of the pool should adequately remove the radioactivity.

Excavation below reactor in Pos #3 & radiography position:



Excavation diameter
← 28" →



Volume: $\pi r^2 l + \frac{1}{2}\left(\frac{4}{3}\pi r^3\right)$

$$\pi\left[(10^2)9 + \frac{2}{3}(9^3)\right]$$

$$V = \pi(900 + 486) \div 1728 = \mathbf{2.52\ ft^3}$$

Activity in concrete below Pos. 3: Activity will not exceed measured maximum of 252 μCi/ft³ by more than a factor of two. Assign 500 μCi/ft³.

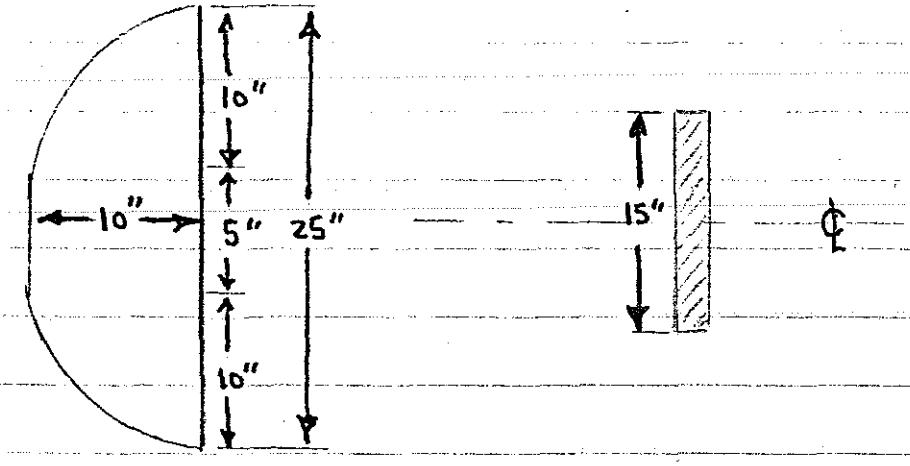
$$2.52 \times 500 = 1260\ \mu Ci$$

Activity in concrete below radiography position: Assign 250 μCi/ft³ (conservative)

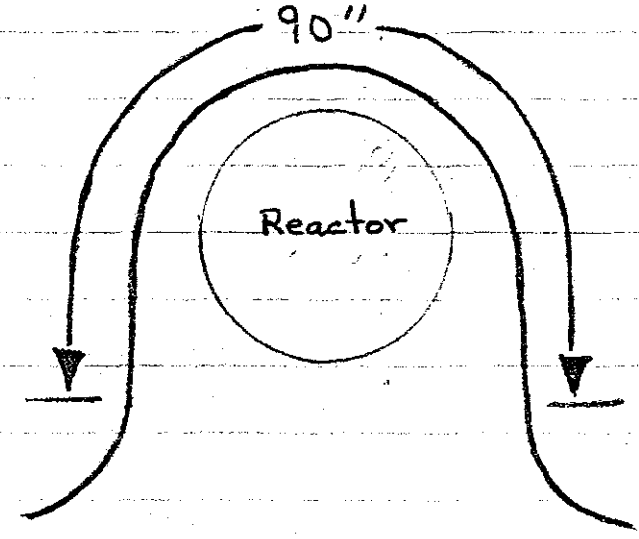
$$2.52 \times 250 = 630\ \mu Ci$$

Excavation from tank-support walls in Pos. #1 & #3:
(active fuel length is 15 inches)

FOR 1 & 3:



Volume for Pos. 3 end of pool:



Volume $L \times (\frac{1}{2}\pi r^2 + t \times d)$

$90\left(\frac{\pi}{2}(10^2) + 5 \times 10\right)$

$V = 90(157 + 50) \div 1728 = 10.8\ ft^3$

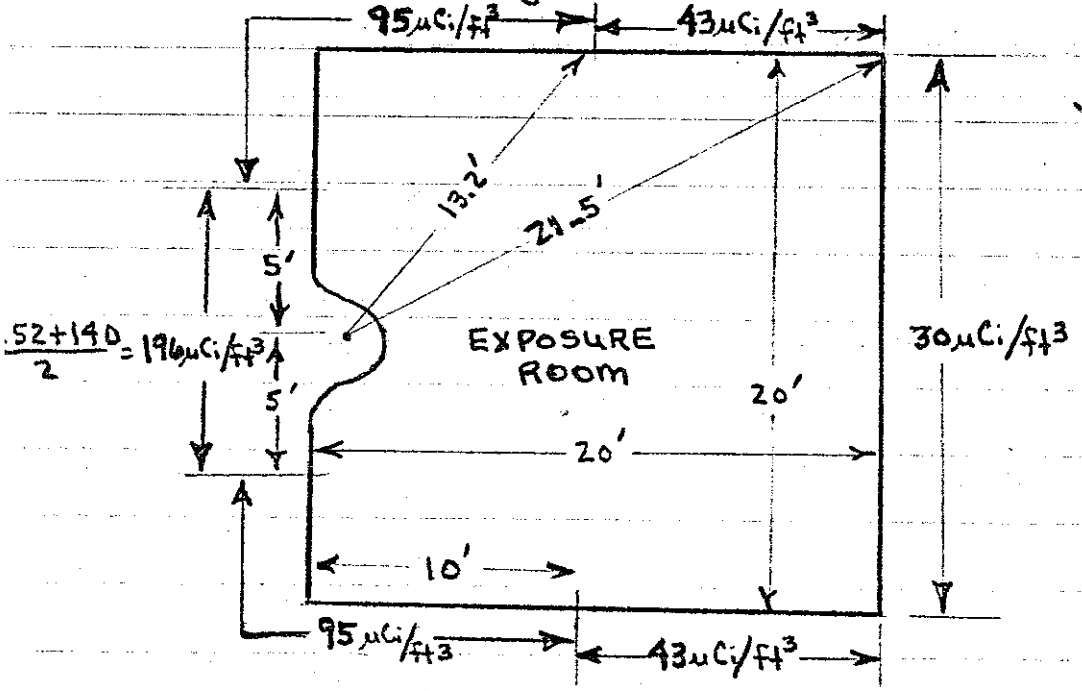
Volume for Pos. 1 end of pool: Because tank projects into the exposure room, L for pos. 1 in taken as 40" rather than 90" and $V = \frac{4}{9}(10.8) = 4.8\ ft^3$

Activity for wall excavation in Pos. 3 & 1: Consistant with conservative estimate for concrete below reactor in Pos. 3, assign $500\ \mu Ci/ft^3$

Activity for Pos. 3 end: $500 \times 10.8 = 5400\ \mu Ci$
Activity for Pos. 1 end: $500 \times 4.8 = 2400\ \mu Ci$

Excavation of concrete from walls, floor and ceiling of the DORF exposure room.

