SEC Petition Evaluation Report
Petition SEC-00224

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Report Submittal Date:	February 18, 2016
Subject Expert(s):	Mitch Findley, Brian Gleckler, Jason Davis, Mike Mahathy
Site Expert(s):	N/A

Petition Administrative Summary
Petition Under Evaluation

Petition Number:	SEC-00224
Petition Type:	83.13
Petition Receipt Date:	December 4, 2014
Qualification Date:	March 13, 2015
DOE/AWE Facility Name:	Argonne National Laboratory-West, Scoville, ID

Petition Class

Petitioner-Requested Class Definition:	All workers who worked in any work location at the Argonne National Laboratory-West from January 1, 1949 through December 31, 1995, with the basis of inadequate monitoring for plutonium, neptunium and fission products during that time period.
Class Evaluated by NIOSH:	All workers who worked in any location of the Argonne National Laboratory-West from April 10, 1951 through December 31, 1979.
NIOSH-Proposed Class(es) to be Added to the SEC:	All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Argonne National Laboratory-West between April 10, 1951 and December 31, 1957 for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort.

Related Petition Summary Information

SEC Petition Tracking Number(s):	N/A
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Report Title:	N/A
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Evaluation Report Summary: SEC-00224, ANL-W

This evaluation report by the National Institute for Occupational Safety and Health (NIOSH) addresses a class of employees proposed for addition to the Special Exposure Cohort (SEC) per the *Energy Employees Occupational Illness Compensation Program Act of 2000*, as amended, 42 U.S.C. § 7384 *et seq.* (EEOICPA) and 42 C.F.R. pt. 83, *Procedures for Designating Classes of Employees as Members of the Special Exposure Cohort under the Energy Employees Occupational Illness Compensation Program Act of 2000*.

Petitioner-Requested Class Definition

Petition SEC-00224 was received on December 4, 2014, and qualified on March 13, 2015. The petitioner requested that NIOSH consider the following class: *All workers who worked in any work location at the Argonne National Laboratory-West from January 1, 1949 through December 31, 1995, with the basis of inadequate monitoring for plutonium, neptunium and fission products during that time period.*

Class Evaluated by NIOSH

Based on its preliminary research, NIOSH modified the petitioner-requested class. NIOSH evaluated the following class: All workers who worked in any location of the Argonne National Laboratory-West from April 10, 1951 through December 31, 1979. The petitioner-proposed class was modified (see Section 3.0 below). The start date was modified because the first reactor facility was completed, and its associated reactor assembly commenced, on April 10, 1951. The end date was modified because abundant plutonium bioassay data has been identified beginning in 1980.

NIOSH-Proposed Class(es) to be Added to the SEC

Based on its full research of the class under evaluation, NIOSH has defined a single class of employees for which NIOSH cannot estimate radiation doses with sufficient accuracy. The NIOSH-proposed class includes all employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Argonne National Laboratory-West between April 10, 1951 and December 31, 1957 for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort. The class under evaluation was modified (see Section 3.0 below) because there are sufficient routine internal and external exposure data beginning on January 1, 1958 to complete dose reconstructions with sufficient accuracy.

Feasibility of Dose Reconstruction

NIOSH finds it is not feasible to estimate internal and external exposures with sufficient accuracy for all workers at the ANL-W site from April 10, 1951 through December 31, 1957 due to insufficient data available during this time period. With the exception of this class, per EEOICPA and 42 C.F.R. § 83.13(c)(1), NIOSH has established that it has access to sufficient information to: (1) estimate the maximum radiation dose, for every type of cancer for which radiation doses are reconstructed, that could have been incurred in plausible circumstances by any member of the class; or (2) estimate

radiation doses more precisely than an estimate of maximum dose. Information available from the site profile and additional resources is sufficient to document or estimate the maximum internal and external potential exposure to members of the evaluated class under plausible circumstances beginning January 1, 1958.

The NIOSH dose reconstruction feasibility findings are based on the following:

- The Argonne National Laboratory-West site, originally known as the Idaho Division of the Argonne National Laboratory or the Idaho Site, was established in 1949 and was operated as a separate site until 2005 when ANL-W was merged with the Idaho National Engineering and Environmental Laboratory (INEEL) to form the Idaho National Laboratory (INL). NIOSH determined that the most effective way of analyzing the ANL-W facilities was by focusing on the two ANL-W operating areas on the INL Site: the EBR-I Complex and the EBR-II Complex. The EBR-I Complex was originally commissioned to test a fuel-breeding reactor; later operations involved building and testing experimental reactors. The EBR-II Complex included test reactor facilities and support facilities for reactor fuel experiments and testing.

- Principal sources of internal radiation for members of the evaluated ANL-W class working in the EBR-I and EBR-II Complexes included exposures to various isotopes of uranium, plutonium, thorium, and other actinides as well as exotic radionuclides (produced from, or as a result of, reactor neutron irradiation) that include mixed fission and activation products (MFP/MAP), radioiodines, and other radionuclides.

- The internal dose potential during the period under evaluation was associated with reactors, laboratories, and other research and support activities as well as decontamination and decommissioning activities and radioactive waste disposal. There was limited personnel internal exposure potential to alpha-emitting radionuclides during the evaluated period. However, for the period from April 10, 1951 through December 31, 1957, NIOSH has not located sufficient personnel or area monitoring documentation to support complete reconstruction of internal personnel exposures. NIOSH has identified personnel monitoring data for mixed fission product exposures that are sufficient to support reconstructing internal doses for ANL-W workers beginning in 1958.

- Principal sources of external radiation for members of the evaluated ANL-W class working in the EBR-I and EBR-II Complexes included beta-gamma and neutron exposures to various isotopes of uranium, plutonium, thorium, and other actinides as well as exotic radionuclides (produced from, or as a result of, reactor neutron irradiation) that include mixed fission and activation products (MFP/MAP), radioiodines, noble gases, and other radionuclides. By policy, ANL-W required external dosimetry (film or TLD) be worn within the operating (fenced) area. This dosimetry was issued at the security guard checkpoint.

- NIOSH has determined that the external dose potential during the period under evaluation was associated with reactors, laboratories, and other research and support activities as well as decontamination and decommissioning activities and radioactive waste disposal. For the period from April 10, 1951 through December 31, 1957, NIOSH has not located sufficient personnel or area monitoring documentation to support complete reconstruction of external personnel exposures. NIOSH has identified personnel monitoring data for external exposures that are sufficient to support reconstructing external doses for ANL-W workers beginning in 1958.

- NIOSH finds that it is likely feasible to reconstruct occupational medical dose for ANL-W workers with sufficient accuracy.

- Pursuant to 42 C.F.R. § 83.13(c)(1), NIOSH determined that there is insufficient information for ANL-W personnel for the period from April 10, 1951 through December 31, 1957 to either: (1) estimate the maximum radiation dose, for every type of cancer for which radiation doses are reconstructed, that could have been incurred under plausible circumstances by any member of the class; or (2) estimate the radiation doses of members of the class more precisely than a maximum dose estimate.

- Although NIOSH found that it is not possible to completely reconstruct radiation doses for the proposed class, NIOSH intends to use any internal and external monitoring data that may become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures). Therefore, dose reconstructions for individuals employed at ANL-W during the period from April 10, 1951 through December 31, 1957, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate.

Health Endangerment Determination

Per EEOICPA and 42 C.F.R. § 83.13(c)(3), a health endangerment determination is required because NIOSH has determined that it does not have sufficient information to estimate dose for the members of the proposed class from April 10, 1951 through December 31, 1957.

NIOSH did not identify any evidence supplied by the petitioners or from other resources that would establish that the proposed class was exposed to radiation during a discrete incident likely to have involved exceptionally high-level exposures. However, evidence indicates that some workers in the proposed class may have accumulated substantial chronic exposures through episodic intakes of radionuclides, combined with external exposures to gamma, beta, and neutron radiation. Consequently, NIOSH has determined that health was endangered for those workers covered by this evaluation who were employed for at least 250 aggregated work days either solely under their employment or in combination with work days within the parameters established for other SEC classes.

For the period January 1, 1958 through December 31, 1979, a health endangerment determination is not required because NIOSH has determined that it has sufficient information to estimate dose for the members of the evaluated class during this time frame.

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Acronyms and Abbreviations

An acronyms and abbreviations list is available in Attachment 2 at the end of this document.

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SEC Petition Evaluation Report for SEC-00224

1.0 Purpose and Scope

This report evaluates the feasibility of reconstructing doses for all workers who worked in any location of the Argonne National Laboratory-West from April 10, 1951 through December 31, 1979. It provides information and analyses germane to considering a petition for adding a class of employees to the congressionally-created SEC.

This report does not make any determinations concerning the feasibility of dose reconstruction that necessarily apply to any individual energy employee who might require a dose reconstruction from NIOSH. This report also does not contain the final determination as to whether the proposed class will be added to the SEC (see Section 2.0).

This evaluation was conducted in accordance with the requirements of EEOICPA, 42 C.F.R. pt. 83, and the guidance contained in the Division of Compensation Analysis and Support's (DCAS) *Internal Procedures for the Evaluation of Special Exposure Cohort Petitions*, DCAS-PR-004.[1]

2.0 Introduction

Both EEOICPA and 42 C.F.R. pt. 83 require NIOSH to evaluate qualified petitions requesting that the Department of Health and Human Services (HHS) add a class of employees to the SEC. The evaluation is intended to provide a fair, science-based determination of whether it is feasible to estimate with sufficient accuracy the radiation doses of the class of employees through NIOSH dose reconstructions.[2]

42 C.F.R. § 83.13(c)(1) states: *Radiation doses can be estimated with sufficient accuracy if NIOSH has established that it has access to sufficient information to estimate the maximum radiation dose, for every type of cancer for which radiation doses are reconstructed, that could have been incurred in plausible circumstances by any member of the class, or if NIOSH has established that it has access to sufficient information to estimate the radiation doses of members of the class more precisely than an estimate of the maximum radiation dose.*

Under 42 C.F.R. § 83.13(c)(3), if it is not feasible to estimate with sufficient accuracy radiation doses for members of the class, then NIOSH must determine that there is a reasonable likelihood that such radiation doses may have endangered the health of members of the class. The regulation requires NIOSH to assume that any duration of unprotected exposure may have endangered the health of members of a class when it has been established that the class may have been exposed to radiation during a discrete incident likely to have involved levels of exposure similarly high to those occurring during nuclear criticality incidents. If the occurrence of such an exceptionally high-level exposure has not been established, then NIOSH is required to specify that health was endangered for those workers

[1] DCAS was formerly known as the Office of Compensation Analysis and Support (OCAS).

[2] NIOSH dose reconstructions under EEOICPA are performed using the methods promulgated under 42 C.F.R. pt. 82 and the detailed implementation guidelines available at http://www.cdc.gov/niosh/ocas.

who were employed for at least 250 aggregated work days within the parameters established for the class or in combination with work days within the parameters established for one or more other SEC classes.

NIOSH is required to document its evaluation in a report, and to do so, relies upon both its own dose reconstruction expertise as well as technical support from its contractor, Oak Ridge Associated Universities (ORAU). Once completed, NIOSH provides the report to both the petitioner(s) and the Advisory Board on Radiation and Worker Health (Board). The Board will consider the NIOSH evaluation report, together with the petition, petitioner(s) comments, and other information the Board considers appropriate, in order to make recommendations to the Secretary of HHS on whether or not to add one or more classes of employees to the SEC. Once NIOSH has received and considered the advice of the Board, the Director of NIOSH will propose a decision on behalf of HHS. The Secretary of HHS will make the final decision, taking into account the NIOSH evaluation, the advice of the Board, and the proposed decision issued by NIOSH. As part of this decision process, petitioners may seek a review of certain types of final decisions issued by the Secretary of HHS.[3]

3.0 SEC-00224 Argonne National Laboratory-West Class Definitions

The following subsections address the evolution of the class definition for SEC-00224, Argonne National Laboratory-West (ANL-W). When a petition is submitted, the requested class definition is reviewed as submitted. Based on its review of the available site information and data, NIOSH will make a determination whether to qualify for full evaluation all, some, or no part of the petitioner-requested class. If some portion of the petitioner-requested class is qualified, NIOSH will specify that class along with a justification for any modification of the petitioner's class. After a full evaluation of the qualified class, NIOSH will determine whether to propose a class for addition to the SEC and will specify that proposed class definition.

3.1 Petitioner-Requested Class Definition and Basis

Petition SEC-00224 was received on December 4, 2014 and qualified on March 13, 2015. The petitioner requested that NIOSH consider the following class: *All workers who worked in any work location at the Argonne National Laboratory-West from January 1, 1949 through December 31, 1995, with the basis of inadequate monitoring for plutonium, neptunium and fission products during that time period.*

The petitioner provided information and affidavit statements in support of the petitioner's belief that accurate dose reconstruction over time is impossible for the ANL-W workers in question. NIOSH deemed the following information and affidavit statements sufficient to qualify SEC-00224 for evaluation:

> The petitioner indicated in Section F of the Special Exposure Cohort Petition – Form B, that there was inadequate monitoring for plutonium, neptunium, and fission products (DSA Ref

[3] See 42 C.F.R. pt. 83 for a full description of the procedures summarized here. Additional internal procedures are available at http://www.cdc.gov/niosh/ocas.

ID: 120480). This serves as the basis for proposing that records and information at ANL-W were inadequate for individual dose reconstruction.

Based on its ANL-W research and data capture efforts, NIOSH determined that it has access to personnel bioassay monitoring data (including urinalysis, fecal sampling, and whole-body counts), and external dosimetry data for ANL-W workers during the time period under evaluation. However, NIOSH also determined that internal and external monitoring records are not complete for all time periods or for all radionuclides. NIOSH concluded that there is sufficient documentation to support, for at least part of the requested time period, the petition basis that internal and external radiation exposures and radiation doses were not adequately monitored at ANL-W, either through personal monitoring or area monitoring. The information and statements provided by the petitioner qualified the petition for further consideration by NIOSH, the Board, and HHS. The details of the petition basis are addressed in Section 7.4.

3.2 Class Evaluated by NIOSH

Based on its preliminary research, NIOSH modified the petitioner-requested class. The class start date was modified because ANL-W's first radiologically-related operation, at EBR-I, did not occur until April 10, 1951, when the reactor facility was completed and its associated reactor assembly commenced. The class end date was modified because of the large number of plutonium bioassay analyses available beginning in 1980. Therefore, NIOSH defined the following class for further evaluation: All workers who worked in any location of the Argonne National Laboratory-West from April 10, 1951 through December 31, 1979.

3.3 NIOSH-Proposed Class(es) to be Added to the SEC

Based on its research of the class under evaluation, NIOSH has defined a single class of employees for which NIOSH cannot estimate radiation doses with sufficient accuracy. The NIOSH-proposed SEC class includes all employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Argonne National Laboratory-West between April 10, 1951 and December 31, 1957 for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort.

4.0 Data Sources Reviewed by NIOSH to Evaluate the Class

Since Argonne National Laboratory-West was made part of Idaho National Laboratory in 2005, records generated by ANL-W have been integrated into INL's records collection. INL uses the Electronic Document Management System (EDMS) to manage records that are in an electronic format. EDMS is made of up of multiple client/server and web applications that run on several different operating systems and hardware to provide thousands of users with electronic documents. Preset as well as user-defined queries can be performed. The NIOSH and ORAUT Leads were provided EDMS accounts to allow for searches of non-classified documents during data capture trips in Idaho. EDMS was the primary resource in identifying and capturing documents of interest for the SEC-00224 Evaluation Report.

As is standard practice, NIOSH also completed an extensive database and Internet search for information regarding ANL-W. The database search included the DOE Legacy Management Considered Sites database, the DOE Office of Scientific and Technical Information (OSTI) SciTech Connect database, the OSTI Energy Citations database, and the Hanford Declassified Document Retrieval System. In addition to general Internet searches, the NIOSH Internet search included OSTI OpenNet Advanced searches, OSTI Information Bridge Fielded searches, Nuclear Regulatory Commission (NRC) Agency-wide Documents Access and Management (ADAMS) web searches, the DOE-National Nuclear Security Administration-Nevada Site Office-search, the Defense Technical Information Center database (DTIC), the DOE website, the DOE Comprehensive Epidemiologic Data Resource (CEDR), the DOE Environmental Management website, the DOE Oak Ridge Operations website, the Energy Employees Claimant Assistance Project (EECAP) website, the Hathitrust website, the Internet edition of the Health Physics Journal, the National Academies Press website, the Oak Ridge National Laboratory online library and report repository, the University of Hawaii, Manoa online collection, the University of North Texas online collection, the U.S. Army Corps of Engineers website, and the U.S. Transuranium and Uranium website. Attachment 1 contains a summary of ANL-W documents. The summary specifically identifies data capture details and general descriptions of the documents retrieved.

NIOSH and ORAUT staff went on five data capture trips to assess and scan ANL-W-related documentation at INL. These five trips resulted in the capture of a wide variety of relevant documentation, including 1,716 documents and 130,661 pages. In addition, ORAUT staff went on two data capture trips to Argonne National Laboratory-East (Lemont, IL), primarily to understand the structure of the employee occupational exposure histories held there, how these exposure history collections are accessed by ANL-E staff when responding to claimant dosimetry requests under EEOICPA, and to retrieve exposure history information on a selected group of early ANL-W workers. These two trips and follow-up inquiries resulted in the capture of exposure histories of 61 early ANL-W workers. The data captures at ANL-E also resulted in the collection of 92 documents and 4,993 pages of relevant information. Table 4-1 lists the ANL-W data capture trips and the types of information obtained.

In addition to the database and Internet searches listed above, NIOSH identified and reviewed numerous data sources to determine information relevant to determining the feasibility of dose reconstruction for the class of employees under evaluation. This included determining the availability of information on personal monitoring, area monitoring, industrial processes, and radiation source materials. The following subsections summarize the data sources identified and reviewed by NIOSH.

Table 4-1: Data Captures Conducted in Support of the SEC-00224 ANL-W Evaluation

Site/Facility of Visit	Dates of Visit	Boxes Reviewed	Docs Captured	Page Count	Brief Summary Of Site Visit
Idaho National Laboratory	09/21/2015 - 09/25/2015	59	95	14,527	Air sample data, radiological data, whole body counts, urinalysis data, and personnel dosimetry data
Idaho National Laboratory	04/27/2015 - 04/30/2015	48	652	40,481	Air sampling data, radiation surveys, daily log books, air monitoring, radiation reports, routine survey reports, smear data, technical specifications, radiation safety log books, personnel skin contamination, health physics log books, (Argonne National Laboratory); air samples, alpha contamination surveys, neutron surveys, safety analysis reports, early waste retrieval, offsite incoming radioactive material shipment records, health physics log books (Idaho National Laboratory).
Idaho National Laboratory	04/13/2015 - 04/16/2015	107	318	37,181	Visitor cards, individual worker's medical files, incidents, film badge reports, air and smear reports, temporary badge reports, radiological surveys, quarterly exposure summaries, bioassay results, contamination surveys, safety training records, health physics weekly reports, urine sample results, personnel exposure questionnaires, appraisal documents, and skin contamination results.
Idaho National Laboratory	03/29/2015 - 04/02/2015	91	489	26,731	Controlled entry procedure, shipping incidents and related contamination, hazards analysis, photographic history of the Argonaut Multiplication Experiment, locating and identifying the source of the 5/24/67 fission product release in EBR-II, air sample data, radiological surveys, sampling and smear data, external dose reports, emergency preparedness memos, inspection reports, gas cooler reactor experiment information, pending whole body count data to be cross-checked against existing data sets, repair of central superheater assemblies, contamination of EBR-II, EBR-II reports, safety procedures, actinide fuel fabrication, health physics logbook, and EBR-II run reports.
Idaho National Laboratory	12/08/2008 - 12/12/2008	62	70	6,748	Monthly HP reports (1968 - 1969), neutron surveys, one month sample of radiological surveys, personnel exposure information (1953 - 1961), routine urinalysis (1963 - 1966), routine whole body counts (1964 - 1971), radiological incidents, air sample data sheets, employee exposure information, whole body and hand exposure (1969 - 19 71).
Argonne National Laboratory - East	10/12/2015 - 10/14/2015	None	None	0	Reconnaissance trip to better understand the structure of the employee occupational exposure histories held there, how these exposure history collections are accessed by Argonne National Laboratory-East staff when responding to claimant dosimetry requests under EEOICPA, and to retrieve exposure history information on a selected group of early Argonne National Laboratory-West workers.
Argonne National Laboratory - East	06/15/2015 - 06/19/2015	82	92	4,993	Radiation safety section monthly reports, air sample data, explosion report at EBR-II, technical specifications for EBR-II, health physic reports, bioassay reports, hazard summary report for EBR-II and addendum, occupational safety and health audit of EBR-II, Idaho reactor division personnel list, safety analysis of the EBR-II reprocessing facility, and tritium in EBR-II.
TOTAL	**N/A**	**449**	**1,716**	**130,661**	**N/A**

4.1 Site Profile Technical Basis Documents (TBDs)

A Site Profile provides specific information concerning the documentation of historical practices at the specified site. Dose reconstructors can use the Site Profile to evaluate internal and external dosimetry data for monitored and unmonitored workers, and to supplement, or substitute for, individual monitoring data. A Site Profile consists of an Introduction and five Technical Basis Documents (TBDs) that provide process history information, information on personal and area monitoring, radiation source descriptions, and references to primary documents relevant to the radiological operations at the site. The Site Profile for a small site may consist of a single document. As part of NIOSH's evaluation detailed herein, it examined the following TBDs for insights into ANL-W operations or related topics/operations at other sites:

- *TBD for INL - ANL-W – Introduction*, ORAUT-TKBS-0007-1; Rev. 03; March 12, 2010; SRDB Ref ID: 79862

- *TBD for INL – ANL-W – Site Description*, ORAUT-TKBS-0007-2; Rev. 04; August 2, 2010; SRDB Ref ID: 84106

- *TBD for INL – ANL-W – Occupational Medical Dose*, ORAUT-TKBS-0007-3; Rev. 02 ; December 21, 2009; SRDB Ref ID: 77829

- *TBD for INL – ANL-W – Occupational Environmental Dose*, ORAUT-TKBS-0007-4; Rev. 02; January 8, 2010; SRDB Ref ID: 78635

- *TBD for INL – ANL-W – Occupational Internal Dose*, ORAUT-TKBS-0007-5; Rev. 03; March 2, 2010; SRDB Ref ID: 79571

- *TBD for INL – ANL-W – Occupational External Dose*, ORAUT-TKBS-0007-6; Rev. 03; April 19, 2011; SRDB Ref ID: 94104

- *TBD for Y-12 National Security Complex – Occupational Internal Dose*, ORAUT-TKBS-0014-5, Rev. 03; Oak Ridge Associated Universities; March 12, 2012; SRDB Ref ID: 109202

4.2 Technical Information Bulletins and Procedures

A Technical Information Bulletin is a general ORAUT or NIOSH working document that provides guidance for preparing dose reconstructions at particular sites or categories of sites. An ORAUT Procedure provides specific requirements and guidance regarding EEOICPA project-level activities, including preparation of dose reconstructions at particular sites or categories of sites. NIOSH reviewed the following procedure as part of its evaluation:

- *OTIB: Internal Dosimetry Organ, External Dosimetry Organ, and IREP Model Selection by ICD-9 Code, ORAUT-OTIB-0005 Rev 05; December 20, 2012; SRDB Ref ID: 121336*

- *OTIB: Technical Information Bulletin: Tritium Calculated and Missed Dose Estimates, ORAUT-OTIB-0011 Rev 00; June 29, 2004; SRDB Ref ID: 19430*

- *OTIB: Assignment of Environmental Internal Doses for Employees Not Exposed to Airborne Radionuclides in the Workplace, ORAUT-OTIB-0014 Rev 00; June 22, 2004; SRDB Ref ID: 19432*

- *OTIB: Bulletin: Interpretation of Dosimetry Data for Assignment of Shallow Dose, ORAUT-OTIB-0017 Rev 01; October 11, 2005; SRDB Ref ID: 19434*

- *OTIB: Internal Dose Overestimates for Facilities with Air Sampling Programs, ORAUT-OTIB-0018 Rev 01; August 9, 2005; SRDB Ref ID: 19436*

- *OTIB: Technical Information Bulletin: Assignment of Missed Neutron Doses Based on Dosimeter Records, ORAUT-OTIB-0023 Rev 01; May 14, 2008; SRDB Ref ID: 43937*

- *OTIB: Application of Internal Doses Based on Claimant-Favorable Assumptions for Processing as Best Estimates, ORAUT-OTIB-0033 Rev 00; April 20, 2005; SRDB Ref ID: 19457*

- *OTIB: Estimating Doses for Plutonium Strongly Retained in the Lung, ORAUT-OTIB-0049 Rev 01 PC-2; November 29, 2010; SRDB Ref ID: 90666*

- *OTIB: Technical Information Bulletin: Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses, ORAUT-OTIB-0054 Rev 02; March 6, 2014; SRDB Ref ID: 130852*

- *OTIB: Technical Information Bulletin: Internal Dose Reconstruction, ORAUT-OTIB-0060 Rev 00; February 6, 2007; SRDB Ref ID: 29984*

- *OTIB: Guidance on Assigning Occupational X-Ray Dose under EEOICPA for X-Rays Administered Off Site, ORAUT-OTIB-0079 Rev 00; January 3, 2011; SRDB Ref ID: 89563*

- *OTIB: Technical Information Bulletin: Dose Reconstruction Method for Chronic Lymphocytic Leukemia, ORAUT-OTIB-0082 Rev 00 PC-1; December 20, 2012; SRDB Ref ID: 121337*

- *TIB: Estimations of Ingestion Intakes, OCAS-TIB-009; Rev. 00; April 13, 2004; SRDB Ref ID: 22397*

- *Procedure: Occupational On-Site Ambient Dose Reconstruction for DOE Sites, ORAUT-PROC-0060 Rev 01; June 28, 2006; SRDB Ref ID: 29986*

- *Procedure: Occupational Medical X-Ray Dose Reconstruction for DOE Sites, Rev 03, ORAUT-PROC-0061 Rev 03; March 3, 2010; SRDB Ref ID: 79758*

4.3 Facility Employees and Experts

To obtain additional information, NIOSH interviewed 19 former and current ANL-W employees.

- *Personal Communication, 2015a, Personal Communication with a former ANL-W chemist; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 18, 2015; SRDB Ref ID: 147021*

- *Personal Communication, 2015b, Personal Communication with a former ANL-W engineer; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 19, 2015; SRDB Ref ID: 147018*

- *Personal Communication, 2015c, Personal Communication with a former ANL-W auditor; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 19, 2015; SRDB Ref ID: 147248*

- *Personal Communication, 2015d, Personal Communication with a former ANL-W operator; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 19, 2015; SRDB Ref ID: 147009*

- *Personal Communication, 2015e, Personal Communication with a former ANL-W manager; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 19, 2015; SRDB Ref ID: 147010*

- *Personal Communication, 2015f, Personal Communication with a current ANL-W operator; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 19, 2015; SRDB Ref ID: 147014*

- *Personal Communication, 2015g, Personal Communication with a former ANL-W supervisor; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 20, 2015; SRDB Ref ID: 147015*

- *Personal Communication, 2015h, Personal Communication with a former ANL-W manager; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 20, 2015; SRDB Ref ID: 147016*

- *Personal Communication, 2015i, Personal Communication with a former ANL-W manager; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 20, 2015; SRDB Ref ID: 147017*

- *Personal Communication, 2015j, Personal Communication with a former ANL-W information staffer; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 20, 2015; SRDB Ref ID: 147019*

- *Personal Communication, 2015k, Personal Communication with a former ANL-W manager; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 20, 2015; SRDB Ref ID: 147020*

- *Personal Communication, 2015L, Personal Communication with a former ANL-W trainer; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 21, 2015; SRDB Ref ID: 147011*

- *Personal Communication, 2015m, Personal Communication with a current ANL-W operator; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 21, 2015; SRDB Ref ID: 147012*

- *Personal Communication, 2015n, Personal Communication with a former ANL-W engineer; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 21, 2015; SRDB Ref ID: 147013*

- *Personal Communication, 2015o, Personal Communication with a former ANL-W pipefitter; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; May 21, 2015; SRDB Ref ID: 147022*

- *Personal Communication, 2015p, Personal Communication with a former ANL-W supervisor; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; July 22, 2015; SRDB Ref ID: 147028*

- *Personal Communication, 2015q, Personal Communication with a former ANL-W manager; In-person Interview by NIOSH and ABRWH; July 22, 2015; SRDB Ref ID: 147249*

- *Personal Communication, 2015r, Personal Communication with a former ANL-W technician In-person Interview by [ORAU Team or NIOSH]; July 22, 2015; SRDB Ref ID: 147446*

- *Personal Communication, 2015s, Personal Communication with a former ANL-W designer; In-person Interview by ORAU Team, NIOSH, ABRWH, and SC&A; July 22, 2015; SRDB Ref ID: 147445*

- *Personal Communication, 2015t, Personal Communication with a former ANL-W reactor operator; In-person Interview by ORAU Team and NIOSH; November 4, 2015; SRDB Ref ID: 150163*

4.4 Previous Dose Reconstructions

NIOSH reviewed its NIOSH DCAS Claims Tracking System (referred to as NOCTS) to locate EEOICPA-related dose reconstructions that might provide information relevant to the petition evaluation. Table 4-2 summarizes the results of this review. (NOCTS data available as of January 15, 2016)

Table 4-2: No. of ANL-W Claims Submitted Under the Dose Reconstruction Rule

Description	Totals
Total number of claims submitted for dose reconstruction	346
Total number of claims submitted for energy employees who worked during the period under evaluation (April 10, 1951 through December 31, 1979).	273
Number of dose reconstructions completed for energy employees who worked during the period under evaluation (i.e., the number of such claims completed by NIOSH and submitted to the Department of Labor for final approval).	237
Number of claims for which internal dosimetry records were obtained for the identified years in the evaluated class definition	161
Number of claims for which external dosimetry records were obtained for the identified years in the evaluated class definition	209

NIOSH reviewed each claim to determine whether internal and/or external personal monitoring records could be obtained for the employee.

4.5 NIOSH Site Research Database

NIOSH also examined its Site Research Database (SRDB) to locate documents supporting the assessment of the evaluated class. Three thousand nine hundred eighteen (3,918) documents in this database were identified as pertaining to ANL-W. These documents were evaluated for their relevance to this petition. The documents include historical background on data types, monitoring programs, smear sampling, air monitoring, bioassay data, radiological control programs, medical monitoring, process materials, and process descriptions.

4.6 Documentation and/or Affidavits Provided by Petitioners

In qualifying and evaluating the petition, NIOSH reviewed the following documents submitted by the petitioners:

- *Special Exposure Cohort Petition – Form B*, submitted by petitioner representative Atomic Weapons Employee Consultants, LLC; submitted November 25, 2014; DSA Ref ID: 120480

5.0 Radiological Operations Relevant to the Class Evaluated by NIOSH

The following subsections summarize the radiological operations at Argonne National Laboratory-West from January 1949 through December 1979, and the information available to NIOSH to characterize particular processes and radioactive source materials. From available sources NIOSH has gathered process and source descriptions, information regarding the identity and quantities of each radionuclide of concern, and information describing processes through which radiation exposures may have occurred and the physical environment in which they may have occurred. The information included within this evaluation report is intended only to be a summary of the available information.

5.1 Argonne National Laboratory-West Plant and Process Descriptions

Since its beginning in 1949 until February 2005, the Argonne National Laboratory-West (ANL-W) was operated by the University of Chicago under the AEC\ERDA\DOE‘s Chicago Operations Office. In February 2005, the Argonne National Laboratory-West facilities were merged with the rest of the Idaho National Laboratory under the DOE’s Idaho Operations Office; the facilities that were still operational were collectively named the Materials and Fuels Complex (MFC).

NOTE: ANL-W was originally known as the “Idaho Division” (ID) of the Argonne National Laboratory (ANL) and was also referred to as the “Idaho Site” in ANL documents. For the purposes of this evaluation report, the site will be referred to as Argonne National Laboratory-West or ANL-W (with the exception of source document titles).

The ANL-W facilities are physically located on portions of the 890-square-mile Idaho National Laboratory (INL) Site, which is located in the high desert of eastern Idaho, west-northwest of the city of Idaho Falls.

NOTE: The INL site was originally known as the National Reactor Testing Station (NRTS). The AEC established NRTS in 1949 on the site of a 1940s U.S. Navy bombing and artillery range. Later, the site was called the Idaho National Engineering and Environmental Laboratory (INEEL). On February 1, 2005, INEEL and ANL-W merged to become INL.

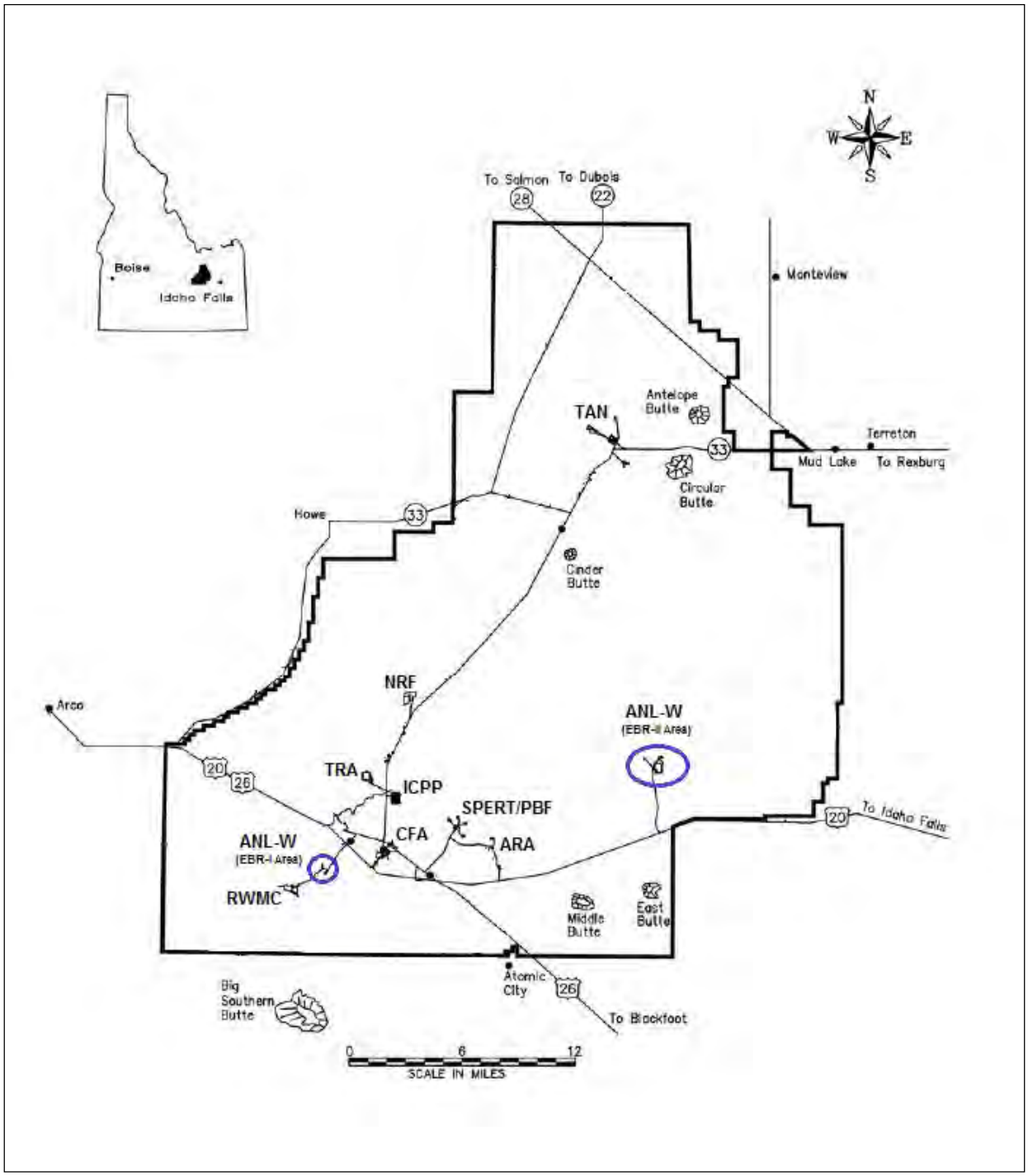
The ANL-W facilities primarily occupy two main areas on the INL Site:

1. EBR-I Complex (a.k.a. ANL West, West Area) (see Section 5.1.1)

2. EBR-II Complex (a.k.a. ANL West, East Area) (see Section 5.1.2)

NOTE: The structure of this evaluation report will reflect the above geographical division.

Figure 5-1 depicts the INL location in the State of Idaho, and the location of ANL-W facilities on the INL site (i.e., the EBR-I Complex and the EBR-II Complex).



Source: Sehlke, 2010

Figure 5-1: Locations of ANL-W Facilities (EBR-I Complex and EBR-II Complex)

5.1.1 EBR-I Complex

The EBR-I Complex is located approximately 20 miles west of the EBR-II Complex. The origins of all the EBR-I Complex facilities can be directly traced back to Argonne National Laboratory in Lemont, Illinois. The facilities were located on the INL reservation due to its remote location. EBR-I was commissioned to test the theory of a fuel-breeding reactor. Later, a series of experimental boiling-water reactors were built to test steam formation and reactor operating characteristics after a steam explosion in Lemont in 1952. The Zero Power Reactor-III facility was part of a large fast reactor program initiated at ANL. The Argonne Fast Source Reactor was the last facility built at the EBR-I Complex and served as a small-scale reactor for experimental measurements.

EBR-I Complex primary radiological facilities included:

- Experimental Breeder Reactor-I (EBR-I) (see Section 5.1.1.1)
- Zero Power Reactor-III (ZPR-III) (see Section 5.1.1.2)
- Boiling Water Reactor Experiments I-V (BORAX-I – BORAX-V) (see Section 5.1.1.3)
- Argonne Fast Source Reactor (AFSR) (first location) (see Section 5.1.1.4)

NOTE: The AFSR was located in the EBR-I Complex until the late-1970s when it was relocated to the EBR-II Complex.

Figure 5-2 shows an aerial photograph of the main part of the EBR-I Complex.

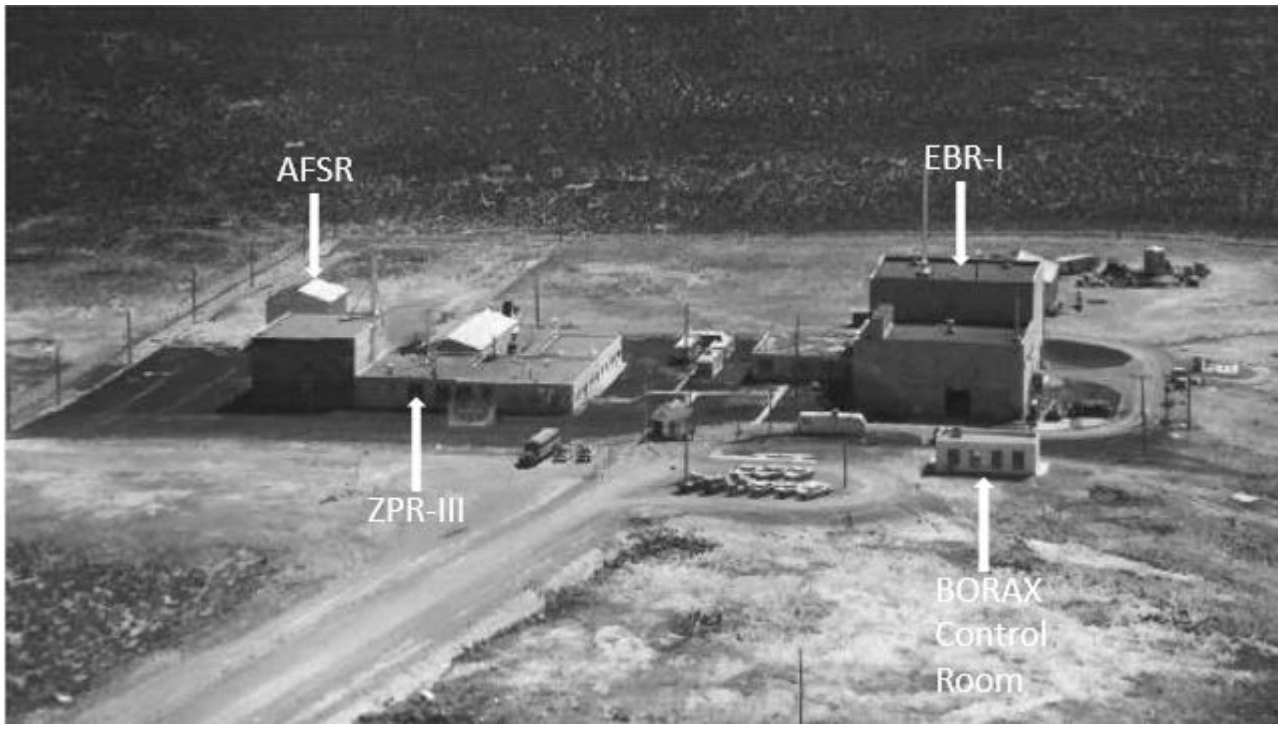


Figure 5-2: Aerial View of the Main Part of the EBR-I Complex

Figure 5-3 shows an aerial photograph of the BORAX locations relative to the main part of the EBR-I Complex.

Source: Google Earth

Figure 5-3: Aerial View of the Entire EBR-I Complex

5.1.1.1 Experimental Breeder Reactor-I (EBR-I)

ATTRIBUTION: Section 5.1.1.1 was completed by Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding EBR-I were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to EBR-I are explained in the associated text.

In 1946, the Manhattan Engineer District (MED) approved a proposal to build the Experimental Breeder Reactor No. 1 (EBR-I). Construction began in late 1949 at the National Reactor Testing Station in Idaho, now called the Idaho National Laboratory. Early in 1951, a few months before the EBR-I facility was completed, nine staff members from the Argonne National Laboratory in Lemont, Illinois arrived to install the reactor vessel that was designed and built at Argonne (see Figure 5-4).

Source: Stacy, 2000

Figure 5-4: Installation of EBR-I Reactor Vessel

The first attempt to operate the new reactor in May 1951 proved to be unsuccessful. It was determined that there was not enough fuel in the core to achieve criticality. Acquiring additional uranium and refabricating slightly larger fuel rods (in Chicago) took nearly three months. In August 1951, EBR-I achieved criticality with a core about the size of a football. Four months of low-power operation followed (Progress Report, 1953).

On December 20, 1951, the first historic EBR-I experiment began. The reactor was started and the power gradually increased over several hours. At 1:50 PM, the first usable amount of electricity ever generated from nuclear power began flowing from the turbine generator, successfully illuminating four light bulbs (see Figure 5-5). The next day, the experiment was repeated and sufficient electricity was generated to power the entire EBR-I facility (EBR-I, undated).

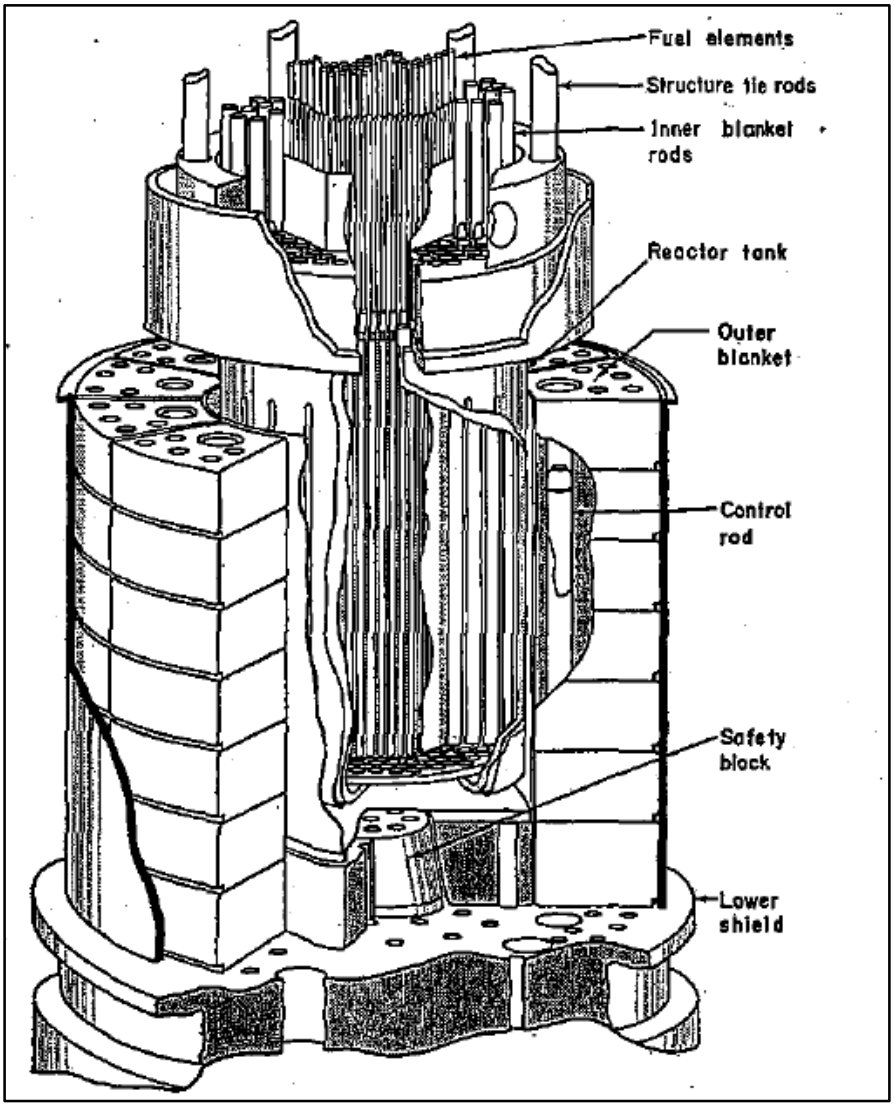
Source: Stacy, 2000

Figure 5-5: Four Light Bulbs Illuminated by EBR-1 (Dec. 20, 1951)

EBR-I's real mission was not to show that electricity could be generated by a nuclear reactor, but rather, to determine whether scientists' theoretical calculations on fuel-breeding could actually be achieved. Early in 1953, analysis showed that EBR-1 was creating one new atom of nuclear fuel for each atom it burned. The concept of fuel-breeding had been proven (Summary Report, 1953). Three additional cores were developed over the next ten years to increase the breeding ratio. The last core, Mark IV, achieved a breeding ratio of 1.27.

The EBR-I reactor was an unmoderated and heterogeneous design with a thermal output of approximately 1 MW at full power. The reactor-tank assembly, and all piping leading from it through the shielding, were double-walled (see Figure 5-6). The inner-tank assembly consisted of several hexagonally-shaped, stainless-steel subassemblies filled with either fuel or blanket rods. The portion of the tank containing the core and inner blanket was only 28 inches long and 15.9 inches ID (interior diameter). The vessel weight was supported by a shoulder formed by an increase in diameter at the junction of the lower tank assembly. The wall of the lower portion of the inner-reactor tank was

5/16-inch thick. An outer Inconel tank with a wall thickness of 1/16-inch served as an emergency receiver in the event of an inner-tank failure. The tanks were separated by a system of ribs running vertically along the inside surface of the outer tank (Inner Tanks, 1957).



Source: Radionuclide Concentrations, undated

Figure 5-6: Cut-away View of EBR-I

The reactor shielding contained a series of horizontal and vertical cross sections. Six horizontal beam holes were used to locate various nuclear instruments. A single three-inch hole at centerline elevation, and off-center with respect to the core and blanket, passed completely through the shielding. All beam holes were steel-lined and closed with concrete-filled steel plugs during operation. An extension of the graphite reflector into the concrete-shielded region served effectively as a thermal column. Five vertical holes penetrated the graphite reflector from the reactor top. These could be used for the irradiation of samples in a thermalized neutron flux. Special thimbles designed for

installation in the core or inner-blanket regions were used for the irradiation of samples in a fast-neutron atmosphere (Mark IV Terminal Report, 1965).