The following diagram shows the estimated volume activity in the concrete of the exposure room walls. Because of thermalization of fast flux by the wood and scatter from the walls, the effective thermal flux is assumed to fall off more nearly as $1/R$ than as $1/R^2$.



room heighth 8 feet.

Excavation depths:

No.	D (inches)	Assign Activity/ft³
1	3.7"	196 μCi/ft³
2	3.2	95 "
3	2.7	43 "
4	2.5	30 "

Volumes of exposure room section 1 thru 4:

No. 1: Area: $8' \times 10' = 80\ ft^2$
Volume: $80 \times 3.7/12 = 24.6\ ft^3$
Activity: $24.6 \times 196 = 4822\ \mu Ci$

No. 2: Wall Area: $2(5' \times 8') + 2(10' \times 8') = 240\ ft^2$
Ceiling/Floor Area: $2(20' \times 10') = 400\ ft^2$
} $640\ ft^2$
Volume: $640 \times 3.2/12 = 171\ ft^3$
Activity: $171 \times 95 = 16245\ \mu Ci$

No. 3: Wall Area: $2(10' \times 8') = 160\ ft^2$
Ceiling/Floor Area: $2(20' \times 10') = 400\ ft^2$
} $560\ ft^2$
Volume: $560 \times 2.7/12 = 126\ ft^3$
Activity: $126 \times 43 = 5418\ \mu Ci$

No. 4: Area: $20' \times 8' = 160\ ft^2$
Volume: $160 \times 2.5/12 = 33\ ft^3$
Activity: $33 \times 30 = 1000\ \mu Ci$

Exposure Room Totals:
Volume: $354.6\ ft^3$
Weight: 70,920 lbs.
Activity: $27.49\ mCi$

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE IV. CONCRETE SAMPLE FROM "FRONT" OF EXPOSURE ROOM

$M = 2.47$ g $T = 54,252$ seconds $(MT = 1.34 \times 10^5)$

ISOTOPE	Half Life $T_{1/2}$	Activity dis/s.g	% of Total Activity
^{60}Co	5.27y	31.66	40.55 %
^{152}Eu-^{154}Eu	12.2y-16y	30.15	38.62 "
^{46}Sc	84 days	5.44	6.97 "
^{134}Cs	2.07 y	5.06	6.48 "
^{65}Zn	245 d	2.29	2.93 "
^{182}Ta	115 d	1.19	1.52 "
^{124}Sb	60 d	0.63	0.81 "
* Annihilation	—	1.65	2.12 "
		TOTAL: 78.07 d/s.g	(100.00)

Nat. Bkg in concrete
(^{226}Ra series)

^{214}Bi 1622y 0.84 d/s.g.

(i.e., concrete is 93 times its own natural background)

Activity per cubic foot of concrete

$(12 \times 2.54)^3 = 28317$ cm^3/ft^3

ρ concrete $= 2.35$ g/cm^3

6.65×10^4 g/ft^3

$$\text{Activity} = \frac{6.65 \times 10^4 \times 78.07}{3.7 \times 10^{10}\ \text{d/s·Ci}} = 1.40 \times 10^{-9}\ \text{Ci} \text{ or } \boxed{140\ \mu\text{Ci/ft}^3}$$

Estimate per "ml" of H_2O

$(78.07) / 3.7 \times 10^{10} \times 2.35 = 8.98 \times 10^{-10}$ or 8.98×10^{-4} μCi/[illegible]g

* Note: All activities except annihilation radiation (511 kev) were determined from sample with background substracted.

TABLE IV-A CONCRETE SAMPLE FROM FRONT OF EXPOSURE ROOM

WEIGHT = 2.47 g (BKG. SUBSTRACTED FOR EQUAL TIME: 54,252 seconds)

* Indicated more that one isotope has a peak in this channel 9.71±0.05 kev/ch+2.5

SOURCE ISOTOPE & $T_{1/2}$	keV ENERGY	CHANNEL	PEAK GROSS COUNTS	COMPTON BKG COUNTS	PEAK NET COUNTS	ε c/d	Total	
^{152}Eu – ^{154}Eu (12.2y – 16year)	122 ^{152} ^{154}	10	23500	3364	20136	2.50×10^{-2}	805440	
	245 & 248 ^{154} ^{152}	23	4889	2030	2859	1.16×10^{-2}	246465	
	344 ^{152}	33	8441	1756	6685	8.10×10^{-3}	825308	
	411 ^{154} ^{152}	40	1894	1370	524	6.66×10^{-3}	78797	
	488 ^{152}	47.8	1658	1340	318	5.58×10^{-3}	56989	
	689 ^{154}	68.8	1366	1230	136	3.90×10^{-3}	34872	
	720 & 723.1 ^{154}	72 *	1357	1240	117	3.63×10^{-3}	32231	
	779 ^{152}	78	2562	1250	1313	3.35×10^{-3}	391940	
	867 & 873 ^{154}	87	1788	1325	463	2.95×10^{-3}	156949	
	966 ^{152}	97	2457	1150	1307	2.66×10^{-3}	491353	
	1006 ^{154}	101	1130	940	190	2.57×10^{-3}	73930	
	~~1092 ^{152}~~	~~110~~	~~1728~~	~~975~~	~~753~~	~~2.45×10^{-3}~~	~~307347~~	delete because of Zinc
	1277 ^{154}	129	321	135	186	1.95×10^{-3}	95385	
	1412 ^{152}	143	1220	50	1170	1.74×10^{-3}	672913	
	1457 ^{152}	146.7	73	32	41	1.67×10^{-3}	24551	
	1528 ^{152}	155	59	18	41	1.60×10^{-3}	25625	
	1595 ^{154}	162.5	60	17	43	1.52×10^{-3}	28289	
						TOTAL =	4.040×10^{6}	
^{60}Co (5.27 years)	1173.1	119	4857	589	4268	2.11×10^{-3}	2.023×10^{6}	
	1332.4	135	4187	126	4061	1.86×10^{-3}	2.219×10^{6}	
						TOTAL =	4.242×10^{6}	
^{134}Cs (2.07 year)	563 & 569	56	1580	1300	280	4.72×10^{-3}	5.932×10^{4}	
	605	60 *	1612	1293	319	4.36×10^{-3}	7.317 "	
	795 & 802	80	2101	1300	801	3.25×10^{-3}	24.646 "	
	1040	105 *	870	840	30	2.50×10^{-3}	1.200 "	
	1168	119 *	(370)	120	250	2.15×10^{-3}	11.628 "	
	1365	139	370	60	310	1.81×10^{-3}	17.127 "	
						TOTAL =	6.785×10^{5}	
^{65}Zn (245 day)	1115.4	110 *	1728	975	753	2.45×10^{-3}	307347	
						TOTAL	3.073×10^{5}	

TABLE IV-A. (continued)

ISOTOPE	kev ENERGY	channel	Gross	Bks	Net	ε c/d	Total
^{46}Sc (84 days)	889.4	89 *	2491	1400	1091	2.70×10^{-3}	3.762×10^{5}
	1120.3	113	2330	(1550)	780	2.21×10^{-3}	3.529×10^{5}
						TOTAL	7.291×10^{5}
^{124}Sb (60day)	602.6	60 *			200	4.40×10^{-3}	45.451×10^{3}
	646	65 *			25	4.17×10^{-3}	5.995 "
	722.8	72 *			25	3.63×10^{-3}	6.887 "
	968	97 *	1120	1100	20	2.66×10^{-3}	7.518 "
	1045	105 *	870	853	17	2.50×10^{-3}	6.800 "
	1691	172	41	28	13	1.43×10^{-3}	9.091 "
	2088	209	25	22	3	1.15×10^{-3}	2.609 "
						TOTAL =	8.435×10^{4}
^{182}Ta (115 day)	100.3	8	4577	4102	475	3.10×10^{-2}	15.322×10^{3}
	152.9&156	13.6	3054	2700	354	1.42×10^{-2}	24.930 "
	198.4	17.5 *	2620	2450	170	1.50×10^{-2}	11.333 "
	223	20.5	2348	2210	138	1.35×10^{-2}	10.222 "
	1121	113 *	870	810	60	2.21×10^{-3}	27.149 "
	1189	119	(710)	650	60	2.09×10^{-3}	28.708 "
	1221.6	123	337	252	85	2.05×10^{-3}	41.463 "
						TOTAL =	1.591×10^{5}
nihilation	511	50	4100	3100	1,000	4.50×10^{-3}	2.22×10^{5}
						TOTAL	2.22×10^{5}

Additional analysis of "natural background in concrete:

^{214}Bi (1622 years*)

* from ^{226}Ra series

1731	177	36	27	9	1.39×10^{-3}	6.475×10^{3}
1767	180	39	29	10	1.24×10^{-3}	8.064 "
2119	217	23	14	9	1.13×10^{-3}	7.965 "
2204	226	32	23	9	1.08×10^{-3}	8.333 "
2293	234	28	12	16	1.03×10^{-3}	15.539 "
2497	250	36	25	11	9.40×10^{-4}	11.702 "
					Actual about 2x this →	TOTAL 5.807×10^{4}

because can see four energies

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE V. TAR-PAPER* SAMPLE FROM FRONT OF EXPOSURE ROOM

* CALLED PHENOLINE COATING

M = 3.69 g T = 20,183 s (MT = 7.45×10^4)

ISOTOPE	HALF-LIFE $T_{1/2}$	Activity dis/s.g	% of Total Activity
^{65}Zn	245 day	185.2	29.1 %
^{60}Co	5.27 year	363.8	57.2 %
^{152}Eu	12.2 year	86.8	13.7 %
	TOTAL:	635.8 d/s.g	100.0 %

From gross counts, this sample was 423 ÷ 11 = 38.5 times more active than the natural background in the Ge(Li) crystal cave. NOTE: It is 635.8/78.1 = 8.1 times more radioactive than the concrete from the front part of the room.

Activity per square foot of paper:
(assuming a 1/8-inch thickness)

$(12 \times 2.54)^2 \times 0.125 \times 2.54 = 295\ cm^3$ per square foot

$\rho = 0.7\ g/cm^3$

Activity $\frac{635.8\ d/s \cdot g \times 206.5\ g}{3.7 \times 10^{10}\ d/s \cdot Ci} =$ **3.55 μCi/sq. foot**

TABLE IV-A. TAR-PAPER SAMPLE FROM POS #1 END OF ALUMINUM TANK (IN EXPOSURE ROOM).

TIME = 20,183 sec.

M = 3.6875 grams

423 sample/10 sec
11 like

22183 000004 032000

	1	2	3	4	5	6	7	8	9
000000									
022183	000004	065213	016877	013805	014030	014273	013571	013285	013616
000010									
048361	011699	011727	011230	011878	010214	010054	009864	009489	009693
000020									
009953	009577	009525	013469	008729	008657	008524	008433	008402	008066
000030									
008237	008377	007927	019884	007346	007546	007144	006983	006859	006789
000040									
007690	006823	006536	007492	006603	006469	006496	006540	006508	006670
000050									
011253	006750	006323	006526	006497	006456	007013	006515	006682	006515
000060									
007831	006514	006633	006741	006702	006907	006732	006904	007032	006989
000070									
006667	006954	007301	007106	007305	007392	007381	007520	009923	007635
000080									
008622	008056	008383	008368	008992	008968	009128	010116	009901	012019
000090									
010301	009768	008668	008394	008263	008354	008177	009248	006260	005804
000100									
005583	005638	004958	004642	004680	004475	004422	004334	004357	004701
000110									
005809	004901	009792	037642	002790	002421	002438	002329	007215	028249
000120									
001547	001386	001187	001140	000822	000787	000644	000546	000530	000822
000130									
000530	000744	000621	000538	000840	027671	000259	000203	000178	000189
000140									
000194	000178	000271	000253	000157	000170	000186	000296	000244	000122
000150									
000139	000137	000141	000132	000135	000193	000148	000144	000137	000121
000160									
000126	000137	000143	000130	000094	000106	000124	000107	000116	000104
000170									
000114	000116	000092	000124	000107	000099	000118	000084	000094	000087
000180									
000094	000080	000089	000100	000093	000115	000099	000102	000088	000080
000190									
000082	000107	000089	000090	000090	000090	000095	000092	000078	000081
000200									
000093	000089	000093	000095	000085	000116	000081	000082	000079	000068
000210									
000090	000064	000065	000062	000069	000070	000053	000076	000058	000074
000220									
000066	000064	000068	000052	000064	000070	000059	000076	000054	000062
000230									
000072	000062	000049	000079	000056	000047	000041	000045	000043	000034
000240									
000033	000027	000023	000019	000027	000016	000012	000013	000012	000016
000250									
000013	000006	000009	000011	000016	000019				

TABLE V-B.