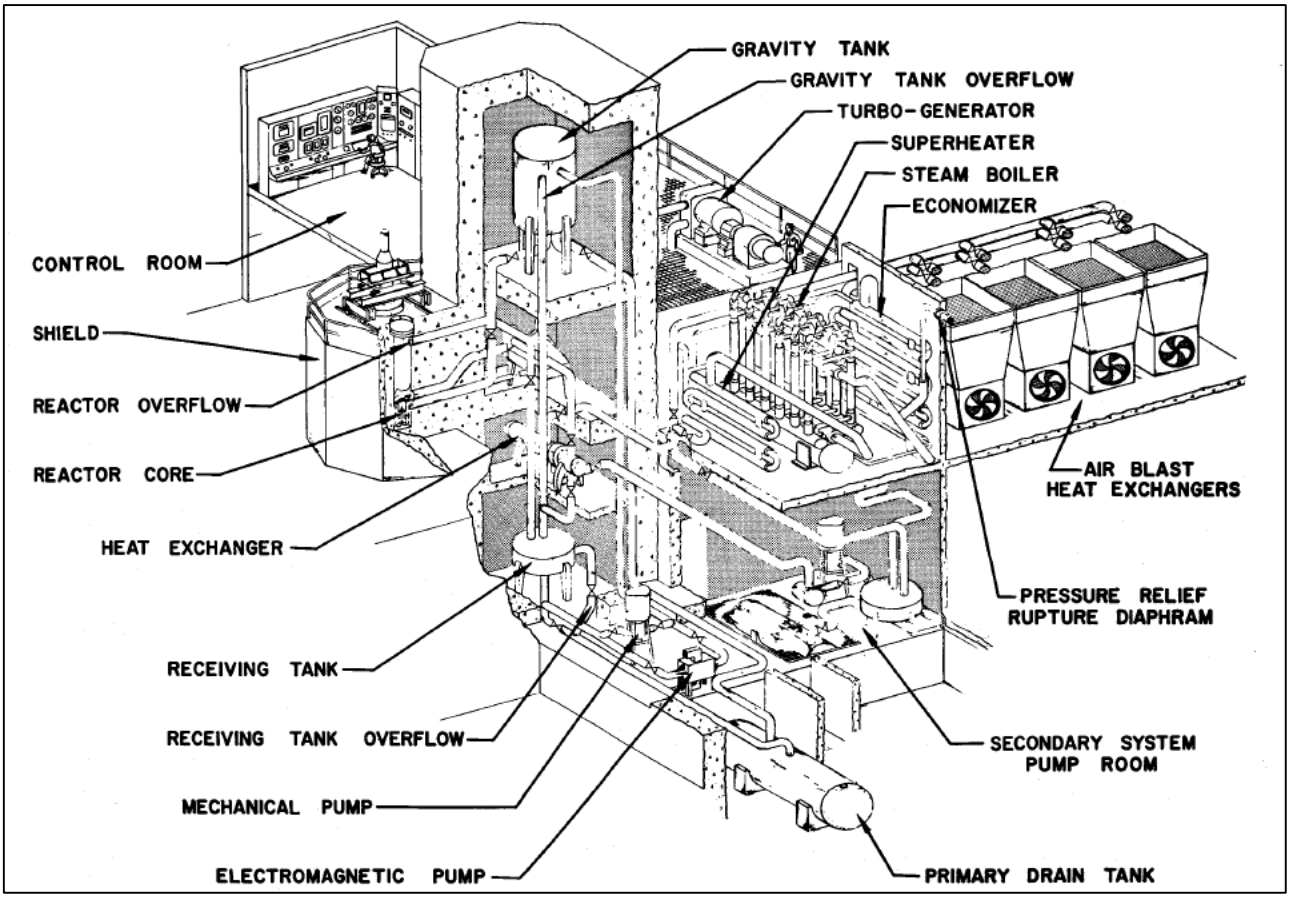
Immediately surrounding the lower portion of the reactor tank was the outer blanket (reflector). Outside the reflector was an annular layer of graphite, 18 inches thick and 35 inches high. Graphite was used in this region to thermalize and to reflect back into the breeding region a fraction of the neutrons normally lost through leakage. On the outside of the graphite was a four-inch-thick cast iron wall followed by eight feet of concrete (Mark IV Terminal Report, 1965). This thickness of concrete reflects additional concrete added in 1952 after it was observed that the dose rates surrounding the reactor during operation were higher than expected (Graham, 1959).

The bottom of the reflector was shielded by four feet of concrete. Radiations penetrating this shield streamed into equipment cells located in the basement directly beneath the reactor. Radiations emitted upward were absorbed, for the most part, in the coolant and rod extensions. Transmitted radiations were additionally absorbed by two shields on the top of the reactor tank. The lower shield consisted of two feet of high-density concrete; the upper shield consisted of eight-inch laminate of Masonite and iron (Mark IV Terminal Report, 1965). When accessing the reactor, both shields were handled with the 20-ton overhead crane (see Figure 5-7).

Source: EBR-I History, 1979, PDF p. 11

Figure 5-7: View of EBR-I

To permit the efficient removal of heat under high-power-density conditions (upwards of 180 kW/liter), and to minimize neutron moderation, a sodium-potassium (NaK) coolant was used. Both the primary and secondary systems used NaK which, with a melting point of -12.5°C, was liquid at room temperature. Under the effect of gravity, the NaK coolant flowed from an elevated supply tank through the inner blanket and core, through a primary-secondary heat exchanger, and into a receiving tank in the basement (see Figure 5-8). A pump, operating at a capacity slightly greater than the main coolant flow, returned the coolant to the gravity supply tank. An overflow system, connected from the supply tank to the receiver, guaranteed a constant level of delivery. From a safety viewpoint, the elevated supply tank provided approximately nine minutes of full coolant flow in the event of pump failure. If necessary, flow from the gravity tank could be throttled to provide up to eight hours of flow at a rate sufficient to remove decay heat (Mark IV Terminal Report, 1965).



Source: Mark IV Terminal Report, 1965

Figure 5-8: EBR-I Reactor Systems

Heat from the secondary coolant (also NaK) was removed in either of two ways: (1) through the generation of steam used to drive a conventional turbine-generator; or (2) through a fan-cooled load dissipater. Because the primary coolant was intensely radioactive during operation and immediately following shutdown (due to activation of the coolant), all primary components were enclosed in concrete-shielded cells. On the other hand, under normal operations, the secondary coolant was not radioactive and required no shielding.

Of the total thermal power generated in the system, approximately 16% was produced in a massive movable reflector that surrounds the primary reactor tank. The reflector consisted of an assembly of 84 stainless-steel-clad, keystone-shaped bricks of natural uranium weighing 100 lbs. each (Inner Tanks, 1957). The entire assembly (weighing approximately five tons) was mounted on a pedestal that could be raised or lowered *in toto* relative to the core. A forced circulation of air through cooling holes in the blanket bricks removed heat generated through fission and gamma absorption. There were twelve stainless-steel-clad uranium rods that served as a coarse control of reactivity and as a means of rapid reduction in reactivity in the event of a scram signal.

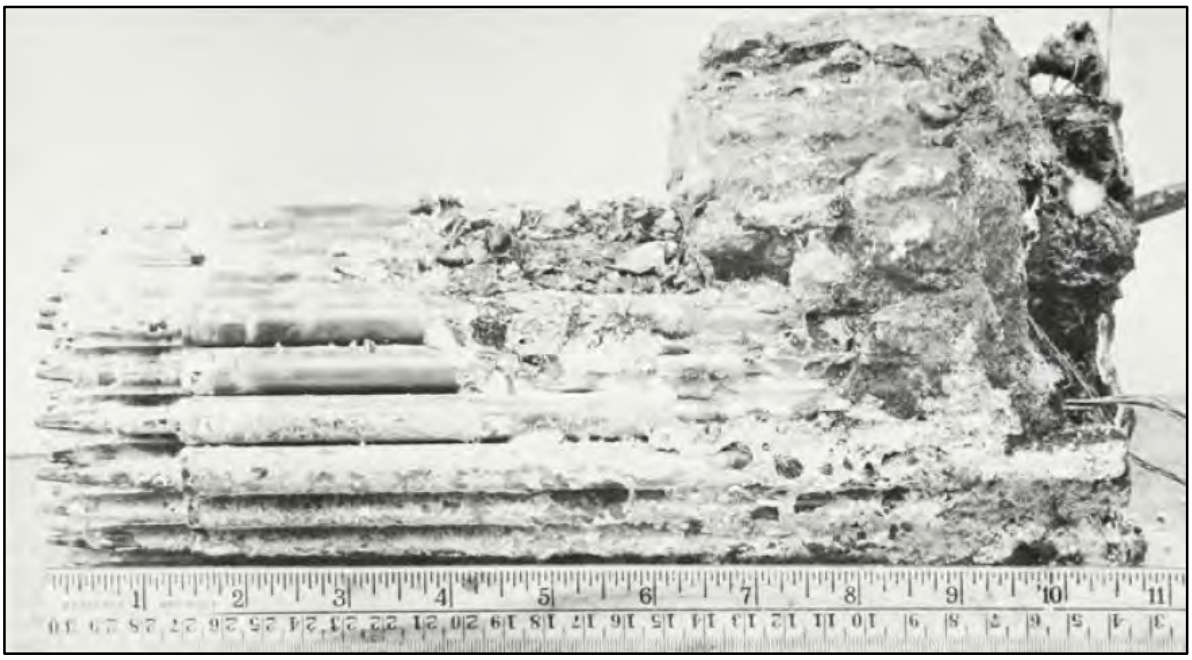
The reactor air supply and outer-blanket-cooling and shield-cooling exhaust systems were interconnected. A blower provided the supply air, which was taken from the outside the building and forced through an electrostatic filter. Cooling air for the shield was pulled from the supply plenum by a centrifugal blower located on the outlet side of the shield-cooling system. Before its release through the exhaust stack, the cooling air was passed through a bank of high-efficiency filters. The reflector-cooling system also received air from the same plenum. A positive-displacement blower drew air downward through cooling holes, through the reflector-support shielding, through telescopic tubes in the elevator structure, and into a plenum chamber. From there, it was drawn through high-efficiency filters and released through the stack.

All NaK storage tanks, both mechanical circulating pumps, and the top of the reactor operated under a controlled atmosphere of purified argon at a pressure of approximately 5 lb./in^2. A similar system operating at a slightly higher pressure was associated with the secondary NaK system. Feed gas, consisting of welding-grade argon, was passed through pressure-reducing valves to a purifier that consisted of a stainless-steel-mesh-packed column of heated NaK. Impurities of oxygen and water were removed through chemical reaction with the NaK.

EBR-I history can be traced by its four core loadings. Cores one through three had cores between 50 and 70 kg of enriched uranium, while the fourth core was plutonium.

- Mark I: This core was 93% enriched unalloyed uranium in stainless-steel jackets. It was processed at the Chemical Processing Plant (CPP) in 1965.

- Mark II: This core was 98% enriched uranium with a 2% zirconium alloy. On November 29, 1955, this core experienced a partial meltdown. Interestingly, EBR-I was originally scheduled to be shut down by 1956 because researchers thought all useful experiments had been carried out at EBR-I; the design of the ZPR-III reactor (operational in 1955) offered greater versatility. As a last EBR-I experiment, it was decided to take measurements of the transient temperature coefficients, which involved briefly shutting off the NaK coolant. The meltdown occurred because the reactor operator used the slower control rods instead of the fast-acting shut-off rods (as intended) to scram the reactor. The two-second time difference in using the slower control rods allowed for an overshoot in temperature and the resultant partial core meltdown (Radionuclide Concentrations, undated). Some of the blanket material melted as well, resulting in an average enrichment of 69% for the melted core assembly. Figure 5-9 shows the core assembly after partial removal of the lower blanket section. A custom dissolution of the core was performed at the Idaho Chemical Processing Plant (ICPP) (IN-1088).



Source: Kittel, 1957

Figure 5-9: View of Melted Mark II Core Assembly

- Mark III: This core was also composed of 98% enriched uranium with a 2% zirconium alloy and was used to prove that there was nothing intrinsically unsafe about a breeder reactor.

- Mark IV: This core was composed of 98.6% plutonium, 1.25% aluminum, and some impurities. The fuel was fabricated and canned at Rocky Flats in a joint effort with ANL-E (Fuel Elements, 1956). The purpose of the study was to better understand the properties of a plutonium core and determine if the expected breeding improvements over a uranium core could be achieved. Eleven of the plutonium fuel slugs were sent to the Metallurgy Division in Chicago for examination.

In 1966, President Lyndon B. Johnson designated EBR-I as a Registered National Historic Monument with the understanding that custody of the facility would be transferred to the National Park Service after operations. An EBR-I Complex Decontamination and Decommissioning (D&D) Program was prepared in September 1973 to make the EBR-I Complex safe for the public prior to transfer to the National Park Service. The primary effort consisted of removing and processing 5500 gallons of NaK coolant from the EBR-I reactor system. D&D was performed throughout the EBR-I complex and was completed on June 13, 1975 (Kendall, 1975; Brown, 1987).

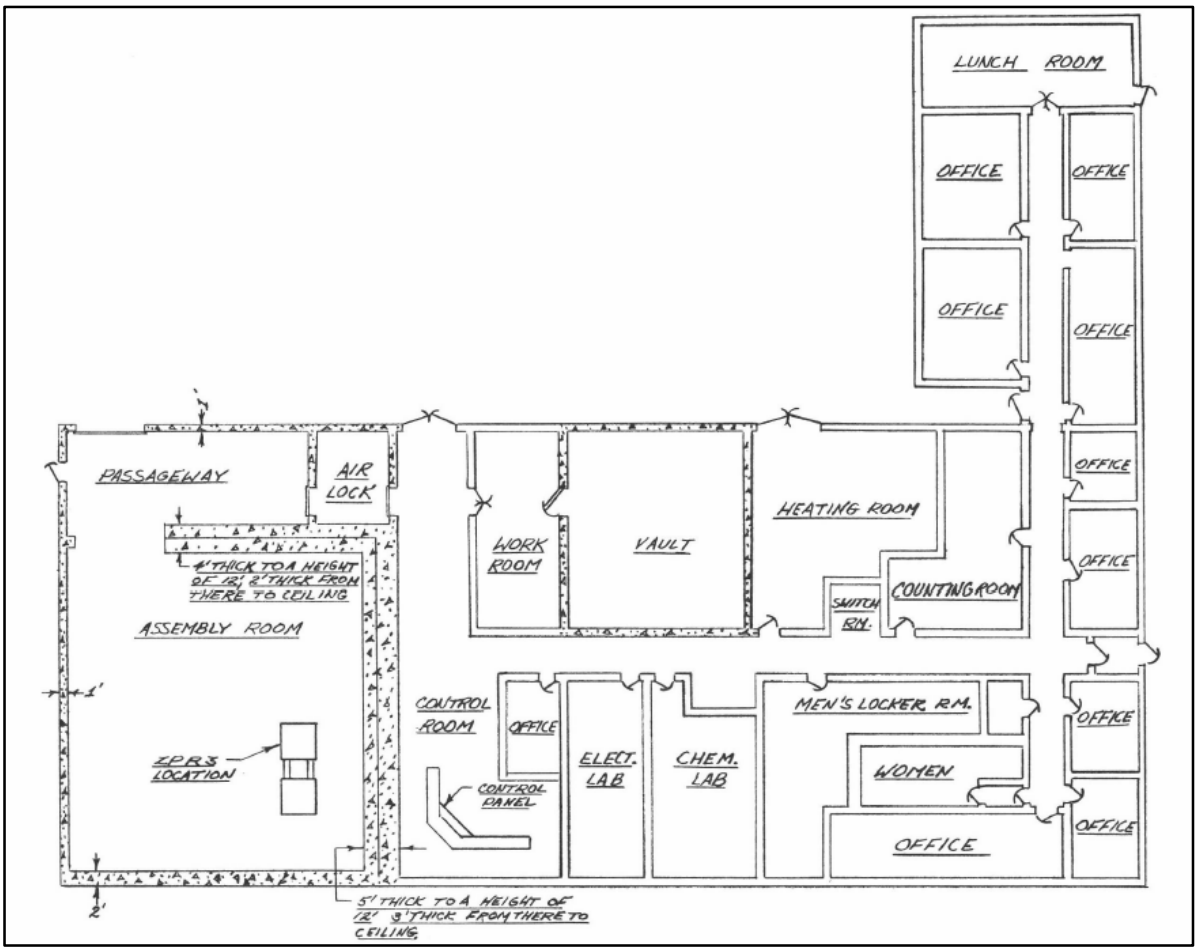
5.1.1.2 Zero Power Reactor-III (ZPR-III)

ATTRIBUTION: Section 5.1.1.2 was completed by Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding ZPR-III were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to ZPR-III are explained in the associated text.

From 1955 to 1990, more than a hundred Zero Power Reactor (ZPR) critical assemblies were constructed by Argonne National Laboratory (Lemont, IL) to support fast reactor development, validate nuclear data, and establish criticality safety benchmarks. ZPR-III and ZPPR were the "chronological bookends" for these critical assemblies and were located at ANL-W. The ZPR-III facility began operations in 1955 and ceased operations in 1970. ZPPR began operations in 1969 and was the last of the ZPR critical assemblies to be shut down.

The primary purpose of the ZPR-III facility was to establish the feasibility of a sodium-cooled, unmoderated fast-power breeder reactor for the production of electrical power, and to facilitate the design and operation of the then-prototype Experimental Breeder Reactor-II (EBR-II). In addition, the cores for the Fermi, Rapsodie, and the Southwest Experimental Fast Oxide Reactor (SEFOR) reactors were originally mocked up in ZPR-III (Stacy, 2000). There were a total of 63 separate assemblies during the operating history of ZPR-III. (An assembly is a critical reactor configuration - typically a mock-up of a particular reactor design.) Some assemblies existed for short periods of time; they were loaded to critical, a short series of measurements were performed, and the assembly was unloaded. Other assemblies lasted much longer; they were entire programs with several major phases and hundreds of experiments (LeSage, 2001).

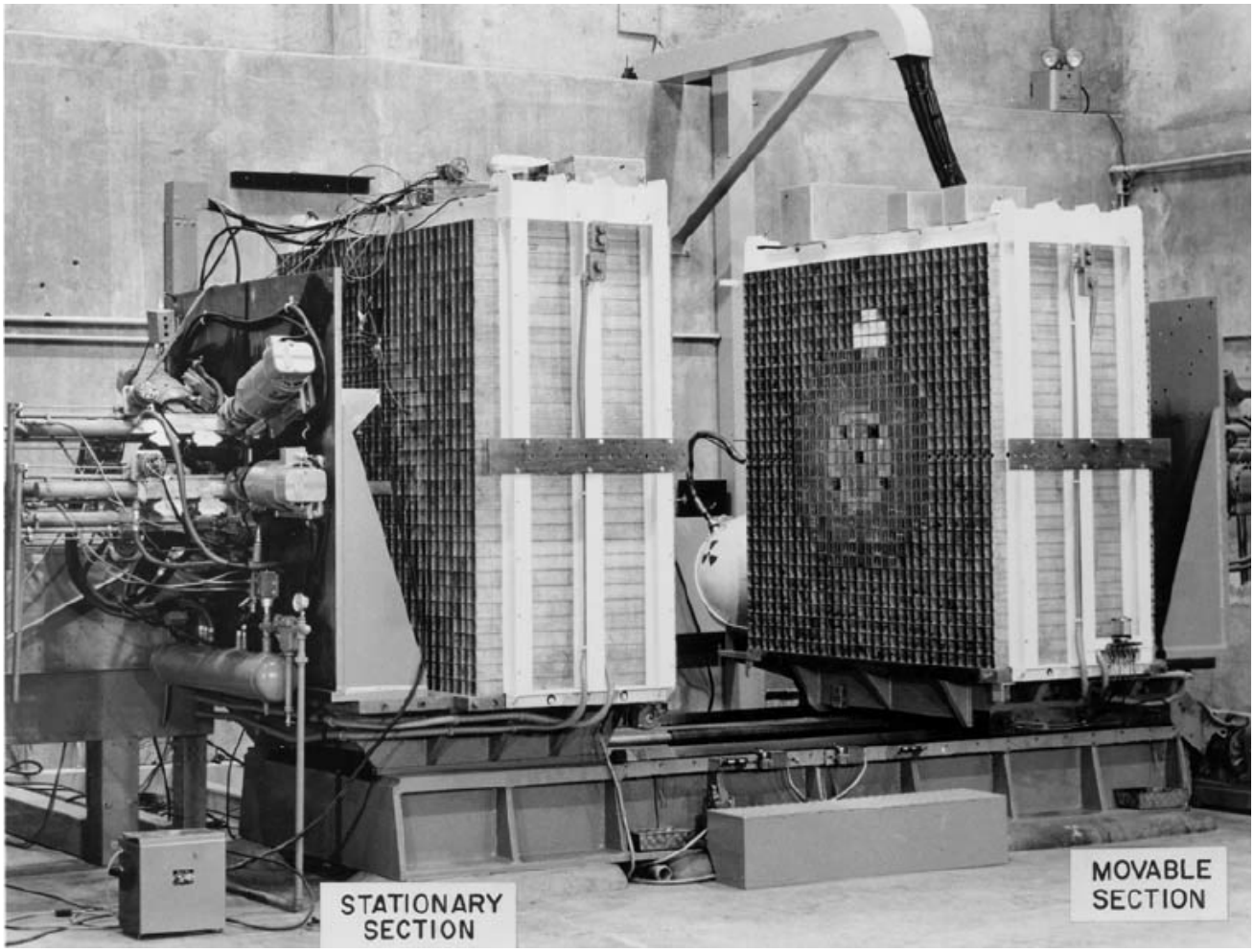
The ZPR-III fast critical facility was a single-story concrete building located in the EBR-I Complex (building designations ANL-713 and RTF-601) (Long, 1961). The building consisted of a high-bay assembly room where ZPR-III was located, a control room, work room, fuel storage vault, laboratory rooms, and offices (see Figure 5-10). The assembly room was 45.5 ft. long by 42 ft. wide by 29.5 ft. high. It was kept at a negative air pressure of two inches of water. An airlock separated the assembly room from the rest of the facility. The facility walls were composed of two to three feet of concrete for shielding purposes. The wall separating the assembly and control rooms was reinforced to five feet of concrete to an elevation of 12 feet (Cerutti, 1956).



Source: Long, 1961

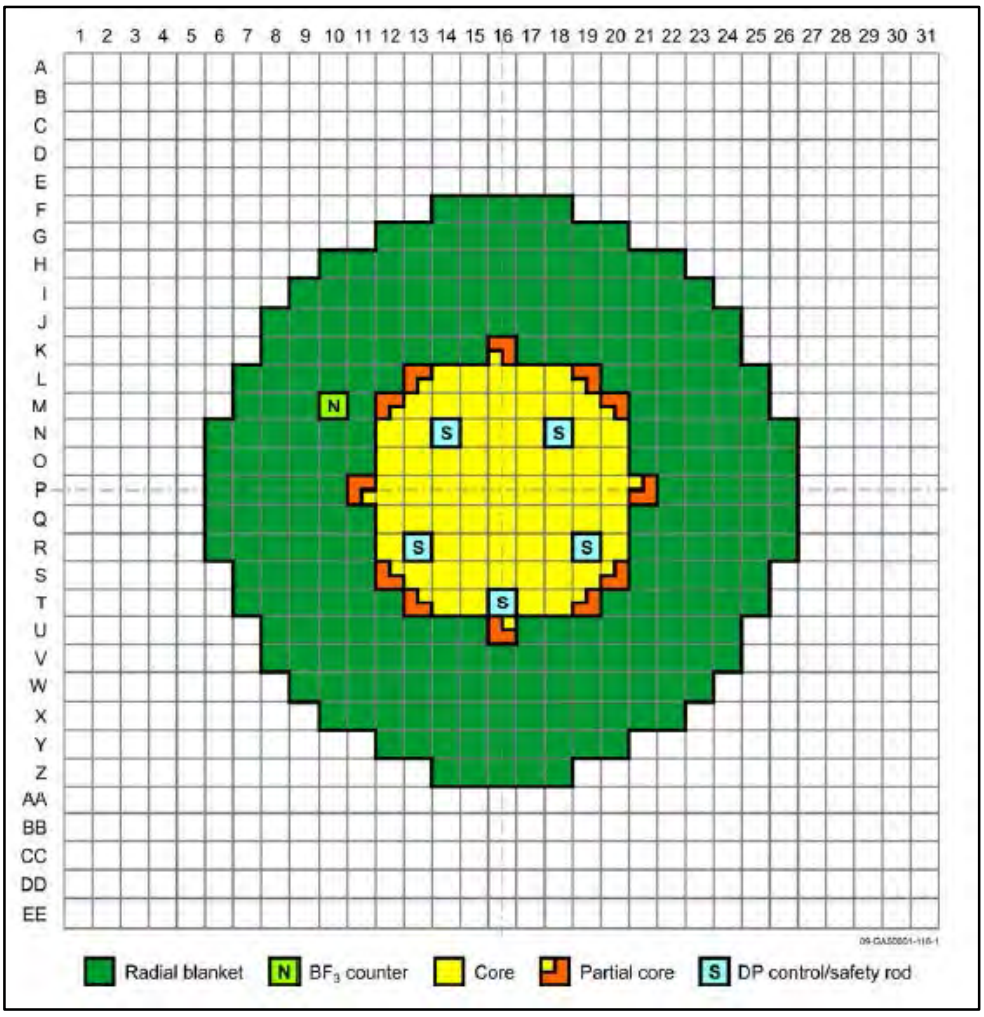
Figure 5-10: ZPR-III Floor Plan

The critical assembly machine was a horizontal split-table apparatus consisting of a large steel platform upon which two tables (or carriages), one stationary and the other movable, were positioned (see Figure 5-11). The matrix assembly was composed of stainless steel or aluminum square tubes, nominally 2.18 inches square, which were arranged horizontally on each carriage to form a 31-row-by 31-column square "honeycomb" matrix (roughly 67 inches by 67 inches). The 31 x 31 array of matrix tubes was held together by the steel structural members of the carriage. The movable carriage allowed for the assembly to be split in half - both as a safety measure and for convenience in loading and unloading. A matrix position in each core half was specified by three parameters: matrix half (S for stationary or M for mobile), row letter (A-Z and AA-EE starting from the top), and column number (1-31 starting from the left looking from the movable half towards the stationary half) (see Figure 5-12). Except during reactor operation, the tables were separated by five feet. For reactor operation, the movable table was driven against the stationary tale by a screw mechanism (Cerutti, undated). Free access of authorized personnel to the Assembly Room was strictly prohibited during operations.



Source: ZPR-III, 2010, PDF p. 5

Figure 5-11: ZPR-III Critical Assembly Machine

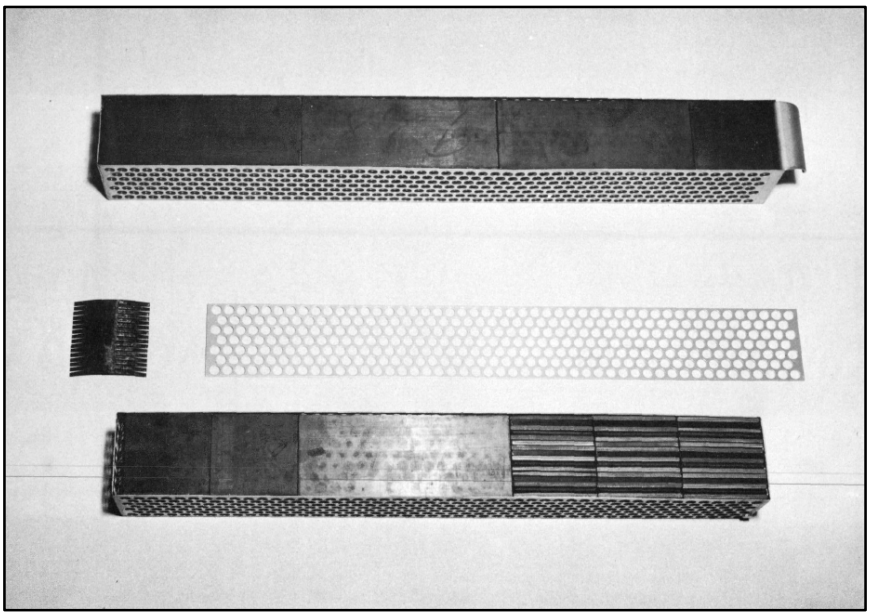


Source: ZPR-III, 2010, PDF p. 9

Figure 5-12: Example of a ZPR-III Core Loading

Each half of the assembly contained a multiple-curie polonium-beryllium source unless there was an inherent source like Pu-240 in the core. The source was driven in or out of the assembly by a motor drive. Each half also initially contained five safety-control rods. On a scram, the rods were driven by compressed air (Cerutti, 1956). Later, five additional safety and control rods were added for plutonium cores.

ZPR-III had no system to cool the matrix loading until the mid-1960s. At that time, a rudimentary forced-air cooling system was devised because plutonium fuel plates came into use which contained a substantial fraction of heat-emitting Pu-240. Before plutonium fuel was used at ZPR-III, there was only one thermocouple per assembly half; five more thermocouples per half were added when plutonium fuel was employed.



Source: Cerutti, Jan1956

Figure 5-13: Typical Loaded ZPR-III Fuel Drawer

The basic building blocks for the core region were composed of 1/8-inch-thick plates two inches high of various lengths that were loaded into 2-inch by 2-inch by 15-3/16 inch fuel drawers (see Figure 5-13) (Cerutti, 1956). Fuel-drawer loading took place in the work room adjacent to the vault where fuel plates were stored. The loading was performed under the direction of the project supervisor or chief physicist (Operations Manual, 1956). The desired average composition was achieved by loading the matrix with drawers containing rectangular plates of different materials (e.g., highly-enriched uranium, depleted uranium, graphite, plutonium). The uranium plates had a 1-mil-thick Teflon-based protective coating (called Kel-F) colored red with an iron oxide powder in the case of the enriched uranium. All plutonium plates were clad in stainless steel. The loading of uranium plates was performed on a downdraft table whereas plutonium loadings were done in a small glovebox that was moved to the table (Personal Communication, 2015c).

Before each drawer was loaded, a drawer-loading chart was made as a permanent record. Each drawer was confirmed for correct loading by a checker before it was removed from the work room. A special cart designed to carry no more than two fuel drawers was used to take drawers to the assembly room for assembly-loading under the direction of the project supervisor or chief physicist. A master assembly loading chart (also known as a matrix loading map), showing a cross-section of each half of the assembly, was kept in the control room. In each of the assemblies, a given matrix position had two drawers, a front drawer and a back drawer (the front drawers in the stationary and moveable halves were adjacent to the matrix interface between halves). Unloading was the reverse of the loading procedure with the custodian of the fissile material checking all material back into the vault as it was unloaded (Long, 1961).

The first reactor built in ZPR-III was assembled on October 18, 1955 with the assembly going critical on October 20. The critical mass was composed of 142.4 kg of U-235 - a core composition roughly that of the then-proposed EBR-II (Cerutti, 1956). Critical masses of the U-235 fuel assemblies ranged from 27 to 575 kg during ZPR-III's operating life (Long, 1963). Although ZPR-III used uranium fuel

plates exclusively during its earliest years, the facility was built with the anticipation of plutonium loadings so that few facility modifications would need to be made (Long, 1963). On June 28, 1961, the first sub-critical Pu-239 loading in ZPR-III began (Safety Monthly Report, 1961). Plutonium plates used at ZPR-III were composed of 95% Pu-239, 4.5 % Pu-240, and 0.5% Pu-241 (Long, 1963; Personal Communication, 2015c). In addition, MOX fuel was used for some assemblies, such as the SEFOR (Southwest Experimental Fast Oxide Reactor) assemblies. These assemblies were used to conduct tests for the experimental sodium-cooled test reactor located in northwest Arkansas and operated from 1969 to 1972 (when the program ended) (Plutonium Loading, 1965). In April 1969, ZPR-III began shipping fuel plates to ZPPR as it began winding down operations (Personnel Surveys, 1970). The reactor was put in cold standby in November 1970 and was later moved into the EBR-I facility for display.

5.1.1.3 Boiling Water Reactor Experiments I-V (BORAX-I – BORAX-V)

ATTRIBUTION: Section 5.1.1.3 was completed by Mike Mahathy, Oak Ridge Associated Universities. All conclusions drawn from the data regarding the BORAX Reactors were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the BORAX Reactors are explained in the associated text.

At the Argonne National Laboratory (in Lemont, Illinois) on June 2, 1952, a critical assembly mockup of a reactor was accidentally placed into a prompt critical reaction resulting in a steam explosion within the assembly. The critical assembly was damaged. Four operators received considerable radiation exposure. Nonetheless, an ANL staff member suggested that direct boiling reactors might be practical in that steam formation would help stabilize the reaction. On July 23, 1952, ANL authorized the design and construction of the first boiling water reactor to test that theory and reactor operating characteristics.

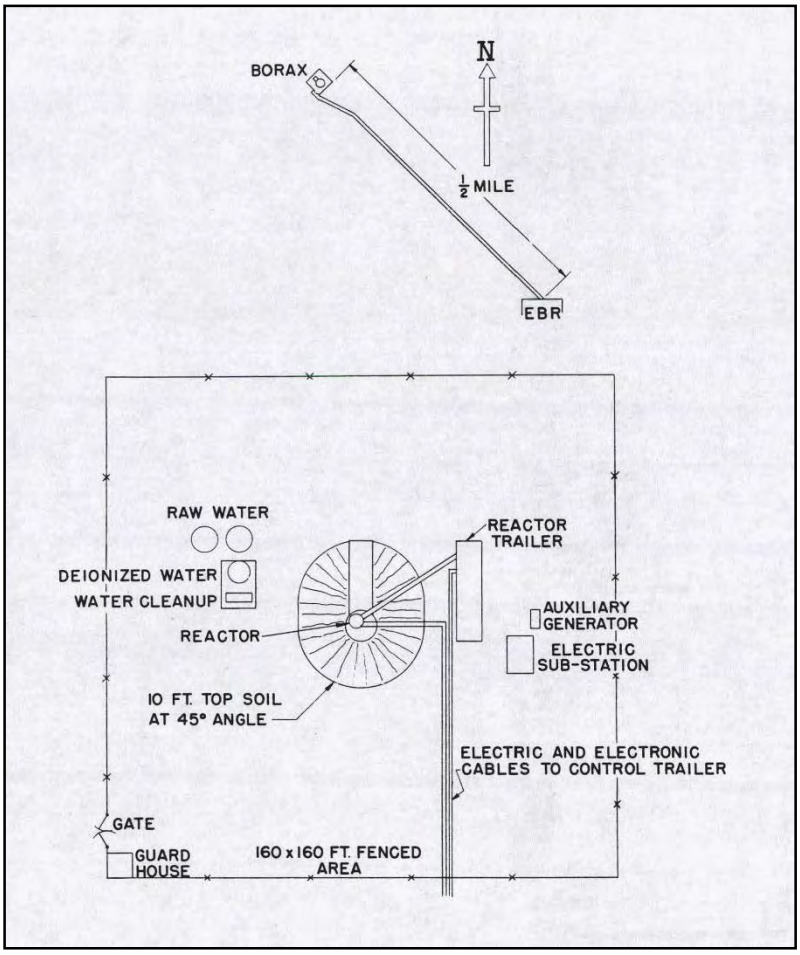
Five reactor configurations were built at the INL and were known as the BORAX (Boiling Water Reactor Experiment) series (NARA 071; Dietrich, 1954). The BORAX reactors were part of the EBR-I Complex. A control building for BORAX-V was constructed about one-half mile from that reactor (Borax-V Description, 1961). Figures 5-1 and 5-3 show the location of the BORAX reactors on the INL site and relative to the main part of the EBR-I Complex. The five BORAX reactors were built and operated in the periods listed in Table 5-1. Detailed results of the experiments are described in the following subsections.

Table 5-1: BORAX Reactors Operational Periods

Reactor	Dates of Operation	Source
BORAX-I	July 1953 to July 1954 (summer and fall months only)	Borax Experiments, 1954
BORAX-II	October 1954 to March 1955	Reactor Quarterly, 1954, SRDB 143253
BORAX-III	July 1955 to April 1956	Reactor Quarterly, 1956a
BORAX-IV	December 1956 to June 1958	Reactor Quarterly, 1956b, Defective Fuel Experiment, 1959
BORAX-V	March 1962 to August 1964	Progress Report, Sep1964

BORAX-I

In 1952, ANL designed an experimental reactor to be used for determining self-limiting characteristics of water-cooled reactors and the operating characteristics of boiling reactors. The reactor was fabricated at ANL and was shipped to and set up at ANL-W in early summer 1953. The reactor could only be used in summer and warm fall months because it was not totally enclosed within a structure. The first BORAX reactor was 2730 feet northwest of the entrance to EBR-I. The reactor was controlled from a trailer placed just outside the EBR-I entrance; during BORAX-I operation, an exclusion radius of one-half mile was maintained around the reactor (Dietrich, 1954). Figure 5-14 shows a drawing of the BORAX-I site plan.



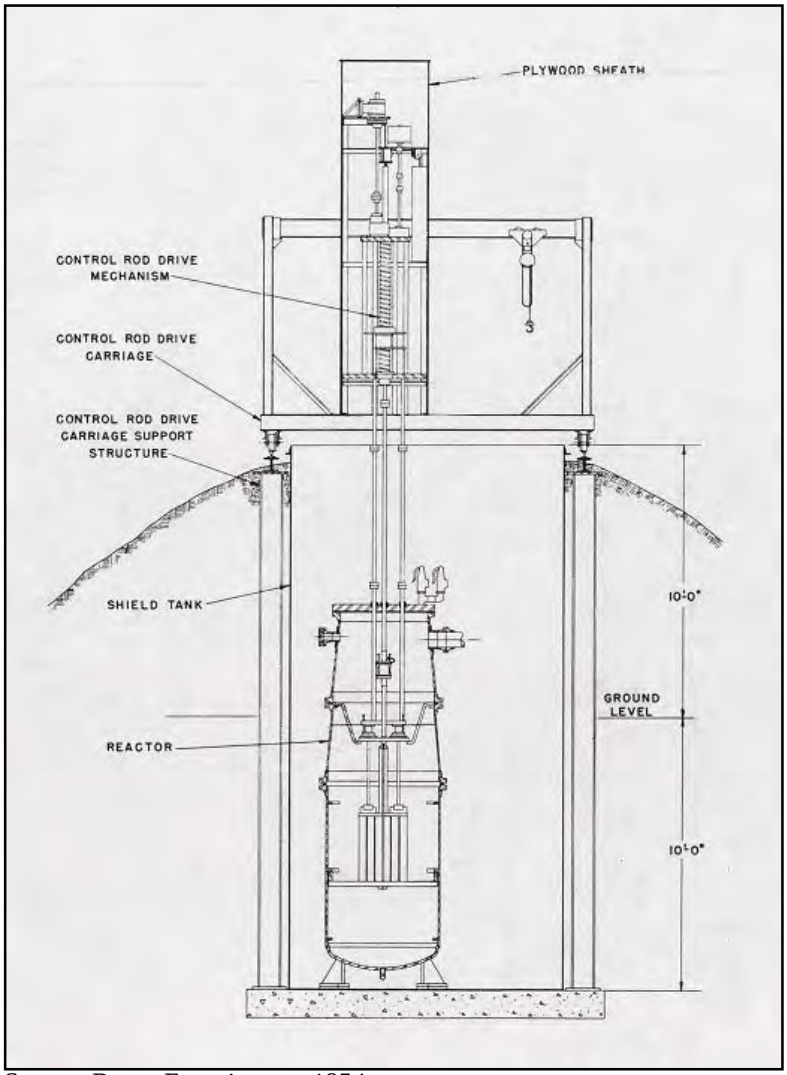
Source: Borax Experiments, 1954

Figure 5-14: BORAX-I Site Plan

The BORAX-I reactor core was built up from a lower grid into which were placed a combination of twenty-six fuel assemblies and ten non-fuel (dummy) plugs. Criticality was reached with the 26 fuel assemblies; the dummy plugs were used to fill the core grid to permit relatively-constant geometry for water flow. Fuel assemblies were made of curved fuel plates containing a U-235-aluminum-alloy core that was clad with aluminum. Each assembly contained 18 fuel plates joined to aluminum side plates to form units about three inches square. The fueled portion of the fuel plates was two feet in

length (Dietrich, 1954). The core was divided into quadrants with control rods located in the openings. The grouped rods were raised to increase reactivity but could also be rapidly forced downward by springs for reactor shutdown. The central control rod was moved downward to increase reactivity but could be ejected downward for rapid reactivity increases as needed by experiment criteria.

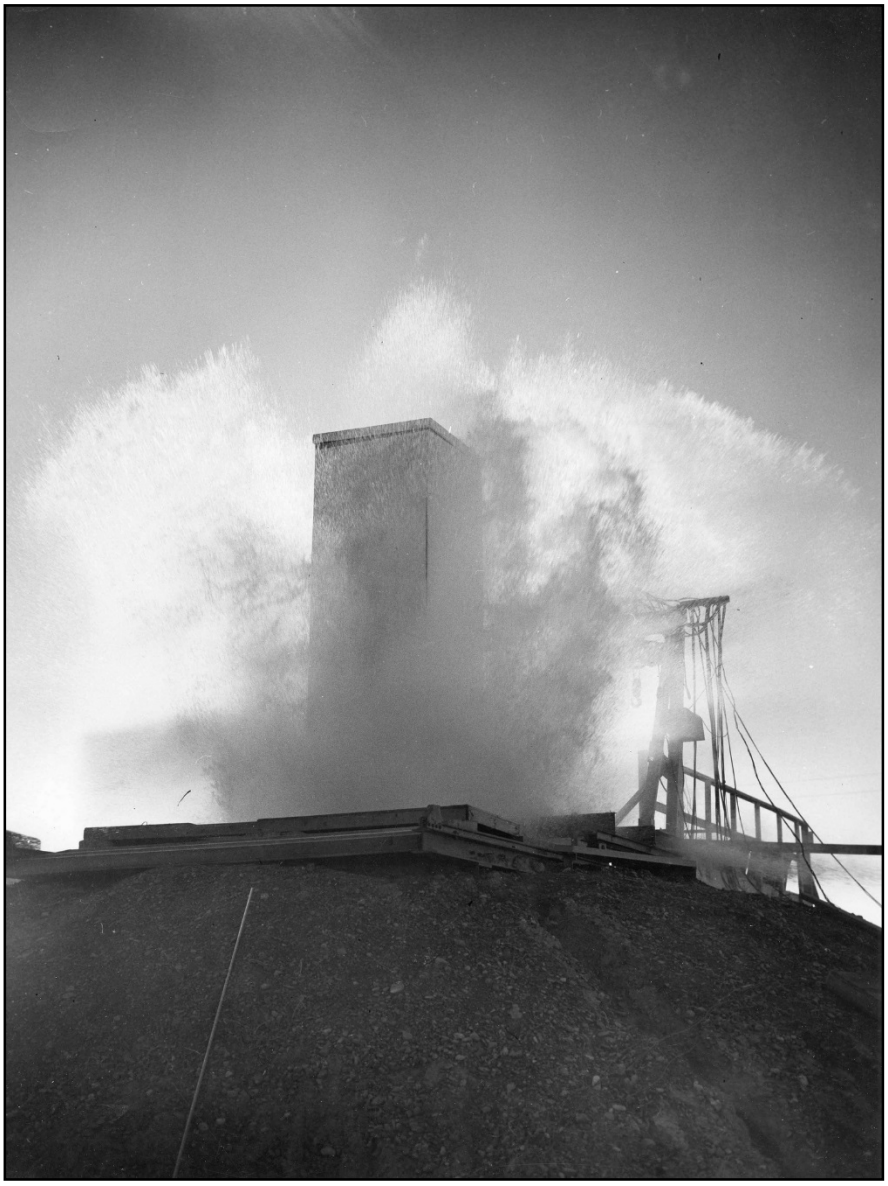
The reactor and associated mechanisms and structure are shown by vertical diagram in Figure 5-15. The reactor was placed in a 20-foot deep shielded tank sitting on a concrete slab ten feet below ground level. The reactor core was located approximately three feet below ground level. Sand and gravel were piled up around the shield tank at about a 45-degree angle which provided at least eleven feet of sand and gravel shielding for workers situated at ground level. That arrangement permitted area access at all times immediately after termination of a run. Sometimes when it was necessary to work on the reactor, the shield tank was filled with water; sub-tolerance radiation levels were reached with a few hours decay. Control rods were moved and positioned by an external drive mechanism.



Source: Borax Experiments, 1954

Figure 5-15: BORAX-I, Vertical Diagram

Through the summer and early fall of 1953, BORAX-I was put through a series of intentional excursions of periods shorter than 25 milliseconds, as well as periods of steady boiling operation; these latter tests covered operating pressures up to 130 psig (pounds per square inch, gauge). These excursions caused steam to escape from the reactor, somewhat resembling a geyser (Figure 5-16). While it was not possible to determine all the characteristics of boiling in such a short time, researchers at ANL-W did gain conclusive proof that steam formation was an effective and rapid power-limiting process. They further deduced that steam would protect properly-designed reactors against short millisecond reactivity excursions (Transients, 1954).



Source: Till and Chang, 2011

Figure 5-16: Steam from Excursion of BORAX-I

By the summer of 1954, ANL-W staff had designed a replacement for BORAX-I. However, they proposed that before the first reactor was replaced, it be subjected to a single, very quick, destructive excursion to determine the inherent safety under extreme conditions. After discussion with the AEC Reactor Development Division and the Advisory Committee on Reactor Safeguards, ANL-W staff decided that such a final experiment was worthwhile. Consequently, BORAX-I was deliberately destroyed in July 1954. ANL-W staff encountered a hitch at the start of the destructive test. A test run was successful without complication. However, upon inspection, they learned that a fuel plate had bowed out so that it was rubbing against the control rod. The staff was under pressure to start the final test. Looking down into the reactor, they could see that the fuel subassemblies were giving off a Cerenkov glow. Even so, staff members lifted the bowed fuel subassembly up out of the reactor, fixed it, and put it back in place. They vacated the reactor and were successful in starting the test.

At the moment of explosion, fuel plate fragments were scattered for a distance of 200-300 feet, but there was no widespread dangerous dispersal of radioactive materials (ANL-W History, 2006). Gold foils had been placed at various locations in the reactor core so that staff could determine the power the reactor attained before exploding. These foils were about one centimeter square and attached to the outside of the fuel subassemblies. The foils were embossed with numbers to record their placement location in the reactor. Over several days, reactor workers were rotated into the reactor pit to recover the gold foils. Site staff also retrieved materials and usable equipment from the reactor and surrounding area over the next two months (Haroldsen, 2008).

The final test revealed that the predictions of total energy and fuel plate temperatures were considerably too low. Instead of melting a few fuel plates, the test melted a major fraction of the entire core. The discrepancy was attributed to the uncertainties of extrapolation. The results of this energy liberation in regard to peak pressures and explosive violence were in a region without previous experimental data.

ANL published a video in March 2013 about BORAX-I called Borax - Safety Experiment on a Boiling Water Reactor. This includes actual footage taken during experiments and provides more detail on reactor installation and operation (ANL, 2013).

BORAX-II

BORAX-II was designed and built by ANL and operated at ANL-W to replace BORAX-I. The goal was to extend the BORAX-I experiments on the transient and steady-state operation of boiling reactors with a new reactor that more closely approximated the characteristics of a practical power reactor operating on the boiling principle. BORAX-II was larger than BORAX-I and designed for operation at higher pressure (300 psig). It was installed in August 1954 on a new site about 200 yards east of the BORAX-I site (see Figure 5-3). The first BORAX-II operation was conducted in October 1954 (Reactor Quarterly, 1954). Figure 5-17 shows a photo of BORAX-II.