ISOTOPIC ANALYSES OF TAR-PAPER SAMPLE FROM ON TANK WALL AT EXPOSURE-ROOM END AT APPROXIMATELY CORE CENTERLINE.

COUNT TIME = 20,183 sec.; WEIGHT = 3.6875 g

SOURCE ISOTOPE / $T_{1/2}$	ENERGY* (keV)	CHANNEL #	PEAK GROSS COUNTS	COMPTON BKG. COUNTS	PEAK NET COUNTS	XTAL EFF. (c/d)	TOTAL d
^{65}Zn (245 day)	1115.4	113	37642	3700	33942	2.45×10^{-3}	1.38×10^{7}
					TOTAL =		(1.38×10^{7})
^{60}Co (5.27 year)	1173	119	28249	1700	26549	2.11×10^{-3}	1.26×10^{7}
	1332	135	27671	750	26921	1.86×10^{-3}	1.45×10^{7}
					TOTAL =		(2.71×10^{7})
^{152}Eu (12.2 year)	121.8	10	48361	12600	35761	2.50×10^{-2}	1.43×10^{6}
	244.6	23	13469	9140	4329	1.16×10^{-2}	0.30 "
	344.2	33	19884	7700	12184	8.10×10^{-3}	1.50 "
	411.0	40	7690	6790	900	6.65×10^{-3}	0.14 "
	443.9	43	7492	6500	992	6.60×10^{-3}	0.15 "
	778.6	78	9932	7550	2382	3.35×10^{-3}	0.71 "
	964.1	97	9248	7206	2048	2.66×10^{-3}	0.77 "
	1086.0	110	5809	4800	1009	2.45×10^{-3}	0.41 "
	1407.5	143	2053	210	1843	1.74×10^{-3}	1.06 "
					TOTAL =		(6.97×10^{6})

* Energy is approximately (±1.5%) 9.69 x (CHANNEL + 2.4) keV/channel

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VI. CONCRETE SAMPLES FROM UNDER TANK IN EXPOSURE ROOM

M = 11.63 g T = 40,000 sec (MT = 4.65×10^5)

ISOTOPE	HALF-LIFE $T_{1/2}$	Activity dis/s·g	% of Total Activity
^{60}Co	5.27 y	63.2	44.5 %
^{152}Eu-^{154}Eu	12.2y – 16y	55.4	39.0 "
^{46}Sc	84 day	10.5	7.4 "
^{65}Zn	245 day	4.0	2.8 "
^{182}Ta	115 day	3.1	2.2 "
^{134}Cs	2.07 y	1.3	0.9 "
^{124}Sb	60 day	0.2	1.2 "
Annihilation	—	2.8	2.0 "
	TOTAL	140.5 d/s·g	(100.0)

Note: This concrete sample is 1.8 times more radioactive than the sample from the front of the room.

Activity per cubic foot of concrete:

Density: 6.65×10^4 g/ft^3

$$\text{Activity} = \frac{6.65 \times 10^4 \times 140.5}{3.7 \times 10^{10}\ \text{d/s·Ci}} = 2.52 \times 10^{-4}\ \text{Ci}$$ or **252 μCi/ft^3**

Estimate per "mL" of H_2O: 1.62×10^{-3} μCi/g

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VII. LEAD FROM BRICK IN MOVABLE LEAD SHIELD

$M = 2.531$ g $T = 40{,}000$ sec $(MT = 1.01 \times 10^5)$

ISOTOPE	HALF-LIFE ($T_{1/2}$)	ACTIVITY d/s.g	% of TOTAL ACTIVITY
^{124}Sb	60 days	181.8	71.0%
^{110m}Ag	253 day	67.3	26.3%
^{152}Eu-^{154}Eu	12.2y - 16y	1.5	0.6%
Annihilation	—	5.4	2.1%
	TOTAL	256.0 d/s.g	(100.0%)

Activity per pound of lead:
454 g/lb

$$\text{Activity} = \frac{454 \times 256}{3.7 \times 10^{10}} = 3.14 \times 10^{-6} \text{ or } 3.14\ \mu\text{Ci/lb}$$

NOTE: About 44% of this radioactivity will decay by spring 1978 because of the ^{124}Sb contribution. Therefore Eff. Activity = 1.76 μCi/lb.
Futhermore, the top and bottom quarter sections of the lead shield should have activities more like that of the lead curtain (Eff. Act = 0.62 μCi/lb) so a better value for the whole shield is $0.6 \times 1.76 + 0.4 \times 0.62 =$ **1.30 μCi/lb**

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VIII. LEAD FROM CURTAIN

M = 0.602 T = 40,000 sec ($MT = 2.41 \times 10^4$)

ISOTOPE	HALF-LIFE $T_{1/2}$	ACTIVITY (d/s.g)	% of TOTAL ACTIVITY
^{124}Sb	60 days	4.96	62.1%
^{110m}Ag	253 days	2.28	28.5%
^{46}Sc	84 days	0.45	5.6%
^{152}Eu-^{154}Eu	12.2y-16y	0.05	0.7%
Annihilation	———	0.25	3.1%
		7.99 d/s.g	(100.0)

Activity per pound of lead:
(454 g/lb)

$$\text{Activity} : \frac{454 \times 7.99}{3.7 \times 10^{10}} = 0.98 \times 10^{-7} = 1.0\,\mu Ci/lb$$

NOTE: About 38% of this activity will have decayed by spring 1978.
Therefore: Eff Activity = **0.62 μCi/lb**

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE IX. WOOD FROM FRONT OF EXPOSURE ROOM

$m = 36.58$ g $T = 8967$ sec. $(mT = 3.28 \times 10^5)$

ISOTOPE	HALF-LIFE $T_{1/2}$	ACTIVITY d/s.g	% of TOTAL ACTIVITY
^{152}Eu-^{154}Eu	12.2y-16y	0.588	46.7 %
^{60}Co	5.27 y	0.534	42.4 %
^{46}Sc	84 day	0.055	4.4 "
^{129m}Te	41 day	0.040	3.2 "
^{127m}Te	110 day	0.026	2.1 "
Annihilation	——	0.017	1.2 "
	TOTAL	1.26 d/s.g	(100.0)

Activity per cubic foot of wood

$28317\ cm^3/ft^3 \times 0.42\ g/cm^3 = 1.189 \times 10^4\ g/ft^3$

$$\text{Activity} = \frac{1.26 \times 1.189 \times 10^4}{3.7 \times 10^{10}} = 4.04 \times 10^{-7} \text{ or } \boxed{0.40\ \mu Ci/ft^3}$$

Less than 10% of this activity will decay by spring 1978.