With BORAX-II, ANL tested new core combinations using varying enrichments of U-235 in the metal fuel plates. The boiling water system was operated at 300 psig, making it essentially a power experiment. Initially, BORAX-II had no turbine generator; thus, no electricity was produced. As with BORAX-I, the energy produced was released in the form of steam.

The reactor tank was made of stainless steel and was installed completely below ground level in a concrete-lined pit that could be flooded with water for shielding for reactor work conducted during

shutdown. A pit was provided with a removable lid of three-foot-thick concrete. The control rod mechanism was mounted above the pit with extension rods passing through the concrete to drive the control rods. Much of the ancillary equipment used outside the reactor tank was originally used on BORAX-I. Those items were decontaminated and repaired after the final BORAX-I destructive experiment. Several new instruments were required.

A machinery pit containing feed pumps, water preheat circuitry, and steam controls was located adjacent to the reactor pit but shielded from the pit by five feet of concrete. The deionizer system for the reactor water, and the nuclear and electronic instrumentation for recording results of transient experiments were mounted at ground level. The entire installation was covered by a 40-foot by 60-foot Butler building. The reactor was operated remotely from a control trailer one-half mile south of the reactor.

Source: Till and Chang, 2011

Figure 5-17: BORAX II

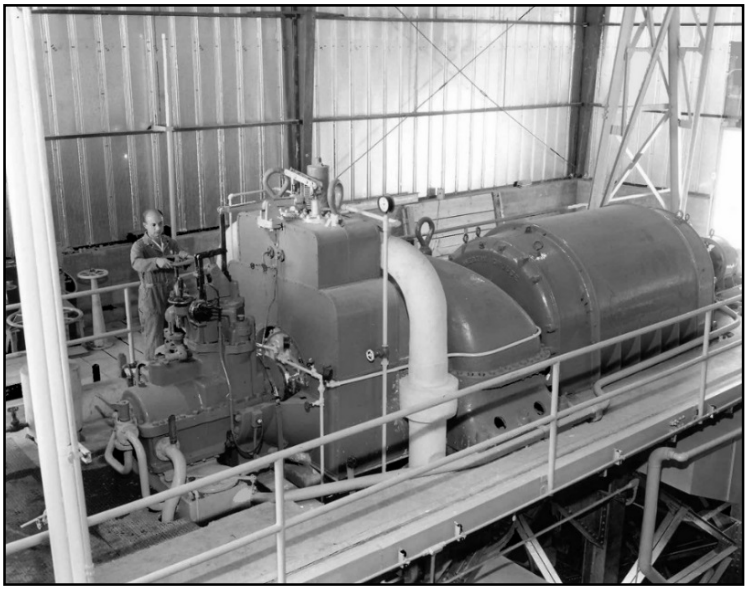
The reactor core was mounted near the bottom of the reactor tank. The core consisted of an adjustable number of modified MTR-type fuel elements held in a closely-packed assembly by an aluminum grid. The core was completely submerged in water that served as coolant, moderator, and reflector. Like BORAX-I, water was circulated in the reactor tank during reactor operation by natural convection (Reactor Quarterly, 1954). The active sections of the fuel elements had the exact external dimensions as standard MTR fuel elements, but they contained only ten fuel plates, with a total loading of 93 g U-235 per element. The reactor core was divided into quadrants by four pairs of cadmium control rods

with a cadmium control rod at the center of the core. The five control elements were operated independently.

BORAX-III

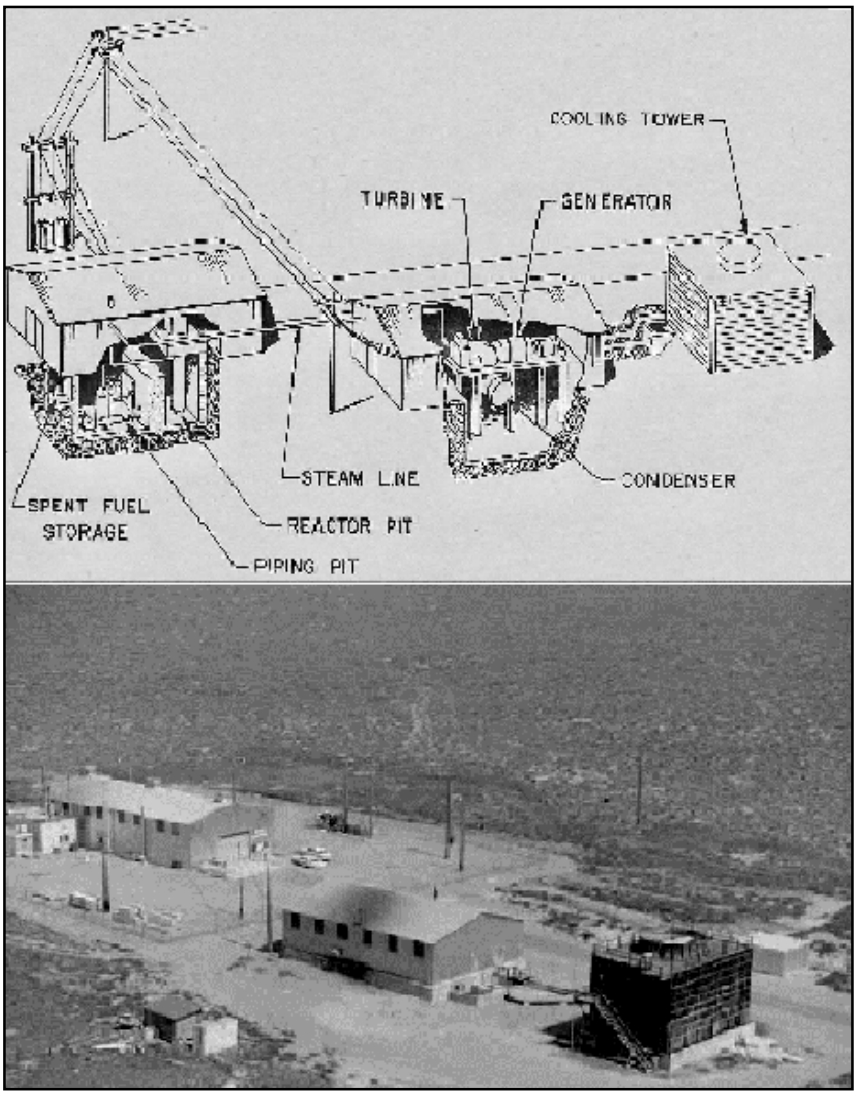
In 1955, ANL replaced the reactor core and added a turbine-generator to BORAX-II to test power distribution and whether turbine contamination would be a significant problem in a boiling water reactor. The revised reactor configuration was designated BORAX-III. The core consisted of uranium fuel enriched to about ninety percent U-235. The uranium was alloyed with aluminum. The uranium-aluminum alloy was then clad with aluminum. Plate-type fuel elements were used. The fuel plates had a short span to avoid distortion from pressure resulting from steam formation in the coolant channels during transient tests (Revised Reactor Configuration, 1956; Fuel Plate Fabrication, 1958). The reactor vessel, made of stainless steel, was unchanged from BORAX-II. New quadrant control rods fabricated from boron-hafnium were added to replace the BORAX-II control rods. The center control rod made from cadmium was not replaced. The reactor, control rod mechanisms, water control systems, canal for spent fuel storage, and steam piping was housed in the sheet metal building constructed for BORAX-II (Control Rods, 1957).

The mission of BORAX-III was to generate electrical power for Arco, Idaho on a one-time basis. A turbine generator, condenser, and equipment to return cooled water to the reactor were installed in a new building. The steam pipe in the reactor building outlet was connected to the turbine via a switching mechanism (see Figure 5-18). Water was condensed and pumped through an adjacent cooling tower, then back to the reactor building for use as feed water (Water Pumps, 1956). Figure 5-19 shows a diagram and photo of the BORAX-III facility. The steam could also be diverted as outdoor exhaust by closing the turbine switch.



Source: ANL, undated

Figure 5-18: BORAX-III Turbine



Source: Till and Chang, 2011

Figure 5-19: BORAX-III Facility

ANL-W workers were successful in preparing and operating BORAX-III to generate electricity for an entire city for the first time anywhere in the world. On July 17, 1955, electricity produced by BORAX-III supplied the town of Arco, Idaho with its entire supply of electrical power (500 kW), generated 500 kW to power the BORAX facility, and 1000 kW to power the Central Facilities Area at INL.

ANL West continued to operate the BORAX-III reactor until April 7, 1956 to test operating characteristics during steady-state operation and to identify fission and activation product contaminants in the plant's air and water (Reactor Steady-State Operation, 1956). The core was removed in June 1956 in preparation for fuel loading for BORAX-IV (Reactor Quarterly, 1956a). The BORAX-III reactor was operated for a total of 1170 hours.

BORAX-IV

ANL-W replaced the BORAX-III core with a core of a maximum seventy-two ceramic uranium-thorium oxide fuel elements to test and demonstrate the feasibility of stable operation with a fuel that: (1) could operate at higher temperatures than the uranium core; and (2) was considered less reactive with water coolant in case of cladding rupture (Fuel Fabrication, 1957; Fuel Element Testing, 1958; Reactor Quarterly, 1956b; Fuel Element Testing, 1958). The new reactor configuration was known as BORAX-IV. Other than the core, all other BORAX-III components were unchanged with BORAX-IV. Like the BORAX-III reactor, BORAX-IV was operated at 300 psig and had the potential to produce 2.5 MW of electricity. Operation of BORAX-IV began on December 3, 1956.

BORAX-IV was first brought to criticality under atmospheric pressure, with water temperatures ranging from cold to boiling (93 degrees C at the elevation of BORAX-IV). It was operated with this core until April 17, 1957. A revised core with uranium-thorium oxide elements was installed in late April 1957. BORAX-IV was operated with that core at 300 psig and 216 degrees C from May 1957 intermittently through December 5, 1957. BORAX-IV also produced measurable quantities of the artificial, thorium-derived fuel, U-233.

The core was again revised in order to increase the maximum power for continuation of stability studies. Boron steel poison rods were removed with ten new elements added to the periphery of the core. On February 19, 1958, the reactor was restarted with the revised core. BORAX-IV was operated during early 1958 to evaluate the effect of operating a direct-cycle boiling water reactor with a fuel element defect, and for locating defective elements in the core. The initial test performed in February was done with simulated ruptured elements (Request for Pellets, 1957; Defective Fuel Experiment, 1959). On March 11-12, 1958, the BORAX-IV reactor was operated at 2.4 MW, even though a large number of fuel elements contained defects through the cladding (Borax-IV Test, 1959). Measurements made during the reactor operations included radioactivity levels of the steam plant equipment; quantitative determination of the fission gases Xe-138 and Kr-88 that were released through the air ejector; analyses of reactor water, condensed steam before the turbine, and condensed steam for fission products; and area contamination downwind from the reactor. BORAX-IV was operated until June 1958. Clean-up and salvage operations began in July and continued through 1959 (Safety Monthly Report, 1958; Air Sample Data Sheets, 1965). Some equipment was salvaged for use with BORAX-V. The reactor vessel was entombed in place using sand capped with concrete (Clean-up Request, 1959; Arave, 1992). Most other equipment and facilities were removed and buried in 1960 and 1961. Fuel plates were shipped to CPP at INL for processing. Personnel involved in the test were monitored for external radiation, and for intakes of fission products by urinalysis; urinalysis detected no activity above background (Graham, 1959).

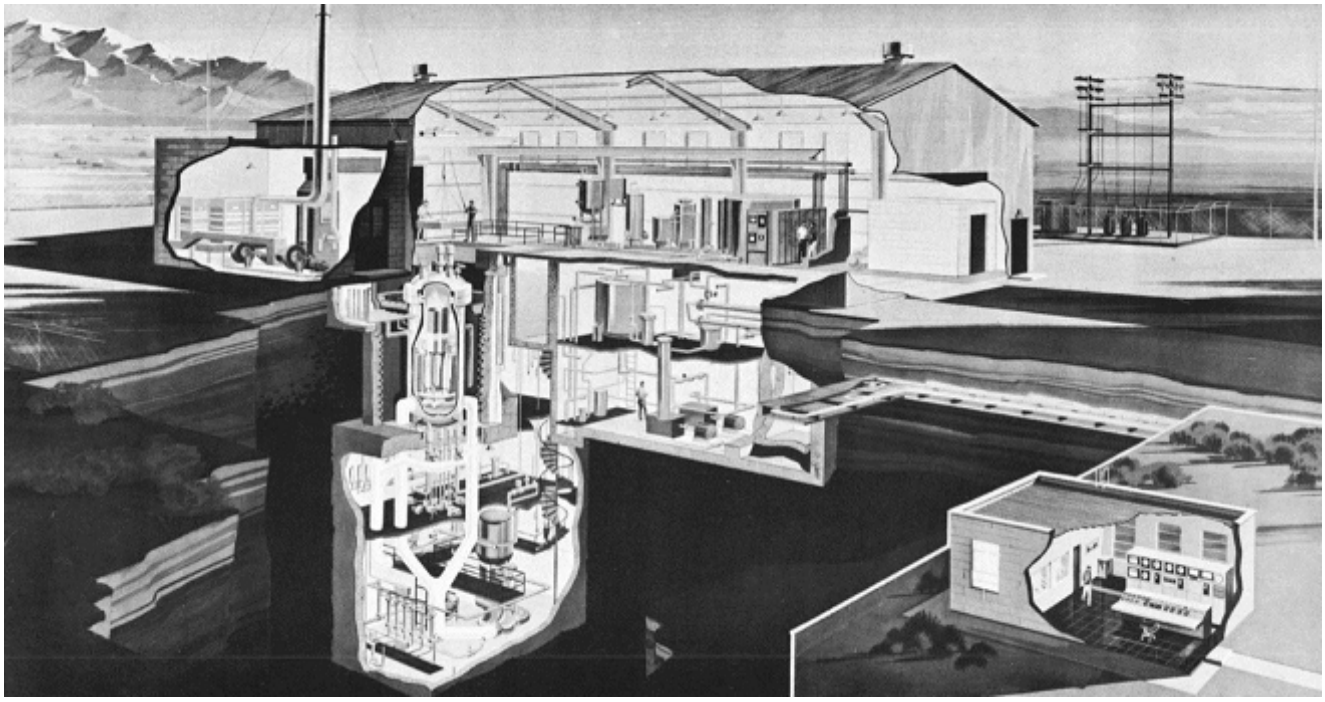
BORAX-V

In 1957, the concept for BORAX-V was derived by ANL's Reactor Engineering Division to test nuclear superheating concepts and to advance the art of boiling-water-reactor design. The greatest departure in design for BORAX-V was in the superheater assembly.

Some of the plant equipment used in the previous BORAX experiments was incorporated in the BORAX V facility, but a new reactor building and reactor assembly with new reactor vessel were constructed. A reactor control building was constructed about one-half mile away (Borax-V

Description, 1961). Figure 5-20 shows a drawing of the reactor facility; the control building shown in the lower right is one-half mile away from reactor (Borax-V Design and Hazards, 1961).

The reactor vessel was a cylinder with ellipsoidal heads made of carbon steel clad internally with stainless steel. The top portion of the vessel was used as a steam dome. The core is centered about four feet from the vessel bottom. Control rods made of Boral and canned in stainless steel were manipulated from below the vessel (Uranium Core Configurations, 1961; Holtz, 1964).



Source: Brunson, 1959

Figure 5-20: BORAX-V Facility

BORAX V was designed to support three separate uranium core configurations: (1) a boiling core without superheater; (2) a boiling core with a centrally-located superheater; and (3) a boiling core with a peripherally-located superheater. In each case, ANL assumed it was possible to operate the reactor with either natural or forced circulation of water through the core. Water served as the moderator in both the boiling and superheater regions of the core and as coolant in the boiling region. The superheater was cooled by steam (Uranium Core Configurations, 1961; Holtz, 1964).

Each of the three cores was 24 in. high with a maximum diameter of 39 in. (depending on the number of fuel assemblies loaded). The central superheater contained twelve fuel assemblies in the central position, while the peripheral superheater contained sixteen fuel assemblies, and four in the middle of each outer row (Borax-V Design and Hazards, 1961). The three core structures were readily exchangeable after removal of fuel and control rods. A core structure was easily withdrawn from the vessel as a unit by a remote- controlled crane and stored in the dry-storage pit.

Each of the three cores was used in the reactor. Subcritical experiments were performed on each core. Two configurations of boiling core without a superheater operated intermittently from March 1962 through February 1963. Two configurations of boiling core with a centrally-located superheater operated between April and November 1963. The boiling core with a peripherally-located superheater was operated between April and July 1964. At the end of July 1964, some fuel elements were replaced with prepared defective fuel elements to test power generation using defective fuel. The core was operated until August 14, 1964, producing 266 MW with the defective fuel. That final experiment in BORAX V demonstrated that negligible contamination to turbo-generator equipment resulted from operating with an experimentally-defective fuel element in the superheated core. The reactor was shut down and disassembled by the end of September 1964. The vessel was stored in the basement of the reactor building. The vessel used in BORAX-II/III/IV was also transferred to that basement for long-term storage (Progress Report, Sep1964).

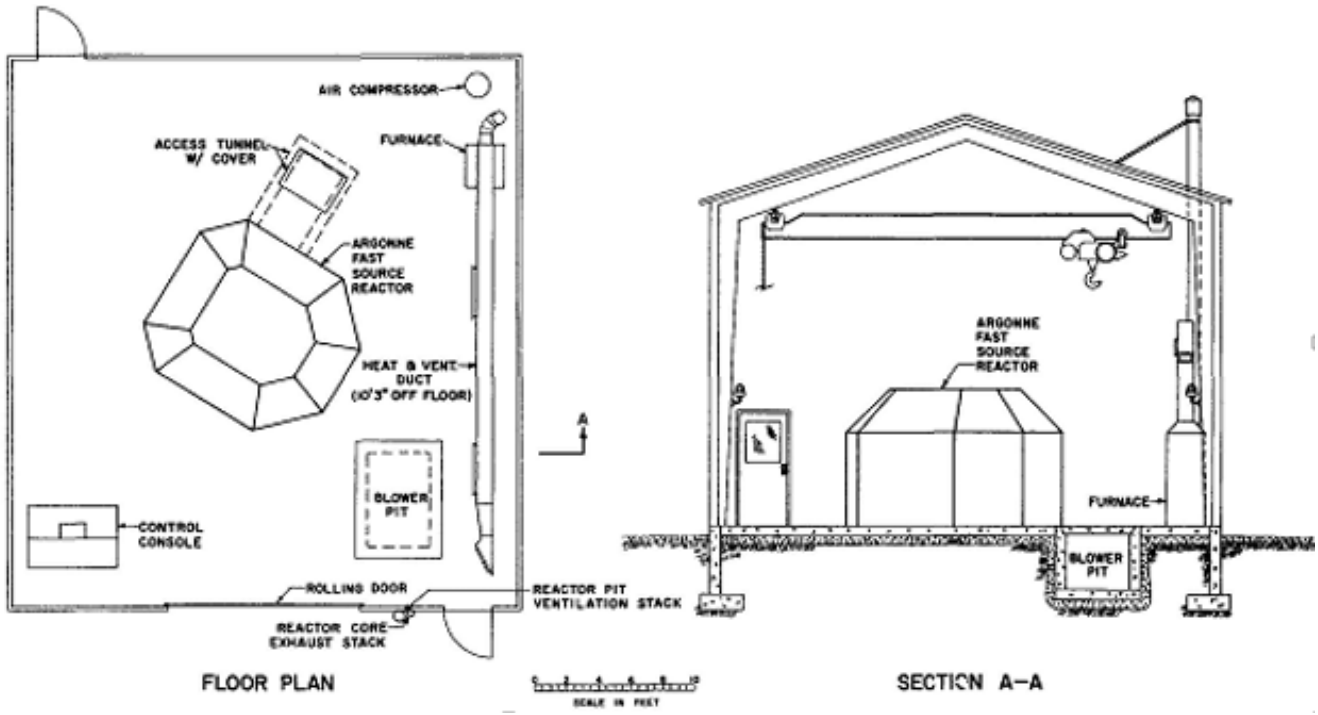
5.1.1.4 Argonne Fast Source Reactor (AFSR) (1st location: 1959-1970)

ATTRIBUTION: Section 5.1.1.4 was completed by Mike Mahathy, Oak Ridge Associated Universities. All conclusions drawn from the data regarding AFSR (first location) were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the AFSR (first location) are explained in the associated text.

The AFSR was designed and built in 1958 near EBR-I and ZPR-III. AFSR was placed in operation in October 1959 with a design power of one kilowatt. AFSR was a laboratory-scale reactor designed to supplement ANL-W's existing facilities by providing a readily-available source of reproducible fluxes of fast and thermal neutrons for developing, testing, calibrating, and standardizing various counters; preparing radioactive metallic foils for developing counting and radiochemical techniques; testing complex experimental systems before operation in other reactors; and developing potential experiments in the fast reactor field (Brunson, 1959; IOO Information Packet, 1962).

AFSR was housed in a prefabricated Butler-type building measuring 32 x 32 x by 20 feet high. The reactor incorporated a cylindrical core of solid, highly-enriched uranium approximately 4.5 inches in diameter by 4.25 inches high. The core was divided into upper and lower sections. The upper section constituted about three-fifths of the critical material. It consisted of two uranium discs separately canned and held in position by the upper cage. There was a horizontal one-half-inch diameter "glory hole" at the approximate reactor center line. The lower section of the core and the supporting blanket section to which it was attached comprised the safety plug which could be raised and lowered using an air cylinder. The lower core section was composed of discs of various thicknesses so that the core loading could be varied. Discs used in the lower section were attached to the supporting blanket section by means of a second stainless steel cage (Brunson, 1959).

The reflector was made of solid depleted uranium with a minimum thickness of eight inches. The outer form was cylindrical. The reactor was freestanding on the reactor floor inside a shield of high-density concrete. A graphite thermal column 4 x 4 x 6 feet was provided for experiments. All control and safety mechanisms were located in a pit beneath the reactor (Brunson, 1959). A 15-curie polonium-beryllium neutron source was located in the reflector. The source was manually retracted into the shield by means of a mechanical drive (Brunson, 1959; IOO Information Packet, 1962; Summary Report, 1960). Figure 5-21 shows the reactor building layout. Figure 5-22 shows a photo of the building.



Source: Brunson, 1959

Figure 5-21: AFSR Building Layout (1st location)

There were three experimental holes in the reflector. All experimental and instrument holes through the shield were provided with stepped plugs. The reactor biological shield was composed of magnetite-aggregate concrete with a density of 215 pounds per cubic foot. The nominal shielding thickness was 4.5 feet around and above the reactor. The thermal column was shielded by a one-inch steel plate faced with 1/8-inch of Boral. As the sub-reactor pit was not to be occupied during operation, the region below the reactor was only partially shielded. A minimum of two feet of magnetite concrete was provided between the pit and the floor outside the shield to absorb neutrons scattered off the pit floor.

Source: Till and Chang, 2011

Figure 5-22: AFSR Building (1st location)

AFSR was shut down in October 1961 when staff discovered deterioration in the lower fuel section. After disassembly of the lower section, small pimples were visible under the thin nickel can. The fuel was shipped to ANL (Chicago) where replacement cans were fabricated and installed around the core pieces. The fuel was returned in two shipments, one by the end of May 1962 and the other in July 1962 when full use of AFSR was restored (Reactor Operation, 1962).

Various documents discuss some of the irradiations and tests performed at AFSR (Summary Report, 1960; Progress Report, 1962; Borax Superheater, 1965; Progress Report, 1965; AFSR, 1965; Progress Report, 1967; Annual Report, 1968). To accommodate the close of the EBR-I and BORAX reactor areas, AFSR was moved to a new location adjacent to the ZPPR facility in the EBR-II Complex in fall 1970.

A small, consistent number of people (4 to 8) periodically worked at AFSR. The work area for personnel is identified in each individual's monitoring records (AFSR Quarterly Reports, 1971).

5.1.2 EBR-II Complex

EBR-II Area primary radiological facilities during the period under evaluation (April 10, 1951 to December 31, 1979) included:

- Transient Reactor Experiment and Test Facility (TREAT) (see Section 5.1.2.1)
- Experimental Breeder Reactor-II (EBR-II) (see Section 5.1.2.2)
- Fuel Cycle Facility (FCF) (see Section 5.1.2.3)
- Fuel Assembly & Storage Building (FASB) (see Section 5.1.2.4)
- Inspection & Test Facility (ITF) (see Section 5.1.2.5)
- Hot Fuel Examination Facility- North (HFEF-N) (see Section 5.1.2.6)
- Zero Power Plutonium Reactor (ZPPR) (see Section 5.1.2.7)
- Laboratory & Operations Building (L&O Building) (see Section 5.1.2.8)
- Argonne Fast Source Reactor (AFSR) (2nd location, 1970 to 1993 (see Section 5.1.2.9)
- Miscellaneous Facilities (see Section 5.1.2.10)

NOTE:The AFSR was located in the EBR-I Complex until the late-1970s when it was relocated to the EBR-II Complex.

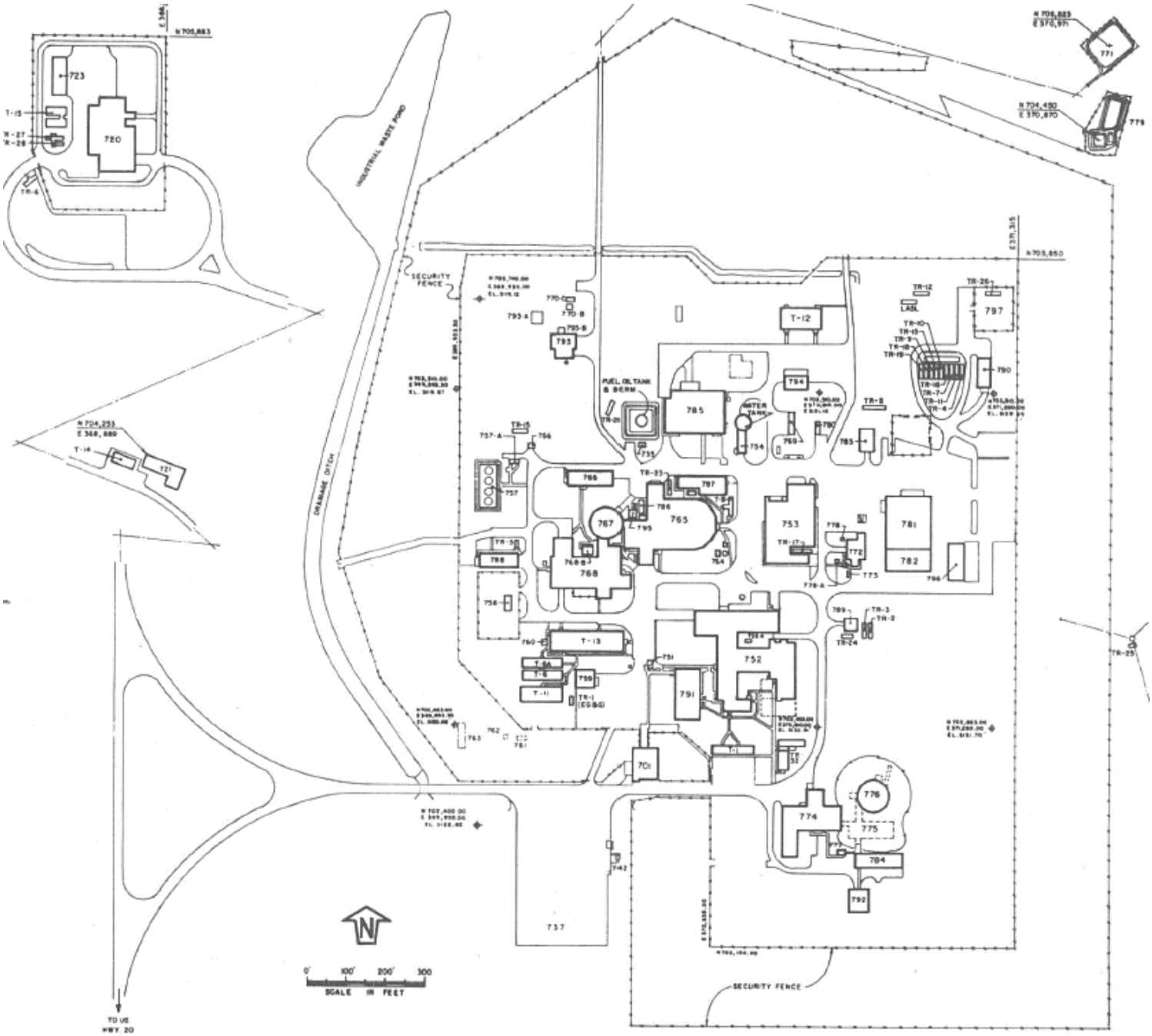
Facilities in the EBR-II Complex were not constructed until several years after the EBR-I Complex facilities began operation. The first radiological operations in the EBR-II Complex took place when the TREAT Reactor went into operation in 1959. TREAT is approximately one mile (1.6 km) outside the main part of the EBR-II Complex (Site Plan, 1981). The first occupancy of facilities within the main part of the EBR-II Complex occurred in 1960, when ANL received “beneficial occupancy” of areas in the EBR-II Power Plant and EBR-II Reactor while the construction contractor finished constructing the initial EBR-II Complex facilities. By November 1960, ANL had complete occupancy of the EBR-II Power Plant and EBR-II Reactor. On July 10, 1961, the construction contractor finished all deficiency items and ANL accepted the completed buildings (Summary Report, 1961, PDF p. 103).

Figure 5-23 shows an aerial photograph of the EBR-II Complex in the early 1960s.

Source: ANL-PIO, undated

Figure 5-23: Aerial View of the EBR-II Complex (circa early-1960s)

Figure 5-24 below shows the physical location of each of the EBR-II Complex buildings in 1981. NOTE: the distance to the TREAT buildings (upper left corner) is not to scale; they are actually about one mile away.



Source: Site Plan, 1981

Figure 5-24: Map of EBR-II Complex in 1981

5.1.2.1 Transient Reactor Experiment and Test Facility (TREAT)

ATTRIBUTION: Section 5.1.2.1 was completed by Jason Davis, Oak Ridge Associated Universities. All conclusions drawn from the data regarding TREAT were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to TREAT are explained in the associated text.

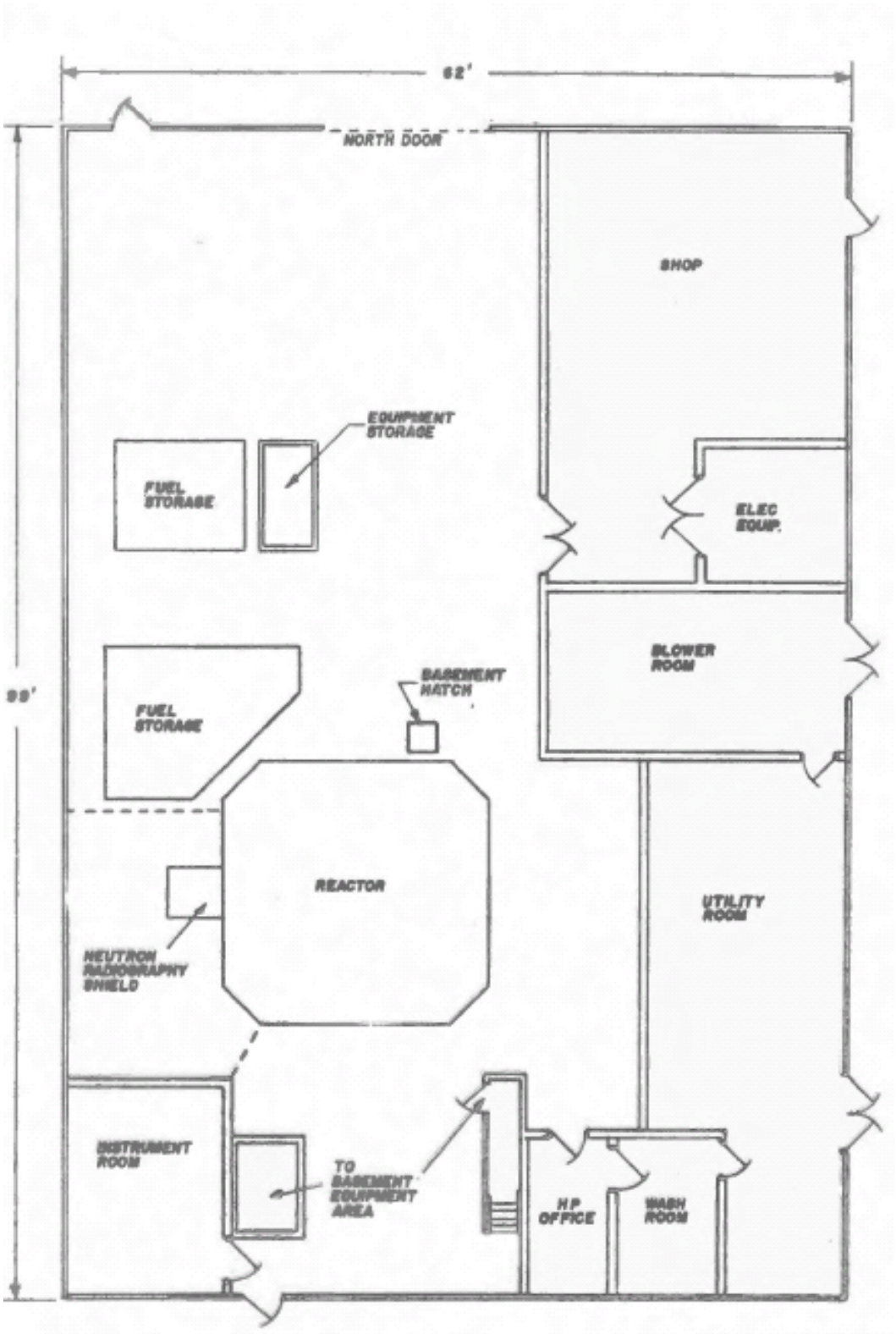
The Transient Reactor Test (TREAT) Facility was an air-cooled, thermal, heterogeneous system designed to evaluate reactor fuels and structural materials under conditions simulating various types of nuclear excursions. TREAT was designed for two modes of operation: steady-state and transient. In steady-state mode, it served as a large neutron-radiography facility (TREAT Manual, 1971). In the transient mode, TREAT's primary objective was to provide quantitative data and visual information on the mechanism of the melting of fast reactor fuel elements by nuclear heating analogous to a power excursion in a fast reactor core.

The TREAT complex was comprised of a reactor building and a control building located about 4250 ft. and 1750 ft., respectively, northwest of the EBR-II containment vessel. The site was selected because the topography permits an unobstructed view between the two buildings. Construction was started in February 1958 and completed in early November 1958. The reactor first achieved criticality on February 23, 1959.

The Reactor Building, shown in Figure 5-25, was an aluminum-sided steel-frame structure featuring a high-bay section and adjacent service wing. The high-bay section (35-ft. ceiling) contained the reactor, fuel storage pit, instrument room, and basement sub-reactor and equipment rooms. The control rod drive mechanisms were located in the sub-reactor room, which measured 16 ft. square by l3.5 ft. high. An adjacent equipment room housed auxiliary apparatus (e.g., pumps, storage tanks, piping) required for experiments. Access to both rooms was by a stairway from the main floor into the equipment room, and through a door into the adjacent sub-reactor room. In addition, there was an escape ladder leading from the sub-reactor room to the main floor.

Fuel transfer was accomplished via a lead-shot and steel-shielded cylinder, known as a coffin. The motor and gearing for performing refueling operations were mounted at the top of the coffin and controlled from a single portable control box. The base of the coffin was positioned above the desired opening in the radial slot of the rotating plug and locked in place with an indexed carriage. The installation of control-rod fuel assemblies was accomplished by removing the adjacent fuel assemblies and rotating an internal tube within the coffin. Operations within the core were accomplished with the aid of a shielding window block in the slot of the rotating shield plug when radiation levels exceeded those permissible for direct viewing through the slot. A periscope was also available for viewing operations within the core (TREAT Facility, 1971).

The Control Building was a single-story, concrete-block structure that contained the control panels and instrumentation for remote-controlled reactor operation.



Source: TREAT Facility, 1971

Figure 5-25: TREAT Reactor Building Plan

In the steady-state mode of operation, TREAT served as a large neutron radiography facility. The neutron radiography facility (Figure 5-26) consisted of a neutron collimator and a massive shield for positioning the radiographic subject. The collimator was a double-tapered, boral-lined channel extending from the edge of the core to the center of the radiography shielding, which was located outside the normal reactor shielding. A massive rotating shutter provided off-on control of the neutron beam. A timer-controlled air solenoid opened and closed the shutter as required for particular samples. The beam covered an area 4 in. x 24 in. at a density of approximately 107 neutrons/cm^2-sec when the reactor was operating at 50-90 kW of steady-state power. The cast-lead radiography shielding permitted immediate access to the facility while the reactor was operating at steady state and a highly-radioactive subject was in position. Subjects to be radiographed were lowered into the shield through an opening in the top of the shield. Highly-radioactive subjects were lowered from a coffin that was positioned on top of the shield (TREAT Manual, 1971).

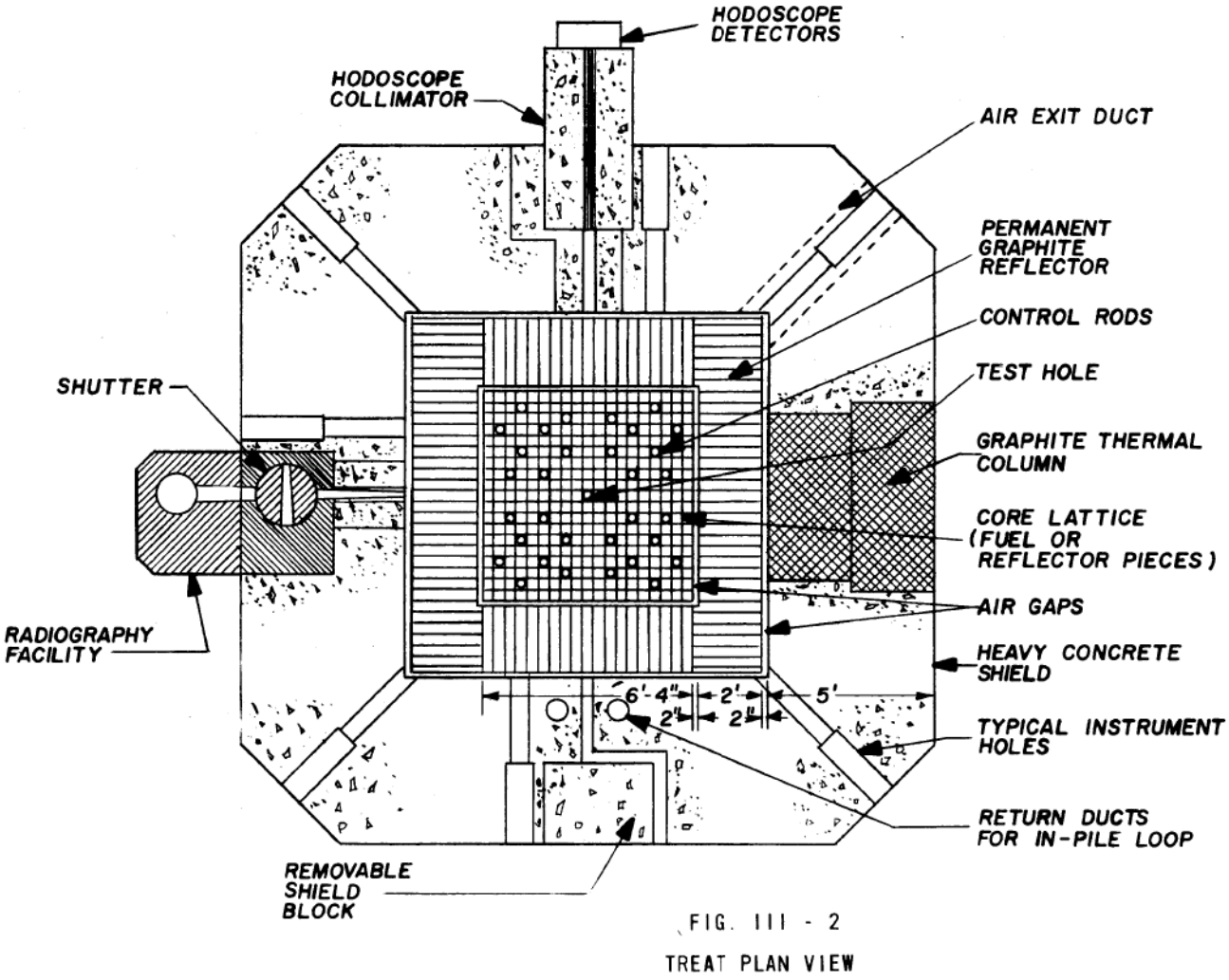
The Mark-I integral sodium loop was intended for experimental use in TREAT meltdown experiments of relatively low-melting-point metallic fast reactor fuels contained in flowing sodium. These loop experiments were sponsored by the Reactor Analysis and Safety Division (RAS) of ANL-East, and were designed to simulate either loss-of-coolant flow accidents or transient-overpower accidents in the Fast Flux Test Facility (Cook, 1975). The loops were used successfully and routinely for such tests. However, the design pressure of 30 atm at 500°C and steady-state temperature limitation below 400°C were not adequate for Liquid Metal Fast Breeder Reactor (LMFBR) oxide-fuel transient safety experiments. The counterflow design of the Mark- I loop was required to reduce the loop size to fit within a single TREAT fuel-element can. However, this counterflow feature made measurement of pressure, flow, and temperature difficult.

The Mark-II integral loop was used for testing LMFBR-type oxide-fuel elements in the TREAT

Facility under transient conditions of coolant flow, coolant temperature, and fuel element power representative of those that may occur in a reactor. The loop was positioned within the TREAT reactor so that the test section penetrated into the reactor core region. The apparatus arrangement was such that no portion of the loop became a permanent or integral part of the reactor itself, with the expectation that the Mark-II loop would not affect reactor operation or reactor control once routine provisions were made to compensate for the removal of two reactor fuel elements (Stephens, 1976).

During 1973 and 1974, several Mark-IIA loop experiments were sectioned in the HFEF-South argon cell as part of the HFEF post-test support of the RAS-TREAT loop experiments. The purpose of sectioning the loops was to reduce the loop experiments into smaller segments for shipment to the

Materials Science Division (MSD) at ANL-East, where the detailed post-test examination of the loop experiments was performed. Following irradiation in TREAT, the loop experiments were transferred to HFEF in the HFEF loop cask and transferred into the HFEF Air Cell and finally into the Argon Cell. In the Argon Cell, the loop insulation, heaters, neutron filters, and miscellaneous loop outfitting were stripped from the loops (Cook, 1975). Further discussion of these processes is reserved for the HFEF section of this report.



Source: TREAT Manual, 1971

Figure 5-26: TREAT Reactor Plan View

The TREAT facility operated from 1959 to 1994, undergoing 6,604 reactor start-ups and 2,885 transient irradiations before being placed on standby. TREAT was used to study fuel meltdowns, metal-water reactions, interactions between overheated fuel and coolant, and the transient behavior of fuels for high-temperature systems. During the first few years of operation, a large number of fuel-rod meltdown experiments (85 during the first two years) were performed on unirradiated and irradiated fuel, most of them in dry capsules. These were largely phenomenological experiments in that they were used to identify and study the basic physical processes occurring during rapid overheating of the fuel-rod samples, rather than simulating specific accident conditions. Although sodium-bonded uranium-alloy fuel was the predominant type tested, various fuel and cladding materials were also studied. By the mid-1960s, integral experiments in flowing-coolant loops were being conducted.

In the late 1960s, the major emphasis shifted to testing of oxide fuel for application in the Fast Flux Test Facility (FFTF), and eventually, in later commercial fast reactors. The Mark-III sodium loop was designed for use in these tests. The original Mark-III loops could accommodate up to seven fuel pins of the FFTF design of 5.84 mm diameter. These loops used a "package" concept, in which the whole experiment was assembled outside the reactor, inserted into an appropriate location in the core for transient irradiation, and then removed for disassembly and post-test examination. Supporting facilities were established at both Argonne sites to assemble test trains (using both fresh and pre-irradiated pins), to insert the test trains into the loops, to disassemble the loops after the tests, and to perform post-test macroscopic and microscopic examinations (Deitrich, 1998).

Also in the late 1960s, the fast neutron hodoscope was developed. This instrument enabled real-time tracking of the location of fissile material within an experiment by counting collimated fission neutrons emitted from the fuel. TREAT experiments done with the fast neutron hodoscope were the primary source of experimental information to validate fuel-motion models used in safety analysis (Deitrich, 1998).

Through the decade of the 1970s, the primary emphasis in TREAT experiments was support of FFTF and the Clinch River Breeder Reactor Plant (CRBRP) safety analysis and licensing. For FFTF, the primary emphasis was on simulations of transient overpower (TOP) and unprotected loss of flow (LOF) accidents. The primary interest was in determining the time and location of fuel pin failure, and the nature of the subsequent fuel motion. The principal finding was that there was no strongly-compactive fuel motion that would lead to a reactivity excursion that would exacerbate the assumed input ramp. The LOF experiments, conducted at essentially constant power with decaying flow, also showed an absence of strongly-compactive fuel motion, implying an energetically-benign transition into a core melt (Deitrich, 1998).

Free access of authorized personnel was permitted to the Reactor Building during shutdown, critical operation, or steady-state operation. The dose rate at normally-occupied locations in the reactor building was less than 7.5 mr/hr during 100 kw operation (TREAT Facility, 1971). One exception to the free-access policy was that access to the subpile room at any time required a check of radiation level, and access to this area during reactor operation required the approval of the Supervisor in Charge. A warning sign, operated from the radiation alarm system and indicating high activity in the basement, was posted at the basement stairway. Prior to prolonged work in sub-pile room, the Health Physics representative was to make a complete survey and post any restrictions in conspicuous places. Access to the Reactor Building during transient operation was not permitted. Prior to transient operation, the Supervisor-in-Charge cleared the building of all personnel, personally inspected all areas in the building, and announced over the public-address system that the Reactor Building was being cleared for a transient operation (TREAT Manual, 1971, PDF p. 82).

5.1.2.2 Experimental Breeder Reactor-II (EBR-II)

ATTRIBUTION: Section 5.1.2.2 was completed by Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding EBR-II were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to EBR-II are explained in the associated text.

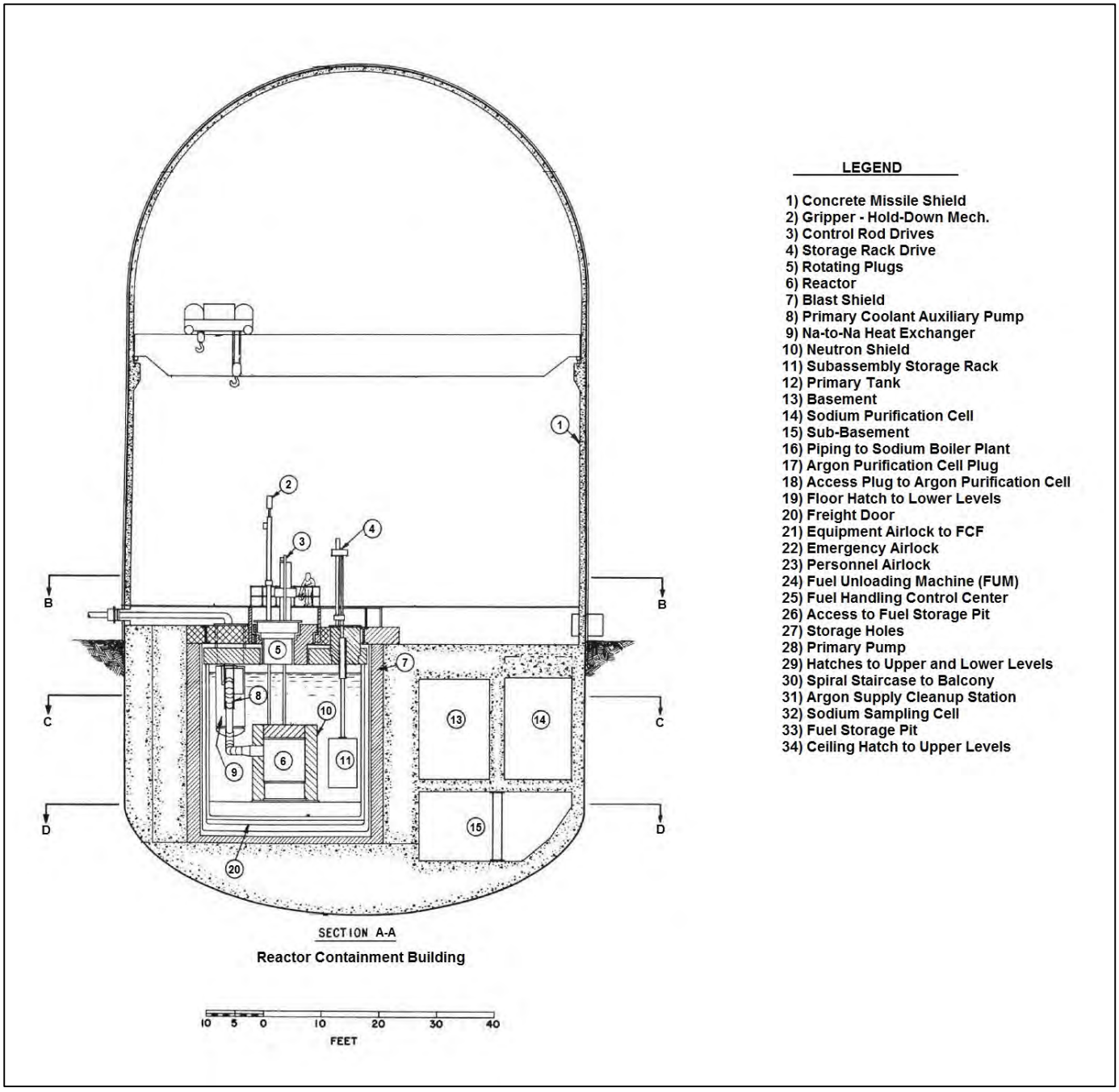
The EBR-II reactor was a second generation Liquid Metal Fast Breeder Reactor that operated from September 30, 1961 to September 30, 1994 (Stacy, 2000). The EBR-II reactor was basically a scaled-up version of the first generation LMFBR (i.e., EBR-I); it was approximately 20 times larger than EBR-I (Stacy, 2000). Another difference was their liquid-metal coolants. EBR-I used a eutectic alloy of sodium-potassium metal (NaK); EBR-II used only sodium metal (Stacy, 2000).

The EBR-II concept was strongly influenced by the limited availability of highly-enriched fuel at that time due to military applications having priority. It appeared that these materials would have limited availability in the future and would be costly. Consequently, the EBR-II concept was based on the need to use all means available to achieve a high power density in the reactor while minimizing the total fuel inventory (EBR-II, undated).

In Figure 5-24, the EBR-II reactor is Building 767. The EBR-II reactor building is also known as ANL-767, EBR-II-601, and MFC-767. Cross-Sections of the various portions of the reactor building are depicted in Figures 5-27 and 5-28.

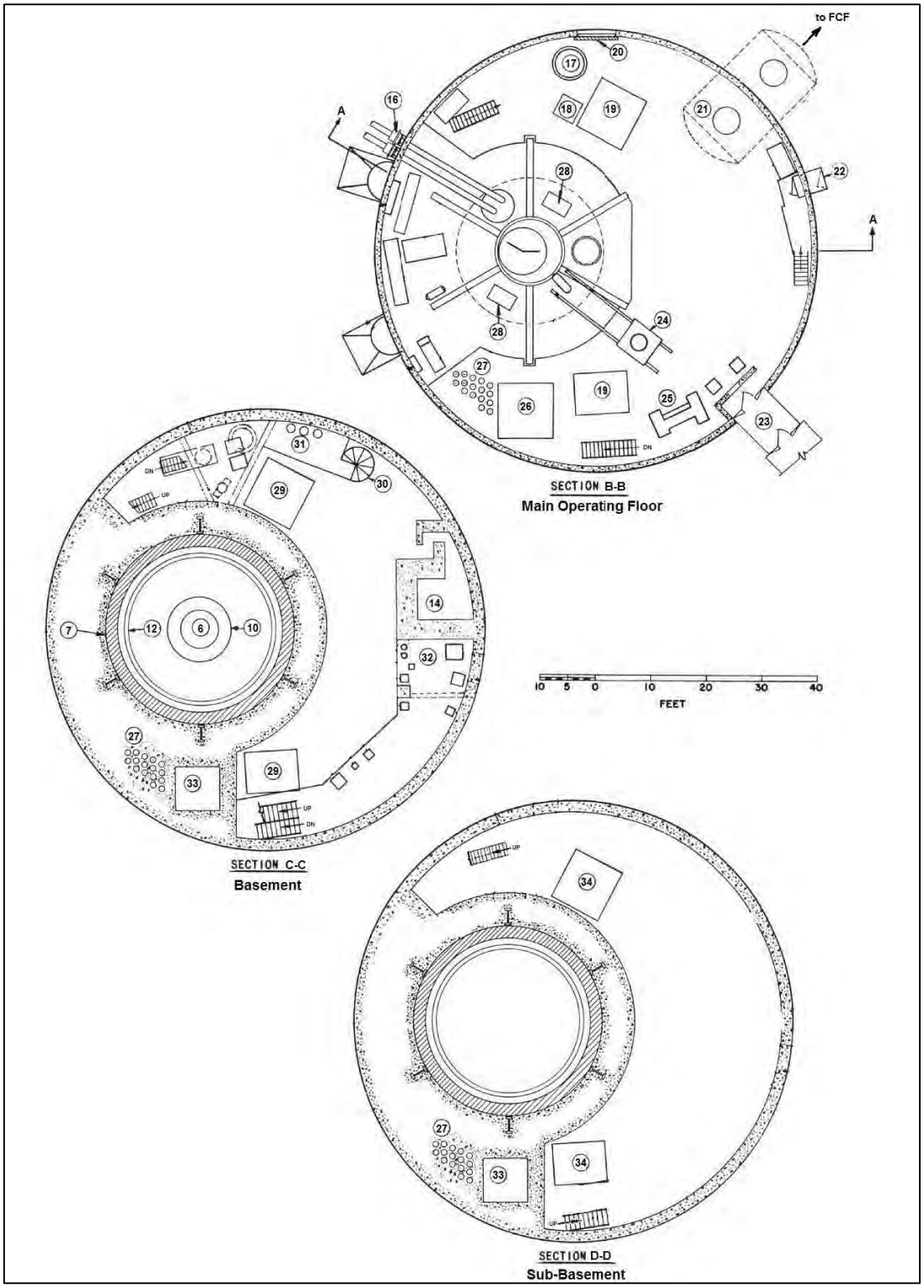
NOTE: Figures 5-27 and 5-28 were assembled from components of images from two historical source documents (Koch, 1964, PDF p. 109; Survey Reports, Aug-Dec1969, PDF p. 62).

NOTE: The legend included in Figure 5-27 applies to Figure 5-28 as well.



Source: Koch, 1964

Figure 5-27: EBR-II Reactor Building (side view)

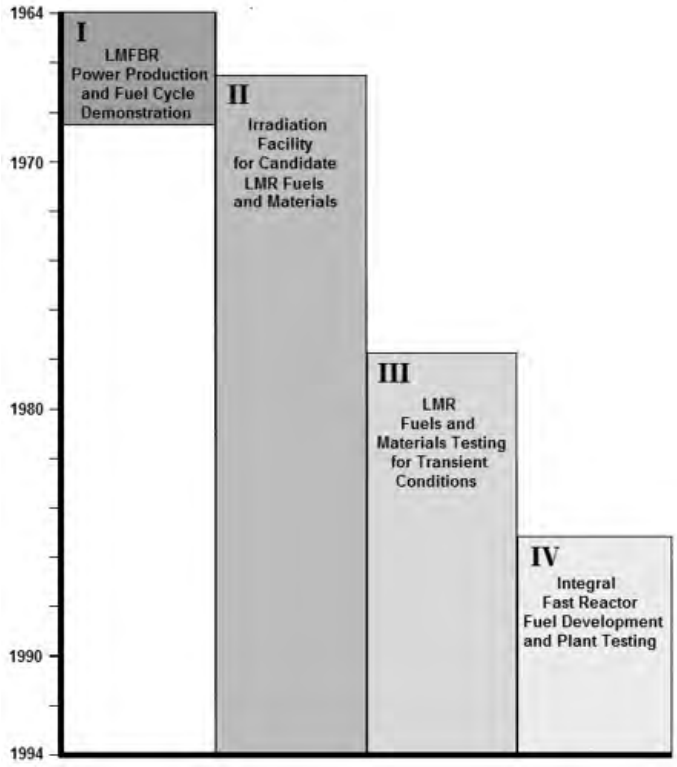


Source: Koch, 1964, PDF p. 109; Survey Reports, Aug-Dec1969, PDF p. 62
NOTE: The legend included in Figure 5-27 applies to Figure 5-28 as well.

Figure 5-28: EBR-II Reactor Building (top view by levels)

EBR-II Missions

Beginning in 1964 and throughout its remaining history, EBR-II had several overlapping operating missions (Sackett, undated). The reactor's first mission effectively ended in April 1969 when the last recycled fuel subassembly was loaded into the reactor (Sackett, undated; EBR-II, undated). The latter three missions continued beyond 1979 (the end date of the period under evaluation) (Sackett, undated). Figure 5-29 portrays EBR-II's four operating missions, followed by a summary of each mission.



Source: Sackett, 2010, derived from figure on PDF p. 15

Figure 5-29: The EBR-II Reactor's Four Operating Missions

Mission I: EBR-II, along with its support facilities (Sodium Boiler Plant, Power Plant, and Fuel Cycle Facility), were originally built to demonstrate the feasibility of on-site fuel reprocessing as an adjunct to a liquid-metal-cooled, fast-breeder-reactor power plant (Stacy, 2000; EBR-II Fact Sheet and Memo, 1968). Together, the EBR-II reactor and the Fuel Cycle Facility formed a closed fuel-cycle operation (i.e., spent fuel from the reactor was transferred to the Fuel Cycle Facility for pyrochemical re-processing, re-enrichment, and re-fabrication, and then it was returned to the reactor) (EBR-II Fact Sheet and Memo, 1968). During the reactor's initial mission, the primary emphasis was on verifying and demonstrating the operating characteristics of the power cycle and the fuel cycle (EBR-II, undated). Particular attention was given to the feasibility of the unique EBR-II fuel cycle, especially the onsite processing and fabrication of highly-radioactive fuel and the power performance of recycled fuel (EBR-II, undated). The objectives of that initial mission were met within the first few years of EBR-II's operation (Stacy, 2000).

Mission II: EBR-II's second mission began in May 1965 when the first irradiation experiment was installed in the reactor's core (EBR-II, undated). In 1969, the experimental focus was reoriented from a demonstration plant to an irradiation facility for candidate LMFBR fuels and materials (EBR-II, undated; OSTI 6345424). In September 1970, some of the reactor's depleted-uranium blanket subassemblies were replaced with stainless-steel reflector subassemblies to enhance the reactor's irradiation capabilities for its fuel-testing and materials-development work (Sackett, undated; Case, 1971). During the second mission, a wide variety of uranium, plutonium, and thorium reactor fuels were tested, including metal, oxide, carbide and nitride fuels (Sackett, undated). Those tests also helped develop the reactor fuels for both the Fast Flux Test Facility and the Clinch River Breeder Reactor Plant (Sackett, undated). It was from that experience that the benefits of metal fuel irradiation performance were established (Sackett, undated). In the late-1970s, the breeding part of the second mission was dropped and the mission was modified to include irradiating candidate fuels and materials for all types of Liquid Metal Reactors (LMRs).

Mission III: In the late-1970s, EBR-II's third mission began. It expanded the effort from a conservative steady-state fuels testing program to an operational reliability testing (ORT) program (Sackett, undated; Lentz, undated). The ORT program used EBR-II for a more aggressive irradiation program consisting of: (1) run-beyond-cladding-breach (RBCB) tests; (2) operational transient tests simulating duty-cycle transients and mild over-power transients on fuel elements; and (3) thermal-hydraulic testing of clusters of fuel elements under both normal and natural convection modes of cooling for testing and verification of shutdown heat-removal codes (Lentz, undated). For the reactor transient tests, plant upgrades included providing special instrumented test facilities for in-core measurements, fission-gas-handling systems for breached fuel testing, and computer-controlled power-shaping for transient testing (Sackett, undated). The most dramatic result of this work was the series of tests conducted in 1985–1986 that culminated in a demonstration of the reactor's ability to accommodate loss-of-flow and loss-of-heat-sink transients without needing to scram the reactor (Sackett, undated).

Mission IV: In the mid-1980s, EBR-II's fourth mission involved fuel development and testing activities for the Integral Fast Reactor (IFR) Program (Lentz, undated). No further details are provided in this document regarding these activities because all of them occurred after the period under evaluation.

EBR-II Reactor Description

EBR-II was an unmoderated, sodium-cooled, fast-neutron power reactor, which was designed to produce 62.5 megawatts of heat (62.5 MW_t) and 20 megawatts of electricity (20 MW_e). EBR-II is a piped-pool design with the entire primary system and reactor vessel submerged in a large pool of sodium contained in the primary tank (Sackett, undated). Heat produced in the reactor is removed by the primary sodium system and transferred to the secondary sodium system by the heat exchanger. From the secondary system, the heat is transferred in the steam generator to produce superheated steam, which is then delivered to a conventional condensing turbine at 850 ^oF and 1250 psi (EBR-II Fact Sheet and Memo, 1968; Lentz, undated).

Until September 1970, EBR-II consisted of a high-enriched uranium core surrounded on all sides by a blanket of depleted uranium (EBR-II Fact Sheet and Memo, 1968; Sackett, undated; Case, 1971). While producing power, the reactor was designed to manufacture (breed) plutonium, another fuel, from the U-238 in the blanket material. Furthermore, the reactor was designed to breed more fuel than it consumed. The net effect of this breeding process is to extend many-fold the United States' natural and valuable supplies of uranium (EBR-II Fact Sheet and Memo, 1968). In September 1970, some of the EBR-II Reactor's depleted-uranium blanket subassemblies were replaced with stainless-steel reflector subassemblies (Sackett, undated; Case, 1971).

EBR-II Timeline Through 1979

Table 5-2 summarizes the important dates in EBR-II's operating history from 1957 through 1979.

Table 5-2: EBR-II Timeline (1957-1979)

Date	EBR-II Activity
December 19, 1957	Construction of the EBR-II's containment vessel started (EBR-II, undated; Smith, 1977).
November 1960	Construction of EBR-II's power plant and reactor plant was completed (EBR-II, undated).
April 1961	Fabrication of the fuel for the first core completed in Chicago, Illinois at ANL-E Site (EBR-II, undated; Chakraborty, 1971).
September 30, 1961	Dry criticality (i.e., low-power operation without sodium coolant) of reactor achieved (EBR-II, undated).
October 30, 1963	Loading of the reactor for the approach to wet critical was started with the transfer of 17 enriched uranium subassemblies from the storage basket to the reactor grid (Progress Report, 1963, PDF p. 16; Kirn, 1964).
November 11, 1963	Wet criticality (i.e., with sodium coolant) of reactor achieved (EBR-II, undated; Progress Report, 1963, PDF p. 16; Kirn, 1964).
December 1963 – May 22, 1964	Reactor did not operate due to repairs being completed on both primary sodium pumps and some reconfiguration of the coolant system. (Progress Report, May1964, PDF p. 52; Familiarization Effort, 1970)
July 16, 1964	Approach-to-power started. Approval was received to begin the approach-to-power experiments on July 16, 1964, and the reactor was made critical at 20:30 hours that day. (Progress Report, Jul1964, PDF p. 41; EBR-II, undated; Perry, 1978)
September 1964	First irradiated (0.1% burnup) fuel was removed from EBR-II for examination and reprocessing (EBR-II Fuel Cycle Story, 1987, PDF p 248).
May 1965	The first recycled fuel subassembly (subassembly C162) was made at the FCF in April 1965 and installed in the reactor core in May 1965 (EBR-II, undated; EBR-II Fuel Cycle Story, 1987, PDF p 248; Progress Report, Dec1965, PDF p. 18).
May 10, 1965	EBR-II began its role as an irradiation facility when the first two experimental subassemblies were inserted into the reactor core for irradiation experiments during Run 5. One experimental subassembly contained mixed oxides for an experiment sponsored by General Electric, and the other subassembly contained U-Pu-Fissium alloy for an experiment sponsored by ANL's Metallurgy Division. (Staker, 1975; EBR-II, undated; Chakraborty, 1971, PDF p.15; Smith, 1977)

Date	EBR-II Activity
May 24, 1967	The first fuel assembly failure occurred in an experimental fuel assembly, which was signaled by a release of fission gases (Familiarization Effort, 1970; EBR-II Fuel Cycle Story, 1987, PDF p. 229; Progress Report, May1967, PDF pp. 15-17). This resulted in radioactivity being released into occupied areas of the reactor building. The six workers that were in the reactor building during the release event received whole body counts, but no intakes of radioactive material were detected. Xe-133, Xe-135, Xe-138, and Cs-138 were detected in samples of the reactor's cover gas. (Progress Report, May1967, PDF pp. 15-17)
June 10, 1967	A small radioactive sodium fire occurred in the primary sodium sampling room. Thereafter, access to the sampling room was restricted until the Na-24 radioactivity had decayed. Some Na-24 was transported out of the sampling room by the ventilation system, and caused minor air and surface contamination in the reactor building. (Progress Report, Jun1967, PDF pp. 47-48)
April 1969	Last recycled fuel assembly installed in reactor core (EBR-II, undated).
Mid-November 1970 – March 1971	Reactor was shut down for about four months between Runs 47A and 47B (Chakraborty, 1971, PDF p. 22).
1972	The first of the In-Core Test (INCOT) Facilities installed in reactor core. The INCOT Facilities provided direct access into the reactor core to accomplish operational testing of in-core instrumentation and materials under simulated LMFBR conditions. (Staker, 1975) Also, some of the experimental fuel elements were tagged with "xenon tags," which were to be used to identify the fuel element that failed (Smith, 1977).
April-June 1972	Reactor was shut down for about three months between Runs 55B and 56A (Case, 1974).
June 20, 1977	The first Run-Beyond-Cladding-Breach (RBCB) experiment was irradiated. The experiment contained a pre-irradiated mixed oxide fuel element with a machined slit in its cladding. (Smith, 1977)

5.1.2.3 Fuel Cycle Facility (FCF)

ATTRIBUTION: Section 5.1.2.3 was completed by Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding FCF were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to FCF are explained in the associated text.

The FCF performed radiological activities at various capacities from July 1963 through 1979 and beyond. The FCF was constructed as a pyrometallurgical reprocessing plant to meet EBR-II's nuclear fuel needs by reprocessing the breeder reactor's driver and blanket fuels. The intent was to reprocess the irradiated depleted-uranium blanket fuels to extract the plutonium that had been bred. However, the equipment for this reprocessing was never installed.

The various steps to the pyrometallurgical reprocessing process took place in the FCF's two hot cells (i.e., the Air Cell and Argon Cell), and this irradiated fuel production line was later referred to as the FCF's "Hot-Line." The Hot-Line included processes for disassembling, refining, refabricating, and reassembling EBR-II fuel subassemblies. In December 1966, a program was announced that would significantly increase EBR-II's fuel production in order to meet the increasing need for EBR-II reactor fuel. This program included plans to procure reactor fuel from off-site commercial sources, and the

on-site fabrication of unirradiated fuel elements outside of the FCF's hot cells (i.e., Cold-Line fuel fabrication). This resulted in Cold-Line fuel fabrication processes being set up in the outer rooms of the FCF's main (ground) floor (i.e., those rooms surrounding the Argon Cell). In October 1968, an announcement of programmatic changes for the FCF's Air and Argon Cells was received (Progress Report, Oct1968). Those changes included a gradual decrease in Hot-Line fuel production and an increase in support for the experimental irradiation programs (Progress Report, Oct1968). By March 1969, the FCF's Hot-Line was phased out and the Air and Argon Cells would only be used for shipping, inspecting, and testing irradiated reactor fuels (Progress Report, Mar1969; Progress Report, Oct1968).

In Figure 5-24, the FCF is Building 765. The FCF is also known by the following building numbers: ANL-765, EBR-II-602, and MFC-765. Maps of the various areas in the FCF are depicted in Figures 5-30 through 5-32.

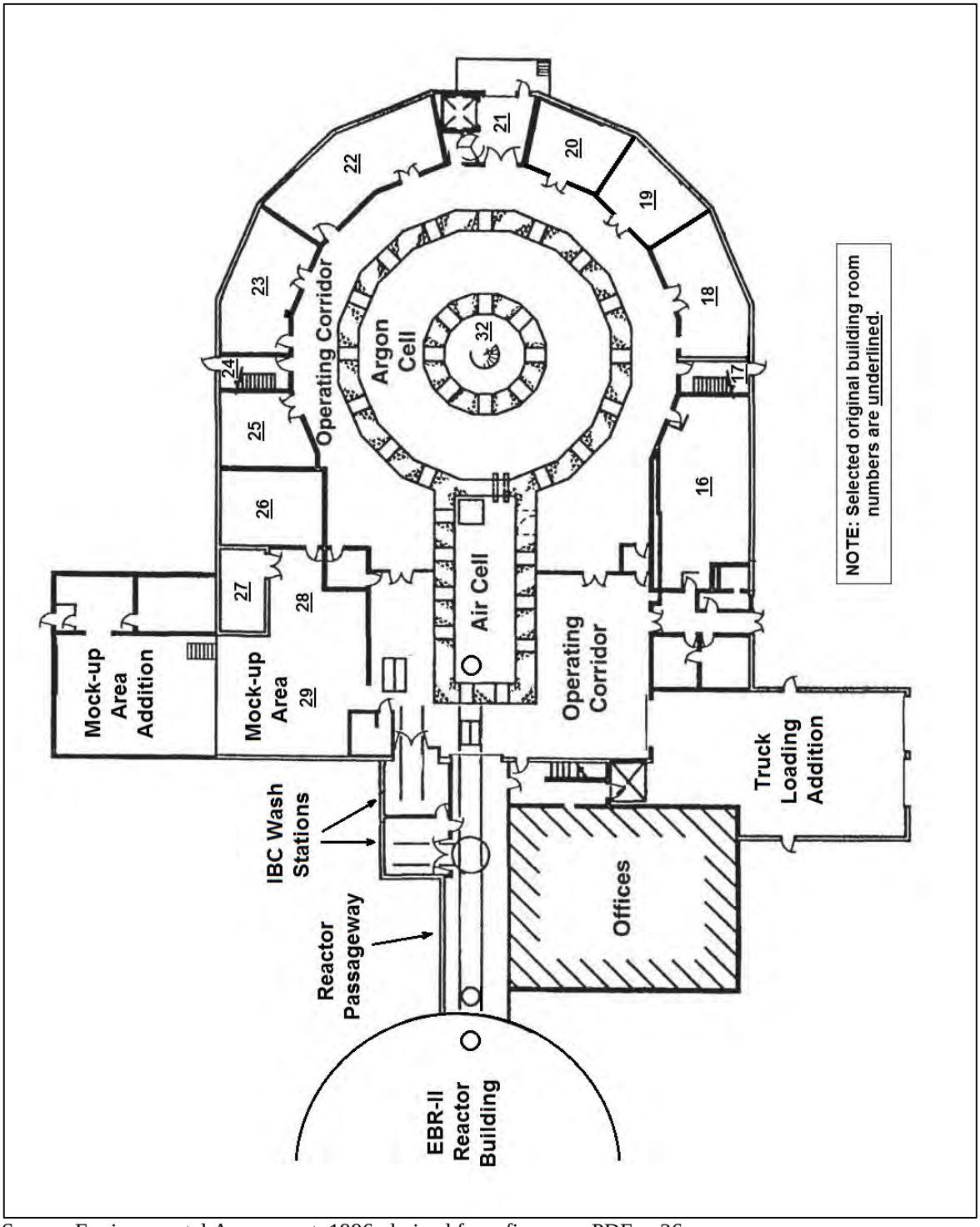
The FCF has had a number of name changes to reflect changes in the type of work being performed at the facility. Between 1964 and 1969, when a demonstration of technology for recycling nuclear reactor fuel was being performed, the facility was called the Fuel Cycle Facility. Between 1970 and 1971, the facility was sometimes referred to as the Fuels and Examination Facility (FEF) (Progress Report, 1971, PDF p. 22; Cushman, 1970, PDF p. 32). The facility was often referred to as the FCF and FEF in the same documents. In various 1972 documents, the FCF was often referred to as the FEF Complex, and then the HFEF Complex, even though the other main part of that complex (HFEF-N) was still under construction. By the time HFEF-N construction was completed in 1972, the FCF was renamed the Hot Fuel Examination Facility-South (HFEF-S and/or HFEF/S) (Operations Training, 2009; Site Plan, 1981). In 1993, the FCF was renamed the Fuel Conditioning Facility (Operations Training, 2009).

FCF Missions

FCF Mission I: Between 1964 and 1969, the FCF was operated as a demonstration of Liquid Metal Fast Breeder Reactor (LMFBR) technology for the remote reprocessing and re-fabrication of nuclear reactor fuel from a closed cycle with the EBR-II reactor (HFEF History, 1979; Stevenson, 1987). This irradiated fuel process was referred to at the FCF as Hot-Line fuel production. By 1969, the FCF had successfully completed its mission of demonstrating the processing and recycling of spent fuel to support EBR-II reactor operations (Courtney, 1991).

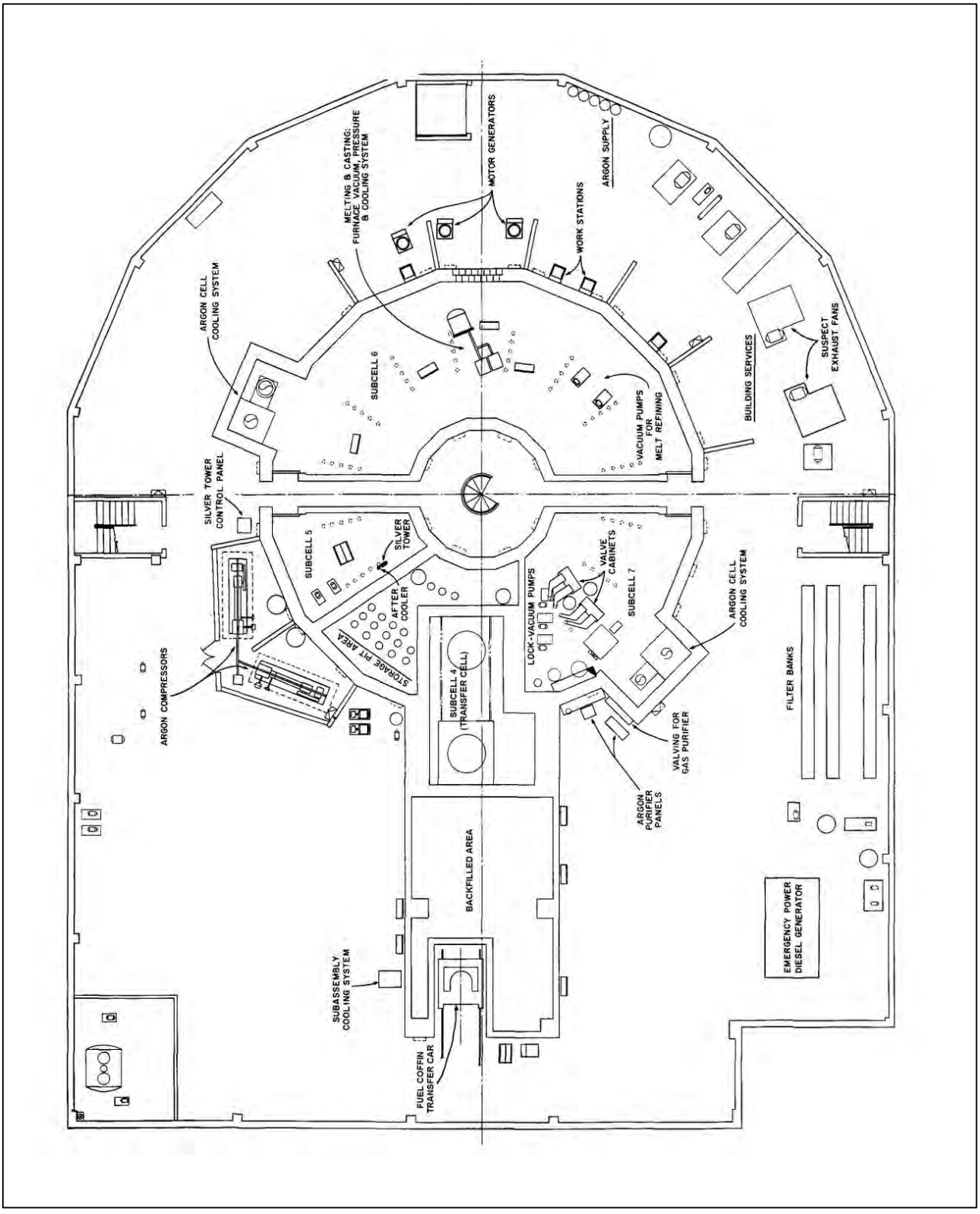
FCF Mission II: In October 1968, an announcement of programmatic changes for the FCF's Air and Argon Cells was received (Progress Report, Oct1968). Those changes included a gradual decrease in Hot-Line fuel production and an increase in support on the experimental irradiation programs (Progress Report, Oct1968). By March 1969, the FCF's Hot-Line was phased out, and the Air and Argon Cells would only be used for shipping, inspecting, and testing irradiated reactor fuels (Progress Report, Mar1969; Progress Report, Oct1968). In support of EBR-II reactor operations, the FCF handled, packaged, and shipped spent fuels to the Idaho Chemical Processing Plant (ICPP) for reprocessing, and packaged blanket and reflector subassemblies for disposal (HFEF History, 1979). The FCF also provided hot cell services for the qualification of EBR-II reactor driver fuel, blanket, and reflector subassemblies, and for handling and examining irradiated components (HFEF History, 1979). The FCF also provided interim and post-irradiation handling, along with destructive and nondestructive examinations of fuel and

material experiments from the EBR-II and TREAT reactor irradiation programs (HFEF History, 1979; 111648).



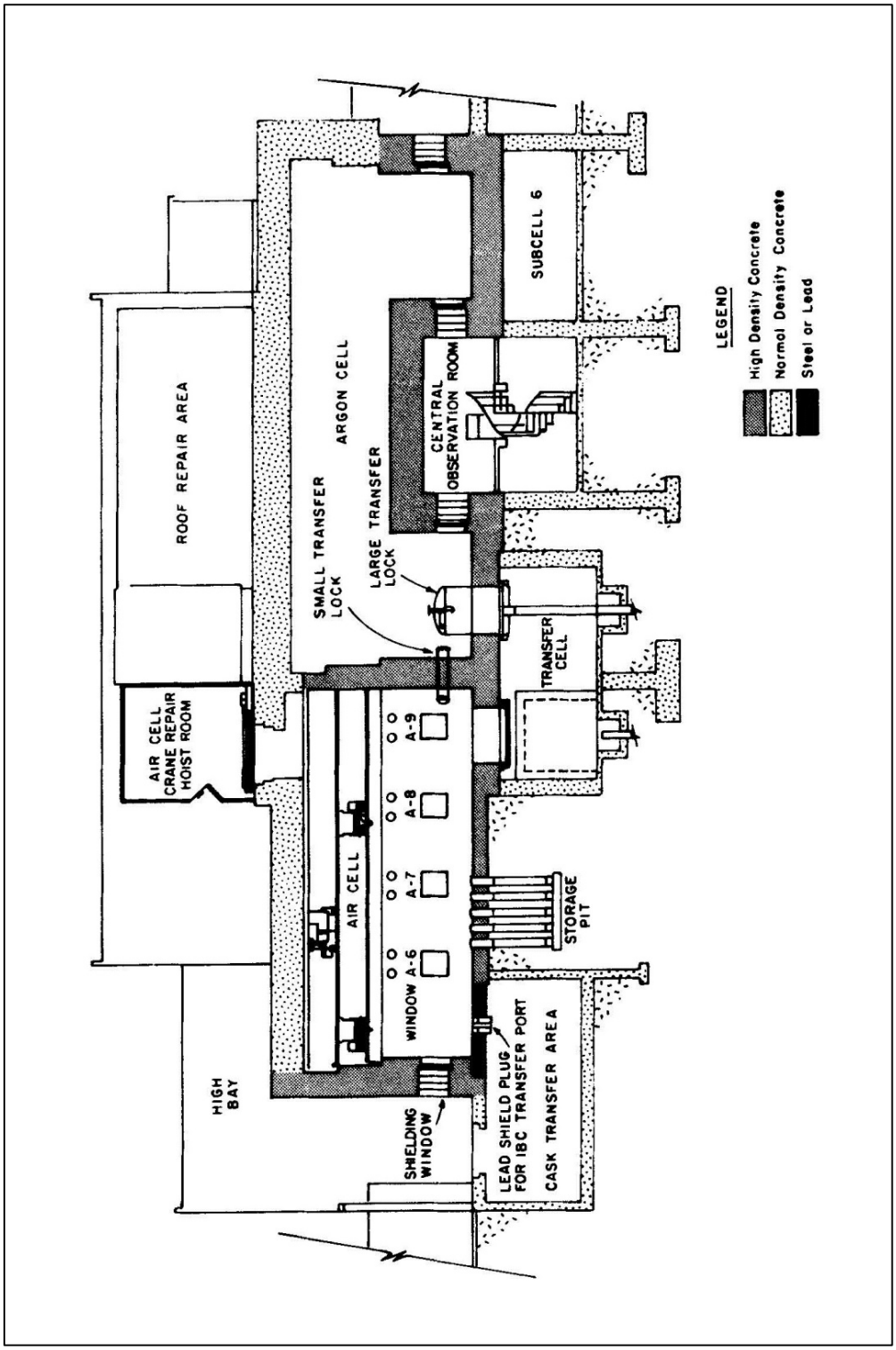
Source: Environmental Assessment, 1996, derived from figure on PDF p. 26

Figure 5-30: General Layout of FCF's Main Floor (ground floor)



Source: Hesson, 1963, derived from figure on PDF p. 52

Figure 5-31: Original Layout of FCF's Basement



Source: Courtney, 1991, PDF p. 5

Figure 5-32: Elevation Cross-Section of FCF

FCF Mission III: Starting in late-1969, the FCF began supporting RAS-TREAT sodium-loop experiment work, which was an intermittent activity that was initially performed in the FCF's Sodium Cleanup Room (Room 27) (FCF Rad Survey, 1970, PDF pp. 208-211; 145251). The RAS-TREAT sodium loop experiments were sponsored by ANL-E's Reactor Analysis and Safety (RAS) Division, and were irradiated in the TREAT Reactor (Cook, 1975). Because the work performed in FCF Rooms 26–29 (and eventually Room 22) was outside of the hot cells, only unirradiated and slightly-irradiated RAS-TREAT sodium loops were initially handled at the FCF. Initially, the FCF's RAS-TREAT sodium-loop experiment work only involved the Mark-I type loops that only contained uranium test fuels, and only involved draining and washing the NaK alloy metal coolant from the loops. However, in 1970, the RAS-TREAT sodium loop experiment work at the FCF started to include the Mark-II type loops, which could contain uranium-plutonium and possibly plutonium test fuels (FCF Rad Survey, 1970, PDF pp. 208-211).

An April 1970 survey is the first indication of Mark-II loop work taking place at the FCF, which was performed in the FCF's Mock-Up Area (Survey Reports, Apr1970). However, this initial Mark-II loop work only involved draining and washing the NaK alloy metal coolant from the loops. By April 1970, Mark-II loop work at the FCF started to include the removal of the test sections from the Mark-II loops in the Sodium Cleanup Room (Survey Reports, Apr1970, PDF pp. 36-48). The start of the potential exposures at the FCF to plutonium without mixed fission products being present would have begun with removing and handling the test sections from the Mark-II loops. In early-1971, that work was moved to a newly-installed inert-atmosphere glovebox in Room 22, which was the former FCF Machine Shop (Progress Report, AprMay1971). In early-1973, the FCF's Argon Cell was decontaminated in preparation for handling and processing more highly-irradiated sodium-loop experiments (Cook, 1975). During 1973 and 1974, the first Mark-IIA sodium-loop experiments were sectioned in the Argon Cell. The purpose of the sectioning was to reduce the sodium-loop experiments into smaller segments for shipment to the Materials and Science Division (MSD) at ANL-E (Cook, 1975).

Background: The integral sodium loop is a package unit in which fast-reactor fuel pins can be tested to destruction in stagnant or flowing sodium coolant while being irradiated within different types of reactors (e.g., air-cooled and water-cooled). The loops contained a sodium reserve and flow system, electromagnetic pump, power transformer, sodium flowmeter, sodium heaters, expansion chamber, fuel-test section, pressure and temperature instrumentation, along with power and control wiring. The sodium loops fit into either a standard single- or double-size dummy fuel assembly for the reactor in which they were to be irradiated (Annual Report, 1965; Robinson, 1971).

FCF Mission IV: In 1990, it was proposed that the FCF be modified to once again reprocess EBR-II reactor fuel as part of a test for the Integral Fast Reactor (IFR) concept. The ultimate purpose of the proposed action was to provide essentially-complete fuel-cycle service for the EBR-II reactor. The proposed fuel cycle program was needed to establish the feasibility of the IFR concept, which includes proving the feasibility of a high-temperature-metal fuel cycle for advanced liquid-metal-cooled reactors. The IFR program was considered important to the national interest because of its potential to improve nuclear power economics and because of the inherent safety achievable through use of liquid-metal-cooling and metallic fuels. Other potential benefits of this fuel-cycle included proliferation resistance of the processed fuel because of residual inherent radioactivity (DOE, 1990). No further details are provided in this document regarding this mission, because it occurred after the evaluation period for this report.

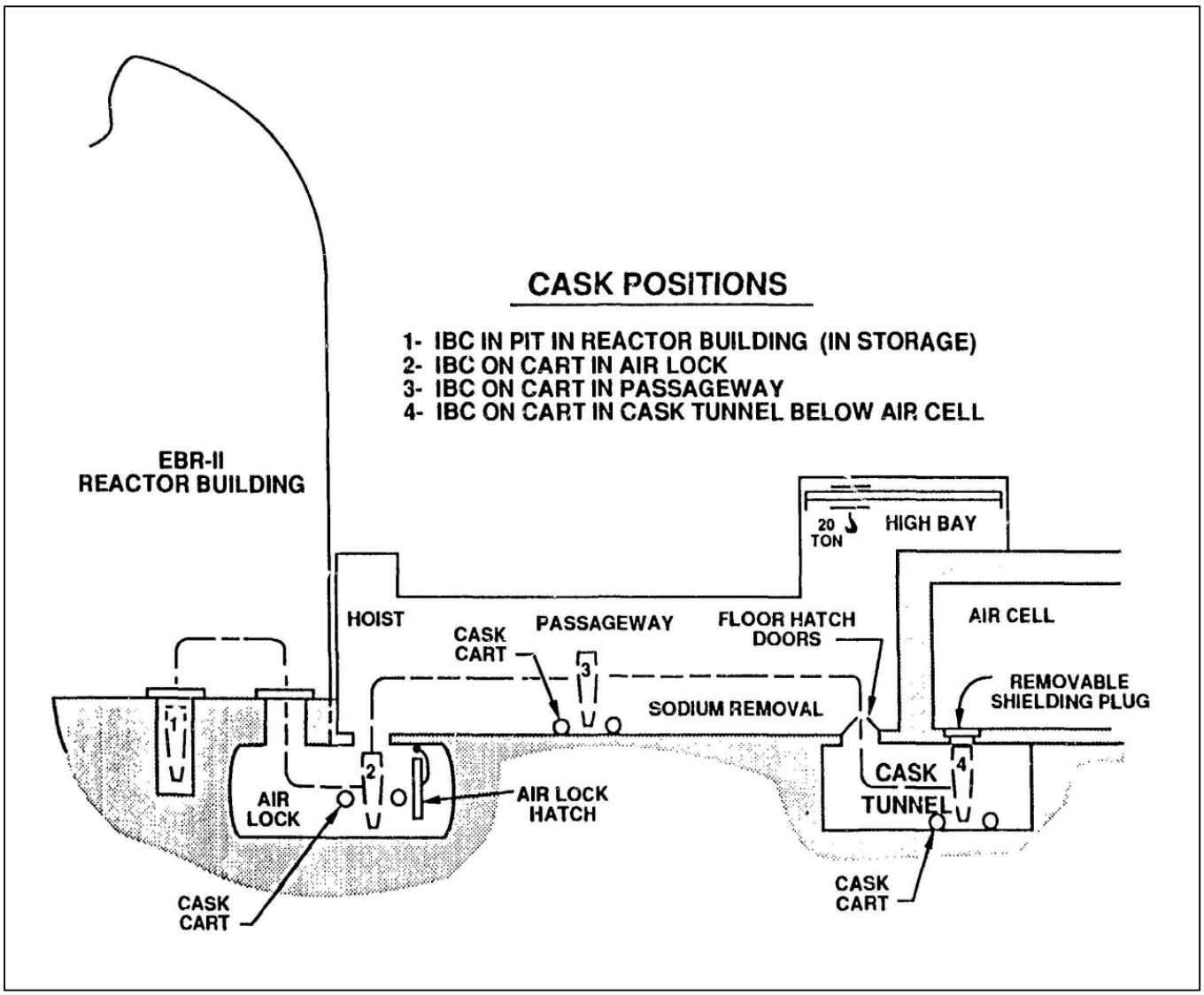
FCF Facility Description

The FCF consists of four main operating areas: Reactor Passageway, Air Cell, Argon Cell, and the Auxiliary Areas. The Auxiliary Areas are further subdivided into three main areas: the Operating Floor Auxiliary Areas, Service Floor Auxiliary Areas, and Roof Area. (Hesson, 1963; Stevenson, 1987)

Over time, manipulator and slave decontamination, maintenance, and repair became a major activity performed throughout the FCF. Based on the available survey records, decontamination work and manipulator and slave repair work was at times performed in nearly every room of the FCF's Auxiliary Areas. This included some rooms originally designated as office areas. As fuel reprocessing came to an end at the FCF, more and more of the offices areas were converted over to work areas and production areas for new projects and activities.

Reactor Passageway

The Reactor Passageway was the passageway between the FCF and the EBR-II reactor. It was used to transfer reactor fuel and experiments (irradiated and unirradiated) between the FCF and the EBR-II reactor. Those reactor fuels and experiments were transferred via an Inter-Building Coffin (IBC). The Reactor Passageway also served as a sodium decontamination area for materials being removed from the EBR-II reactor. A description of the activities performed in the Reactor Passageway is provided later in this section in Hot-Line Step 1. The pathway that the IBCs would travel between EBR-II and the FCF is shown in Figure 5-33.



Source: Forrester, undated

Figure 5-33: Elevation Cross Section of Reactor Passageway

Air Cell

The FCF's Air Cell is a large (15 ft. wide x 47ft long x 21 ft. high) hot cell with an air atmosphere. The original primary purpose of the Air Cell was to provide a radiation-shielded area with an air atmosphere where reactor subassemblies could be disassembled and reassembled, and where fuel elements could be inspected and tested by remote methods. Initially, the Air Cell was the only terminal for the air locks to the Argon Cell. The Air Cell also served as a place for servicing Argon Cell equipment and for preparing scrap for disposal. The Air Cell is constructed of reinforced high-density (3.5 g/cm^3) barite concrete. The east end of the Air Cell shares a wall with the Argon Cell. The Air Cell is equipped with nine viewing windows, one periscope opening, one optical peephole sleeve in the walls, a roof hatchway with stepped removable plugs served by a jib crane, and two plugged viewing positions in the roof. The jib crane was later removed and the roof hatch was modified. Each window position is provided with two sleeves for master/slave manipulators and with

sleeves through the cell floor to the basement for service leads. The Air Cell is also equipped with one five-ton crane and two bridge-type manipulators. The bridge manipulator operates on rails along the cell walls below the crane's rails. Processing equipment inside the Air Cell is limited to 10 ft. in height due to the elevation of the bridge manipulators (Hesson, 1963; Stevenson, 1987).

The Air Cell is also equipped with a two-part entry door for personnel, which was normally sealed, one large transfer port to the basement (a.k.a. Service Floor), and one small transfer port to the operating area. Irradiated fuel subassemblies from the EBR-II reactor were brought into the Air Cell via the large transfer port. There are also one large and two small transfer locks to the Argon Cell. The two smaller transfer locks were used to transfer small items (e.g., fuel elements, fuel cans, crucibles, fume traps, samples) directly into the Argon Cell. Larger materials and equipment were transferred through the Air Cell's large (6-ft diameter) transfer lock to the Argon Cell via the Transfer Cell, which is located below the Air and Argon Cells. The Transfer Cell (a.k.a. Subcell 4) served at the air lock between the Air and Argon Cells (Hesson, 1963; Stevenson, 1987).

The Air Cell is maintained at a negative pressure relative to the occupied areas in the FCF, and is ventilated at the rate of 5000 ft^3/min. Building air from outside the Air Cell enters the cell through dampers at the upper part of the cell wall at the west end and is exhausted through the Transfer Cell, which is below the east end of the cell floor. This creates a downward airflow throughout the cell, which minimized the spread of contamination. After passing through a bank of fine-particle filters, the air leaving the Air Cell is released to the atmosphere through a 200-ft-tall stack (see Figure 5-34) (Stevenson, 1987).

Argon Cell

The FCF's Argon Cell is an annular-shaped hot cell with an inert-argon atmosphere. The original primary purpose of the Argon Cell was to provide a radiation-shielded area with an inert-argon atmosphere where metallic reactor fuels and sodium can be safely exposed (Hesson, 1963; Stevenson, 1987).

The Argon Cell is in the shape of a 16-sided regular polygon that has a central control area (see Figure 5-30). The west side of the Argon Cell shares a wall with the Air Cell, each of the other 15 sides is provided with a shielded window or viewing device. The central control area is also in the shape of a 16-sided regular polygon, but with only eight shielded windows (one window in alternate faces). The Argon Cell is 22 ft. high, has a 16-ft-wide annular area between the two polygons with a total floor area of 2,000 ft^2 and a total volume of about 60,000 ft^3 (Stevenson, 1987). The walls on the operating level of the Argon Cell are made of reinforced high-density (3.5 g/cm^3) barite concrete (Hesson, 1963; Stevenson, 1987). The Argon Cell is also completely lined with zinc-coated sheet steel (Stevenson, 1987).

On the west end of the Argon Cell, there are two small transfer locks to the Air Cell in the wall between the cells, and one large (6-ft diameter) transfer lock in the floor, which also leads to the Air Cell via the Transfer Cell. There are also six viewing plugs, a plugged-and-sealed overhead hatchway, and three periscope openings in the roof (Stevenson, 1987).

The cell is provided with two five-ton cranes and six manipulators (Hesson, 1963; Stevenson, 1987). The crane and manipulator bridges pivot in the center of the cell above the central control area, and the outer ends operate on circular rails. The crane bridges operate on outer rails 17 ft. above the floor

and on inner rails suspended from the roof. The manipulators operate on outer rails 11.5 ft. above the floor and on inner rails mounted on the central control area roof. Process equipment height is limited to 10 ft. The cranes cannot pass each other but can pass above the manipulators. The manipulators cannot pass each other (Hesson, 1963).