NOTE: From WRAMC Health Physics survey of this sample we find there is a β/γ ratio of approximately two.

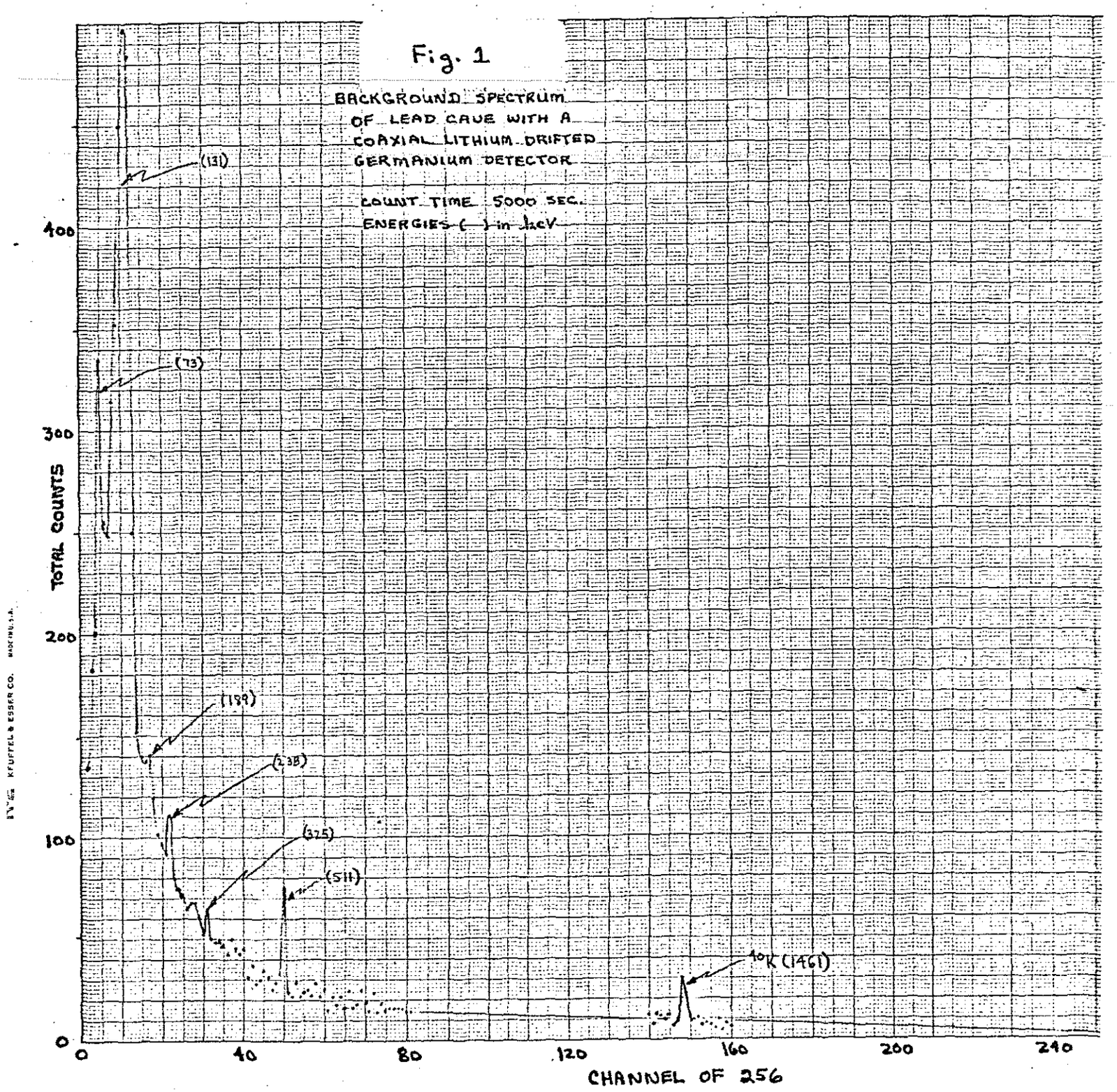


Fig. 1
BACKGROUND SPECTRUM OF LEAD CAVE WITH A COAXIAL LITHIUM DRIFTED GERMANIUM DETECTOR
COUNT TIME 5000 SEC.
ENERGIES ( ) in keV
TOTAL COUNTS
CHANNEL OF 256
0
100
200
300
400
0
40
80
120
160
200
240
(131)
(73)
(189)
(238)
(325)
(511)
40K (1461)