There is a suspect stack. Its purpose is to provide a high point of discharge to the atmosphere of all ventilation atmosphere suspected of containing radioactive contamination. There is a hold-up tank to provide a place for storing active gases and vapors from the melt-refining operations (and possibly skull oxidation) to await favorable meteorological conditions for stack release. As a secondary effect, some decay of activity can take place in the hold-up tank. Xe-133 and possibly I-131 were the chief radionuclides to undergo decay (Hesson, 1963).

The ventilation air is filtered before being discharged to the stack. Gases from process operations are filtered through high-efficiency filters and, in most cases, are further processed (e.g., through a tower for iodine removal) before being released to the stack. Each of the locations has booster fans and filters. The FCF has two 38,000-cfm main booster fans. The air flow in the ductwork, filters, and piping is controlled at a negative pressure.

Operating Level Auxiliary Areas

The major radiological work areas on the Operating Level are described below.

Mock-Up Area: The Mock-Up Area was initially an area in which the feasibility of proposed remote operations could be determined (Hesson, 1963). Later, the area was also used to test and develop equipment for cladding the fuel pins and testing the completed elements (Stevenson, 1987). Eventually, the area was used for a multitude of activities, some involving highly radioactive equipment and materials (e.g., manipulator repairs.

Room 25: This is the Mold Preparation Room, and was sometimes referred to as the Thoria Room. Thoria (ThO_2) was used to coat the molds until Cold-Line (unirradiated) fuel production operations began in July 1967. After Cold-Line operations began, zirconium oxide was used instead of thoria to coat the molds (Progress Report, Dec1967). As early as January 1967, survey records indicate that depleted uranium (DU) fuel pins were being sent to this room for cleaning prior to being sent elsewhere for radiography. Survey records also indicate that enriched uranium was being handled in this room and that the room was also being used for manipulator decontamination and repairs.

Room 20: Until 1967, Rooms 19 and 20 were just office areas. By August 1967, Room 19 was incorporated into Room 20 to create the Fuel Pin Manufacturing Facility (Batte, 1986; Stoddart, 1967). The reconfigured Room 20 was the initial and primary location for the Cold-Line for EBR-II fuel production for several years (i.e., the unirradiated fuel production line). Operations in Room 20 included: (1) casting fuel pins; (2) shearing fuel pins to length; (3) removing glass mold material; and (4) dimensionally inspecting the fuel pins prior to encapsulating them into elements and subassemblies (Batte, 1986; Stoddart, 1967). The first injection casting run was made in Room 20 on August 26, 1967 (Stoddart, 1967).

At the end of December 1969, the Cold-Line fuel pin and fuel element production equipment in FCF Rooms 20 and 26 was put on stand-by (Progress Report, Dec1969, PDF p. 88). In the mid-

1970s, the FCF's Cold-Line fuel pin and fuel element production equipment was deactivated. Fuel pin fabrication in the FCF was not reinitiated until 1982 (Batte, 1986, PDF p. 9).

Room 26: Prior to July 1967, Room 26 was referred to as the Degas Room, Measurements Laboratory, and Precision Measurements Lab, and contained a vacuum-type induction furnace, air gauges, an optical comparator, a polariscope, and various components for equipment in the hot cells (i.e., in-cell equipment). During that period, Room 26 appears to have been used only for: (1) melt-refining-crucible preparation; (2) cladding and Vycor mold inspection; and (3) calibrating and testing of in-cell equipment (Hesson, 1963).

By July 1967, Room 26 had been reconfigured to be part of the FCF's Cold-Line. The available information indicates that Room 26 was used for preparing the uranium-fissium alloy for injection-casting in Room 20 and removing the skull (dross) from the casting crucibles. As a result, this room was sometimes referred to as the Alloy Prep Room. In July 1967, five test runs on the alloy-preparation furnace (a.k.a. melt refine furnace) were completed using chopped-up depleted-uranium-fissium alloy pins (Progress Report, Jul1967). The first enriched uranium ingot for the Cold-Line was melted and poured in Room 26 on August 17, 1967 (Stoddart, 1967).

Room 23: Initially, Room 23 was referred to as the Glove Box Laboratory or Dry Boxes Room, and contained a sodium extruder, a degreasing tank, and low-moisture, low-oxygen argon gloveboxes (dry boxes) (Hesson, 1963). At that time, Room 23 appears to have been used only for: (1) sodium extrusion and loading into fuel element cans; and (2) performing maintenance and service of processing equipment (Hesson, 1963). Survey data indicates that depleted and possibly enriched uranium were handled in Room 23.

After August 1967, Room 23 was used for some of the Cold-Line fuel production activities, but it is uncertain what was done in this room. Given that one of the gloveboxes in Room 23 was converted into the "Cold-Line glovebox," that glovebox was likely used to assemble the Cold-Line fuel elements and perform the sodium-bonding step in the process.

Room 27: Room 27 was a steel-lined room with a blowout wall and sodium-disposal hood (Hesson, 1963). The room was used for sodium disposal and for decontaminating various items (sodium and radiological contaminants) (Hesson, 1963). The room was also known as the Sodium Cleanup Room, Sodium Disposal Room, and Decontamination Room. Prior to 1970, potential actinide exposures without fission products present were limited to depleted and enriched uranium.

Based on the survey records, work with the Mark-I TREAT Sodium Loops in Room 27 began in February 1970 (Survey Reports, Feb1970, PDF pp. 81-85; FCF Rad Survey, 1970, PDF p. 208). However, this initial TREAT Sodium Loop work only involved loops containing metallic uranium fuel. The work was performed under Non-Routine FCF Operation Proposal No. 1-391 (a.k.a. NR1-391) (Survey Reports, Feb1970, PDF pp. 81-85).

Later in 1970, work involving the disassembly of Mark-II TREAT Sodium Loops, which sometimes contained "PuO and UO_2", began in this room (FCF Rad Survey, 1970, PDF pp. 208-211). This work was performed under Non-Routine FCF Operation Proposal No. 1-398 (a.k.a. NR1-398) (Survey Reports, Apr1970, PDF pp. 36-48). Based on the Propose to Carry Out date on NR1-398, this activity did not begin until April 14, 1970 (Survey Reports, Apr1970, PDF pp.

36-48). Based on the survey and air monitoring records, this Room 27 work was sporadic and periodically resulted in some alpha contamination (surface and airborne). However, the Health Physics Instructions for NR1-398 required that, upon job completion, the area be surveyed and that the area and equipment be decontaminated, as necessary (Survey Reports, Apr1970, PDF pp. 36-48). This activity appears to have taken place in Room 27 until around April 1971 (Progress Report, AprMay1971). Based on the available information, a glovebox was installed in Room 22 for this activity and this work was transferred to Room 22 after the glovebox became operational (Progress Report, AprMay1971).

Room 22: This room was initially the FCF Machine Shop until sometime between 1968 and 1971. In 1968, construction was completed on Building 782, which was built to serve as a centralized Machine Shop and receiving inspection facility for the EBR-II Area (Site Plan, 1981). A November 4, 1970 survey is the first instance found where Room 22 was referred to as the "Old Machine Shop." As early as January 1971, Reactor Development Program Progress Reports were indicating that a glovebox for handling the Mark-II TREAT Sodium Loop was being installed in Room 22. While Room 22 operated as the FCF Machine Shop, survey records indicate that materials containing depleted and enriched uranium were machined there, and that some of those materials were irradiated.

Sometime after May 1971, the argon atmosphere glovebox in Room 22 became operational and work with the Mark-II TREAT Sodium Loops started (Progress Report, AprMay1971). A survey record from August 17, 1972 is the first indication of Pu in the Room 22 glovebox, which is also when they started collecting much more air monitoring data for Room 22.

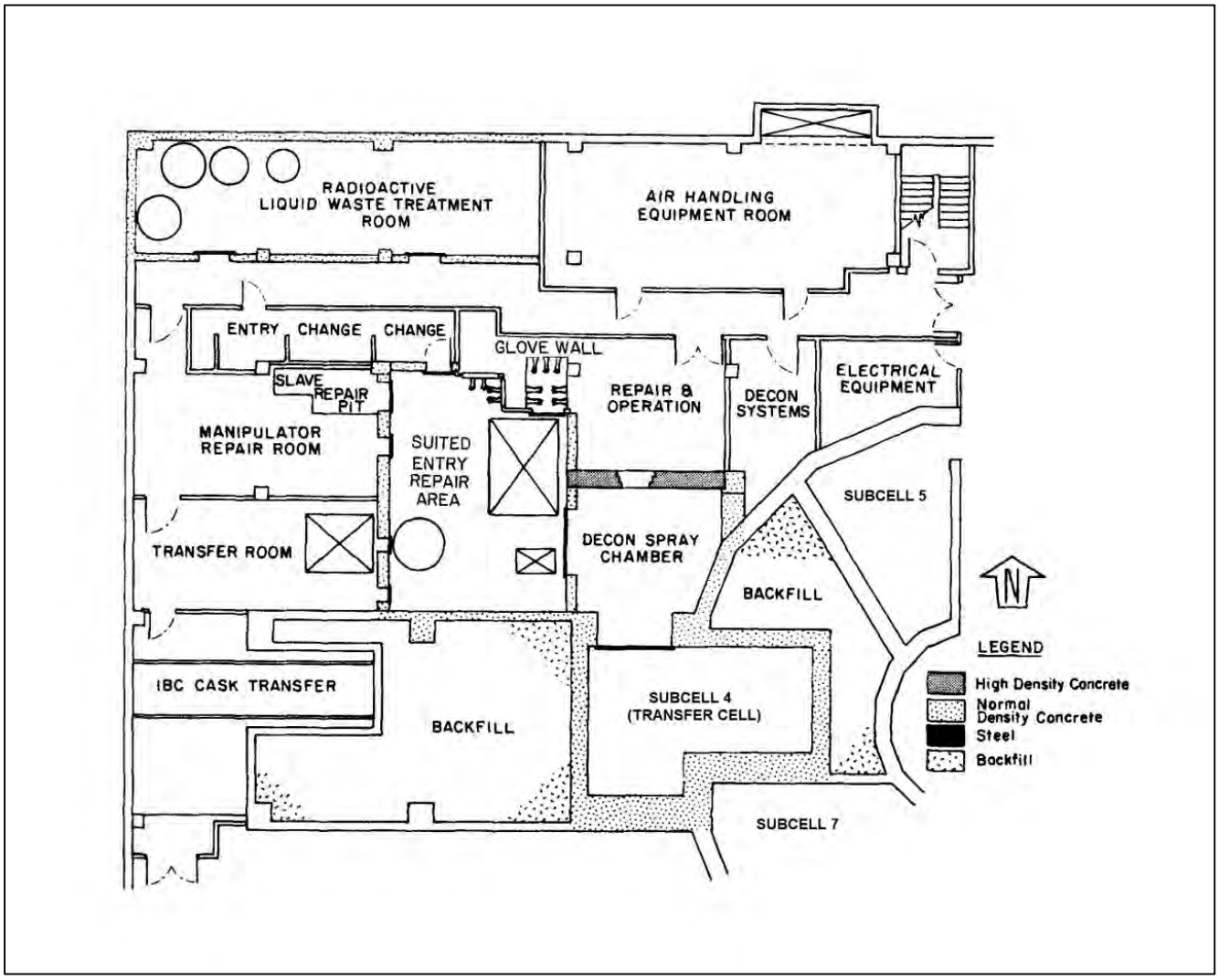
Room 30: This was a small triangular-shaped room that was equipped with a sink and was used to decontaminate small materials and equipment. The room was called the Small Decontamination Room (Health Physics Monthly Report, Jan1968, PDF p. 32), but it is suspected that it may have also been the room that is periodically referred to as the "Janitor's Closet" in a number of the survey records. Survey records indicate that this room was normally maintained as a contamination area.

Room 12: This was designated as an office. Starting in April 1970, survey records and Non-Routine FCF Operation Proposal No. 1-400 (NR1-400) indicate that high-pressure rupture tests were being performed on irradiated tubing using a BRATT furnace in Room 12. NR1-400 also indicated that no fuel was involved with those tests, and the primary radionuclides of concern were Co-58 and Mn-54. (Survey Reports, Apr1970, PDF pp. 74-76).

Service Floor Auxiliary Areas

The primary feature of the FCF Service Floor (a.k.a. Basement) was the Sub-Cells, which were directly below the Air and Argon Cells. The Sub-Cells were areas where the auxiliary equipment for the Air and Argon Cells was maintained. In December 1968, modifications to Sub-Cell 4 (a.k.a. Transfer Cell), which was used to transfer large items between the Air Cell and Argon Cell, were completed. These modifications allowed items (mostly manipulators) to be transferred directly between Sub-Cell 4 and the basement (Progress Report, Dec1968, PDF p. 72). This also eliminated the need for all large items to pass through the Air Cell before going into the Argon Cell. In late

1970, a manipulator decontamination and repair area was constructed in the basement (Survey Reports, Oct1970, PDF p. 63). Figure 5-34 shows the New Hot Repair Area.



Source: Courtney, 1991, derived from figure on PDF p. 8

Figure 5-34: New Hot Repair Area in FCF's Basement

Roof Area

Directly above the ceiling hatches to the Air Cell, the FCF's Roof Area (a.k.a. the Penthouse) was used as a radioactive equipment repair area (mostly for manipulators), and initially was completely exposed to the outside environment. In time, a "Wind and Weather Enclosure" was built around this area on the roof. In May 1968, construction of a full-blown manipulator decontamination and repair facility was started (Progress Report, May1968, PDF pp. 81, 91). This facility would provide a place to decontaminate in-cell cranes, manipulators, and other large pieces of equipment prior to attempting their contact maintenance. Construction of this roof-top facility was not completed until sometime after December 1968 (Progress Report, Dec1968, PDF p. 72).

Overview of the FCF Hot-Line Fuel Production Process

The FCF used a pyrometallurgical process to reprocess EBR-II reactor fuel into usable fuel. Hot-Line (irradiated) fuel production involved 14 major processing steps that occurred almost entirely inside the FCF's Air and Argon Cells. The primary feed for the Hot-Line was irradiated fuel subassemblies (a.k.a. driver subassemblies) from the EBR-II reactor. To a much lesser extent, fresh (unirradiated) fuel was also fed into the process as make-up fuel, to replace the U-235 that was burnt up in the reactor.

The Hot-Line's 14 processing steps are summarized below (Hesson, 1963; Koch, undated; Stevenson, 1987; Stevenson, 1969; Fuel Cycle Facility, undated):

Hot-Line Step 1 – Transfer of Subassemblies to FCF & Sodium Removal: After a 14-day cooling period, irradiated fuel subassemblies were ready to be transferred from the EBR-II reactor to the FCF. At the reactor, an individual subassembly was transferred from the reactor's subassembly storage rack into a shielded portable cask called an Inter-Building Coffin (IBC). The IBCs were transferred from the EBR-II Reactor to the FCF's Reactor Passageway on a cart. While in the Reactor Passageway, a simple procedure was performed to remove the residual sodium on the subassemblies while still in the IBC. The transfer of the IBCs from the EBR-II Reactor to the FCF's Air Cell is depicted in Figure 5-33 (Stevenson, 1987, PDF pp. 259-260; Fuel Cycle Facility, undated, PDF p. 8).

Hot-Line Step 2 – Subassembly Disassembly: In the FCF's Air Cell, the irradiated fuel subassemblies were received and remotely-disassembled into individual fuel elements. The fuel elements were then loaded into magazines and then passed through one of the small transfer locks into the Argon Cell (Hesson, 1963; Koch, undated; Fuel Cycle Facility, undated).

Hot-Line Step 3 – De-canning & Pin Chopping: After the irradiated fuel elements were received in the Argon Cell, the fuel elements went through the de-canning process (Koch, undated; Stevenson, 1969; Fuel Cycle Facility, undated). The de-canning process reduced the fuel elements to bare fuel pins, and mechanically removed the cladding from the fuel (Koch, undated; Stevenson, 1969; Fuel Cycle Facility, undated). Prior to a melt-refining run, approximately 150 irradiated fuel elements were selected for a given melt-refining run (Stevenson, 1987, PDF p. 104). The selected fuel pins with their adherent sodium were then chopped into approximately 1.5-inch sections by a mechanical chopper (Koch, undated; Stevenson, 1969; Fuel Cycle Facility, undated). The chopped fuel pins were then transferred to a pin charger before going to the Melt Refining Station (Stevenson, 1987, PDF p. 105).

Hot-Line Step 4 – Melt Refining: Melt refining is also known as "oxidative slagging" or "drossing." As indicated above, approximately 150 irradiated fuel elements were selected for a given melt-refining run. Outside the Argon Cell, the required quantities of the alloying constituents (U-235, U-238, fissium metals, a master silicon alloy, if required) were weighed out from unirradiated stocks and added to a new zirconia (ZrO_2) crucible. The partially-charged ZrO_2 crucible was then transferred into the Argon Cell and placed in the melt-refining furnace. Next, the pin charger that came from the pin chopper was positioned over the ZrO_2 crucible and tipped by the in-cell manipulator, allowing the pin charge to slide into the crucible. With the crucible's charge completed, the Fiberfrax® fume trap was then placed over the crucible and the

bell jar was placed over the furnace and sealed. These runs typically involved 10–12 kg charges, which yielded an average ingot size of 10 kg. The charge in the crucible was then melted and poured into a graphite ingot mold with a pyrolytic carbon coating (Stevenson, 1987, PDF pp. 95-129, 259-260).

After the melt-refining furnace had cooled down, the ingot was then dumped from the crucible, and the sample protrusions that had been cast at the base of the ingot were sheared off and collected in aluminum sample containers. The capped sample containers in the hexagonal aluminum block were transferred to the Junior Caves (hot cells) in the L&O Building for analysis (Stevenson, 1987, PDF pp. 95-129).

The FCF's average throughput for the melt-refining process was about 100 kg of fuel per month, with a peak throughput of 245 kg/month occurring in mid-1967 (Stevenson, 1987, PDF p. 115).

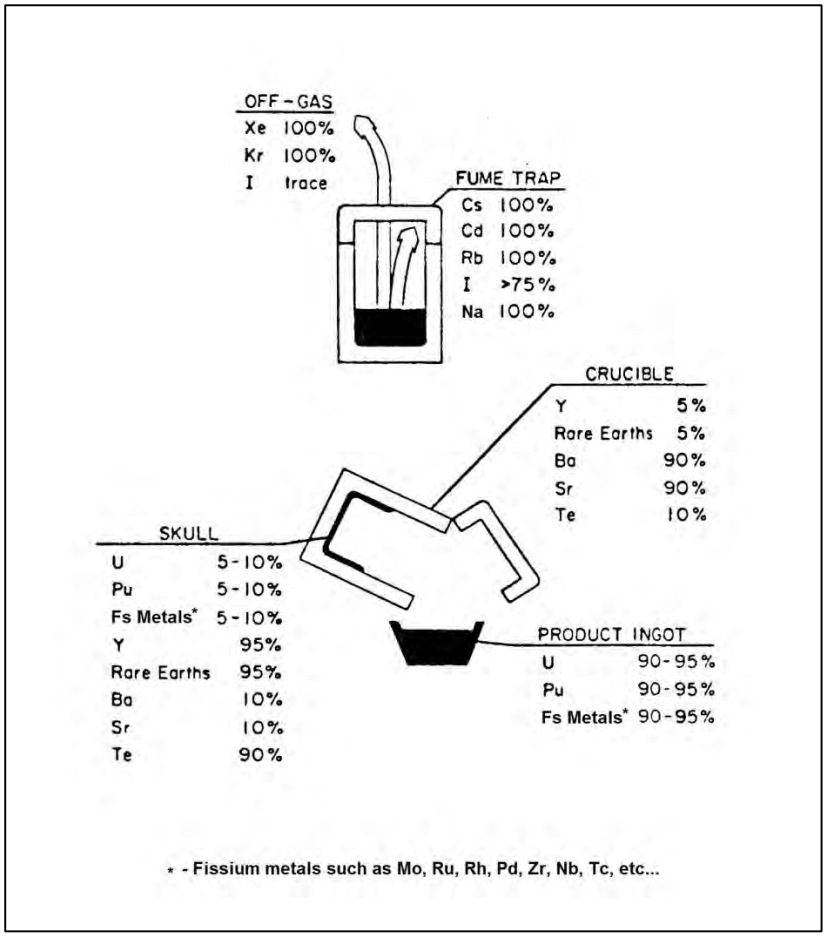
Fission products can be divided into three major groups, according to their behavior during melt refining. Group 1 includes elements that are either gases or are highly volatile at 1400°C. These elements separate from the melt because they are much more volatile than the molten metal alloy. Group 2 consists of the elements that react chemically with the ZrO_2 crucible at process temperature more readily than does uranium. This interaction results in diffusion of the Group 2 fission products into the crucible matrix, its adherence to the crucible surface, or its separation to form a hard slag at the upper-melt surface. Group 3 fission products include those that do not separate from the molten alloy because they are not volatile and are less chemically-reactive toward ZrO_2 than uranium. Fortunately, the Group 3 fission products provide some stability to the uranium by reducing swelling during irradiation. The bullet list below shows how the fission product elements fall into each of these three groups during melt refining:

- Group 1 (gases or highly-volatile): Krypton, Xenon, Bromine, Iodine, Rubidium, Cesium, and Cadmium

- Group 2 (chemically-reactive with the crucible): Strontium, Yttrium, Tellurium, Barium, Lanthanum, Cerium, Praseodymium, Neodymium, Promethium, Samarium, and Europium

- Group 3 (not volatile and less chemically-reactive): Molybdenum, Ruthenium, Rhodium, Palladium, Zirconium, Niobium, Technetium, Silver, Indium, Tin, and Antimony

Figure 5-35 depicts where all of the key elements go during the melt-refining process.



Source: Stevenson, 1987, PDF p. 107

Figure 5-35: Approximate Distribution of Elements in Melt-Refining Process

Hot-Line Step 5 – Skull Recovery: After melt refining, the ZrO_2 crucibles, which had a uranium-bearing melt residue (a.k.a. skull) adhered to them, were sent to the skull recovery area. In the Hot-Line, the skulls were recovered from the ZrO_2 crucibles using an oxidation process. The ZrO_2 crucibles were charged to the skull oxidation furnace. The furnace oxidized the enriched-uranium skull adhering to the crucible's inner wall into a free-flowing powder. The oxide was then dumped into a suitable storage can, and the crucible residue discarded as waste. Cans of the skull oxide were periodically shipped to the Idaho Chemical Processing Plant (ICPP) via a heavily-shielded "oxide coffin" to recover the enriched uranium. The skull oxidation furnace was also used to recover uranium from unpourable melt-refining charges and slag materials (Hesson, 1963, PDF p. 77; Stevenson, 1987, PDF pp. 98, 102-104)

Hot-Line Step 6 – Pin Fabrication: Fuel-pin fabrication involved an injection-casting process. A ThO_2 (thoria)-coated graphite crucible was charged with a melt-refining ingot, which averaged 10 kg. Broken pieces of heel from previous injection-casting runs were usually added to the charge to reach the proper charge weight, which varied between 11.5 and 14 kg. The charged graphite crucible was then placed into the injection-casting furnace. Also located in the furnace, in a vertical attitude above the crucible, was a cluster of approximately 100 precision-bored thoria-coated Vycor® glass (high-silica glass) fuel-pin molds (a.k.a. Vycor® glass pin molds), that were held in position by a pallet stand. Once the casting temperature was attained and the charge was molten, the graphite crucible was raised to a position that immersed the lower end of the Vycor® glass pin mold cluster. The furnace was then rapidly pressurized to 1.7 atmospheres to drive the melt upward into the evacuated molds. The pregnant Vycor® glass pin molds were then placed in a feeder bucket for transfer to the pin-processing operation, or for temporary storage. The injection-casting heels were removed from the crucibles and used in future castings (Stevenson, 1987, PDF pp. 140-141; Fuel Cycle Facility, undated, PDF p. 14)

A significant step in this process (from a radiological perspective) took place in Room 25 of the FCF (the Mold Preparation Room), which was outside the Air and Argon Cells. Because the crucibles used in the injection-casting process were graphite, and because graphite is soluble in molten uranium, the inside of the graphite crucibles had to be protected with an insoluble coating of refractory material to prevent carbon pick-up by the charge. The refractory coating on the inside of the graphite crucibles also facilitated skull removal and crucible clean-up. This coating was also applied to the outside of the graphite crucibles to reduce radiant-heat losses during the injection-casting process. The precision-bored Vycor® glass fuel pin molds also needed to be coated with a refractory material to prevent metal adherence and to help create a smooth fuel-pin surface.

From the start of the FCF's Hot-Line injection-casting process to the start-up of the Cold-Line injection-casting process in late-1967, thoria was used as the primary refractory material for coating the graphite crucibles and Vycor® glass pin molds (Progress Report, Aug1967, PDF p. 18; Hesson, 1963; Stevenson, 1987). Only about 35 g of thoria were used per graphite crucible, and only about 60 g of thoria were used per 120 Vycor molds (Hesson, 1963, PDF pp. 46, 81, 160-161, 183; Stevenson, 1987, PDF pp. 134-135, 139-140). By December 1967, zirconia (ZrO_2) was satisfactorily substituted for thoria in coating graphite crucibles and Vycor® glass pin molds (Progress Report, Dec1967, PDF p. 102; Progress Report, Aug1967, PDF p. 18; Stevenson, 1987, PDF pp. 139-140). This change eliminated the radiological control problems associated with naturally-radioactive thoria. Once Room 25 was decontaminated after this change, the potential for thorium exposures at the FCF would have been eliminated.

Hot-Line Step 7 – Pin Processing: The first step in the pin-processing cycle consisted of breaking the pregnant Vycor® glass pin molds away from the cast fuel pins. The molds were placed over a slit that was narrower than the molds but wider than the cast fuel pins. A pneumatically-driven flat-edge blade forced the mold against the slit, cracking the glass, and forcing the fuel pins through the slit. The fuel pins were then sheared to the required length. At the length-and-weight measuring station, the pins were checked for weight, length, diameter, porosity, and surface defects and voids. (Fuel Cycle Facility, undated, PDF pp. 15-16)

Hot-Line Step 8 – Cladding Fabrication and Sodium Loading: Cladding fabrication and sodium loading were two non-radiological processes that were performed outside of the FCF's Air and Argon Cells. The completed jacket assemblies (empty fuel-cladding tubes) were visually inspected for evidence of superficial damage (e.g., faulty welds, scratches) and leak-checked. Acceptable jacket assemblies were then loaded with sodium in an argon-filled glovebox located in FCF Room 23. Batches of sodium-loaded jackets were sealed in polyethylene bags before transferring them from the glovebox in Room 23 to the Argon Cell (Fuel Cycle Facility, undated, PDF pp. 16–17).

Hot-Line Step 9 – Fuel Element Assembly: All final fuel-element-closure operations were conducted remotely in the Argon Cell. After the sodium-loaded jackets were received in the Argon Cell, fuel pins were then inserted in their respective jackets and the sodium was melted in a shell furnace. End plugs (caps) were inserted into the jackets and peripherally-welded to the jackets. The finished product (now called a fuel element) was visually inspected for welding flaws. With the fuel material and bond sodium protected by the end plugs, the finished elements were transferred to the Air Cell for leak-testing (Fuel Cycle Facility, undated, PDF pp. 17).

Hot-Line Step 10 – Leak Testing: The closure welds on the newly-welded fuel elements were leak-tested for integrity with the helium-pressure leak detector located in the Air Cell. The accepted elements were placed in the bonding magazine located in the bonder magazine turntable. (Stevenson, 1987, PDF pp. 177–178)

Hot-Line Step 11 – Sodium Bonding: All finished elements were subjected to an impact-bonding process. Following this process, the elements were cooled in a manner that allowed the bond sodium to freeze from the bottom upward, thus eliminating the possibility of shrinkage voids and contractional tearing (Fuel Cycle Facility, undated, PDF p. 18).

Hot-Line Step 12 – Sodium Bond Testing: The final step in testing activities consisted of an examination of bond quality. Elements, one at a time, were raised upward through an eddy-current coil that sensed voids as an electrical disturbance in an otherwise passive signal. The acceptable elements were then removed from the magazine and placed in a 91-element storage holder. When the 91-element holder was filled, it was transferred to the final assembly station for loading into a subassembly. Elements rejected for one reason or another were fed back to the fuel stream by stripping away the jackets and returning the fuel material to the front end of the melt-refining cycle. (Stevenson, 1987, PDF pp. 178–179; Fuel Cycle Facility, undated, PDF pp. 19-20)

Hot-Line Step 13 – Subassembly Fabrication: In the Air Cell, completed fuel elements were loaded into fuel assemblies using master-slave manipulators. The completed subassemblies were tensile-tested and tested for straightness prior to going to the EBR-II Reactor (Fuel Cycle Facility, undated, PDF p. 20).

Hot-Line Step 14 – Scrap Reclamation & Waste Disposal: Two scrap streams from the process contained a sufficient quantity of uranium to warrant recovery. The first stream was the mixture of uranium and fission product oxides obtained from the used melt-refining crucibles, and uranium oxide in slags from scrap consolidation or alloy-preparation melts. The uranium from this stream represented from 3% to 8% of the total uranium in the charge (Stevenson, 1969). As

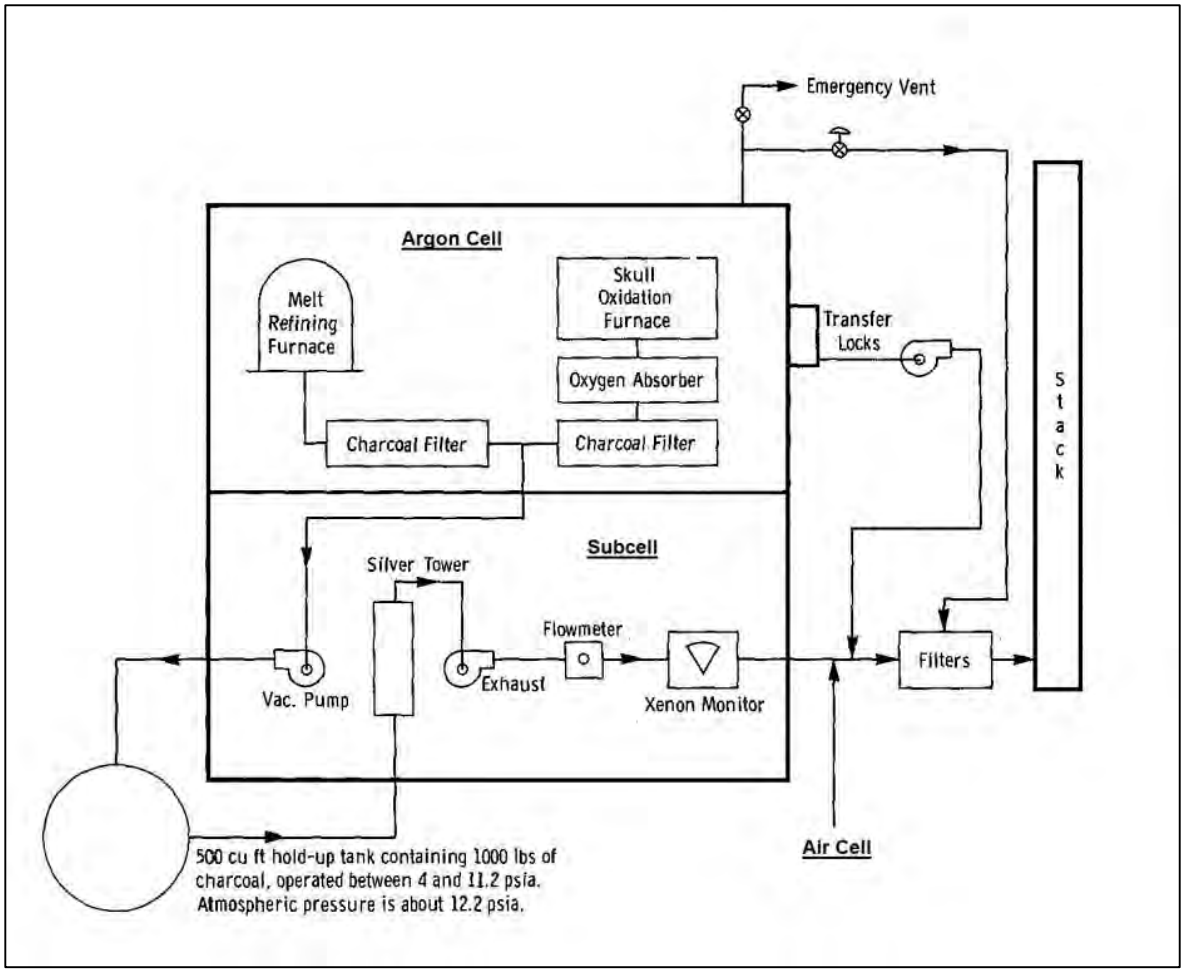
previously indicated in Step 5, cans of the uranium oxide were periodically shipped to the ICPP via a heavily-shielded "oxide coffin" to recover the enriched uranium (Hesson, 1963, PDF p. 77; Stevenson, 1987, PDF pp. 98, 102-104). A large part of the fission-product activity is assumed to have gone with the skull oxide sent to the ICPP (Stevenson, 1987, PDF p. 212). The second stream consisted of crushed Vycor® glass pin molds containing on the order of 1% of the uranium charged to injection-casting as uranium-fissium alloy. To facilitate concentration of FCF effort on fuel production, the crushed Vycor® glass pin molds were also shipped to the ICPP for uranium recovery by acid leaching (Stevenson, 1969).

Process wastes, except for the caustic solution obtained by washing sodium-contaminated subassemblies, were all in form of solids or gases. The principal solid radioactive wastes were:

- Activated stainless-steel components of the fuel subassemblies that were removed in disassembly
- Activated stainless-steel cladding (and possibly some chips of fuel) and adherent sodium that was separated from the fuel pin in the de-cladding operation
- Zirconia crucibles, fume traps, and graphite ingot molds used to melt and cast the ingot of purified fuel in melt refining
- Graphite crucibles and stainless-steel mold pallets used to re-melt the purified ingot and to cast fuel rods by injection-casting
- Various contaminated tools and defective in-cell equipment

All solids, including the stainless-steel subassembly hardware, are placed in 11-inch diameter by 6-foot high steel containers equipped with lug clamp lids. Some of the wastes are secondarily contained in smaller steel containers with similar lids. These containers were then sent to the Radioactive Scrap and Waste Facility (RSWF) for disposal. (Stevenson, 1987, PDF p. 212; Stevenson, 1969)

The principal radioactive gaseous wastes were the off-gases from the melt refining, skull oxidation, and injection-casting furnaces; the scrubbing gas stream from sodium removal; the contaminated argon exhausted from transfer locks; the air cell exhaust; and the potentially-contaminated building exhaust from operating areas. Figure 5-36 depicts the gaseous effluent treatment system for the FCF's Air and Argon Cells. Iodine is largely absorbed, and krypton and xenon are held for up to 30-days' decay before release through a 200-foot stack under favorable wind conditions. Any residual iodine is absorbed on a hot $AgNO_3$ bed (a.k.a. silver tower). (Stevenson, 1987, PDF p. 212; Stevenson, 1969)



Source: Hesson, 1963

Figure 5-36: Gaseous Effluent Treatment System for FCF Air and Argon Cells

Overview of the FCF Cold-Line Fuel Production Process

Cold-Line (unirradiated) fuel-production processes (which took place outside the FCF's Air and Argon Cells) were very similar to the Hot-Line processes; the only major differences being in the first four steps. The Cold-Line process did not include the Transfer of Subassemblies to FCF, Sodium Removal, Disassembly, Decanning, and Pin Chopping steps. The Cold-Line's alloy preparation was very similar to the Hot-Line's melt-refining step, with the main difference being that fission products and other impurities were not in the Cold-Line's feedstock (Stoddart, 1967; Hass, 1979). Therefore, Cold-Line fuel production only involved 12 major process steps.

The 12 Cold-Line fuel production process steps are summarized below.

NOTE: The Hass document (Hass, 1979) appears to only describe Cold-Line fuel production at ANL-E versus the Cold-Line fuel production at ANL-W. Even though the processes at ANL-E and ANL-W were nearly identical, there were some differences. The most notable differences included: (1) the mold coatings; (2) alloy preparation crucibles; and

(3) the make and design of some of the equipment. It should also be noted that many of the pictures provided in the Hass document appear to be from the Cold-Line facilities at ANL-E versus ANL-W.

Cold-Line Step 1 – Alloy Preparation: For the Cold-Line melt runs, either 10 kg or 20 kg charges were loaded into a new zirconia (ZrO_2) crucible. Each charge was melted to create the alloy, and then tilt-poured into a graphite ingot mold. After cooling, the ingots were removed, weighed, sampled, and held for injection-casting of the fuel pins. (Stoddart, 1967; Hass, 1979)

Prior to each run, the graphite ingot mold was coated with either a thoria/ethyl alcohol or zirconia/ethyl alcohol wash to preclude the melt from reacting with and degrading the graphite ingot mold. The Reactor Development Program Progress Reports for September and December 1967 indicate that thoria was being used through September 1967; however, by December 1967, the use of thoria had been phased out (Progress Report, Dec1967, PDF p. 102; Progress Report, Sept1967, PDF p. 19).

Cold-Line Step 2 – Skull Recovery: After alloy preparation, the crucibles, which had a uranium-bearing melt residue (a.k.a. skull) adhered to them, were processed to remove the skulls. Early Cold-Line production information indicates that a mechanical process was used, and that process may have been as simple as breaking the crucibles, or may have involved chipping the skull material out with a hammer and screwdriver (Stoddart, 1967). A Health Physics Report for October 1967 indicates that this process was performed in FCF Room 26 (Health Physics Monthly Report, Dec1968, PDF p. 23; Health Physics Monthly Report, Jan1968, PDF p. 25).

Cold-Line Step 3 – Pin Fabrication: The Cold-Line's pin-fabrication process, which was performed in FCF Room 20, was nearly identical to the Hot-Line's pin-fabrication process that was performed in the Argon Cell (Hass, 1979, PDF pp. 15-20). Details of this process were previously described in Hot-Line Step 6.

Cold-Line Step 4 – Pin Processing: The Cold-Line's pin-processing step was also performed in FCF Room 20 (Health Physics Monthly Report, Dec1968, PDF p. 14). After the cast fuel pins had cooled, they were moved to a hood, where the Vycor® glass pin molds were broken and the fuel pins removed. The pins were sheared to length in preparation for inspection. The pins were then checked for length, weight, diameter, internal flaws, texture, and compositional requirements. (Hass, 1979, PDF pp. 20-22)

Cold-Line Step 5 – Cladding Fabrication and Sodium Loading: The Cold-Line's cladding-fabrication and sodium-loading processes were identical to Hot-Line's processes, as previously described in Hot-Line Step 8, with the only exception being that materials were not transferred to the Argon Cell (Hass, 1979, PDF pp. 22-31). As with the Hot-Line, all sodium-loading operations were carried out in an argon-filled glovebox located in FCF Room 23.

Cold-Line Step 6 – Fuel Element Assembly: The Cold-Line's fuel element assembly was performed in a glovebox with an inert-argon atmosphere, instead of the FCF's Argon Cell (Hass, 1979, PDF pp. 27-34). It is uncertain where this process step was performed. However, based on the available survey records, this step was most likely performed in the Cold-Line glovebox located in FCF Room 23 (Survey Reports, Feb1969). The general process was similar to the Hot-Line's processes, as previously described in Hot-Line Step 9 (Hass, 1979, PDF pp. 35–35).

Cold-Line Step 7 – Leak Testing: Because the newly-welded fuel elements would need to be leak-tested before they were removed from the inert-argon atmosphere, the leak-testing was likely also performed in the Cold-Line glovebox located in FCF Room 23. The general process was similar to the Hot-Line's processes, as previously described in Hot-Line Step 10 (Hass, 1979, PDF pp. 36–38).

Cold-Line Step 8 – Sodium Bonding: Based on the available survey records, this step was performed in the Low Bay of the Inspection Test Facility (Survey Reports, Feb1969). The general process was similar to the Hot-Line's processes, as previously described in Hot-Line Step 11 (Hass, 1979, PDF pp. 38–39). However, the process for producing Mark-II fuel elements included the extra step of dimpling the fuel elements. Because the Mark-II fuel elements did not rely on internal restrainers to keep the fuel pins in position within the elements, three dimples were impressed into the Mark-II fuel elements 120^{o} circumferentially apart from each other (Hass, 1979, PDF p. 40).

Cold-Line Step 9 – Sodium Bond Testing: Based on the available survey records, this step was performed in the Low Bay of the Inspection Test Facility (Survey Reports, Feb1969). The general process was similar to the Hot-Line's processes, as previously described in Hot-Line Step 12 (Hass, 1979, PDF pp. 41–39).

Cold-Line Step 10 – Element Final Inspection: This was an added step that was not performed in the Hot-Line. Based on the available survey records, this step was performed in a new room that was added on to the southwest end of the Inspection Test Facility (Survey Reports, Feb1969). All fuel elements were inspected for dimensional conformance and surface quality. Special calibrated inspection gauges were used to verify conformance-to-length, straightness, and dimple location. All fuel element cladding surfaces had to be free of splits, cracks, seams, and inclusions and also had to be free of scratches, dents, or off-specification marks (Hass, 1979, PDF pp. 46–47). In addition, one fuel element was selected from each batch for destructive evaluation. The cladding from that element was stripped from the fuel and prepared for metallographic examination. The bare fuel pins were evaluated for texture, grain size, and extraneous phases (Hass, 1979, PDF p. 47).

Cold-Line Step 11 – Subassembly Fabrication: This step was performed in the High Bay of the Inspection Test Facility. The general process was assumed to have been similar to the Hot-Line's processes, as previously described in Hot-Line Step 13.

Cold-Line Step 12 – Scrap Reclamation & Waste Disposal: Because unirradiated feedstock was used for the Cold-Line, a much higher percentage of the Cold-Line materials could be reclaimed and recycled than from the Hot-Line. In the Cold-Line, the primary uranium reclamation was from the recovery of, and reuse of, the alloy-preparation skulls and injection-casting heels, which were in forms that could be returned directly back into the Cold-Line processes. No information is available on what was done with the crushed Vycor® glass pin molds from the Cold-Line, which would have contained about the same amount of residual uranium as the crushed glass from the Hot-Line. Because no acid-leaching process was set up at ANL-W to recover that uranium, it is assumed that crushed Vycor® glass pin molds from the Cold-Line were also shipped to the ICPP for uranium recovery by acid-leaching.

Details regarding waste disposal for the Cold-Line are not available.

FCF Timeline Through 1979

The timeline in Table 5-3 summarizes important dates in the FCF's operating history through 1979.

Table 5-3: Important Dates in FCF Operating History

Date	Activity
December 1962	Construction of the FCF completed (Stevenson, 1987, PDF p. 15).
March 1963	Air Cell significantly contaminated from removal of a several-thousand-curie antimony source for the EBR-II Reactor from its capsule, and while attaching it to a suitable fixture for handling it in the reactor. Contamination was due to some loose highly active deposit on the surface of the tantalum-clad source. Decontamination of the Air Cell was performed and completed. (Progress Report, Mar1963, PDF p. 30)
August 1963	Start of operations in the Argon Cell (Research Highlights, 1965, PDF p. 22). The first melting of uranium (depleted) was accomplished satisfactorily in the melt-refining furnace system (Progress Report, Aug1963, PDF p. 18).
Early-1964	Installation of process equipment was completed. Unirradiated dummy fuel elements and subassemblies containing unenriched uranium fuel alloy were fabricated and carried through all the process steps to determine where modifications might be needed and to verify equipment operability. (Stevenson, 1987, PDF p. 248)
January 31 to February 27, 1964	On January 31, 1964, the Argon Cell was filled with air to allow personnel to enter the cell to perform maintenance on cranes and manipulators, to speed installation of additional equipment, and to remove mercury which had spilled into the cell from a blown out blister seal located near the cell's ceiling. On February 27th, the Argon Cell was refilled with argon. (Progress Report, Feb1964; Progress Report, Mar1964; Research Highlights, 1965, PDF p. 23)
March 1964	All the fuel element fabricating equipment in the Argon Cell became operational (i.e. the melt-refining equipment, injection-casting equipment, pin-processing equipment, and fuel element settling and welding equipment). Fuel element can now be made in the Argon Cell. (Progress Report, Mar1964)
September 1964	One core subassembly and one blanket subassembly were transferred from the EBR-II Reactor to the FCF, and processing of the first irradiated reactor fuel from the EBR-II reactor was initiated in the FCF's Hot-Line (Progress Report, Sep1964, PDF p. 52; Stevenson, 1987; Koch, undated; 148941).
April 1965	The manufacturing of the first subassembly containing irradiated fuel was completed. (Koch, undated; Stevenson, 1987, PDF p. 248)
June 1965	The first two cans of FCF waste were shipped to the Radioactive Scrap and Waste Facility (RSWF), which was also the first use of the RSWF (Progress Report, Jun1965, PDF p. 12).
April 1967	To reduce the number of transfers through the Air Cell, a new opening was made in the wall of the transfer subcell between the Air and Argon Cells. The creation of the new openings included the installation of a window and transfer lock. This transfer system was created to allow all the unirradiated materials, such as crucibles, molds, and waste cans, to be introduced directly into the Argon Cell via the large lock rather than having to traverse the whole length of the Air Cell. When the mobile decontamination facility is completed, this lock system will also be used to remove equipment components for cleaning and repair. (Progress Report, Apr1967, PDF p. 30)

Date	Activity
July 1967	Installation of the FCF Cold-Line's alloy-preparation furnace was completed. Five test runs were completed using depleted uranium-fissium alloy. (Progress Report, Jul1967, PDF p. 66) Because the FCF's in-cell storage capacity is now fully utilized, subassemblies being fabricated with unirradiated fuel are being stored in the L&O Building vaults (Progress Report, Jul1967, PDF p. 65).
November 1967	The Cold-Line equipment became available for routine operations, and routine Cold-Line production began (Progress Report, Nov1967, PDF pp. 108-109).
January 1968	The first four uranium-5 w/o fissium ingots manufactured by a commercial vendor were received by ANL. The ingots were used in the FCF's Cold-Line injection-casting furnace. (Progress Report, Jan1968, PDF p. 96)
April 1968	Construction of the truck lock and the second-story addition to the FCF office building was started. The truck lock provided a ventilation barrier for loading and unloading casks from trucks, and an area for decontaminating truck beds and casks. (Progress Report, Apr1968, PDF p. 104) The first neutron radiographs were performed in the Argon Cell (Progress Report, Apr1968, PDF p. 92).
October 1968	Announcement of programmatic changes for the FCF, and that there will be a gradual decrease in Hot-Line fuel production and an increase in support on the experimental irradiation programs (Progress Report, Oct1968, PDF p. 94).
December 1968	Modifications of the transfer cell (a.k.a. Subcell 4) below the Air and Argon Cells were completed. These included: (1) placing into operation the pneumatic transfer system for conveying samples to the analytical caves in the L&O Building; (2) installing and testing the movable shield plug for the entrance to the subcell; (3) installing two master-slave - manipulators; and (4) installing additional lighting equipment in the subcell. (Progress Report, Dec1968, PDF pp. 75, 84)
1969	Permanent decontamination and repair facilities were constructed on the FCF roof and in the FCF basement (Cook, 1969, PDF p. 17).
March 1969	Phasing out of Hot-Line production activities in the Argon Cell was completed (Progress Report, Mar1969). Fuel elements remaining in the Air Cell were being processed for use in subassemblies (Progress Report, Mar1969). During the operation of the FCF's Hot-Line roughly 35,000 fuel elements (the equivalent of 5 EBR-II Reactor core loadings) were recycled and returned to the EBR-II reactor, in the form of over 400 fuel subassemblies (Cook, 1975; Stevenson, 1987, PDF p. 260). Cold-line equipment was modified for fabrication and inspection of Mark-II fuel elements. Preparation of uranium-fissium metal ingots for the Mark-II fuel alloy started in the alloy-preparation furnace (FCF Room 26) of the Cold-Line. (Progress Report, Mar1969)
December 1969	Cold-Line fuel pin and fuel element production equipment put on stand-by (Progress Report, Dec1969, PDF p. 87). However, Cold-Line continued to fabricate unirradiated subassemblies from vendor produced fuel elements and perform impact-bonding of the unbonded vendor fuel elements (Progress Report, Dec1969, PDF p. 87). Fuel-pin fabrication in the FCF was not reinitiated until 1982 (Batte, 1986, PDF p. 9).
1969	The first refurbishment of the Air Cell, which was assumed to take place after all irradiated fuels had been removed from the Air and Argon Cells (HFEF History, 1979).
July 1970	Reactor Development Program Progress Report for July 1970 identifies the FCF as being part of both the Hot Fuel Examination Facilities and EBR-II Operations (Progress Report, Jul1970).

Date	Activity
August 1970	The FCF was renamed as the Fuel Examination Facility (FEF), to reflect the change in emphasis and nature of the work being done in the facility and planned for it (Progress Report, Aug1970). However, after that date, documents still refer to this facility as being both the FCF and FEF, sometimes in the same document.
1972	The FCF was renamed the Hot Fuel Examination Facility-South (HFEF-S) after construction on the Hot Fuel Examination Facility-North (HFEF-N) was completed. (Stevenson, 1987, PDF p. 260)
Early-1973	FCF Argon Cell was remotely decontaminated in preparation for RAS-TREAT sodium loop experiment work. The RAS-TREAT sodium loop experiments were sponsored by ANL-E's Reactor Analysis and Safety Division, and were irradiated in the TREAT Reactor. (Cook, 1975, PDF p. 24)
July 1973–April 1974	Several plutonium bearing Mark-IIA loop experiments were sectioned in the FCF's Argon Cell as part of the posttest support of the RAS-TREAT loop experiments. The purpose of sectioning the loops is to reduce the loop experiments into smaller segments for shipment to the Materials Science Division (MSD) at ANL-E, where the detailed posttest examination of the loop experiments is performed. During this period, loops L3, L4, E6, E7, H4, and H5 were sectioned and their ALIPs (annular linear-induction pumps) were recovered. This activity contaminated the Argon Cell with plutonium. (Cook, 1975)
1976	The FCF equipment mockup/qualification area was substantially enlarged (HFEF History, 1979). The second refurbishment of the Air Cell was performed in 1976 (HFEF History, 1979).
1977	Argon Cell was shut down for to correct several problems that had developed over its 13-year life. Principally, the overhead handling systems had deteriorated, but in-cell illumination and the viewing windows also needed improvement. (Stevenson, 1987, PDF p. 260)
June 23, 1978	First entry into the Argon Cell since February 1964 (HFEF History, 1979; Entry Log; 1979; Research Highlights, 1965, PDF p. 23). Operations involving irradiated fuels began in the Argon Cell began after February 1964 (Entry Log; 1979). The Argon Cell was entered in February 1964, which was after some work involving unirradiated dummy fuel elements and subassemblies containing unenriched uranium fuel alloy was performed in the Argon Cell (Stevenson, 1987, PDF p. 248; Research Highlights, 1965, PDF p. 23).
1987	By 1987, operations in the Argon Cell had not restarted. (Stevenson, 1987, PDF p. 260)

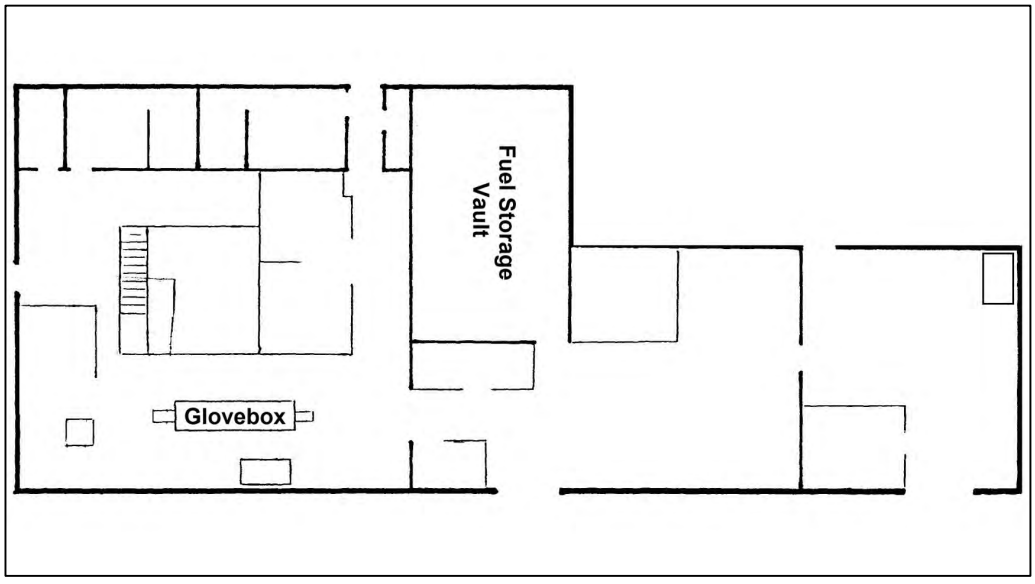
5.1.2.4 Fuel Assembly & Storage Building (FASB)

ATTRIBUTION: Section 5.1.2.4 was completed by Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding FASB were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to FASB are explained in the associated text.

In Figure 5-24, the FASB is Building 787 located adjacent to the north side of the FCF. The FASB is also known by the following building numbers: ANL-787, EBR-II-627, and MFC-787. The building was also sometimes referred to as the "FAS Building." In later years (i.e., sometime after 1979), this facility was renamed the Fuels and Applied Science Building, which preserved the acronym "FASB." Figure 5-37 is based on a radiological survey map that shows the general FASB layout.

Construction of this 6,025 ft^2 (560 m^2) building was completed in 1971 (Site Plan, 1981). The structure features masonry exterior walls, built-up roof on pre-stressed concrete beams, a reinforced concrete floor, and no windows (Site Plan, 1981). The building was equipped for assembling EBR-II driver-fuel pins into fuel elements, fabricating EBR-II driver and experimental subassemblies, performing inspections on fuel elements and finished subassemblies, and storing fuel elements and finished subassemblies (Site Plan, 1981). The facility was also use to perform air-flow tests on subassemblies (Site Plan, 1981).

Based on radiological survey records, Cold-Line (unirradiated) fuel production at the FASB appears to have begun around February 1972. An October 26, 1972 safety review for the FASB's Fuel Storage Vault indicates that the vault was not used until sometime after October 1972 (FASB Vault, 1972).



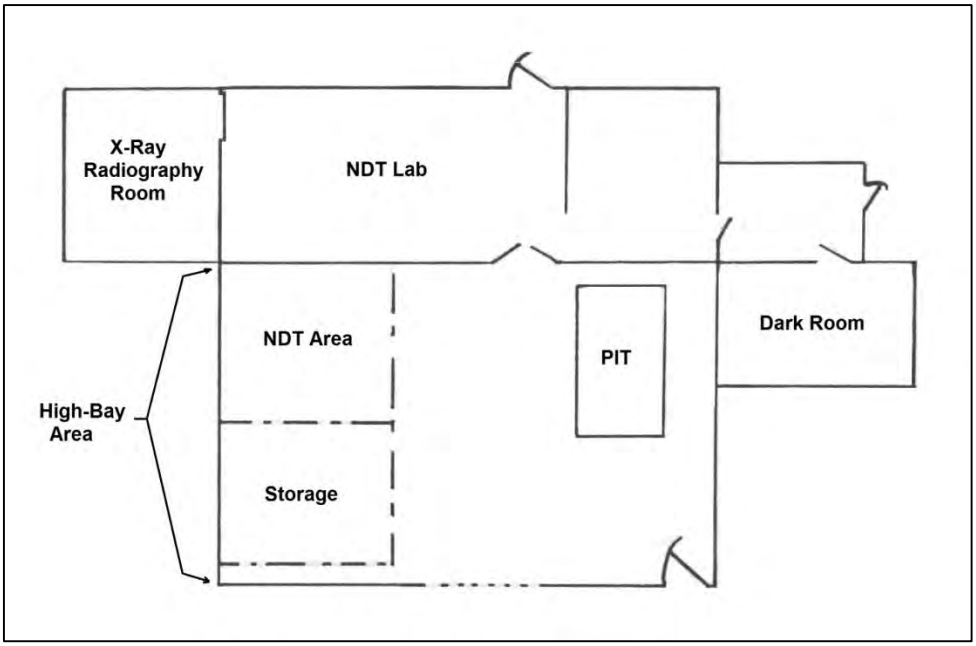
Source: FASB Surveys, 1983, derived from figure on PDF p. 35

Figure 5-37: General Layout of FASB

5.1.2.5 Inspection & Test Facility (ITF)

ATTRIBUTION: Section 5.1.2.5 was completed by Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding ITF were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to ITF are explained in the associated text.

In Figure 5-24, the ITF is Building 772 located approximately 350 ft. (107 m) east of the FCF. The ITF is also known by the following building numbers: ANL-772, EBR-II-616, and MFC-772. Construction of this 3,455 ft^2 (320 m^2) building was completed in 1966 (Site Plan, 1981). The structure features masonry exterior walls, steel framing, and corrugated sheet metal siding on the high-bay area (Site Plan, 1981). Figure 5-38 is based on a radiological survey map showing the general ITF layout.



Source: ITF Survey Data, 1976, derived from figure on PDF p. 29

Figure 5-38: General Layout of ITF

The ITF was originally constructed to inspect and test EBR-II reactor components (including in-sodium tests), and to perform nondestructive testing of experimental irradiation specimens and reactor components (Cold-Line Facility, 1969; Site Plan, 1981). By August 1967, those activities were curtailed by the pressing demand for a temporary Cold-Line fuel production facility, which included parts of the ITF and FCF (Cold-Line Operation, 1967; Cold-Line Facility, 1969; Site Plan, 1981). Because of that effort, most of the high-bay and half of the low-bay of the ITF were converted for Cold-Line fuel production (Cold-Line Facility, 1969). Based on a review of ITF radiological survey records, most of its Cold-Line work involved installing fuel elements and experiments into subassemblies for the EBR-II Reactor. The radiological survey records also indicate that irradiated fuels and materials were sometimes handled at the ITF.

By July 1969, plans were being made to convert the ITF back to its original tasks as soon as possible (Cold-Line Facility, 1969). Cold-Line operations at the ITF appear to have been significantly scaled back after the permanent Cold-Line fuel production facility (FASB) started operating in February 1972. After fuel production operations began at FASB, radiological survey records and health physics log books indicate that the Cold-Line operations at the ITF were reduced to performing X-ray radiography on fuel pins.

Prior to the HFEF-N's NRAD Facility becoming operational in March 1978, the 20-ft (6.1-m) deep pit in the ITF's high-bay area was fitted with a tank for receiving the NRAD Reactor fuel (Neutron Radiography, 1979; Site Plan, 1981). Radiological survey records indicate that the pit in the high-bay may have been present when the ITF was constructed, and initially may have been used to perform X-ray radiography on subassemblies (ITF Survey Data, PDF pp. 20-21).

By February 1981, ITF activities included: (1) testing non-radioactive EBR-II components using X-ray, ultrasonic, eddy-current, thermal, and other nondestructive methods; and (2) rigging, tackle, and load testing (Site Plan, 1981).

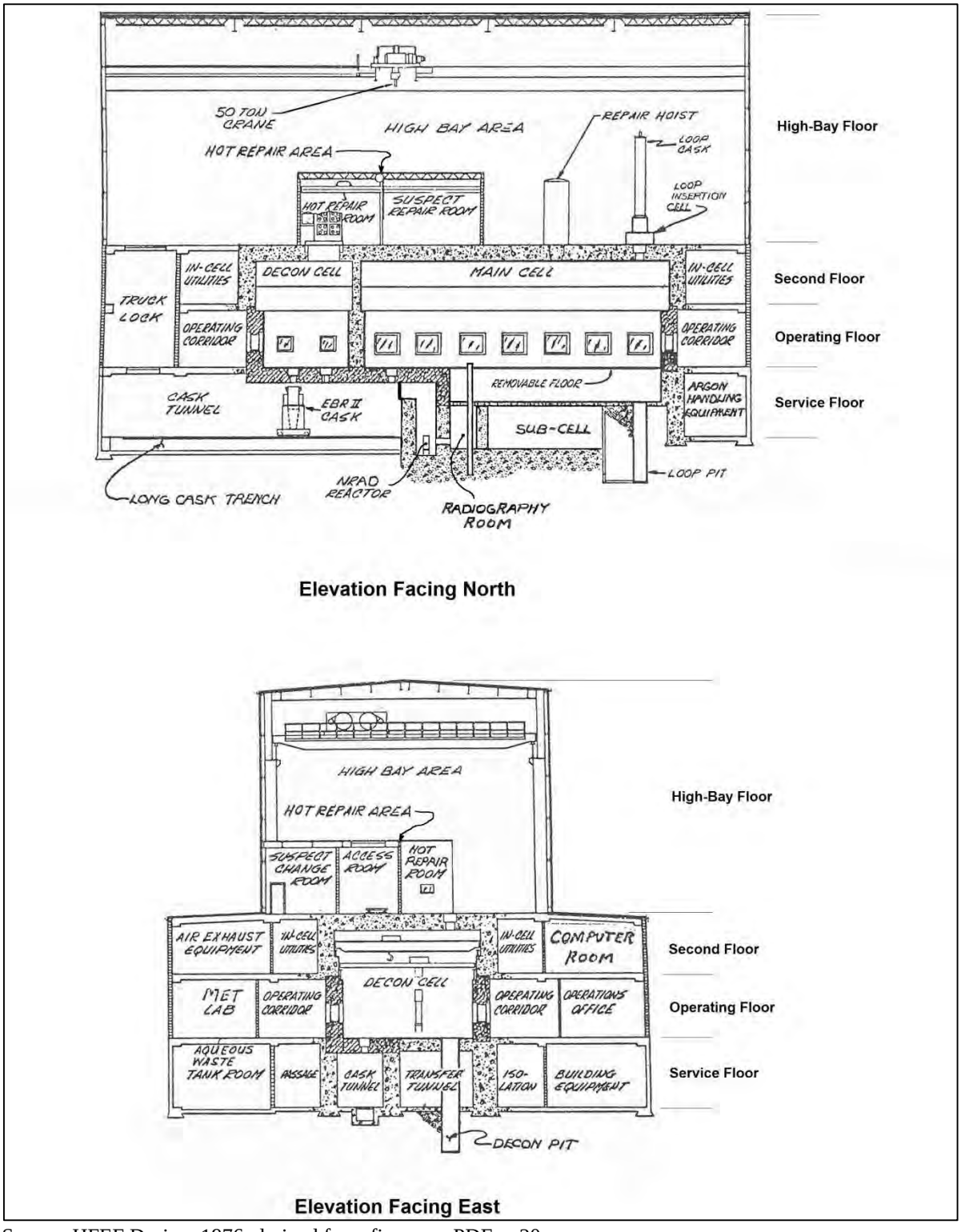
5.1.2.6 Hot Fuel Examination Facility-North (HFEF-N)

ATTRIBUTION: Section 5.1.2.6 was completed by Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding HFEF-N were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to HFEF-N are explained in the associated text.

HFEF-N is a large hot-cell facility consisting of heavily-shielded cells, support laboratories, and support equipment (HFEF Training, 2009; HFEF Design, 1976). The facility was specifically designed to remotely characterize highly-irradiated fuel and structural materials (HFEF Training, 2009). Radiological operations started at HFEF-N in August 1972 and continue to the present.

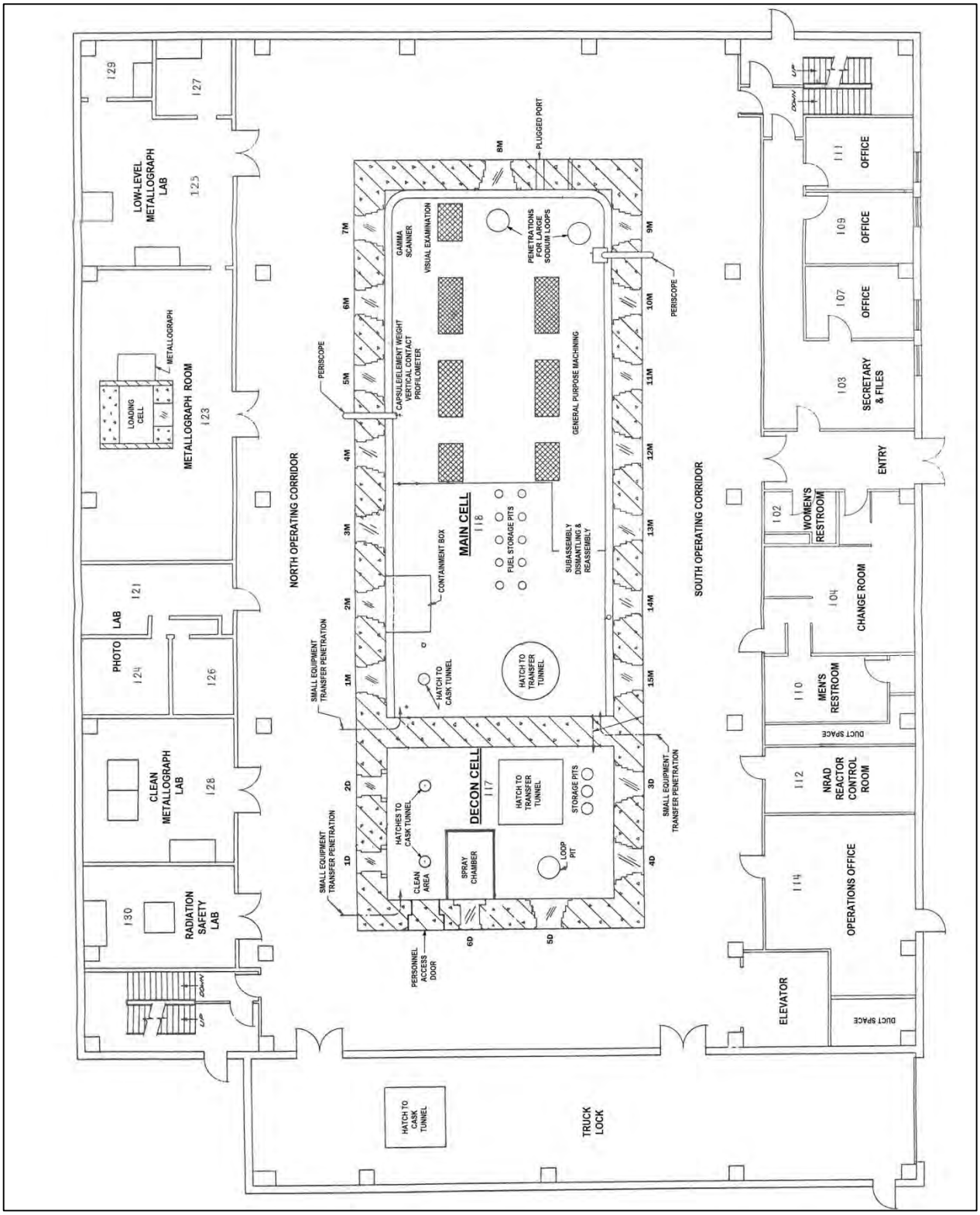
The initial function of the HFEF-N was to prepare and examine irradiation experiments in support of the Liquid Metal Fast Breeder Reactor (LMFBR) Program (HFEF Safety Report, 1975). To support LMFBR fuel and structural material development, the materials examined at the HFEF-N were usually irradiated in the EBR-II Reactor (HFEF Safety Report, 1975). To support the LMFBR safety programs, HFEF-N was used to prepare test specimens and examine sodium loop experiments irradiated in the TREAT Reactor, INL's Engineering Test Reactor (ETR), or INL's Power Burst Facility (PBF) (HFEF Safety Report, 1975). Also, at times, irradiated materials from other programs and facilities were accommodated at HFEF-N (HFEF Safety Report, 1975).

In Figure 5-24, HFEF-N is Building 785, and is located approximately 175 ft. (53 m) north of the FCF. The HFEF-N is also known by the following building numbers: ANL-785, EBR-II-626, and MFC-785. After HFEF-S was renamed the Fuel Conditioning Facility in 1993, HFEF-N was only referred to as HFEF, which was what the FCF was sometimes referred to prior to the HFEF-N being constructed. Figure 5-39 shows an elevation cross-section of HFEF-N. Figure 5-40 is a floor plan for the HFEF-N Operating Floor. Figure 5-41 is an elevation detail of the NRAD Reactor and Radiography Room at HFEF-N.



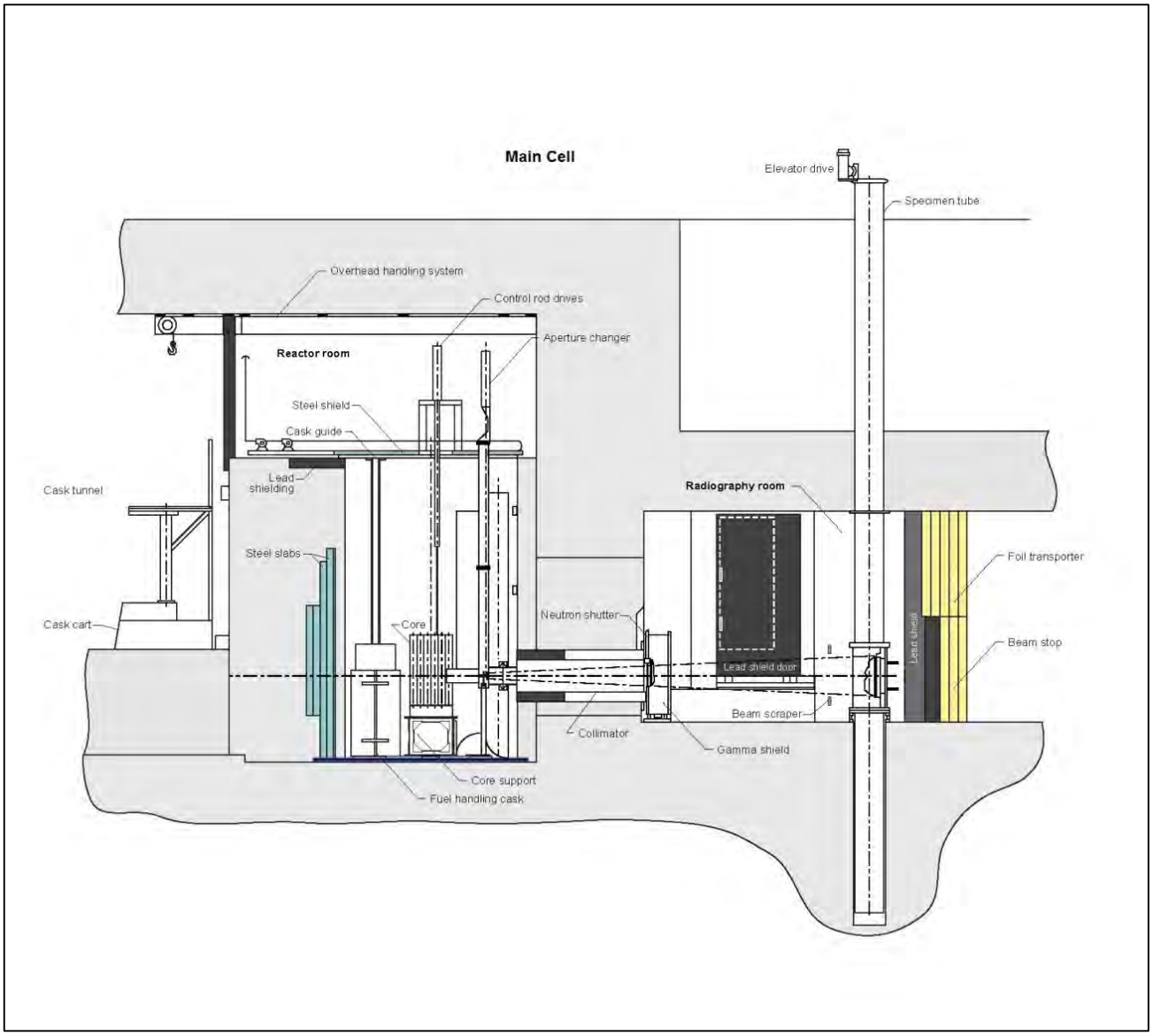
Source: HFEF Design, 1976, derived from figure on PDF p. 20

Figure 5-39: Elevation Cross-Sections of HFEF-N



Source: HFEF Design, 1976, derived from figure on PDF p. 21

Figure 5-40: Floor Plan of HFEF-N Operating Floor



Source: Bess, 2014, PDF p. 16

Figure 5-41: Elevation Detail of the NRAD Reactor and Radiography Room at HFEF-N

Facility Description

HFEF-N is structurally designed to accommodate support equipment on all sides of the centrally-located hot cells. This large heavily-shielded hot-cell facility includes two adjacent hot cells (the Main Cell and Decon Cell). The building is framed with concrete that is heavily-reinforced with steel beams. From ground-level to the top of the second-floor area, the exterior walls are concrete block. From the top of the second floor upward, the exterior walls are covered in sheet metal, insulation, and an interior lining. The building floors are made of steel-reinforced concrete (HFEF Design, 1976).

The three-story HFEF-N building is 112 ft. wide x 154 ft. long, has a gross floor area of 56,570 ft^2, and a gross volume of 1,337,200 ft^3. The building is designed with ample space and heavy-duty handling equipment to accommodate large irradiated loops, such as the Sodium Loop Safety Facility (SLSF), loops from INL's Engineering Test Reactor, as well as smaller equipment for handling reactor subassemblies (see Figure 5-39). Numerous penetrations into the cells provide easy access in transporting and handling irradiated materials into and out of the heavily-shielded cells. In-cell equipment and out-of-cell laboratories provide for both destructive and nondestructive examination of the irradiated reactor components. (HFEF Design, 1976)

HFEF-N was designed as a post-irradiation examination facility and is fully equipped for destructive and nondestructive examinations. Following receipt of irradiated test materials or components, the specimens can be visually inspected, measured, documented, and subjected to gamma scanning, ultrasonic testing, neutron radiography, and other examinations. Destructive testing begins with cutting and preparation for metallographic and ceramographic optical examination, micro-hardness testing, and mechanical properties testing. Further preparation can be performed and samples transferred to other facilities for continued analytic chemistry or electro-optical analysis using transfer casks or a pneumatic transfer system between facilities (HFEF Training, 2009).

As shown in Figure 5-39, the HFEF-N building has four floors: (1) Service Floor (basement); (2) Operating Floor; (3) Second Floor; and (4) High-Bay Floor (HFEF Design, 1976):

The major HFEF-N radiological work areas are listed below and described in the following subsections.

- Main Cell
- Decon Cell
- Out-of-Cell Metallographic Areas
- Radiation Safety Laboratory
- Third Floor Repair Area
- NRAD Facility
- Truck Lock
- Cask Tunnel
- Transfer Tunnel

Main Cell

The heart of the HFEF-N facility where most of the examinations were carried out is the Main Cell (HFEF Safety Report, 1975). The Main Cel1 provides facilities for disassembling, examining, and testing irradiated materials in an inert (argon gas) atmosphere (HFEF Design, 1976). In anticipation of handling plutonium-bearing experiments, the Main Cell was designed for the particle-tight containment of any plutonium contamination (Site Plan, 1981).

The Main Cell has 15 workstations that are each 10-ft wide, and numbered "1M" through "15M." Each workstation can be equipped with a pair of manipulators, a shielding window, lights, a utility distribution system, examination equipment, and/or suitable work tables. A few work stations are equipped with ports for specific purposes (e.g., periscopes, stereomacroscopes, and gamma-scanning equipment). Work stations that are not provided with examination equipment at start-up can be fitted

with sealed and plugged ports for utilities. These items can be added later without releasing radioactive contaminants, polluting the cell atmosphere, or extensively interrupting work at adjacent stations (HFEF Safety Report, 1975).

Decon Cell

The Decon Cell is a decontamination hot cell with an air atmosphere that is equipped for decontaminating and packaging nuclear fuel capsules and elements for shipment (HFEF Design, 1976). It also provides a shielded path for transfers of radioactive material to and from the Main Cell's inert atmosphere, and has work stations for some sodium-loop operations (HFEF Design, 1976). The Decon Cell is separated from the west end of the Main Cell by a 4-ft-thick concrete wall (HFEF Safety Report, 1975; HFEF Design, 1976). The cell has outer-shielding walls equivalent to those of the Main Cell and has inside dimensions of 30 ft. wide x 20 ft. long x 25 ft. high. The cell floor is lined with stainless steel and the walls are lined with carbon steel to a height of 13.5 ft. above the cell floor. Wall, roof, and floor penetrations are provided for windows, utility-service feedthroughs, special feedthroughs, in-cell handling equipment, ducting, and transfer hatches. These penetrations are steel-lined and surrounded by high-density shielding inserts (HFEF Design, 1976).

The Decon Cell has six work stations that are numbered "1D" through "6D." Some are used for decontamination of equipment, irradiated components, and loop hardware; others are used for transfers and future projects (HFEF Safety Report, 1975).

The Decon Cell is maintained at a negative pressure relative to the operating corridor, the cask tunnel, the high-bay area, and the transfer tunnel to prevent the release of particulate contamination from the cell. Remotely-replaceable filters and de-misters are installed in the exhaust ducts within the cell. The spread of vapors from decontamination reagents is minimized by performing wet operations within closed and vented vessels such as the spray chamber (HFEF Safety Report, 1975).

Out-of-Cell Metallographic Areas

The Out-of-Cell Metallographic Areas are used for all but the high-level metallographic specimens. These areas are on the Operating Floor across the north operating corridor from the Main Cell (see Figure 5-36). These Metallographic Areas consist of four sub-areas (HFEF Design, 1976):

1. The Metallograph Room (Room 123) houses the metallograph loading cell and the scanning electron microscope (HFEF Design, 1976).

2. The Low-Level Metallograph Lab is a complex of rooms used for preparation of low-level, metallographic samples and unirradiated fuels. It includes the metallograph lab (Room 125), cutting room (Room 127), and macrophoto room (Room 129) (HFEF Design, 1976).

3. The Clean Metallograph Lab (Room 128) is used for preparation and microscopic examination of cold non-fissile metallographic specimens (HFEF Design, 1976).

4. The Photo Lab (Rooms 121, 122, 124, and 126) provides the support necessary for photographic examinations conducted during hot-cell and out-of-cell lab operations (HFEF Design, 1976).

High-level or "hot" metallographic specimens (fueled and non-fueled capsules and elements and other irradiated components) are prepared remotely in the in-cell containment box located at station 2M of the Main Cell (HFEF Safety Report, 1975).

Metallographic samples are prepared by means of cutting, mounting, grinding, polishing, and etching operations. Then both low-level and high-level specimens are examined by use of a remotely-operated, shielded optical metallograph. This gastight metallograph, with its gastight shielded enclosure (the metallograph loading box) is located in the Metallograph Room. Sample transfers between the in-cell containment box and the metallograph loading box are accomplished by use of a gastight and shielded pneumatic transfer system (HFEF Safety Report, 1975).

Radiation Safety Laboratory

The Radiation Safety Laboratory (Room 130) is on the Operating Floor across the north operating corridor from the Decon Cell (see Figure 5-36). This lab is used to detect and contain the spread of contamination into the environment or prevent undue radiation exposure to personnel (HFEF Design, 1976).

Hot Repair Area

The Hot Repair Area (a.k.a. Third Floor Repair Area, Roof Repair Area, and/or Repair Area) is a complex of rooms on the High-Bay Floor (third floor) directly above Decon Cell and the west end of the Main Cell (see Figures 5-35 and 5-37). The outside dimensions of this facility are 45 ft. x 46 ft. (HFEF Design, 1976). The Hot Repair Area includes two primary work rooms, the Hot Repair Room and Suspect Repair Room. It also includes the following work and support rooms: glovebox room, equipment access and transfer area, isolation area, waste packaging area, service room, and change room (HFEF Safety Report, 1975; HFEF Design, 1976).

The Hot Repair Area provides a containment enclosure in which radioactively-contaminated equipment or suspect (potentially-contaminated) equipment can be repaired or modified by personnel wearing protective clothing (HFEF Safety Report, 1975; HFEF Design, 1976). These operations include rebuilding of mechanical master-slave manipulators, repair of Main Cell and Decon Cell telescoping-tube and crane carriages, and maintenance or modification of examination equipment (HFEF Safety Report, 1975).

NRAD Facility

Through 1979, the Neutron Radiography Facility (NRAD or N-RAD) was mostly located on the HFEF-N Service Floor below the Main-Cell (see Figures 5-38 and 5-39). As shown in Figure 5-36, the NRAD Reactor control room is located in Room 112 on the Operating Floor (HFEF Design, 1976). NRAD uses a 250-kW TRIGA® (Training, Research, Isotopes, General Atomics) reactor as a neutron source for performing neutron radiography irradiations on small test components. By the end of 1979, NRAD was only equipped with a single beam tube; however, in 1982 the facility was expanded outside the Service Floor to incorporate a second beam tube. This eliminated the need for all test components to pass through the HFEF-N's Main Cell, which was significantly contaminated with alpha radioactivity (HFEF FSAR, 1977; NRAD Upgrade, 2014; North Radiography Station, 1982).

The NRAD reactor is a portion of the TRIGA-conversion-type reactor originally located at the Puerto Rico Nuclear Center (PRNC) (HFEF FSAR, 1977; NRAD Upgrade, 2014). In 1971, the PRNC Rector was converted to a TRIGA-FLIP (Fuel Life Improvement Program) system that used 70% enriched uranium (HFEF FSAR, 1977; NRAD Upgrade, 2014). In September 1976, the PRNC Reactor was shut down (HFEF FSAR, 1977; NRAD Upgrade, 2014). Then in 1977, a portion of the PRNC reactor fuel elements and other components (e.g., control rods, control-rod drives, and instrumentation/control console) were moved by the DOE to the ANL-W site and built into the NRAD Reactor (HFEF FSAR, 1977; NRAD Upgrade, 2014). The NRAD Reactor first went critical in October 1977 (NRAD Upgrade, 2014); it became operational in 1978 (HFEF History, 1979; NRAD Upgrade, 2014).

NRAD Facility is primarily used for neutron radiography analysis of both irradiated and unirradiated nuclear fuels and materials. Typical applications for examining the internal features of nuclear fuel elements and assemblies include: fuel pellet separations, fuel central-void formation, pellet cracking, evidence of fuel melting, and material integrity under normal and extreme conditions. Examination of the behavior of large test loops and assemblies can also be performed (NRAD Upgrade, 2014).

Truck Lock

The Truck Lock is the primary location where casks and equipment are transported to and from the HFEF-N (HFEF Safety Report, 1975).

Cask Tunnel

The 70 ft. long x 12 ft. wide x 13 ft. high Cask Tunnel is located on the Service Floor level and extends beneath the Truck Lock, Decon Cell, and a portion of the Main Cell (HFEF Design, 1976). The Cask Tunnel is used for transferring equipment and irradiated components to and from the Truck Lock to the Decon Cell and/or Main Cell.

Partition walls and controlled air-flow patterns between the various Service Floor areas are used to help prevent the spread of radioactive contaminants in the event of an accidental release (HFEF Design, 1976).

Transfer Tunnel

The heavily-shielded Transfer Tunnel is located on the Service Floor level and extends beneath the Decon Cell and a portion of the Main Cell (HFEF Safety Report, 1975; HFEF Design, 1976). The shorter Transfer Tunnel runs parallel to the Cask Tunnel; the two tunnels are separated by a thick concrete shield wall. The Transfer Tunnel is used for the movement of large equipment and irradiated components between the Decon Cell and Main Cell (HFEF Safety Report, 1975).

On the west end of the Transfer Tunnel, there is an entrance that permits infrequent personnel entry. This entry way is normally closed by inner steel doors and a rectangular shielding plug. This door is supported on air-bearing support pads and is manually opened and closed (HFEF Safety Report, 1975). When closed, the inner steel doors are sealed with duct tape to prevent outleakage of radioactive particulates in case the tunnel air pressure inadvertently becomes positive relative to other parts of the Service Floor (HFEF Safety Report, 1975; HFEF Design, 1976).

HFEF-N Operating History

HFEF-N construction was completed in 1972 (Site Plan, 1981). Documents indicate that HFEF-N was not activated until March 1975 (HFEF History, 1979; Bacca, 1985; Site Plan, 1981). However, survey records indicate that radiological work was initiated at HFEF-N well before March 1975. By August 1972, routine surveys of HFEF-N had been initiated; however, there is no indication of radiological work being performed until October 1972 (HFEF Survey Data, 1972-1973, PDF p. 14). A survey record indicates that grinding was performed on depleted uranium in October 1972 (HFEF Survey Data, 1972-1973, PDF p. 49); this was likely done outside the hot cells. A Special Survey Report indicates that "Pu oxide element E-6" was transferred into the Decon Cell on December 21, 1972 (HFEF Survey Data, 1972-1973, PDF p. 32).

In August 1976, a highly-sensitive fixed air-sampling system (FASS) was put into operation at HFEF-N. The system consists of 66 sampling stations dispersed throughout the facility to provide data on any radioactive particulates in the atmosphere. The sensitivity of the FASS enabled early detection of airborne alpha radioactivity in concentrations as low as 1×10^{-15} μCi/cm^3. The FASS data would alert operation personnel to airborne contamination problem and reflect the long-term trend in the effectiveness of contamination controls (HFEF History, 1979; Courtney, 1977).

The NRAD Reactor first went critical in October 1977 (NRAD Upgrade, 2014); the NRAD Facility became operational in early 1978 (HFEF History, 1979; NRAD Upgrade, 2014).

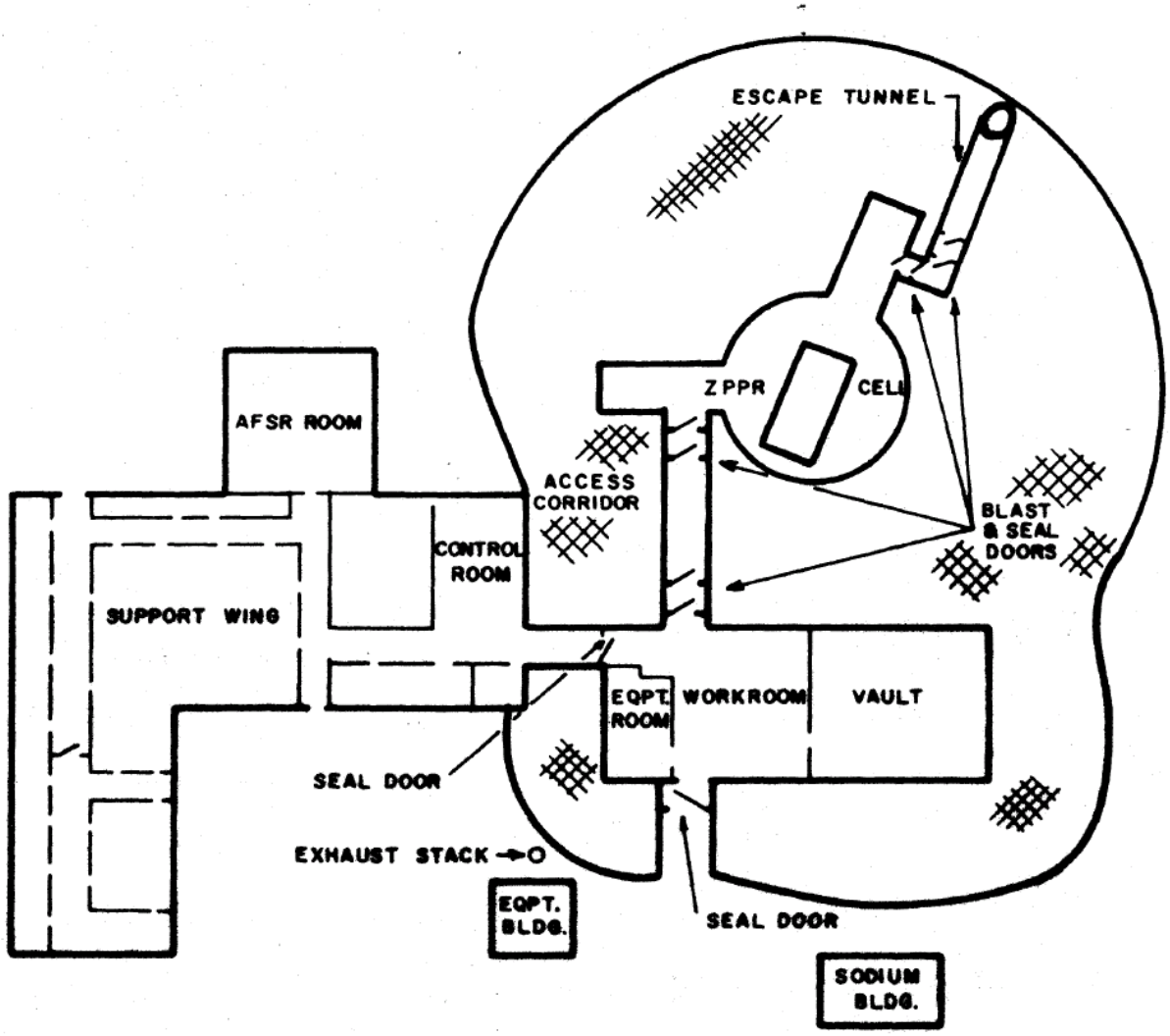
Between 1977 and 1979, the Hot Repair Area was extensively modified and expanded to facilitate the efficient repair of manipulators and other in-cell equipment, and to enhance the controls for preventing radioactive contamination of the High-Bay Area. The modifications to the Hot Repair Area provided (HFEF History, 1979):

- readily-decontaminated surfaces
- a two-station shielded glove-wall equipment repair area
- a glove box system for decontamination and repair of non-sealed master-slave manipulators
- a glove box system for repair of sealed manipulator slaves
- a buffer zone between the Hot Repair Area and the remainder of the High-Bay Area

5.1.2.7 Zero Power Plutonium Reactor (ZPPR)

ATTRIBUTION: Section 5.1.2.7 was completed by Jason Davis, Oak Ridge Associated Universities. All conclusions drawn from the data regarding ZPPR were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to ZPPR are explained in the associated text.

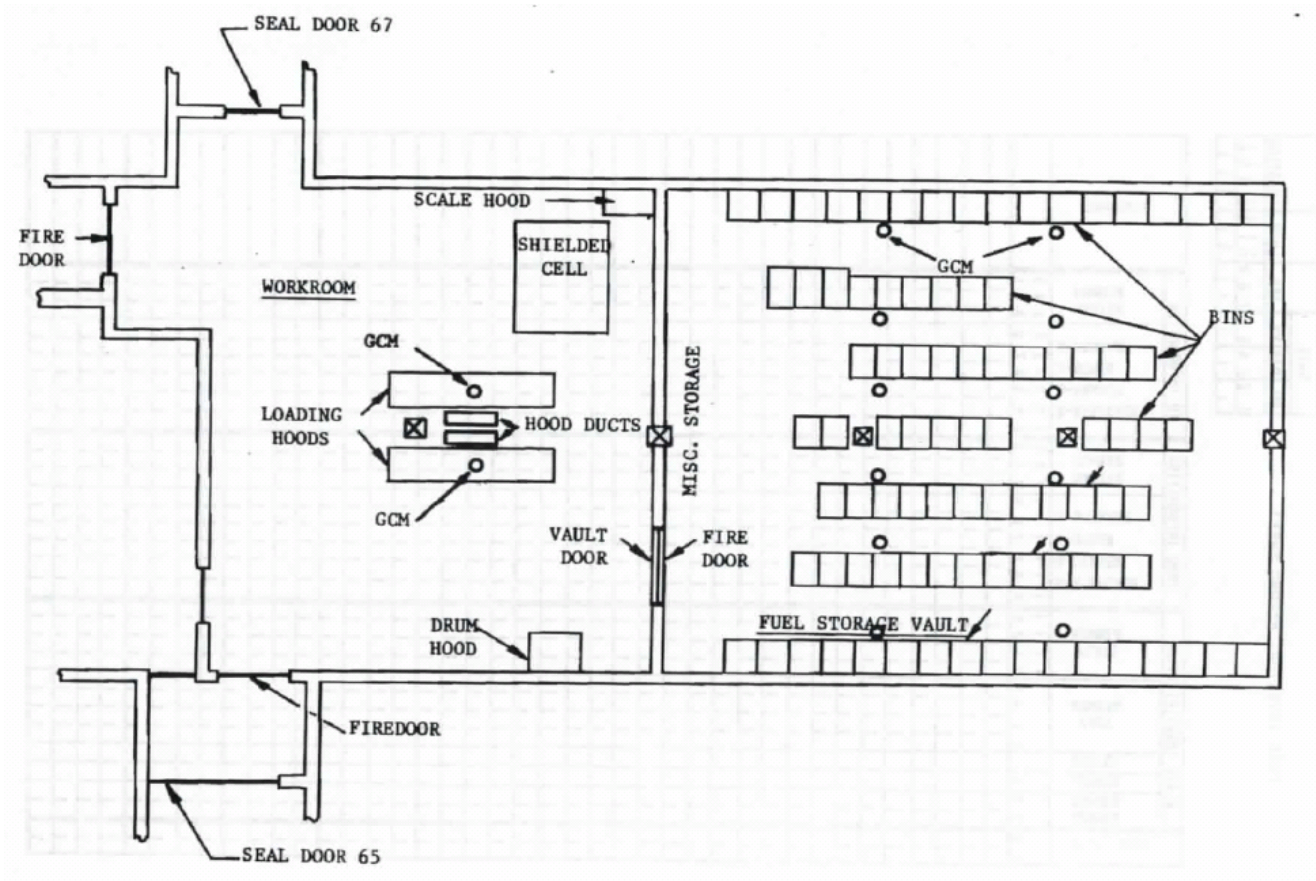
The Zero Power Plutonium Reactor (ZPPR) facility was a research facility designed to test the neutronics of' large, fast-neutron power reactors of' many different sizes, compositions, and shapes. ZPPR helped define many of today's nuclear reactor core designs through the use of standard, 2.54 cm-wide nuclear (e.g., up to DOE Category I) and non-nuclear materials. Many of these materials are still stored at the facility.



Source: ZPPR, undated

Figure 5-42: ZPPR Reactor and Support Areas

The reactor assembly itself' was housed in a cylindrical room with a reinforced concrete wall surrounded by earth fill (see Figure 5-42). The roof structure consisted of steel cables, wire mesh, and earth and gravel fill in the form of an inverted cone. The vault and work room, together with some portions of the control room, were located in and under the earth fill surrounding the assembly room (see Figure 5-43). The various work areas, control room, vault, work room, assembly room cell, were connected by tunnel corridors equipped with blast absorbers, blast- and gas-tight doors, and fire doors (ZPPR, 1963).



Source: ZPPR, 1984

Figure 5-43: ZPPR Vault Workroom Area

The critical assemblies (reactors) were constructed within a large stainless-steel honeycomb structure called the ZPPR matrix consisting of two halves, each 14 ft. x 14 ft. x 5 ft. When the reactor was shut down, the halves were separated by a large gap. This feature, known as a split-table design, allowed easy access to the very center of the core, while at the same time providing a large safety margin for personnel operations within the cell. The matrix halves were driven together by three electric motors, and then the reactor was operated normally by remotely-driven control rods (ZPPR, 1985).

The ZPPR fuel elements were called "drawers." Drawers were 2 in. x 2 in. x 3 ft. long stainless-steel containers loaded with rectangular pieces of reactor materials. These pieces were 2-in. high, from 1 in. to 12 in. long, and varied from 1/16 in. to 2 in. wide. The sodium coolant and the plutonium/uranium fuels were clad in stainless-steel containers (ZPPR, 1985).