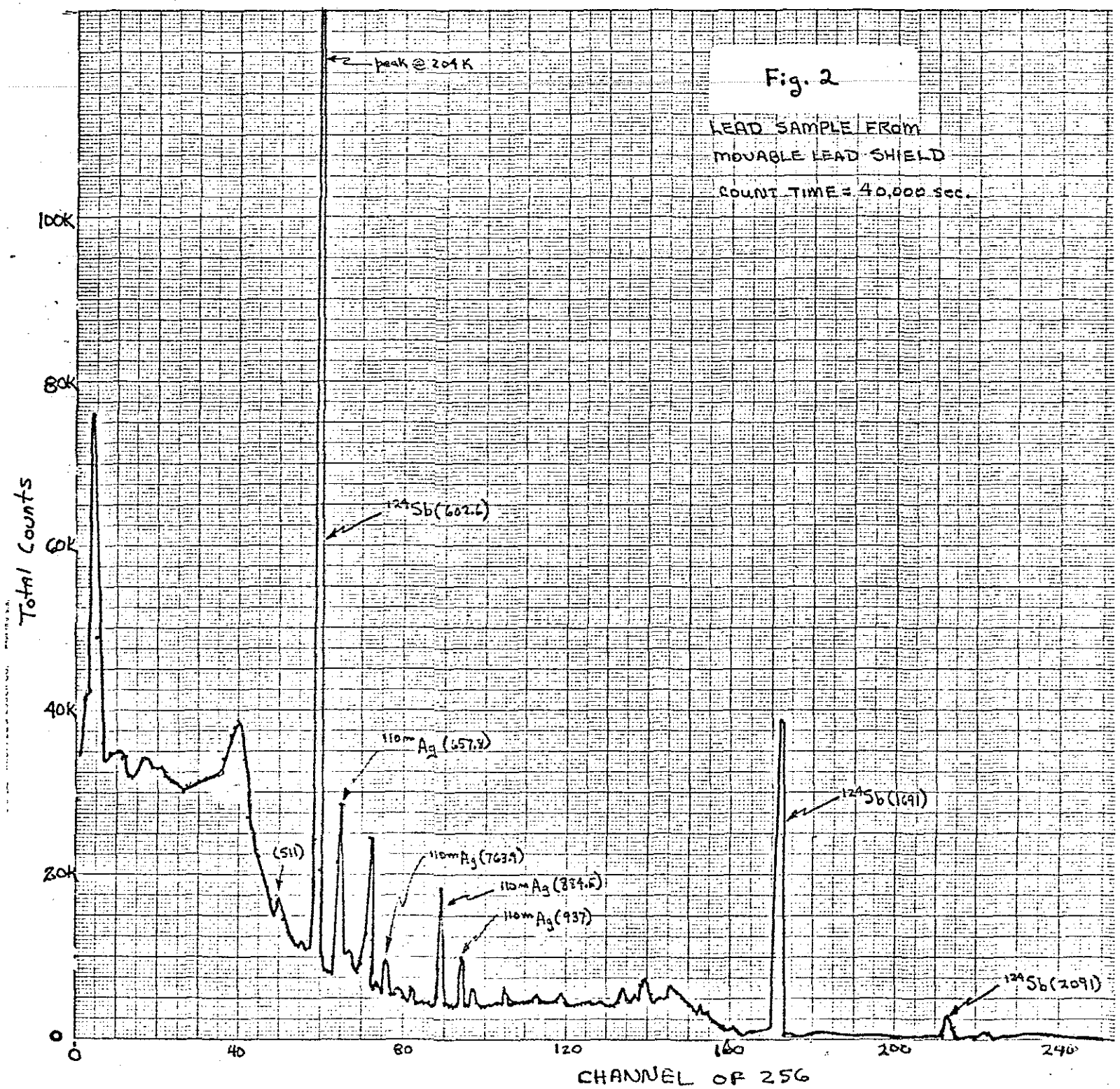
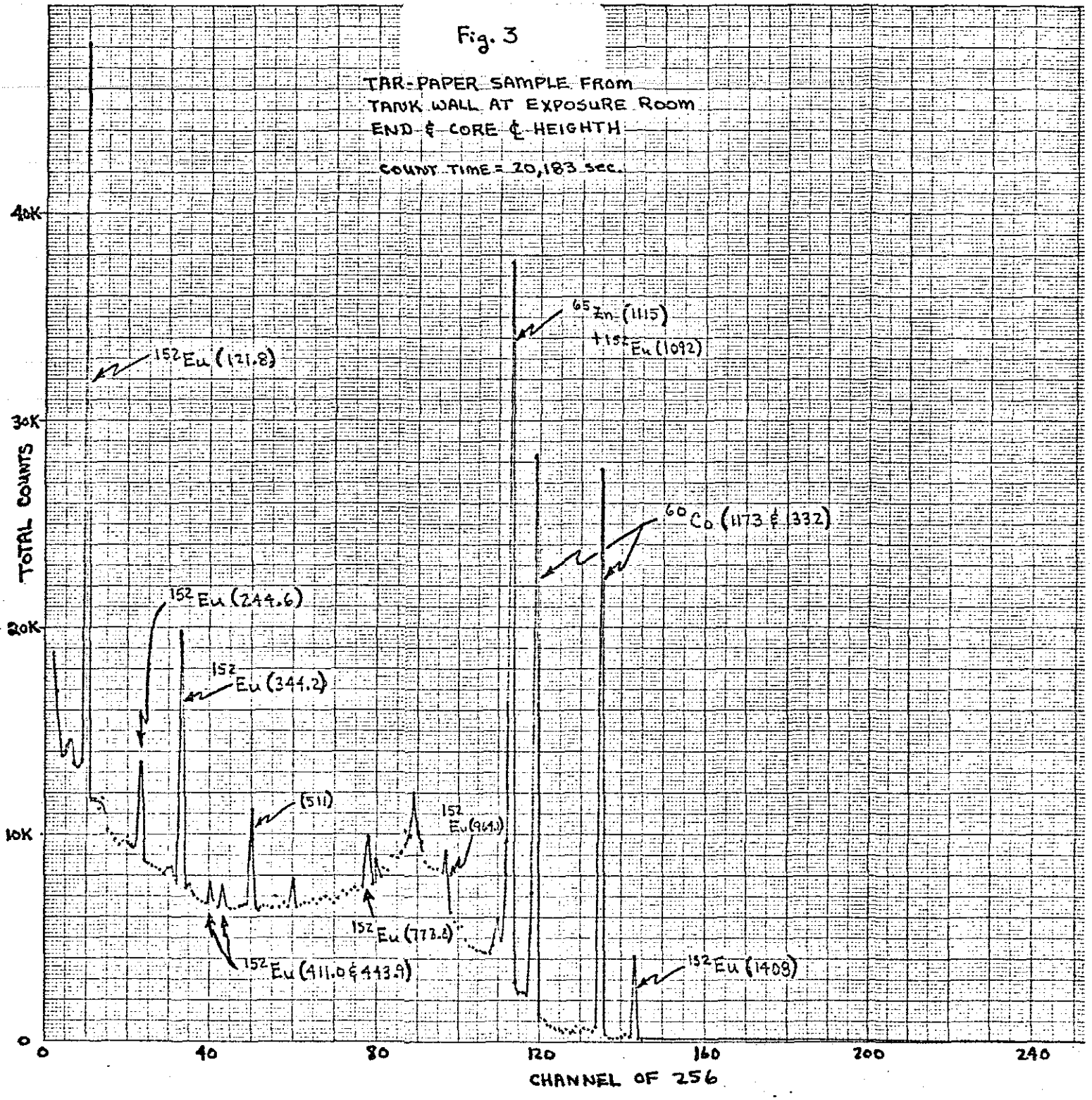

peak @ 209 K
Fig. 2
LEAD SAMPLE FROM
MOVABLE LEAD SHIELD
COUNT TIME = 40,000 sec.
100K
80K
60K
40K
20K
0
Total Counts
0
40
80
120
160
200
240
CHANNEL OF 256
124Sb (602.2)
110m Ag (657.8)
(511)
110m Ag (763.9)
110m Ag (884.6)
110m Ag (937)
124Sb (1691)
124Sb (2091)

Fig. 3

TAR-PAPER SAMPLE FROM TANK WALL AT EXPOSURE ROOM END ℄ CORE ℄ HEIGHTH

COUNT TIME = 20,183 SEC.