Authorization to operate ZPPR was granted on March 27, 1969. Initial criticality was achieved on April 18, 1969, with a reactor loading in the FTR-1 pattern having a total fissile mass (Pu-239, Pu-241 and U-235) of 362.840 kg. In January 1970, loading of ZPPR Assembly 2 began. This assembly was a demonstration-plant benchmark reactor in a size projected for 350-500 MWe. Initial criticality was reached in March 1970 with a volume of 2398 liters containing 1,066 kg Pu-239 and Pu-241 (ZPPR, 1970).

Any change in the composition and/or configuration of the ZPPR critical assembly was documented in a written Loading Change Plan. The Loading Change Plan contained a detailed description of the loading of each drawer or calandria involved in the change, as well as a detailed plan for the changes made in the reactor matrix loading. Forms were also expected to include an estimate of the reactivity change that would have resulted from the loading change. All personnel involved in work activities in the mound area wore, at a minimum, lab coats, gloves, and shoe covers. All personnel within the reactor cell during fuel unloading operations, or any person working in a reactor plenum was required to wear a full-face respirator (ZPPR, 1984).

Fuel plates or rods were normally loaded into pre-loaded drawers or calandria. All normal loading changes were done in the workroom, and the loading of plutonium fuel plates into drawers and plutonium fuel rods into calandria cans were done in the loading hoods in the workroom by gloved personnel (ZPPR, 1984).

Fissile materials were kept in the vault when they were not being loaded or in the reactor cell. Each plutonium plate removed from the canisters was smeared for alpha contamination (at least one large surface and both closure welds per plate), and the results of these smears had to be available before loading the affected plates into drawers. The plates were allowed to be batch-smeared, but if a single batch-smear indicated more than an average of 25 dpm per plate, the plates were to be smeared again individually (all surfaces). Plates exceeding 25 dpm on the individual smears were considered contaminated, placed in a fuel pot, and returned to the Special Materials (SPM) representative (ZPPR, 1984).

After smearing, the fuel plates were passed through a go/no-go gauge and, if a plate was found to be oversized or otherwise deformed, it was placed in a fuel pot and returned to the SPM representative (ZPPR, 1984).

The loaded drawers and calandria cans could be transferred to the reactor in two ways: (1) the drawers were removed from the hood and carried to the transfer tube on a drawer cart, then delivered to the reactor cell via the transfer tube, and then transferred to the reactor via another drawer cart; or (2) the drawers were removed from the hood and placed on a drawer cart, then the cart would be pushed into the reactor cell via the access corridor, and then the drawers placed in the reactor. For safety, two full-time ANL-W employees were required in the reactor cell whenever the cell was occupied (ZPPR, 1984).

Fuel unloading was similarly regimented. At least two people in full-face respirators were required to be in the cell during any unloading operations. An alpha probe was used in the cell for initial PuO_2 surveillance, and neutron shields would be used in such a way that the person between the halves would be exposed to as little of the core surface area as possible. One of the two workers would go between the halves, pull a drawer carefully and expeditiously, and place it on a drawer cart or pass it to another worker to place on a drawer cart. The Health and Safety (HS) representative standing nearby would first survey each drawer for Pu alpha contamination as it was placed on the drawer cart using an alpha probe meter. When the cart was filled, they would smear the top of the fuel column and count the smear before the cart was allowed to leave the cell. The personnel in the cell were prohibited from removing their respirators until they were told that the smear indicated that there was no loose plutonium (<25 dpm/plate). The fuel was subsequently smeared in the workroom. If the

smear taken in the workroom indicated that there was loose plutonium in a drawer, the cell personnel were to be informed and were to evacuate the cell (ZPPR, 1984).

To transport drawers to the workroom via the conveyor, the tube door in the reactor cell was opened, the drawer inserted, and the conveyor advanced. No more than 10 fuel drawers or 13.0 kg of Pu were allowed on the conveyor. At the workroom end, after the cell tube door was closed, the workroom door was opened and the drawers removed. The cell and workroom tube doors were interlocked to prevent simultaneous opening. Similarly, if the drawer carts were used, no more than 10 fuel drawers or 13 .0 kg of Pu were allowed on one cart. The carts were wheeled from the cell through the access corridor and placed near the loading hoods (ZPPR, 1984).

Plutonium plates were to be removed from the drawer and batch-smeared (at least one large surface and both closure welds per plate). If the batch-smear count indicated less than an average of 25 dpm per plate, the diluent materials and the drawer were assumed uncontaminated and removed from the hood and placed into appropriate storage bins. If the batch-smear indicated more than an average of 25 dpm per plate, the plutonium plates would be individually smeared and the diluent columns adjacent to the fuel were batch-smeared. All plutonium plates with smear counts greater than 25 dpm were put into fuel canisters and returned to SPM for further disposition. If the diluent-material smears were less than 200 dpm/U_3O_8 plate, or less than 100 dpm/other plates, all of the diluent materials from the drawer were permitted to be removed from the hood and placed into the appropriate (usable material) storage bins; if not, the smeared materials were put into a plastic bag. The bag was sealed and put into a bin designated for contaminated diluent materials. Drawers that contained contaminated materials were bagged and taken out of the hood and put into appropriate storage containers (ZPPR, 1984).

Contamination control was a primary focus during ZPPR loading and loading operations. Smearable alpha activity on the working surfaces of the hoods was required to be less than 10 dpm/100 cm^2 before loading or unloading operations could begin. Similarly, smearable alpha-particle activity on working surfaces of tables was limited to less than 25 dpm/100 cm^2. A Health and Safety (HS) representative was required to smear each hood and table to assure conformance with these levels. If the contamination level of a hood or table exceeded the control level, it was to be cleaned until that level was achieved. The hoods and tables were to be re-checked for contamination periodically (at least twice per day) during loading and unloading operations in which they were involved (ZPPR, 1984).

Floors and exposed surfaces in the reactor cell and/or workroom were cleaned prior to the beginning of loading or unloading operations whenever the contamination level was above 25 dpm/100 cm^2. At least five smears were used to establish contamination levels in the cell at the following locations: (1) on the loading platform; (2) in the plenum behind Half 1; (3) in the plenum behind Half 2; (4) on the floor along the path of travel from the matrix to the transfer port; and (5) on the floor along the path of travel from the matrix to Door 70. At least three smears were to be obtained in the workroom at the following locations: (1) on the floor along the path of travel from the conveyor tube to the loading hood; (2) on the floor along the path of travel from Door #67 to the loading hood; and (3) on the floor along the path of travel from the loading hood to the utility hood and vault (ZPPR, 1984).

Irradiation of hundreds of plutonium and uranium foils was a standard ZPPR experiment for determining detailed reaction rate and power distributions. The irradiated foils were analyzed in the ZPPR gamma and alpha spectroscopy laboratory situated in the basement of the office complex. The laboratory was also equipped with an alpha-counting system, used mostly for determination of absolute masses on foil deposits (ZPPR, 1985).

5.1.2.8 Laboratory & Operations Building (L&O Building)

ATTRIBUTION: Section 5.1.2.8 was completed by Jason Davis, Oak Ridge Associated Universities. All conclusions drawn from the data regarding the L&O Building were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the L&O Building are explained in the associated text.

On June 23, 1959, Argonne National Laboratory moved its ANL-W offices from the EBR-I Complex to the Laboratory & Services Building (later renamed L&O Building) in the EBR-II Complex. This was the first occupancy at the EBR-II Area (EBR-II, undated). The L&O Building, including the Analytical Laboratory (AL), was built in the late 1950s and was operational in the early 1960s. Important functions of the analytical laboratory staff included assisting reactor personnel in the detection of fuel-element failures in the reactor, and locating the failed element or elements in a subassembly in the core (EBR-II Fuel Cycle Story, 1987).

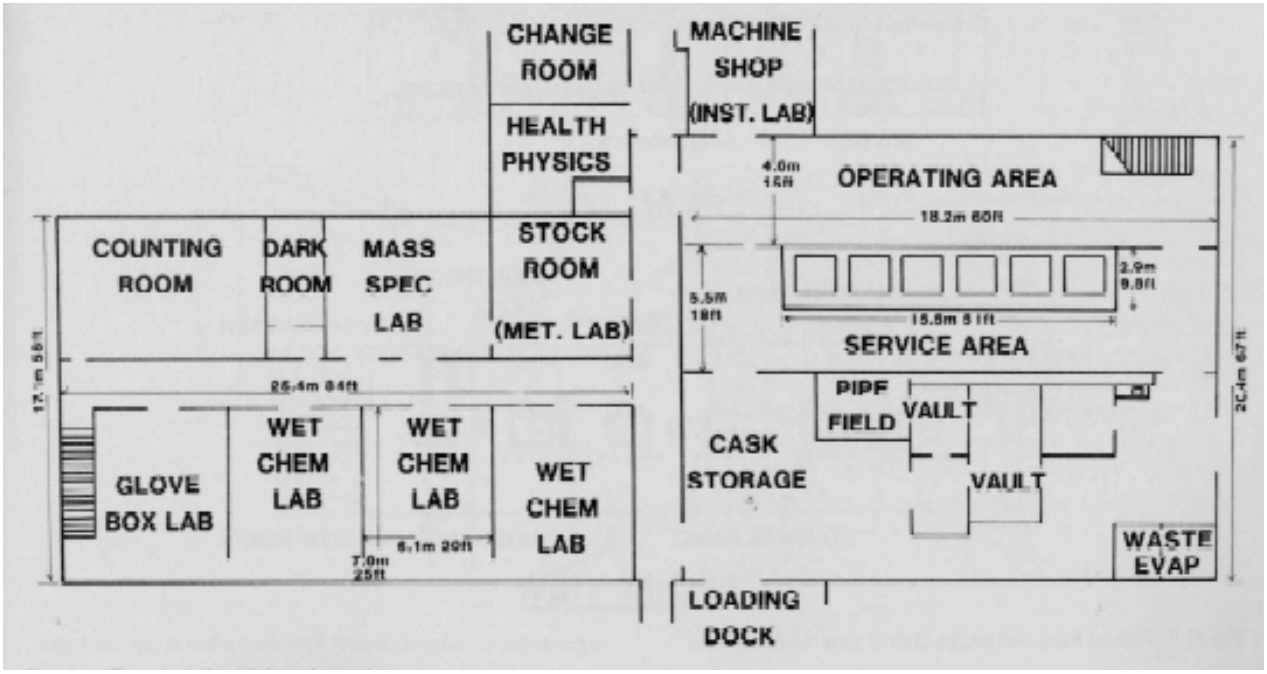
Analytical and control support activities were conducted at the L&O Building. Six analytical caves (hot cells) were provided to handle small radioactive samples, including irradiated fuel samples and radioactive samples of the fuel elements and process materials. These samples included fuel alloy, cladding, oxidized skull, and scrap. Analytical facilities were also provided to support operation and control of argon and sodium systems, and a variety of waste processes (EBR-II, undated). The work initially performed in the hot cells supported the recycle of EBR-II fuel during the late 1960s.

There were three wings on the main floor of the L&O Building: a junior cave wing, a conventional laboratory wing, and an office wing (EBR-II Fuel Cycle Story, 1987). The AL occupies the main floor of the north wings (A & B Wings) and includes the Nondestructive Analysis (NDA) Laboratory located at the east end of B Wing. The AL also included two counting rooms and a storage room in the B Wing basement, in addition to the ventilation exhaust filter areas in the A and B Wings of the basement. The AL complex consists of the following:

A Wing:
- Six shielded hot cells and attached gloveboxes
- A decontamination and manipulator repair room
- Two storage vaults
- One general chemistry/instrumental laboratory
- Gloveboxes and analytical instruments

B Wing:

- Eight general chemistry laboratories
- Analytical glovebox laboratories
- Mass spectrometry laboratories
- Three counting rooms (two in the basement)
- One chemical storage room
- One experimental fuel casting laboratory
- An NDA laboratory
- Basement utility area, including B-50/51



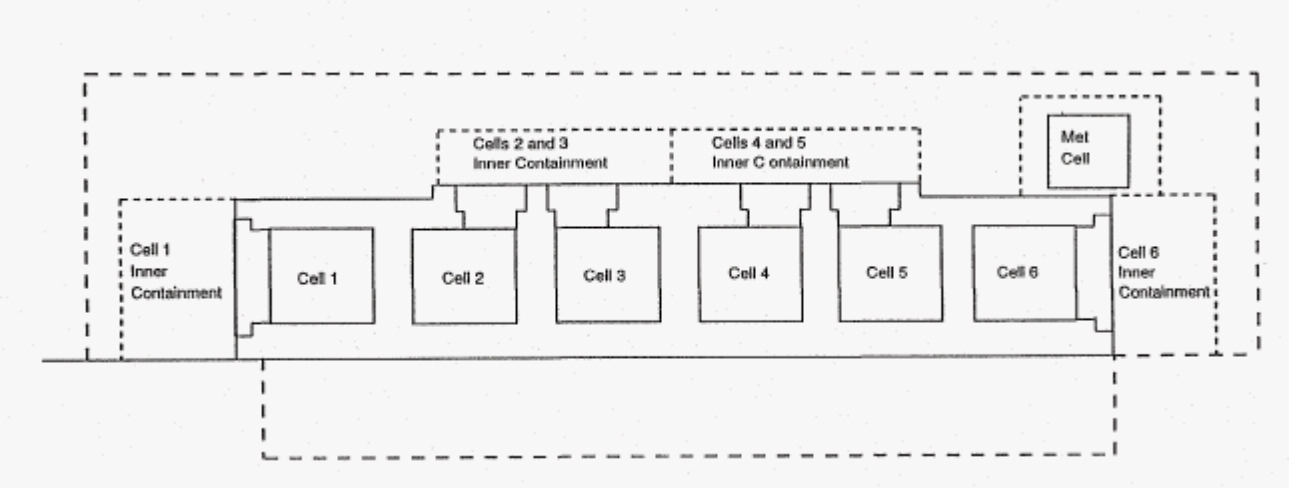
Source: EBR-II Fuel Cycle Story, 1987

Figure 5-44: The Analytical Laboratory Wing of the L&O Building

The cave wing was divided lengthwise by the block of caves into an Operating Area and a Service Area (see Figure 5-44). Another full-length partition separated the cave area from an area containing the upper part of a waste evaporator, storage vaults for special nuclear material, and a decontamination area. Transfer locks for introducing materials to, and removing materials from, the caves are located in the back walls of Caves 1 and 6. A hot-drain system was provided with openings into the system at one point on each work tray with a header leading to a hot-waste tank adjacent to the solidification system in the basement area. However, shortly after work with highly-active samples began, it was determined that the wastes were likely to be too hot to handle in this manner. Since then, highly-active liquid wastes were sorbed on vermiculite in plastic bottles for pneumatic transfer to a shielded cask for disposal (EBR-II Fuel Cycle Story, 1987).

An important accessory to the junior caves was a decontamination room located adjacent to the Service Area. This room was used to reduce the level of activity from malfunctioning cave accessories in order to repair or modify them, as well as to keep the surface activity levels of transfer casks low.

The cells ("caves") of the analytical junior cave facility were designed to safely handle analytical samples with a radioactivity level of 20 Ci of 1-MeV gamma radiation (see Figure 5-45). The basic purpose of the hot cells was to provide a shielded laboratory environment for the analysis of nuclear fuel and cladding samples that were irradiated in EBR-II. In addition, the laboratory was intended to handle most analytical problems occurring during EBR-II operation. Each hot cell contained one 26-inch-thick leaded glass shielding window. The window contained three seven-inch-thick panes of leaded glass cover plates. The leaded glass panes were separated from one another by a void containing mineral oil. Each window weighed approximately 5500 lbs. The walls separating the hot cells from one another had a 1 ft. by 2 ft. hole through which passed a transfer system. The purpose of the transfer system was to transfer tools, samples, chemicals, and other support equipment from one cell to another (ANL, undated).



Source: Michelbacher, 1995

Figure 5-45: The Analytical Laboratory Junior Caves

Fuel was transferred from one cell to another with the aid of the transfer system located in front of the cells. The transfer system consisted of a cart that rode on a stainless-steel conveyor tray supported by a steel stanchion. The cart held the transfer materials and was pushed from cell to cell by the master/slave manipulators (ANL-W S&H Plan, 1992).

Each hot cell had a unique function for the analysis of nuclear fuel samples. Cell #1 was the waste-compacting and disposal cell. The cell contained a storage rack for fuel-pin storage, and the floor sump received contaminated liquid from floor drains in other cells. Cell #2 was the fuel dissolution cell. Dissolved fuel was placed in containers and transferred to Cells #3, #4, #5, or #6 for a specific type of analysis. Cell #2 was also used for waste reduction of liquid fuel samples. Within Cell #3, analytical techniques such as retained fission gas analysis, radiochemistry analysis, sample burn-up analysis, and fuel preparation were performed. A 1000 lb. capacity monorail crane was

installed in Cell #4. Physical characteristics of fuel were analyzed in this cell. Atomic absorption of fuel samples was performed in Cell #5.

From the start of fuel recycling operations in 1964 through most of 1968, all unclad fuel samples from the Argon Cell were transferred (in groups of up to six samples) in capped 0.75-in. by 2.5-in. aluminum cans held in holes bored into a hexagonal aluminum block. The blocks were moved through a transfer lock into the Air Cell transfer port for insertion into the cavity of a top-loading cask for shipment to the Junior Caves of the Hot Analytical Laboratory. These operations were frequently accompanied by the dispersal of contaminated particles on the upper surfaces of the cask, within the cavity, on the outer wall of the cell, and on the floor beneath the port. Particle dispersal was reduced somewhat by taping plastic around the lower part of the cask. However, it was necessary to remove the plastic in order to lower the sample block out of the cask and place it into the breach-loading transfer device of the Junior Cave. This resulted in the spread of contamination at that point. The contamination problem was resolved in the latter part of 1968 by the construction of a pneumatic tube transfer system from the Air Cell and its subcell to the Junior Cave system. Individual sample cans were transferred by this system rather than the blocks of six samples, and worker exposure during these processes was virtually eliminated (EBR-II Fuel Cycle Story, 1987).

Conventional wet chemical laboratories with hoods, in conjunction with the junior caves, were required for several functions: the preparation of reagent solutions and analyses of nonradioactive materials; the development of methods for the analysis of both radioactive and nonradioactive materials; the conduct of routine and special analyses when massive shielding was not required; and for miscellaneous investigations (since there was little other work space available). Two such laboratories (20 ft. x 25 ft.) were initially provided. They were equipped with standard laboratory benches on the side walls and a double-hood island in the middle. Two other substantially-identical rooms were also available, but were initially used as offices for FCF and laboratory staff. As other office space was obtained and laboratory needs increased, these two rooms were converted into a third wet chemical laboratory and a laboratory provided with an inert-atmosphere (argon) and an alpha (air-atmosphere) glove box.

The AL's mass spectrometer was installed in September 1960. The spectrometer provided isotopic analyses of uranium, plutonium and other elements, as required the FCF (Reactor Operation, 1962). Burn-up measurements were made on irradiated fuel using isotopic dilution mass spectrometry techniques to measure the fission products La-139 and Nd-148. Because isotopes of xenon in various proportions were added as identification tags to experimental capsules irradiated in EBR-II, a gas mass spectrometer was also installed in this laboratory for reactor gas analyses to identify leaking experimental subassemblies (Stevenson, 1987).

A small stockroom for laboratory supplies and equipment had been provided adjacent to the mass spectrometer laboratory. Around 1962, it was determined that laboratory space was needed for preparing specimens of structural materials and unirradiated fuel for metallographic examination to resolve certain fuel fabrication and performance problems. Consequently, the stockroom became a metallurgical laboratory. In 1971, when better facilities for that purpose became available in the Hot Fuel Examination Facility/North, the met lab was equipped for preparing gas mixtures for the xenon-tag program.

In 1971, after a larger machine shop was provided in another building, the L&O Building machine shop was converted to office space. A 12 ft. long section of this area was partitioned off and used as an instrument laboratory associated with junior cave work (EBR-II Fuel Cycle Story, 1987). The L&O Building also contained a number of tenant activities, including major office areas, a cafeteria, and a library, none of which contained radioactive or extremely hazardous substances.

5.1.2.9 Argonne Fast Source Reactor (AFSR) (2nd location): 1970-1997

ATTRIBUTION: Section 5.1.2.9 was completed by Jason Davis, Oak Ridge Associated Universities. All conclusions drawn from the data regarding AFSR (second location) were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to AFSR (second location) are explained in the associated text.

In the fall of 1970, the AFSR reactor was moved to a new location adjacent to the ZPPR facility at the ANL East Area site of the NRTS. In the course of (and as a result of) this relocation, some modifications were undertaken for both the reactor and reactor building. Within a 55-meter radius of the new reactor location were located all of the ZPPR facilities and up to 75 personnel. The Lab and Office Building and EBR-II facilities fell within a 300-meter radius. Approximately 550 personnel were employed in this area.

Prior to the move, AFSR was housed in a Butler-type building with a self-contained ventilation system. All building exits were to the outdoors, and there were no ventilation ductwork connections with adjacent facilities. Subsequent to the relocation, AFSR was housed in a portion of the ZPPR support wing. Conditioned air was supplied to the AFSR room at 4800 cfm from the ZPPR service-wing ventilation system. This air exhausted in part (4600 cfm) to the outside via two exhaust fans, one each on the east and west walls. The reactor-pit-ventilating air (280 cfm) was drawn from the AFSR room through the reactor pit and through the blower pit. Since this was a negative-pressure system, airborne radioactive material in the reactor pit or blower pit would be expected to have been retained in the exhaust air system and removed by the HEPA filters.

The AFSR reactor-cooling system drew air from the room. The flow path was around the outside of the depleted-uranium blanket and around the outside of the core. The reactor-cooling blower drew this air through a filter and exhausted it through a 30-ft-high stack on the east wall of the building. This was also a negative-pressure system before the blower and airborne radioactive material originating in the blanket core region would have been removed in the filter system.

The essential differences between the reactor shields at the two locations are that the shield at the new location was straight-sided from top to bottom, in contrast to the pyramidal shape of AFSR West. In addition, the AFSR's cylindrical stepped beam-hole-plug and hole was replaced during relocation with a four-foot-square hole and rollaway concrete plug. The interface between the inner end of the rollaway-plug hole and the depleted-uranium blanket was covered with a flat 1/4- in. thick aluminum plate on the reactor after relocation (Matlock, 1976).

5.1.2.10 Miscellaneous Facilities

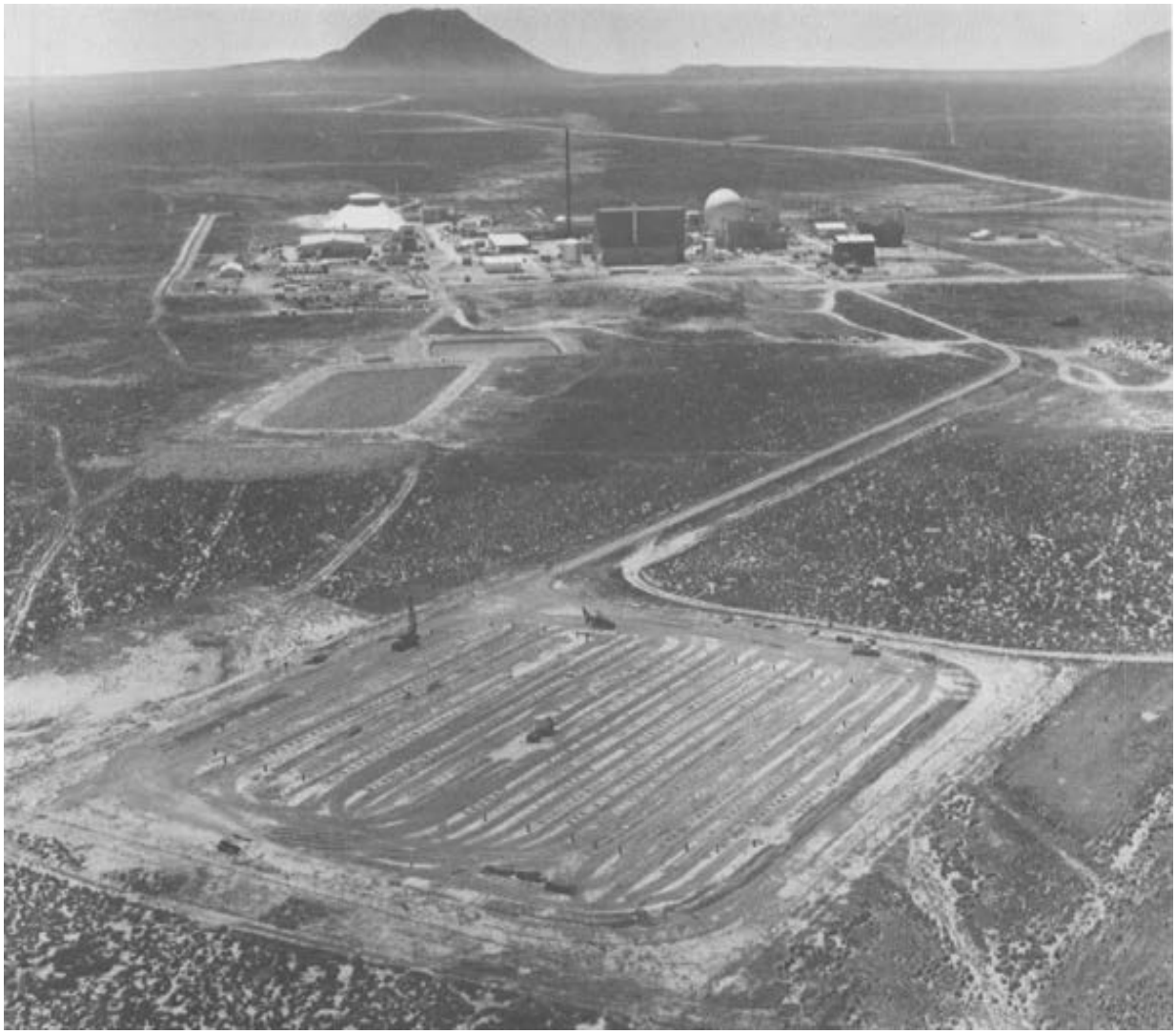
ATTRIBUTION: Section 5.1.2.10 was completed by Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding Miscellaneous Facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to Miscellaneous Facilities are explained in the associated text.

In addition to the primary radiological facilities at the EBR-II complex, there were several miscellaneous support facilities of importance. The Radioactive Scrap and Waste Facility served primarily as a repository for retrievable scrap generated from ANL-W operations, primarily EBR-II. The Radioactive Liquid Waste Treatment Facility did not begin radiological operations until after the end date for the evaluation period for this report. The Industrial Waste Pond (now used for industrial water and storm drainage) has never contained radioactive material.

Radioactive Scrap and Waste Facility

The ANL-W Radioactive Scrap and Waste Facility (RSWF) is a subsurface dry storage facility located approximately one-half mile north of the EBR-II complex, as shown in Figure 5-46). It consists of a four-acre fenced area that employs a grid of vertical storage holes lined with steel liners for storage of solid wastes and scrap from ANL-W activities. There are no permanent buildings located at the RSWF. When it was constructed in 1965, the elevation of the facility was set above the surrounding land by adding several feet of earth. This was done to preclude flooding by surface runoff that had previously caused problems at the INL Burial Ground (Witbeck, 1976).

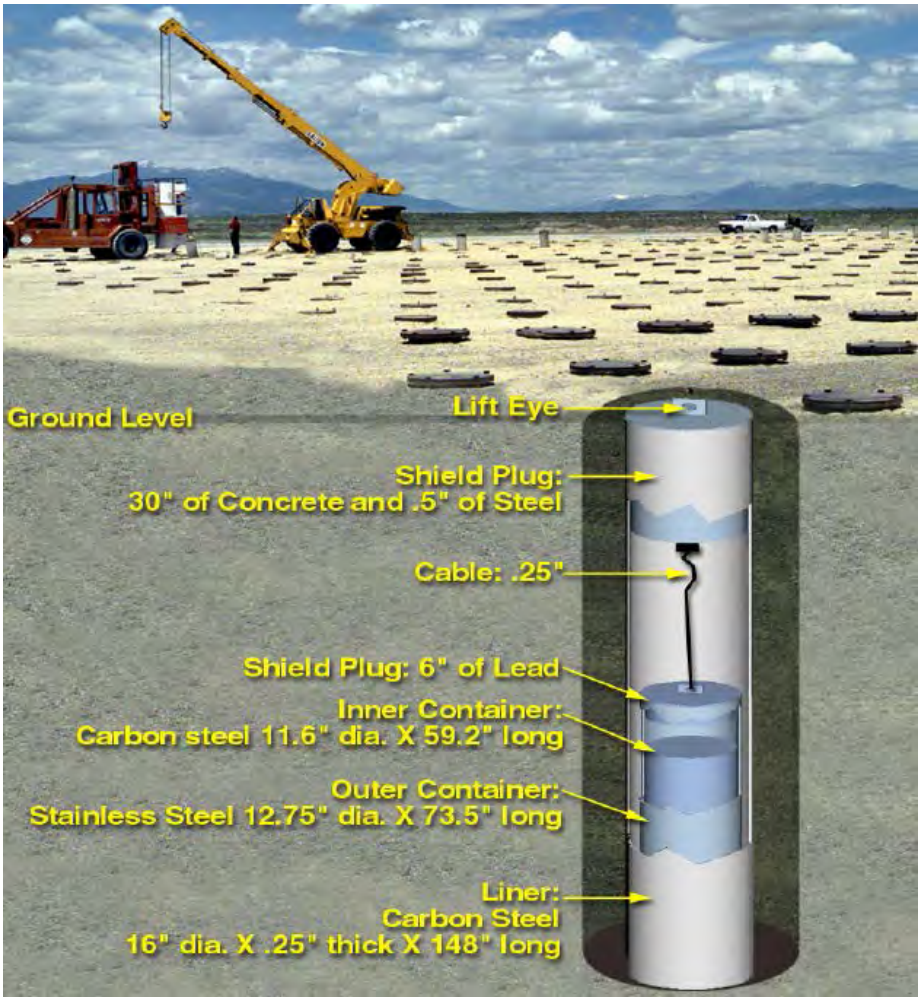
RSWF's initial purpose was to store highly-radioactive solid waste generated primarily from the EBR-II fuel-refining operations performed at the Fuel Cycle Facility. The first waste disposal in the facility was reported in a 1965 Reactor Development Program Report, which stated: "The use of the EBR-II burial ground for waste has been initiated with the disposal of two cans of waste. One can contained an estimated 1500 curies and showed a radiation level of 80 R at 1 ft.; the other contained an estimated 6500 curies and read 250 R at 1 ft." (Progress Report, Jun1965). Later, irradiated fuel element and subassemblies, and other irradiated materials associated with the Liquid Metal Fast Breeder Reactor program, were also stored at the RSWF. The primary advantage to this type of storage is that, if in the future it becomes desirable to do so, it would be possible to retrieve any individual can of scrap. (Radioactive scrap is material that contains enough source or special nuclear materials to be economically recoverable at some time.) The facility has expanded over the years; in 2001, there were 1350 liners with 939 filled to capacity. (Smith, 2001).



Source: Site Plan, 1981

Figure 5-46: ANL-W Radioactive Scrap Waste Facility (foreground)

The storage liners at the RSWF are arranged in rows with about 40 liners per row (see Figure 5-47). The upper end of the liners are nominally four inches above grade. The liners are welded closed at the bottom end and have a top closure plate that is welded on after canned material has been deposited into the storage tube. The storage liners are just over 12 feet in length and are constructed of 16-inch OD (outer diameter) carbon steel. In addition to the 16-inch OD liners, there are also 21 liners constructed of 26-inch OD carbon steel. These oversized liners are used for the storage of wastes from the Sodium Loop Safety Facility (Safety Report, 1983). The containment cans used to store material in the steel liners are modified carbon steel paint cans that are five feet long and 11-3/4 inch OD. Within each of these cans, material is contained in smaller cans. There can be up to five smaller cans loaded and sealed within an inert gas (argon) hot cell. The lid of each five-foot can is machine-crimped (Witbeck, 1976).



Source: Smith, 2001

Figure 5-47: RSWF Storage Liner Array and Schematic

Transport of scrap and waste to the RSWF has always been a fairly simple procedure. The containment cans are loaded into a shielded cask at the Fuel Cycle Facility and transported to the RSWF. The bottom of the transport cask is covered during movements to and from the RSWF so that any contamination from inside the cask or outside the scrap or waste container cannot spread. Deposition of steel cans into steel liners is carried out via the shielded transport cask with a remote door-opening cable device that allows for continuous shielding while the containment can is lowered into the liner. For scrap cans, a cable is left in place for future retrieval operations. Until 1976, waste (but not scrap) storage liners were shielded with six feet of gravel instead of concrete plugs, and no retrieval cable was installed (Safety Report, 1983). Since 1977, the only materials accepted for storage at RSWF have been solid radioactive waste that contains metallic sodium or solid radioactive scrap, which may or may not contain metallic sodium. Before 1977, non-sodium-containing radioactive waste was stored at the RSWF. Thirty of the pre-1977 non-sodium waste-containing liners were retrieved from the RSWF and shipped to the INL Burial Ground (RWMC) (Safety Report, 1983). Since 1977, all non-sodium-containing radioactive waste has been sent to the RWMC.

The Health Physics section had the responsibility of approving entry to the RSWF, designating the hole location for each burial, providing constant monitoring of functions during any burial operation, maintaining official logs and records of the burial ground operations as necessary, and periodically testing holes in the burial ground for water seepage (Hogg, 1971). A 1983 report indicated that over 700 operations have been conducted at RSWF with an average personnel exposure < 0. 01 man-Rem/operation (Safety Report, 1983).

RSWF retrieval operations may be of several different types depending on the type of waste or scrap, the method of packaging at the time of storage, and the method used during storage. Retrieval methods may consist of: (1) removing the whole RSWF liner intact from the soil, thus using the liner as the waste package; (2) removing the waste or scrap can from the liner and leaving the liner in the soil for future use; or (3) removing the scrap EBR-II subassemblies from the liner and leaving the liner in the soil (Safety Report, 1983).

Radioactive Liquid Waste Treatment Facility

The Radioactive Liquid Waste Treatment Facility (RLWTF) processes low-level radioactive liquid for ANL-W. The RLWTF supports EBR II, HFEF, TREAT, ZPPR, and ANL-W support facilities. The RLWTF is capable of evaporating approximately 60,000 gallons of radioactive liquid annually, with the resulting solidified residue packaged and stored in an environmentally-acceptable form for interim storage or shallow land burial. The RLWTF did not become operational until June 1983 and is therefore outside the time period under evaluation. (Black, 1984)

Industrial Waste Pond

The Industrial Waste Pond was subject to chemical contamination during its history but never contained radioactive material. Consequently, it will not be covered in this evaluation. (Personal Communication, 2015e)

5.2 Internal Radiological Exposure Sources from ANL-W Site Operations

Radionuclides with the widest application throughout ANL-W facilities during the period under evaluation were:

- Mixed fission and activation products
- Radioiodines (I-129, I-131, I-132, I-133, I-134, and I-135)
- Radioactive noble gases (Ar-41, Kr-85m, Kr-87, Kr-88, Xe-129m, Xe-135, Xe-135m, and Xe-138)
- Plutonium (Pu-238, Pu-239, Pu-240, Pu-241, Pu-242)
- Uranium (U-232, U-233, U-234, U-235, U-236, U-238)
- Thorium (Th-228, Th-230, Th-232)

- Other Actinides (Ac-227, Pa-231, Am-241, Am-243, Cm-242, Cm-243, Cm-244, Cm-245, Cm-246, Np-235, Np-237)

- Other radionuclides (e.g., H-3, Be-10, C-14, Na-24, Al-28, Fe-55, Co-60, Zn-65, Sr-90, Zr-95, Nb-93, Nb-95, Ag-110, Sb-122, Sb-124, Cs-137, Cs-139, Ce-141, Ce-144, Pr-140, Tm-170, Hf-175, Hf-181, Ta-179, Ta-182, W-185, W-187, Tl-204, Po-210, Ra-226, and others produced by reactor neutron irradiation)

The following subsections provide an overview of the internal exposure sources for the ANL-W class under evaluation.

5.2.1 Internal Radiological Exposure Sources from EBR-I Complex Operations

ATTRIBUTION: Section 5.2.1 was completed by Mike Mahathy, Oak Ridge Associated Universities, and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the EBR-I Complex were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the EBR-I Complex are explained in the associated text.

EBR-I: Radiological operations at EBR-I commenced in 1951 and concluded in 1963. Facility decontamination and decommissioning was completed in 1975. EBR-I was a research reactor with no known radiological separations work, with the exception of the work performed in the EBR-I hot cell. The categories of radionuclides specific to EBR-I operations are discussed in the subsections below and in Section 7.2.

ZPR-III: Radiological operations at ZPR-III commenced in 1955 and concluded in 1970. ZPR-III decontamination and decommissioning was completed in 1975 along with EBR-I. ZPR-III was a research reactor with no known radiological separations work. The categories of radionuclides specific to ZPR-III operations are discussed in the subsections below and in Section 7.2.

BORAX: NIOSH can describe internal exposure sources for all five BORAX reactors due to similar operations. Uranium-only-based cores were used in four of the reactors (BORAX-I, -II, III, and –IV. A core of uranium oxide-thorium oxide was used in BORAX-V. NIOSH has determined that internal doses from uranium and releases of fission products were plausible during the BORAX-1 destructive test, from excursions of the BORAX-I, -II, -III and –IV reactors, and from testing of fuel failures at BORAX-V, although uranium was not detected in air concentrations at BORAX-IV or from BORAX-V fuel failures. NIOSH has determined that the internal dose potential during the period under evaluation is associated primarily with reactor operation with some potential during loading/unloading of fuel cores and from waste handling. The radionuclides specific to BORAX operations are discussed in the subsections below and in Section 7.2.

AFSR: NIOSH has identified documentation listing specific radionuclides generated by irradiation and other operations at AFSR. NIOSH has determined that the internal dose potential during the period under evaluation is associated with irradiations, loading/unloading experiments, and radioactive waste disposal. The radionuclides specific to AFSR operations are discussed in the subsections below and in Section 7.2.

5.2.1.1 Fission and Activation Products

Mixed fission and activation products have been present at the EBR-I Complex since radiological operations started there in 1951. Over 200 fission and activation products were produced by the various reactors at the EBR-I Complex and were received at the complex via shipments of radioactive materials (e.g., irradiated fuels, contaminated equipment). The potential for exposure to those fission and activation products existed throughout the EBR-I Complex.

5.2.1.2 Radioiodines

At the EBR-I Complex, radioiodines (e.g., I-129, I-131, I-132, I-133, I-134, and I-135) were produced by EBR-I and the BORAX reactors and appear to have only been released in relatively small quantities during the evaluation period.

5.2.1.3 Radioactive Noble Gases

At the EBR-I Complex, radioactive noble gases (e.g., Ar-41, Kr-85m, Kr-87, Kr-88, Xe-129m, Xe-135, Xe-135m, and Xe-138) were produced by EBR-I and the BORAX reactors and appear to have been released only in relatively small quantities, with the exception of BORAX-IV after the failed-fuel experiment, which primarily released Xe-138 and Kr-88 (Borax-IV Test, 1959).

Because radioactive argon, krypton, and xenon do not represent a significant internal exposure concern, NIOSH finds there is no need to assess the internal doses attributable to them. Therefore, further discussion of the internal dose contribution from noble gases for the EBR-I Complex are not included in this report. (ICRP 30, 1979, PDF p. 34)

5.2.1.4 Uranium

EBR-I: EBR-I used three uranium cores during its operating history. In 1959, it introduced a plutonium core. While using the plutonium core, there were tentative plans for irradiating encapsulated samples of U-233. Whether the experiment was actually performed is unclear, but it would have been another potential source of uranium exposure (EBR-1, 1959). During its entire operating history, EBR-I had an outer blanket of U-238 bricks as part of its primary goal of proving the breeding reactor concept. The exposure potential to uranium isotopes was small because the fuel rods and blanket bricks were encapsulated. A breech of the cladding (e.g., the ruptured brick incident in 1955 and the partial core meltdown in 1955) would be examples of when uranium exposure was most likely.

ZPR-III: From 1955 to late-1963, ZPR-III exclusively used uranium fuel plates for reactor physics experiments. Even when plutonium fuel plates were introduced at ZPR-III, depleted-uranium plates were still used; enriched-uranium fuel plates were stored in the Special Materials vault. Depleted-uranium plates were uncoated due to their very small specific activity, but they did cause minor contamination. This contamination was typically found on the gloved hands of workers handling the bare plates. The workers' gloves would become blackened from the oxide layer of the depleted uranium plates rubbing off onto the gloves. The potential for uranium exposure to U-234, U-235, and U-238 existed throughout ZPR-III's operating history (Fish, 1967).

BORAX: The potential for uranium exposure to U-234, U-235, and U-238 existed at all five BORAX reactors during the entire period under evaluation. However, the exposure potential was very small because encapsulated fuel was used. With one exception, the fuel was irradiated and either stored as spent fuel or shipped to the CPP at INL without opening the capsules. The exception was the BORAX-I destructive test. In that test, the entire reactor structure blew up, distributing fuel element pieces and contaminated equipment up to 300 feet from the reactor (ANL-W History, 2006). Although the plume traveled up to 1-1/2 miles in a southwesterly direction, results from filter-paper samples, fallout plates, and continuous air monitors (CAMs) detected no significant non-natural alpha activity (Borax HP Documents, 1959).

AFSR: The potential for uranium exposure existed at AFSR but the potential was very small because AFSR received encapsulated materials, irradiated them, and shipped them away from AFSR without opening the capsules. Uranium would include U-233, U-234, U-235, and U-238.

Infrequently, work tables and other equipment were contaminated during the removal of uranium foils but those items were measured and decontaminated. One example occurred in November 1963 when irradiated U-235 foils were removed. A work table used in preparing the foils prior to the irradiation, and unloading the foils after irradiation, was found to be contaminated to 200 dpm/100 cm^2 smearable alpha. The table was decontaminated to less than 10 dpm/100 cm^2 smearable alpha (Safety Monthly Report, 1963, PDF p. 14).

5.2.1.5 Plutonium

EBR-I: EBR-I used a plutonium core (designated Mark IV) from 1959 until the facility ceased operations in 1963. However, throughout its entire operating history there was an outer breeder blanket of encapsulated U-238 bricks. Thus, there was a small potential for plutonium exposure from a breech in the integrity of one of the outer blanket bricks. In fact, there was a breach in the integrity of a blanket brick in April 1955 that precipitated *in-vitro* plutonium bioassay for workers with an elevated exposure potential. That scenario would have represented the only real plutonium exposure potential because there was never an integrity problem with the cladding on the plutonium core (fabricated and canned at Rocky Flats) (Fuel Elements, 1956).

ZPR-III: Experience at ZPR-III with receiving and handling plutonium fuel plates began in 1960. The first batch of plutonium fuel plates for ZPR-III were found to be contaminated during the receipt survey and were sent back to ANL-E (Lemont, IL) without being placed into service. There were no other known problems with receiving contaminated plates after that initial event. The first full-scale plutonium loading of ZPR-III occurred in late 1963. Thus, plutonium exposure potential existed from 1960 until 1969, when the plutonium fuel was shipped to ZPPR (Fish, 1967). The plutonium exposure potential was minimal because the fuel plates were clad in stainless steel and there was no known problem with cladding integrity (Personal Communication, 2015c).

5.2.1.6 Thorium

BORAX: The potential for thorium exposure existed during BORAX-IV operations from 1957 through 1959. However, only encapsulated ceramic fuels of UO_2-ThO_2 were used. Those elements (including the defective elements irradiated in 1958) were manufactured off site. Fuel elements were not opened at ANL-W, although some were shipped to CPP at INL for processing. Alpha monitoring conducted at BORAX-IV did not detect thorium contamination.

5.2.2 Internal Radiological Exposure Sources from EBR-II Complex Operations

ATTRIBUTION: Section 5.2.2 was completed by Jason Davis, Oak Ridge Associated Universities, and Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding the EBR-II Complex were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the EBR-II Complex are explained in the associated text.

Radiological operations at the EBR-II Complex have taken place since 1959. The EBR-II Complex was primarily an experimental nuclear reactors operations area, but also had radiological facilities for: reprocessing and refabricating irradiated nuclear fuel (Hot-Line fuel production), un-irradiated fuel production (Cold-Line fuel production), processing and examining irradiated nuclear fuels, neutron and photon radiography, performing laboratory analyses, machine shops, other research and support facilities, radioactive waste treatment and disposal, and other research and support facilities.

The following categories of radionuclides that were specific to EBR-II Complex operations are discussed in the subsections below and in Section 7.2:

- Mixed fission and activation products
- Radioiodines (I-129, I-131, I-132, I-133, I-134, and I-135)
- Radioactive noble gases (Ar-41, Kr-85m, Kr-87, Kr-88, Xe-129m, Xe-135, Xe-135m, and Xe-138)
- Plutonium (Pu-238, Pu-239, Pu-240, Pu-241, Pu-242)
- Uranium (U-232, U-233, U-234, U-235, U-236, U-238)
- Thorium (Th-228, Th-230, Th-232)
- Other Actinides (Ac-227, Pa-231, Am-241, Am-243, Cm-242, Cm-243, Cm-244, Cm-245, Cm-246, Np-235, Np-237)
- Other radionuclides (e.g., H-3, C-14, Na-22, Na-24, and others produced by reactor neutron irradiation)

5.2.2.1 Fission and Activation Products

Mixed fission and activation products have been present at the EBR-II Complex since radiological operations started there in 1959. Over 200 fission and activation products were produced by the various reactors at the EBR-II Complex and were received at the EBR-II Complex via shipments of radioactive materials (e.g., irradiated fuels, contaminated equipment). The potential for exposure to those fission and activation products existed throughout the EBR-II Complex.

Because radioiodines and radioactive noble gases were periodically present at the EBR-II Complex in significant quantities without any other fission and activation products being present, these fission products are addressed in their own subsections below.

5.2.2.2 Radioiodines

At the EBR-II Complex, radioiodines (e.g., I-129, I-131, I-132, I-133, I-134, and I-135) were produced by the various reactors and, with the exception of the EBR-II Reactor and the FCF, appear to have only been released in relatively-small quantities during the evaluation period. Radioiodines were periodically released from failed fuel in quantities large enough to cause the EBR-II reactor building to be evacuated. Much larger quantities of radioiodines were released from the fuel reprocessing activities at the FCF. However, the FCF had gaseous effluent control systems in place to safely collect and store the radioiodines that were released from fuel reprocessing. Those systems could later release the radioiodines (which would have gone through some radioactive decay) through a 200-ft-stack when meteorological conditions allowed for their safe release.

5.2.2.3 Radioactive Noble Gases

At the EBR-II Complex, radioactive noble gases (e.g., Ar-41, Kr-85m, Kr-87, Kr-88, Xe-129m, Xe-135, Xe-135m, and Xe-138) were produced by the various reactors and, with the exception of the EBR-II Reactor and the FCF, appear to have been released only in relatively small quantities. Radioactive noble gases were periodically released from failed fuel in quantities large enough to cause the EBR-II Reactor building to be evacuated. Beginning in 1972, some of the noble gas releases from the EBR-II reactor included isotopes of radioactive xenon that were used to tag the fuel elements so that a breached fuel element could be readily-identified while still in the reactor (Smith, 1977). Much larger quantities of radioactive noble gases were released from the fuel-reprocessing activities at the FCF. However, the FCF had gaseous effluent control systems in place to safely collect and store the radioactive noble gases that were released from fuel reprocessing. Those systems could later release the radioactive noble gases (which would have gone through some radioactive decay) through a 200-ft-stack when meteorological conditions allowed for their safe release.