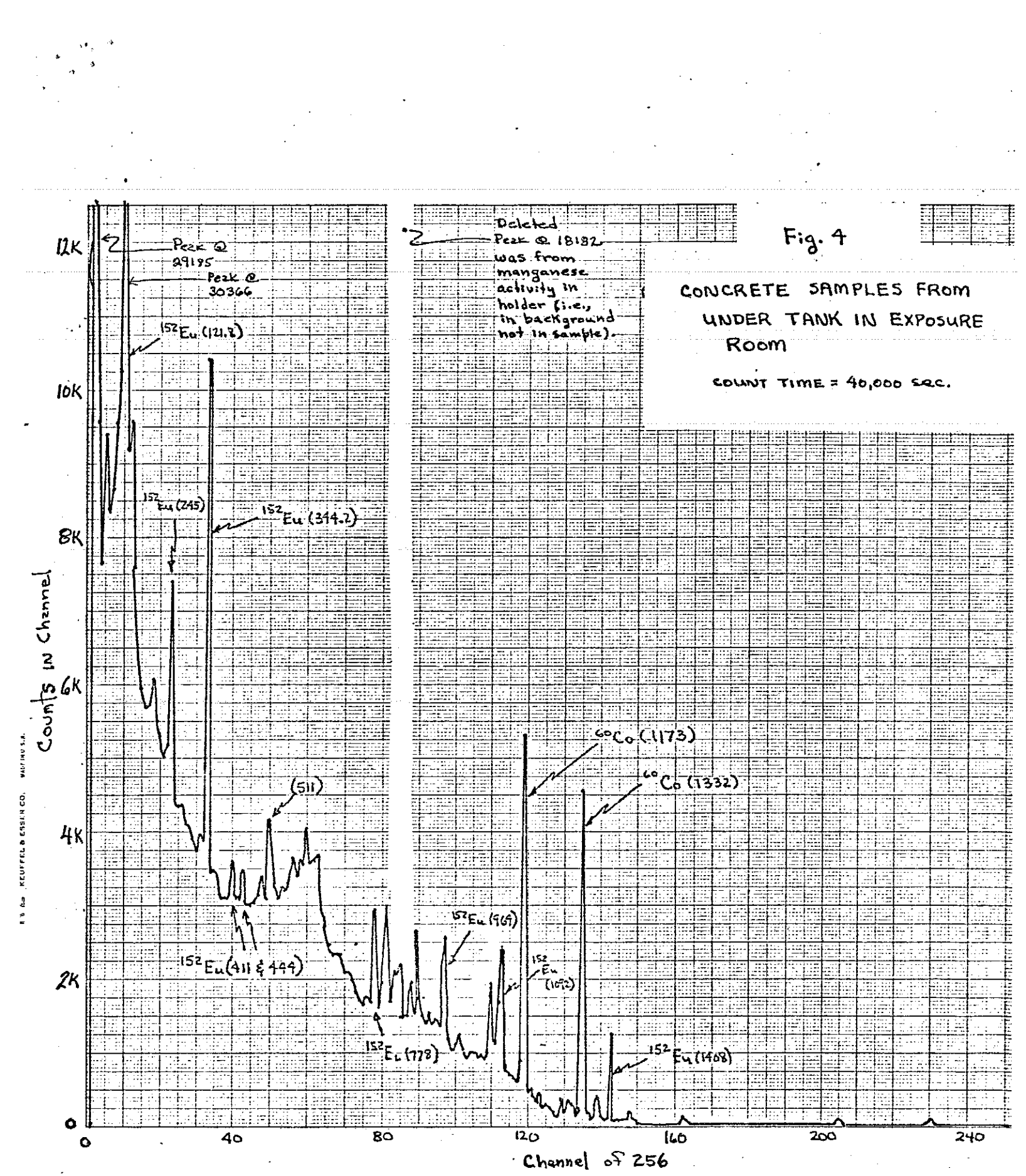


Fig. 4
CONCRETE SAMPLES FROM UNDER TANK IN EXPOSURE ROOM
COUNT TIME = 40,000 sec.
Deleted Peak @ 18182 was from manganese activity in holder (i.e., in background not in sample).
Peak @ 29195
Peak @ 30366
152Eu (121.8)
152Eu (245)
152Eu (344.2)
(511)
152Eu (411 & 444)
152Eu (778)
152Eu (969)
152Eu (1092)
60Co (1173)
60Co (1332)
152Eu (1408)
Counts in Channel
0
2K
4K
6K
8K
10K
12K
0
40
80
120
160
200
240
Channel of 256

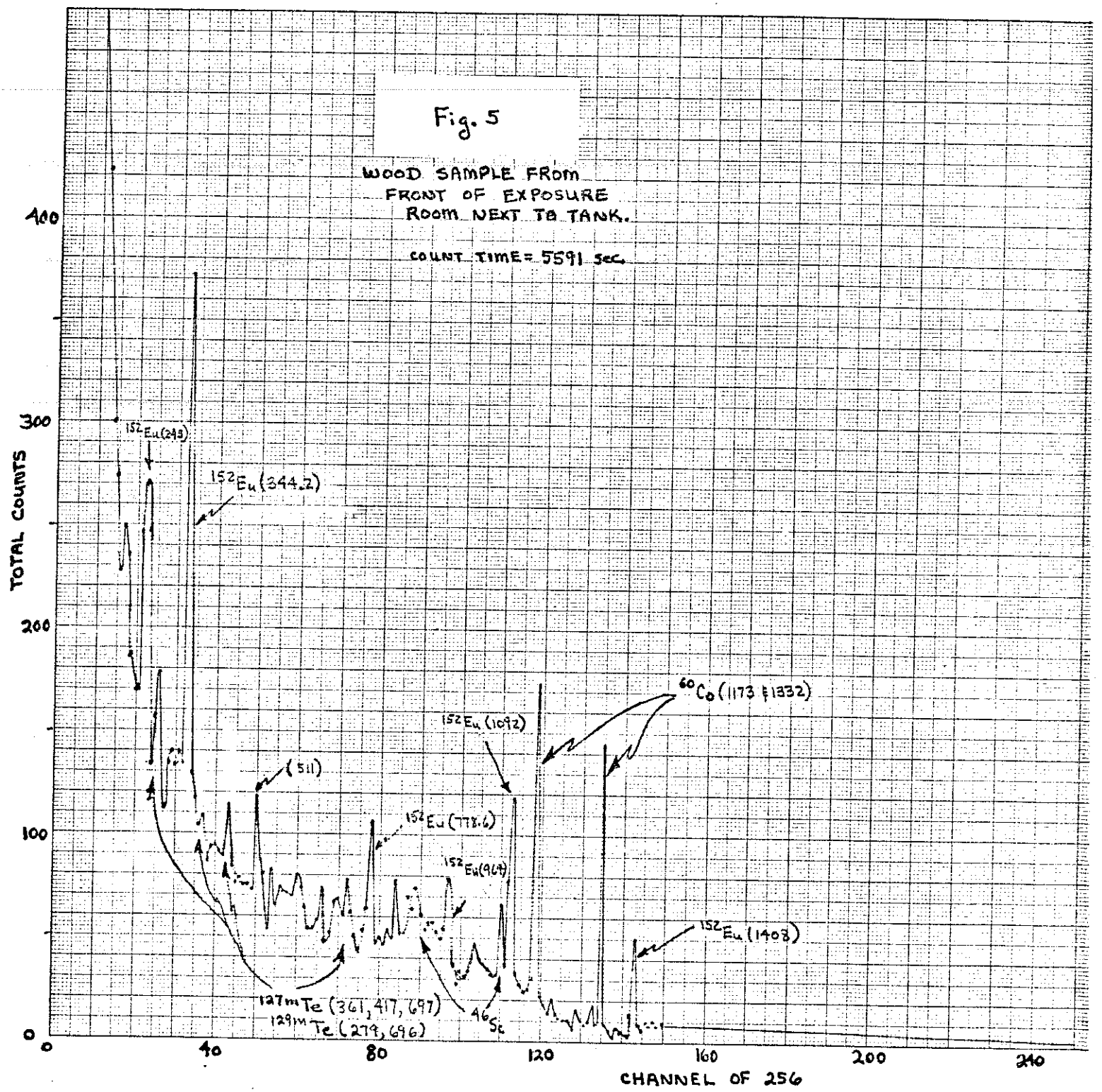


Fig. 5
WOOD SAMPLE FROM FRONT OF EXPOSURE ROOM NEXT TO TANK.
COUNT TIME = 5591 sec.
TOTAL COUNTS
CHANNEL OF 256
0
100
200
300
400
40
80
120
160
200
240
152Eu (248)
152Eu (344.2)
(511)
152Eu (778.6)
152Eu (1092)
152Eu (964)
60Co (1173 & 1332)
152Eu (1408)
127mTe (361, 417, 697)
129mTe (279, 696)
46Sc

CONCLUSIONS & RECOMMENDATIONS

The principal conclusion from this study is that once the reactor-grid support structure has been removed there is very little radioactivity remaining at DORF. Unfortunately the levels are definitely above background, but only by factors of several hundred, and the radioactivity is mainly distributed throughout concrete walls and floors. Deep excavations will not be necessary. However, this is of little consequence if one still has to remove several inch-thick layers from large areas. This is the situation in the exposure room. In fact, the exposure-room decontamination is by far the major problem and several possible methods of attack come to mind.

(1) Excavate and remove the three 5000-gallon waste-water holding tanks, cut off part of the tops and use them as shipping containers for the radioactive debris from DORF. For example, the wood has suffered radiation damage and dry rot so that it crumbles rather easily. It is a big volume (1200 ft^3) but relatively light in weight so it can easily be tossed or shoveled into the tanks and they could then be closure welded for shipment. There will also be much dust, dirt, paper and small concrete chips of radioactive waste, all of which could be put into the tanks.

(2) Mechanically cut, DO NOT CUT WITH A TORCH, the aluminum because of the radioactive "tar-paper" liner which could easily catch on fire and produce contaminated smoke. However by reference to the excavation-of-concrete details in this report, the places where the aluminum liner will be radioactive are easily identified. It does not appear that the liner will produce a problem in other than these areas.

(3) Thought should be given to the possibility of transferring some of the lead to AFRRI or APRF because its radioactivity is really not a serious hazard and these facilities need it for shielding in neutron fields. This could save a few dollars on transportation and disposal costs.

(4) Survey activities are going to be a problem because there just isn't much activity to survey right now. For example, depending on what is going to be done with the exposure room, it may not be necessary to excavate concrete from the rear wall of the room. In any event, thought should be given to how much the survey reading "from the rear wall only", before excavation, must be decreased by material removal to provide an "acceptable" survey level. In view of the expense to breakup and ship concrete, it is prudent to be practical about sealing up or burying very small, but detectable, amounts of radioactivity.

(5) Almost all of the materials exhibit one or two predominant and characteristic photopeaks. Therefore, survey activities could be determined by a sodium-iodide scintillation detector. It is suggested that a portable detector with a 3/4-inch-thick cylindrical lead shield around the sides would be practical. Calibration could be accomplished in a crude, but adequate, manner by measuring the response of a variety of sources simultaneously positioned over a square-meter plane area behind about 1/4-inch thick aluminum. This approximates the following situation. The dose rate to tissue in rads per hour in an infinite medium, of density ρ, uniformly contaminated by a gamma emitter, of energy E (MeV), is

$$2.12\ EC/\rho \qquad (3)$$

where C is in microcuries per cm^3. At the surface, the dose rate is about one half of this and for air a one-centimeter-from-the-surface survey is an adequate representation of the surface rate. By then surveying the "calibration setup at one meter" and correcting for $1/R^2$ to one centimeter, one can estimate the rads per hour efficiency of the scintillation detector. A variety of sources, respositioned should be used and the results averaged.