Because radioactive argon, krypton, and xenon do not represent a significant internal exposure concern, NIOSH finds there is no need to assess the internal doses attributable to them. Therefore, further discussion of the internal dose contribution from noble gases are not included in this report (ICRP 30, 1979, PDF p. 34).

5.2.2.4 Plutonium

At differing times, plutonium has been present at the facilities comprising the EBR-II Complex. TREAT began the first irradiation of U/Pu foils in August 1959, EBR-II started up in September 1961, FCF received its first irradiated fuel in September 1964, and L&O began analyzing irradiated fuels and samples in 1964. Plutonium was present on the complex for the following reasons:

- Plutonium was produced by the various reactors at the EBR-II Complex.
- Plutonium was used as primary and experimental reactor fuels in the EBR-II Complex reactors.
- Plutonium was received at the EBR-II Complex via shipments of radioactive materials (e.g., irradiated fuels, contaminated equipment).
- Plutonium was an impurity in the recycled uranium that was used as a reactor fuel.
- Plutonium was present in non-dispersible forms (e.g., encapsulated sources, electro-plated sources).
- Plutonium may have been present as liquid laboratory standards.

Although plutonium may have been technically present in some forms at an early date, the purpose of this evaluation report is to identify and evaluate both exposure potential and actual personnel exposure. Although plutonium was present in matrices/forms at certain locations for certain operations, mere presence alone does not qualify as a source of personnel exposure requiring accounting for the purpose of this report.

Some sources of plutonium at the EBR-II Complex require evaluation and accounting. In such cases, with a few exceptions, the majority of the dispersible forms of plutonium were present with mixed fission products (e.g., EBR-II and FCF reactors). Plutonium without mixed fission products would have been contained in fresh plutonium fuels that had not aged long enough to allow for the ingrowth of fission or daughter products, as well as in fuels that had undergone chemical separation. The primary areas in which plutonium was present without mixed fission products were HFEF-N, L&O Building, and the TREAT and ZPPR reactors.

At the FCF, intermittent work associated with the Mark-II type RAS-TREAT sodium-loop experiments could have resulted in potential intermittent exposures to plutonium without mixed fission products present when that work included the removal of the test sections and when that work was performed outside of the Air and Argon Cells. At the HFEF-N, the source of plutonium without the mixed fission products was also from work associated with the Mark-II type RAS-TREAT sodium-loop experiments. However, that work appears to have been performed exclusively in the HFEF-N's hot cells, so the potential for worker exposures was minimal. At the EBR-II, TREAT, and ZPPR reactors, the potential for plutonium exposures was minimal because these reactors used encapsulated fuel sources. At the L&O Building, the primary sources of plutonium without mixed fission products were some of the encapsulated fuels being stored in the building as well as radiochemical analysis of samples containing plutonium in order to determine the breeding ratio. Liquid laboratory standards may have also been present in the L&O Building.

5.2.2.5 Uranium

Uranium has been present at the facilities comprising the EBR-II Complex since radiological operations started there in 1959. Uranium was present on the complex for the following reasons:

- Uranium was used by the reactors as fuel.
- Irradiated uranium was reprocessed and refabricated into nuclear fuel (Hot-Line fuel production).
- Non-irradiated uranium was fabricated into nuclear fuel (Cold-Line fuel production).
- Uranium was received at the EBR-II Complex via shipments of radioactive materials (e.g., non-irradiated fuels, irradiated fuels, contaminated equipment).
- Uranium may have been present as non-dispersible radioactive sources.
- Uranium may have been present as liquid laboratory standards.

The uranium at the EBR-II Complex covered the entire spectrum in regards to its U-235 content, ranging from depleted uranium to fully-enriched uranium.

The majority of the dispersible forms of uranium without mixed fission products present at the EBR-II Complex were in the Cold-Line fuel production areas at the FCF, ITF, and FASB. At the end of December 1969, the Cold-Line fuel pin and fuel element production equipment in FCF Rooms 20 and 26 was put on stand-by (Progress Report, Dec1969, PDF p. 88). However, the Cold-Line continued to fabricate unirradiated subassemblies from vendor-produced fuel elements (at ITF) and to perform impact-bonding of the unbonded vendor fuel elements (in FCF Room 23) (Progress Report, Dec1969, PDF p. 88). The FCF's Cold-Line fuel pin and fuel element production was likely shut down due to a large backlog of completed fuel elements, and given that ANL-W was beginning to receive large numbers of acceptable vendor-produced fuel elements. In January 1970, there were a total of 23,766 completed fuel elements that were ready to be put into EBR-II Reactor subassemblies (Progress Report, Jan1970, PDF p. 78). In the mid-1970s, the FCF's Cold-Line fuel pin and fuel element production equipment was deactivated (Batte, 1986, PDF p. 9). Fuel-pin fabrication in FCF Room 20 was not re-initiated until 1982 (Batte, 1986, PDF p. 9). Given that Cold-Line fuel production in FASB did not begin until February 1972, no Cold-Line fuel pin and fuel element production took place at ANL-W during the period of January 1970 – January 1972.

At the HFEF-N, the source of uranium without the mixed fission products was from work associated with the Mark-II type RAS-TREAT sodium-loop experiments. However, that work appears to have been performed exclusively in the HFEF-N's hot cells, so the potential for worker exposures was minimal.

At the L&O Building, the primary dispersible sources of uranium without mixed fission products were the non-irradiated derbies, ingots, and fuel pins being stored in the building. Liquid laboratory standards may have also been present in the L&O Building.

5.2.2.6 Thorium

At differing times, thorium has been present at the facilities comprising the EBR-II Complex. Thorium was present on the complex for the following reasons:

- was produced in small quantities by the reactors.
- was an impurity in the recycled uranium that was used as a reactor fuel.
- was used as an experimental reactor fuel and/or blanket material.
- was primarily used in the form of thoria (ThO_2) as part of the Hot-Line fuel production at the FCF.

Although thorium may have been technically present in some forms at an early date, the purpose of this evaluation report is to identify and evaluate both exposure potential and actual personnel exposure. Although thorium was present in matrices/forms at certain locations for certain operations, mere presence alone does not qualify as a source of personnel exposure requiring accounting for the purpose of this report.

Some sources of thorium at the EBR-II Complex require evaluation and accounting. Thorium that had undergone chemical separation and not aged long enough to allow for the ingrowth of daughter products would not have contained appreciable quantities of mixed fission products. The majority of the dispersible thorium without mixed fission products present was in FCF Room 25, which is where the crucibles and molds used in Hot-Line fuel production were coated with thoria. At the FCF, thoria was used as a coating during the period of August 1963 – November 1967. The experimental reactor fuels and blanket materials were encapsulated so the potential for worker exposures to that thorium was minimal. In the late-1970s, some contaminated thorium blanket materials were received at ZPPR; this is discussed in more detail below.

In the late-1970s, the effects of substituting thorium for uranium in both axial and radial ZPPR blankets were studied (ZPPR, 1985). Integral experiments on ZPPR Assembly 8 were designed to test the effect of thorium in different blanket zones. ZPPR 8C was a uranium-plutonium oxide reactor with 293 kg of thorium loaded into the central blanket to simulate a thorium blanket (Thorium Blanket, 1978).

During the last week of September 1977, ZPPR received a 362.5 kg shipment of thorium from Atomics International. The thorium plates read 120 mR/hr beta-gamma at one inch when a shipping container was opened. Smearable activity of 880 dpm/100 cm^2 alpha was detected on the plates. Five plates were packaged and sent to the L&O Chemistry Labs to see if a way could be found to clean the loose contamination from the plates.

During that same week, a shipment of thorium previously received from ANL-E was opened in a ZPPR hood for inspection. The material read 65 mR/hr beta-gamma at one inch and had smearable activity of 10,000 dpm/100 cm^2 alpha. When the plates were moved for inspection, oxide could be seen falling off the plates. There was no personnel contamination and no activity detected outside the hood. The hood was effectively decontaminated, as was the shipping container, which was returned to storage. An air sample taken in the breathing zone during the inspection showed 3.94×10^{-12} µCi/cc after 52 minutes decay, or 13% of the RCG at the time for Th-232. The sample had decayed to <1% of RCG by the following morning. As with the Atomics International plates, five of these plates were

sent to the L&O Building's chemistry labs to see if they could find an effective way to decontaminate and coat them so they could be used in the reactor.

At the L&O Building, selected thorium plates were coated with Kel-F solution to determine what effect it would have in reducing smearable activity levels. This method reduced smearable alpha activity from an average of 450 dpm/plate to <10 dpm/plate. The cleaning and coating of 4283 thorium plates was completed on November 8, 1977. There was no detectable contamination outside the hoods or coating machine in which all work was done. Smearable alpha activity on coated plates was <25 dpm/plate, which was considered acceptable for use in the reactor. The hoods and coating machine were decontaminated to <10 dpm/100 cm^2 alpha and <100 dpm/100 cm^2 beta-gamma. There were no instances of personnel contamination. In all, smearable alpha activity was reduced from a maximum of 10,000 dpm/plate to <25 dpm/plate by this cleaning and coating technique (Weekly Reports, 1977).

5.2.2.7 Other Actinides

Other actinides (i.e., excluding plutonium and uranium) have been present at the EBR-II Complex since radiological operations started at the EBR-II reactor in 1964. At the EBR-II Complex, other actinides were produced by the various EBR-II Complex reactors in small quantities via neutron absorption, and were received at the EBR-II Complex via shipments of radioactive materials (e.g., irradiated fuels, contaminated equipment). Based on the available information, the other actinides were never present at the EBR-II Complex without mixed fission products being present.

5.2.2.8 Other Radionuclides

The "other radionuclides" category includes any other radionuclides that were not addressed in the radionuclide categories listed above. This category includes fission and activation products that had been chemically separated from the other mixed fission and activation products. With the exception of the radionuclides that are specifically addressed below, the majority of the "other radionuclides" at the EBR-II Complex were likely encapsulated experiments or non-dispersible radioactive sources.

Tritium (H-3) is produced by the neutron bombardment of a number of elements, and was known to be present in the sodium coolant of the EBR-II reactor and was likely present as a contaminant in some the other EBR-II Complex locations. Radioactive isotopes of sodium (Na-22 and Na-24) were produced by the neutron bombardment of stable sodium, and were present in the EBR-II reactor's sodium coolant and in the NaK coolant used in the RAS-TREAT sodium-loop experiments. Carbon-14 (C-14) can be produced from a number of neutron interactions, and was likely present as a contaminant in carbon, nitrogen, and/or oxygen-bearing materials that were irradiated in nuclear reactors at the EBR-II Complex.

5.3 External Radiological Exposure Sources from ANL-W Site Operations

To provide consistency of radiation safety programs at the entire INL site given the large variety of facilities and constantly-changing contractors, the AEC-Idaho Operations Office (AEC-IDO) established a Health and Safety (H&S) Laboratory to provide technical support for internal and external dosimetry programs over the entire site, including ANL-W. The name of this organization was changed to Health Services Laboratory (HSL), then to the Health and Safety Division (HSD), then to the Idaho Center for Radiological and Environmental Sciences, and most recently, to the Radiological and Environmental Sciences Laboratory (RESL) (ORAUT-TKBS-0007-6).

By policy, the INL and ANL-W sites monitored by dosimeter all personnel who were expected to receive any radiation dose or whose work was centered at the site (Cipperley, 1968; Cipperley, 1958). Dosimeters were usually stored at the operational area entrance security gates.

Table 5-4 shows a historical summary of the beta-gamma dosimetry systems used at the INL site, and their corresponding Limits of Detection (LODs).

Table 5-5 shows a historical summary of the neutron dosimetry systems used at the INL site, and their corresponding Limits of Detection (LODs).

Table 5-4: INL Site Beta-Gamma Dosimetry Systems History and LOD Summary

Period of Use[a]	Dosimeter	Exchange Frequency	Beta LOD[b] (mrem)	Gamma LOD[b] (mrem)
August 1951–March 1958	INL initial film, 552 DuPont film	Weekly (n=52) Monthly (n=12)	30	30
August 1951–March 1958	Reactor areas, DuPont 558 film	Weekly (n=52)	30	10
March 1958–December 1966	INL multielement DuPont 508 film	Weekly (n=52) Biweekly (n=26) Monthly (n=12)	30	10
December 1966–February 1974	INL multielement DuPont 508 film	Weekly (n=52) Biweekly (n=26) Monthly (n=12)	30	10
December 1966–February 1974	INL LiF TLD	Quarterly (n=4) Semi-ann(n=2) Annual (n=1)	15	15
February 1974–May 1975[c]	INL Atlas TLD LiF in Teflon	Monthly (n=12) Quarterly (n=4) Semi-ann(n=2) Annual (n=1)	30	30
December 1974-December 1985[c]	INL Harshaw two-chip TLD	Monthly (n=12) Quarterly (n=4) Annual (n=1)	15	15
January 1986–2006	INL Panasonic four-chip TLD	Monthly (n=12) Quarterly (n=4)	15[e]	15[d]
January 1986–2006	INL Panasonic four-chip TLD	Monthly (n=12) Quarterly (n=4)	30[e]	10[d]

Source: This table is derived from Table 6-15 in ORAUT-TKBS-0007-6.

a. For many years, INL workers had a dosimeter assigned to each operating area where they worked, or they were issued visitor dosimetry. An exception existed from March 1970 through December 1974 when a "one badge, all areas" approach was used. After 1974, dosimetry reverted to the original "one badge, one area" system.

b. LODs are based on: Cipperley, 1958; Cipperley, 1968; Cusimano, 1963; Kalbeitzer, 1983; Gesell, 1986; Gesell, 1992; Perry, 1993; Ruhter, 2002.

c. ICPP began using the Harshaw TLD in December 1974, the prime contractor began in February 1975, and ANL-W began in May 1975.

d. The LOD was 15 mrem from January 1, 1986 to July 7 1986; 10 mrem from July 7, 1986 to September 1989; and 15 mrem until 1993, when it returned to 10 mrem.

e. The LOD was 15 mrem from January 1, 1986 to July 7 1986 and 30 mrem after that.

Table 5-5: INL Site Neutron Dosimetry Systems History and LOD Summary

Period of Use	Dosimeter	Exchange Frequency	LOD (mrem)
1951–1958	NTA Film (Kodak Type A Film)	Weekly (n=52)	14[a]
1959–September 1976	NTA Film (Kodak Type A Film)	Weekly (n=52) Biweekly (n=26) Monthly (n=12)	20
January 1986–2006	INL Panasonic four-chip TLD	Monthly (n=12) Quarterly (n=4) Monthly (n=12)	15[b]

Source: This table derived from Table 6-16 in ORAUT-TKBS-0007-6.
a. Cipperley, 1958, PDF p. 20
b. Gesell, 1996, PDF p. 19

When addressing exposure to noble gases, external radiation dose is the overriding concern. As stated in ICRP 30 (when discussing internal dose from noble gases):

> *Therefore, when applying the system of dose limitation described in Chapter 2* [Basic Limits for the Control of Internal Dose], *it is clear that, for exposure by submersion in radioisotopes of the noble gases, external radiation will be of such overriding importance that it alone need be considered.* (ICRP 30, 1979, PDF p. 34)

NOTE: The ICRP 30 wording above is less technical and more straightforward than later ICRP documents. ICRP 68 states: "For this report the behaviour of gases and vapours assumed in *ICRP Publication 30* has been retained, but is represented using the formalism of *ICRP Publication 66*." (ICRP 68, Section 2.3, "Dose Coefficients for the Intakes of Radionuclides by Workers")

Exposures by submersion in noble gases would be accounted for by ANL-W personnel external dosimetry.

The following subsections provide an overview of the external exposure sources for the ANL-W class under evaluation.

5.3.1 External Radiological Exposure Sources from EBR-I Complex Operations

ATTRIBUTION: Section 5.3.1 was completed by Mike Mahathy, Oak Ridge Associated Universities, and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the EBR-I Complex were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the EBR-I Complex are explained in the associated text.

5.3.1.1 Photon

EBR-I: Photon exposure resulted from nuclear fission as well as the decay of fission and activation products produced by fuel irradiation. Within the EBR-I reactor building there were hot cells, a rod storage "farm," and heavily-shielded, remote-handling rooms illustrating cognizance of the high-energy photons present. The workplace photon energy spectra for EBR-I would have been dominated by high-energy photons from mixed fission and activation products. Based on review of

available records, the highest external exposure potential from photons was likely during the removal of the partially-melted core in 1956. Special shielding was installed to reduce personnel exposures during that special work evolution. Most of the photon dose at EBR-I would have been photons with energy greater than 250 keV. (ORAUT-TKBS-0007-6, PDF p. 26)

ZPR-III: When the ZPR-III reactor was in operation, the Assembly Room was vacated to prevent personnel exposures. Due to the "zero power" operation of ZPR-III, there were no significant quantities of fission and activation products produced. Consequently, no special remote-handling procedures were needed at ZPR-III (Long, 1962). Photon exposure of any significance would have occurred while handling fuel plates. Therefore, most of the photon dose was from photons with energy greater than 250 keV prior to the use of plutonium fuel plates. Photon dose from photons after the introduction of plutonium fuel plates likely contained photon energy less than 250 keV (ORAUT-TKBS-0007-6, PDF p. 26).

BORAX: Photon exposure resulted from mixed fission and activation products and isotopes produced by reactors. Exposure rates varied by dates, but generally, exposure potential increased with time due to increased operational power and increased lengths of operation. (Low-power operations lasted only a few days in 1953, but there were longer-term operations at BORAX-IV and BORAX-V from 1956 through 1958, and from 1962 through 1964.) The radiation fields at BORAX were generated primarily by mixed fission and activation products. Therefore, most of the photon dose has been from photons with energy greater than 250 keV (ORAUT-TKBS-0007-6, PDF p. 26).

ASFR: Photon exposure was from mixed fission and activation products resulting from irradiation of target materials. Therefore, most of the photon dose was from photons with energy greater than 250 keV (ORAUT-TKBS-0007-6, PDF p. 26).

5.3.1.2 Beta

EBR-I: EBR-I operations did not have pure beta external hazards. Beta exposure would have resulted from mixed fission and activation products, and isotopes produced by EBR-I. The highest potential for beta external exposure would have been during an incident, such as the 1955 partial-core meltdown during which mixed fission and activation products broke through the argon cover gas and were released into the reactor building. Beta dose would have resulted primarily from exposure to electrons with greater than 15 keV (ORAUT-TKBS-0007-6, PDF p. 26).

ZPR-III: ZPR-III operations did not have pure beta external hazards. Beta exposure would have resulted from the handling of uranium fuel plates. This exposure would be expected to be minimal due to coatings on the uranium plates. This is reflected in the discovered non-routine dosimetry processing results (Film Processing, 1961). Plutonium fuel plates were clad in stainless steel and would not be a source of beta exposure. Beta dose resulted primarily from exposure to electrons with energy greater than 15 keV (ORAUT-TKBS-0007-6, PDF p. 26).

BORAX: Beta exposure resulted from mixed fission and activation products, and isotopes produced by reactors. Exposure rates varied by dates, but generally, exposure potential increased with time due to increased production and the increasing number of reactors (in 1957 and 1967). The radiation fields at the BORAX reactors were generated primarily by mixed fission and activation products. Beta dose resulted primarily from exposure to electrons with greater than 15 keV (ORAUT-TKBS-0007-6, PDF p. 26).

ASFR: Beta exposure resulted from mixed fission and activation products, and isotopes produced by reactors. Beta dose resulted primarily from exposure to electrons with energy greater than 15 keV (ORAUT-TKBS-0007-6, PDF p. 26).

5.3.1.3 Neutron

Operating reactors would have been the primary source of neutron exposure during the period under evaluation. These reactors were experimental facilities for measuring shielding factors, reactor kinetics, and neutron irradiation. Thermal neutron radiation levels 10 feet from the reactor during operations at 100 kW were 50-1,500 n/cm^2/sec (Freund, 1960).

EBR-I: From EBR-I's inception, shielding from fast and thermal neutrons was a consideration. Before operations commenced, neutron flux studies were performed at ANL in Illinois as part of the Experimental Breeder Reactor program. Experiments were performed to measure the anticipated flux at various locations in the reactor. This was done by activating gold and phosphorus foils at points in the reactor and along the outer surface of the uranium control blanket. These studies provided estimates of the resonance flux, thermal flux, thermal leakage, and equivalent fission flux when operating at 1 MW (Butler, 1952; Progress Report, 1953). EBR-I workers may have been exposed during EBR-I operation and any activities involving neutron sources. Neutron surveys were a part of the routine radiological monitoring at EBR-I (Survey Reports, Mar1962). A 1952 survey of the outer surfaces of the reactor shield concluded that the radiation levels were "generally safe with the shielding presently in place" (Butler, 1952). However, as a precaution, additional concrete shielding was added.

ZPR-III: ZPR-III was used to mock-up reactor cores and obtain neutron physics data; thus, understanding the neutron flux within the reactor was of highest importance. Spectral determinations were made at ZPR-III as well as derived by multi-group calculations (Hess, 1960). ZPR-III Operating Instructions required a properly-operating neutron detector and constant monitoring of the neutron flux during operations, demonstrating the prominence of neutron monitoring (Operations Manual, 1956). Neutron activity was monitored by both thermal and fast neutron detectors. With plutonium fuel plates, the neutron flux was a combination of spontaneous fission neutrons and neutrons resulting from the α,n reaction between plutonium alphas and the aluminum alloys used in the fuel trays. Pu-240, with two neutrons per spontaneous fission, was the principal source of neutrons from plutonium. Neutron exposure was considered serious enough that the neutron field between the halves, and near the assembly, were monitored each time a new size assembly was constructed; personnel exposure was limited accordingly (Long, 1962). ZPR-III workers were most likely to have been exposed to neutrons during fuel-plate handling. Neutron surveys were a part of the routine radiological monitoring at ZPR-III (Survey Reports, Mar1964).

BORAX: BORAX reactor workers may have been exposed to neutrons from the nuclear reactors and from transferring neutron sources, including antimony-beryllium and polonium-beryllium. Neutron fields were measured Health Physics staff in the turbine building at BORAX-III, in the BORAX-IV reactor building, and in the BORAX-V reactor building.

ASFR: During very early operations conducted between January and June 1960, neutron shielding was found to be inadequate; consequently, more shielding was added (Site Plan, 1981, PDF p. 148; Safety Monthly Report, 1960). AFSR workers may have been exposed to neutrons from the reactor

and from transferring neutron sources, including polonium-beryllium. Neutron radiation was a concern of Health Physics staff given the use of AFSR for material irradiation through beam ports; therefore, neutron fields at ASFR were measured.

5.3.2 External Radiological Exposure Sources from EBR-II Complex Operations

ATTRIBUTION: Section 5.3.2 was completed by Jason Davis, Oak Ridge Associated Universities, and Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding the EBR-II Complex were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the EBR-II Complex are explained in the associated text.

5.3.2.1 Photon

EBR-II: Based on a review of the available ANL-W records, the photon radiation fields at the EBR-II reactor were generated primarily by the operation of the nuclear reactor, irradiated reactor fuels, activated materials, and releases of radioactive noble gases. Photon energies were predominately greater than 30 keV (ORAUT-TKBS-0007-6). Photon radiation fields at the EBR-II reactor are known to have ranged from immeasurable levels to over 1×10^6 mrem/hr from irradiated fuel assemblies. However, the irradiated fuel assemblies were always heavily shielded inside the reactor pool or inside an inter-building coffin (IBC).

FCF: Based on a review of the available ANL-W records, the photon radiation fields at the FCF were predominately from irradiated reactor fuels, activated materials, and un-irradiated uranium. Photon energies were predominately greater than 30 keV (ORAUT-TKBS-0007-6). Photon radiation fields at the FCF are known to have ranged from immeasurable levels to over 1×10^6 mrem/hr from irradiated fuel assemblies. However, the irradiated fuel assemblies emitting those radiation levels were always heavily shielded inside the FCF's hot cells or inside an inter-building coffin (IBC).

HFEF-N: Based on a review of the available ANL-W records, the photon radiation fields at HFEF-N were predominately from irradiated reactor fuels, activated materials, and by the operation of the NRAD Reactor (October 1977 and later). Photon energies were predominately greater than 30 keV (ORAUT-TKBS-0007-6). Photon radiation fields at HFEF-N are known to have ranged from immeasurable levels to over 1×10^6 mrem/hr from irradiated fuel assemblies. However, the irradiated fuel assemblies emitting those radiation levels were always heavily shielded inside the HFEF-N's hot cells or inside heavily shielded casks.

FASB: Based on a review of the available ANL-W records, the photon radiation fields at the FASB were predominately from unirradiated uranium. Photon energies for uranium facilities are predominately in the range of 30-250 keV. Because the focus of data capture efforts was on contamination survey records versus radiation survey records, little is known about the actual photon radiation levels within the FASB. However, the photon radiation fields in the FASB would have ranged from immeasurable levels to those associated with large quantities of recycled highly-enriched uranium.

ITF: Based on a review of the available ANL-W records, the photon radiation fields at the ITF were predominately from slightly-irradiated reactor fuels, activated materials, X-ray generating equipment, and un-irradiated uranium. Photon energies were predominately greater than 30 keV (ORAUT-

TKBS-0007-6). Photon radiation fields at the ITF are known to have ranged from immeasurable levels up to 700 mrem/hr from slightly-irradiated fuel assemblies.

TREAT: As is the case with any reactor, the potential existed for TREAT personnel to receive external exposures due to emissions from mixed fission and activation products during activities associated with reactor operation and maintenance. Reactor shielding permitted personnel access around and atop the reactor during steady-state operations at 100 kW. Access to the subpile room was controlled during steady-state operation and, prior to transient operations, the building was evacuated of all personnel (TREAT Manual, 1971).

Radiation levels were monitored via a Jordan Electronics Model "RAMS II" with an alarm and metering panel at the Control Building. Detectors were located in the subpile room, on the main floor, on the reactor-coolant exhaust filters, and on the exhaust stack. The monitor was set to alarm any time the main floor radiation level exceeded 7.5 mr/hr or the subpile room radiation level exceeded 50 mr/hr (TREAT Manual, 1971).

Photon exposure resulted from photon radiation produced during nuclear fission. In addition, many of the fission and activation products resulting from fuel irradiation decay by photon emission. The workplace photon energy spectra in the TREAT operational areas would have been dominated by high-energy photons from mixed fission and activation products. General gamma radiation levels at a distance of 10 feet from the reactor during operations at 100 kW were 5-8 mrem/hour (Freund, 1960).

ZPPR: The possibility of external exposure existed at ZPPR as a result of working with reactor processes and fuel loading and unloading. Photon exposure resulted from photon radiation produced during nuclear fission. In addition, many of the fission and activation products resulting from fuel irradiation decay by photon emission. The workplace photon energy spectra in the ZPPR operational areas would have been dominated by high-energy photons from mixed fission and activation products.

L&O Building: Photon exposure potential existed from the mixed fission and activation products from the reactor fuels analyzed within the L&O Building. The workplace photon energy spectra in the L&O Building would have been dominated by high-energy photons from mixed fission and activation products. Pa-234m is a decay product in the U-238 decay chain and emits a 2.29 MeV beta particle. A significant quantity of photons resulting from Bremsstrahlung radiation are produced and contribute photons of intermediate energy (30-250 keV). Although enriched uranium has significantly less in-growth of Pa-234m, U-235 and its decay products emit a 185.7 keV photon 57% of time and a 143.8 keV photon 11% of the time.

The most significant gamma activity would have been from Cs-137 decaying to Ba-137m and its subsequent gamma emission, as well as the decays of Eu-154, Ru-106, and Ce-144. To a lesser degree, gamma activity would also result from the decay of Rh-106 and Pr-144.

In addition to the fission-product gamma rays, gamma rays from the activation of fuel cladding and structural materials (e.g., Mn-54, Co-58 and Co-60) would also be present, but would vary depending on the type of fuel under analysis at the time. The cells (caves) of the analytical junior cave facility were designed to safely handle analytical samples with a radioactivity level of 20 Ci of 1-MeV gamma radiation (EBR-II Fuel Cycle Story, 1987).

5.3.2.2 Beta

EBR-II: Based on a review of the available ANL-W records, the beta radiation fields at the EBR-II reactor were generated primarily by the operation of the nuclear reactor, irradiated reactor fuels, activated materials, releases of radioactive noble gases, and radioactive contamination from those materials. All beta radiation at the EBR-II reactor is attributed to greater than 15 keV electron radiation (ORAUT-TKBS-0007-6).

FCF: Based on a review of the available ANL-W records, the beta radiation fields at the FCF were generated primarily by irradiated reactor fuels, activated materials, and radioactive contamination from those materials. All beta radiation at the FCF is attributed to greater than 15 keV electron radiation (ORAUT-TKBS-0007-6).

HFEF-N: Based on a review of the available ANL-W records, the beta radiation fields at the HFEF-N were generated primarily by irradiated reactor fuels, activated materials, and radioactive contamination from those materials. All beta radiation at the HFEF-N is attributed to greater than 15 keV electron radiation (ORAUT-TKBS-0007-6).

FASB: Based on a review of the available ANL-W records, the beta radiation fields at the FASB were predominately from un-irradiated uranium. All beta radiation at the FASB is attributed to greater than 15 keV electron radiation (ORAUT-TKBS-0007-6).

ITF: Based on a review of the available ANL-W records, the beta radiation fields at the ITF were predominately from slightly-irradiated reactor fuels, activated materials, and un-irradiated uranium. All beta radiation at the ITF is attributed to greater than 15 keV electron radiation (ORAUT-TKBS-0007-6).

TREAT and ZPPR: Reactor operations in the TREAT and ZPPR operational areas did not have pure beta external hazards, but beta exposures would have resulted from the mixed fission and activation products produced by the reactors. Whether beta radiation was considered an external hazard depended on the maximum energy of the beta emission for a given radionuclide, any shielding used, and the use of protective clothing.

L&O Building: As with photon exposures, beta exposure potential resulted from the mixed fission and activation products from the reactor fuels analyzed within the L&O Building. Most fission products are initially rich in neutrons and undergo beta decay. In general, the thick shielding afforded by the cell walls would attenuate any beta particles emitted.

5.3.2.3 Neutron

Operating reactors would have been the primary source of neutron exposure during the period under evaluation. These reactors were experimental facilities for measuring shielding factors, reactor kinetics, and neutron irradiation. Thermal neutron radiation levels 10 feet from the TREAT reactor during operations at 100 kW were 50-1,500 n/cm^2/sec (Freund, 1960).

Locations and periods of where neutron exposures are considered to be likely within the EBR-II Complex are identified in Table 6-11 of ORAUT-TKBS-0007-6. The EBR-II Complex areas listed in that table include TREAT (1958–1994), ZPPR (1969–1992), and the HFEF-N's NRAD Facility (1977–2006) (ORAUT-TKBS-0007-6). More-than-incidental exposures to neutrons were not likely at the L&O Building, FASB and ITF because of the materials and activities being performed at those facilities. The following provides additional details for the EBR-II Complex facilities with potentially-significant sources of neutrons.

EBR-II: Based on a review of the available ANL-W records, the neutron radiation fields at the EBR-II reactor were generated primarily by the operation of the reactor and irradiated reactor fuels. Neutron radiation fields at the EBR-II reactor are known to have ranged from immeasurable levels to over 1 x 106 mrem/hr in the reactor core. In occupied areas of the reactor, neutron radiation fields were normally immeasurable (Neutron Surveys, 1967).

FCF: Based on a review of the available ANL-W records, the neutron radiation fields at the FCF were generated primarily by irradiated reactor fuels, and a Sb-Be neutron radiography source (September 1968 and later). However, these sources of neutrons were always in either heavily-shielded casks or the FCF hot cells. In occupied areas of the FCF, neutron radiation levels were negligible.

HFEF-N: Based on a review of the available ANL-W records, the neutron radiation fields at HFEF-N were generated primarily by irradiated reactor fuels and the NRAD Facility's reactor (October 1977 and later). Neutron energies were generally less than 20 MeV (ORAUT-TKBS-0007-6). Neutron radiation fields at the NRAD Facility's reactor are known to have ranged from immeasurable levels to over 1 x 106 mrem/hr in the reactor core.

ZPPR: The operating reactor would have been the primary source of neutron exposure during the period under evaluation. These reactors were experimental facilities for measuring shielding factors, reactor kinetics, and neutron irradiation. Neutron and gamma dose rates were measured in ZPPR since start-up in 1969. In general, the neutron fields experienced by ZPPR personnel were relatively insignificant (less than 10 mrem/hr.) with the exception of those fields experienced between the reactor halves when the personnel shields were removed (Neutron Exposure Assignments, 1977).

5.4 Incidents at the ANL-W Site

5.4.1 Incidents at the EBR-I Complex

ATTRIBUTION: Section 5.4.1 was completed by Mike Mahathy, Oak Ridge Associated Universities, and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the EBR-I Complex were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the EBR-I Complex are explained in the associated text.

ANL-W facilities and activities involved experimental reactor design and development, closed-cycle irradiated-fuel processing, and analytical support. The ANL-W monitoring and analytical programs were designed to initiate an investigation of any potential internal intake, as indicated by off-normal workplace indicators such as personnel contamination or elevated air sampling. Although incidents involving exposures of regulatory significance were rare, the following incidents are provided as important events in ANL-W's radiological history.

5.4.1.1 Excursion at BORAX-I (July 1954)

A deliberate excursion to destroy the reactor was planned and carried out on July 22, 1954 to determine the inherent safety of the system under extreme conditions (ANL-W History, 2006; QA Assurance Files, 1991). After the test run of the reactor, one of the control rods became bowed in the subassembly, preventing the start of the excursion. Some of the staff were instructed to fix the rod so that the excursion could proceed. Two workers went to the reactor and, looking down, saw the unmistakable Cerenkov glow. One of the workers who was monitored for external dose reached into the pit to straighten the rod (Haroldsen, 2008).

Within a few milliseconds after the start of the excursion (following removal of the control rod from the reactor), a dark grey column of smoke and water was ejected from the reactor to a height of more than 80 feet (see Figure 5-48). Shortly thereafter, the sound of a medium-sharp detonation reached the control room about a half-mile away. Fuel plate fragments were scattered up to 300 feet, but no widespread dangerous dispersal was observed. Most of the heavy debris fell to the ground near the shield pit. The control rod drive mechanism fell on the side of the earth shield after having been thrown about 30 feet into the air. Fragments of fuel plates were thrown up to 200 feet from the reactor site. A photographer was sent to retrieve the film of the explosion; he then had to be decontaminated of fresh mixed fission products. Over the following days, workers were rotated into the reactor pit to recover gold foils so that they could determine the power the reactor attained before exploding. Wearing protective clothing and respirators, site staff retrieved materials and usable equipment from the reactor and surrounding area over the next two months (Haroldsen, 2008; Personal Communication, 2015b). Health Physics monitored clean-up and recovery operations (Griffiths, 1954). Workers would pick up a shovel full of sludge and hold it to a detector to observe the count rate to determine the presence of a gold foil.

Surveys of the total fission-product radioactivity of all the debris revealed that most of the fuel originally in the reactor could be accounted for within a 350 ft.-radius. No large fraction of the reactor core material left the site in the form of airborne material. More than fifty samples were analyzed for alpha activity from near the reactor through the site periphery. The samples included both filter paper samples of air particulates from the continuous air monitors permanently located around the ANL-W periphery, and gummed paper fallout plates from special stations established for the Borax experiment. In most cases, the activity of the air samples was statistically zero. Each sample represented from one to five days of continuous sampling. Health Physics analyzed three samples of contamination from Borax using a sodium iodide crystal gamma spectrometer to determine the energies of the contamination. They compared the spectrums of each to a spectrum of three-year-old fission products, and to a spectrum of six-day-old fission products from a test performed at the Pacific Proving Grounds in May 1954.

Source: http://www.ne.anl.gov/About/reactors/lwr3.shtml

Figure 5-48: Destructive Test of BORAX-I

The BORAX spectra closely compared to the spectrum from the Pacific Proving Grounds (Griffiths, 1954). Figure 5-49 shows spectra from two of the contaminated BORAX-I items (left and center) and from the Pacific Proving Grounds (right). A modeled reconstruction of the excursion conducted by INL in 1991 provided estimates of fission products released by the excursion (QA Assurance Files, 1991).

Samples counted within one day after termination of sampling had alpha activities ranging from 32 to 200 dpm. The recounting of samples after two to three days showed the alpha activity to be the decay of radon and thoron daughters (Griffiths, 1954). A 1980 study stated that about twelve percent of the uranium in the core, or 479 g, was recovered from fuel fragments in the reactor pit and area surrounding BORAX (Smith, 1980). A modeled reconstruction of the excursion conducted by INL in 1991 estimated that 4.7 g of uranium was released to the environment (QA Assurance Files, 1991). The balance of the uranium remained in the melted core (Smith, 1980).

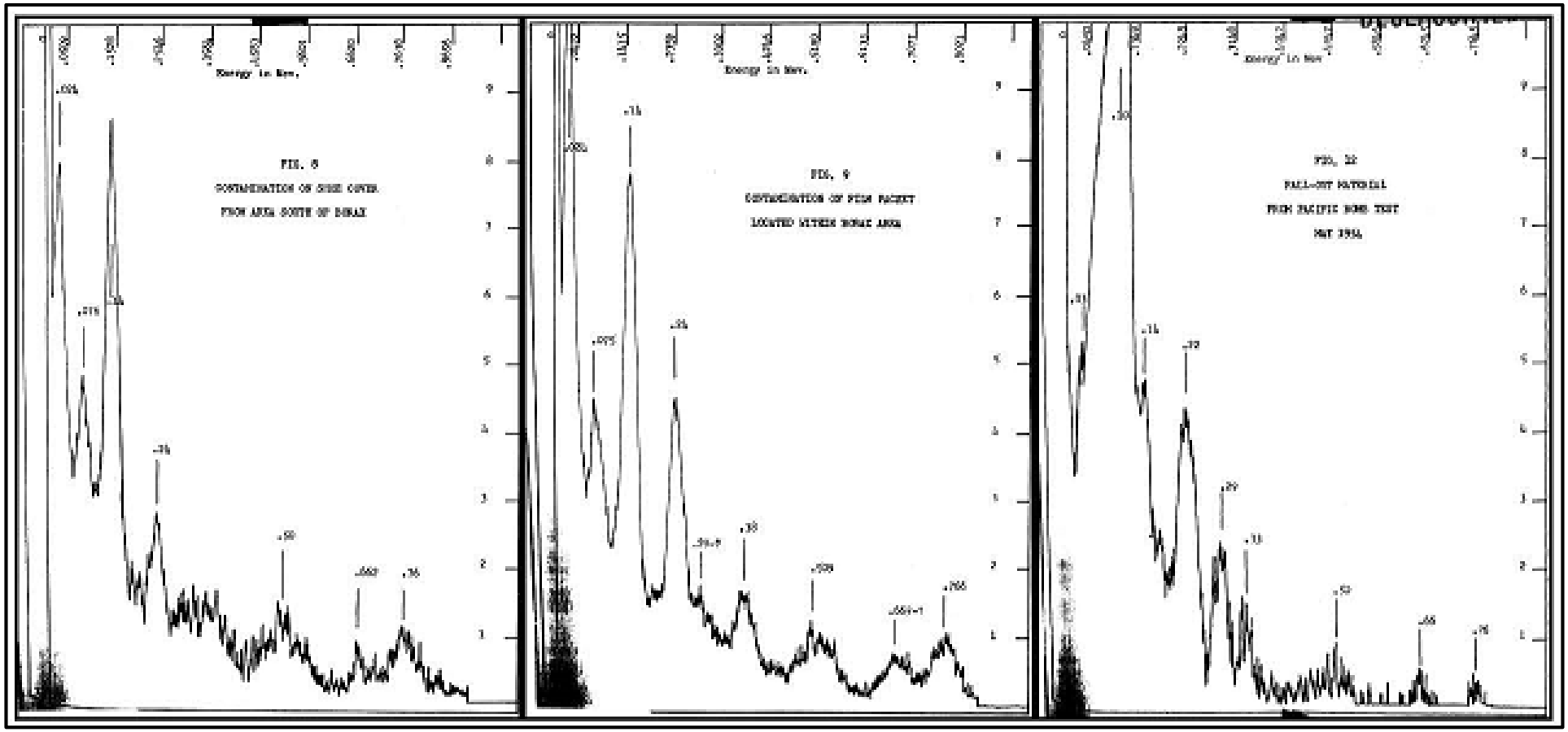


Figure 5-49: Spectra Comparison: BORAX-I and Pacific Proving Grounds

5.4.1.2 Suspected Brick Rupture in EBR-I Breeding Blanket (April 1955)

In April 1955, transferrable contamination was discovered at the EBR-I reactor. A brick in the breeding blanket surrounding the reactor was suspected to have ruptured. Workers of concern were identified and sampled for beta and plutonium urinalysis. The available bioassay results did not indicate intakes of radioactive material occurred due to this incident (Bioassay Sheets, 1963, PDF pp. 32, 54).

5.4.1.3 Partial Meltdown of EBR-I Core (November 1955)

On November 29, 1955, the EBR-I core suffered a partial meltdown while performing a planned experiment for taking the reactor critical at a very low power level without coolant flow. The purpose of the experiment was to obtain transient temperature coefficients on fuel only. The meltdown occurred due to a misunderstanding by the reactor operator to use the slower control rods to shut down the reactor instead of the fast-acting safety rods. The two-second time difference was enough to permit the temperature of the reactor to overshoot to a point at which the alloying of uranium and steel and melting of uranium took place. Although the EBR-I reactor operated with an argon cover gas, fission products from the meltdown broke through and leaked into the reactor building. An order to vacate the reactor building was made. A large comprehensive special bioassay program followed. (Kittel, 1957; Zinn, Jun1956; Bioassay Sheets, 1963; Haroldsen, 2008). The bioassay results indicated there were no internal depositions of radioactive material as a result of the partial meltdown (Bioassay Sheets, 1963).

In March 1956, four months after the meltdown, work began on opening the reactor to examine the extent of damage. The time delay was essential to allow for an extended period of radioactive decay and to ease the procedure for lifting the core holder and fuel elements from the reactor. A concrete "hot" cave was constructed on the top of the reactor to provide shielding from the high radiation levels for the anticipated high external dose rates that were expected when the reactor assembly was lifted out of the reactor tank (Disassembly Compliance Report, 1956). The core was removed and placed in a custom-made transfer coffer that was sent to ANL in Illinois for examination. Air monitoring and contamination surveys were performed during the core removal (Air Sample Data Sheets, 1956).

5.4.1.4 Personnel Contamination at BORAX-IV (July 1957)

On July 17, 1957, a [job category redacted] got a small amount of a radioactive waste water sample in [body part redacted]. Health Physics surveyed the [job category redacted] [body parts redacted] with no measureable activity found. Health Physics collected a bioassay sample that was reported as negative for activity (Safety Monthly Reports, 1957, PDF p. 80).

5.4.1.5 Airborne Contamination at BORAX-IV (September 1957)

A leak in the air ejector filter occurred in September 1957 during an extended run of the BORAX-IV reactor. Airborne contamination was detected by Health Physics. Additional air samples were collected and processed the following day; results indicated no activity above background. The filter on the air ejector was repaired (Safety Monthly Reports, 1957).

5.4.1.6 Fuel Defect Test at BORAX-IV (March 1958)

BORAX-IV was not operated at a power level from December 5, 1957 until February 19, 1958 when a power run was attempted. In the interim, ten new fuel elements were added to periphery of the core. After attempting to start the reactor on February 19, and again on February 25, radioactivity was measured in the reactor hotwell and turbine buildings at five times the maximum design activity (Bailey, 1958). A steam break also occurred in the turbine during the February attempt (Findings Report, 1958). Those start-up attempts were halted before maximum power was reached because ANL-W assumed that one of more fuel elements had rupture, but remained encapsulated by the cans.

ANL-W planned controlled excursions for March 11-12 to determine the extent of the rupture and the limiting effects encountered while powering the reactor with defected fuel (Bailey, 1958). ANL-W Health Physics helped plan the test and provided monitoring during and after the test, which was concluded on March 12, 1958 (Status Summaries - Borax IV, 1958; Memo of Run, 1958; Defective Fuel Experiment, 1959). Dose rates were measured around the reactor and turbine buildings using Juno survey meters and neutron detectors. Health Physics also monitored reactor water, steam, and air ejector gases for radionuclides. Dose rates resulting primarily from $N^{16}[O^{16}(n,p)N^{16}]$ were measured ranging from 27 mr/hr at the main doors to the turbine and 30 mr/hr at the reactor buildings to 50,000 mr/hr at the turbine air ejector exhaust filter (Borax-IV Test, 1959). Xe-138 and Kr-88 accounted for the major portion of radioactivity released from the fuel elements. The major contaminant released from the steam system was Cs-138 (Borax-IV Test, 1959; Findings Report, 1960). I-131 was detected in reactor water, steam, and condensate. Long-lived radionuclides that were identified in trace amounts in the turbine and condenser after operation were Ba-140, La-l40, Sr-90, Y-90, Sr-89, and Cs-l37 (Borax-IV Test, 1959). Alpha radiation was not detected in water, steam, or air. Only trace amounts of fission products without nongaseous precursors (e.g., Mo-99) were detected in the reactor water (Borax-IV Test, 1959, PDF p. 31).

Personnel involved in the February and March releases were monitored for external radiation, and for fission product intake via urinalysis (Memo of Run, 1958; Defective Fuel Experiment, 1959). NIOSH has obtained those urinalysis results (Bioassay Sheets, 1963).

5.4.1.7 Fire in EBR-I Rod Wash Room (September 1959)

On September 17, 1959, an explosion in the rod washroom occurred when a pump and fan unit for the washroom exhaust caught fire. The fire spread to the wooden frame around the unit and part of the media of a filter used in the exhaust system. Air samplers were placed into service to determine if airborne activity escaped into the EBR-I basement area. Air samples taken during and after the event did not show significant airborne contamination. All bioassay of personnel involved were less than the limits of detection (Safety Monthly Report, 1959).

5.4.1.8 Contamination on Plutonium Plates at ZPR-III (March 1960)

On March 14, 1960, ZPR-III staff were preparing for the initial use of plutonium fuel plates. These plates were clad in stainless steel. An inspection of the plates discovered “a small number” of plates with external contamination that was confirmed to be plutonium. While there were no intakes of radiological significance from the incident, it was the only time in the history of ZPR-III that evidence of plutonium contamination was found. Due to the recognized hazards of plutonium, a rigorous

program was established for surveying plutonium fuel plates, not just upon receipt, that sustained for ZPR-III's operating history (Safety Monthly Report, 1960; Fish, 1967).

5.4.1.9 Alpha Contamination at AFSR (November 1963)

On November 29, 1963, while removing irradiated U-235 foils from the reactor, a work table used for unloading the foils was contaminated to 200 dpm/100 cm^2 alpha. The table was subsequently decontaminated to less than 10 dpm/100cm^2 alpha (Safety Monthly Reports, 1963).

5.4.1.10 Defective Fuel Test at BORAX-V (August 1964)

Much like the fuel defect tests conducted at BORAX-IV, ANL-W prepared for power tests at BORAX-V using prefabricated defective fuel. Defective fuel elements were prepared at ANL to test power generation using superheated defective fuel. The canned defective elements replaced some of the normal fuel elements. The core was operated from the first week in August until August 14, 1964, producing 266 MW with the core containing partially-defective fuel. That final experiment in BORAX-V demonstrated that negligible contamination to turbo-generator equipment resulted from operating with an experimentally-defective fuel element in the superheated core (Borax Superheater, 1964).

ANL-W Health Physics monitored the August power generation. The major contaminants released from the steam system (as measured by air monitoring or smear analysis) were: Co-58, Kr-88, Sr-91, I-131, Xe-138, Ba-140, and La-140 (Progress Report, Aug1964). Personnel working at BORAX-V were routinely monitored by film badge and bioassay, including whole-body counting and urinalysis.

5.4.1.11 Spontaneous Combustion of U-235 Foils at ZPR-III (November 1966)

On November 23, 1966, enriched-uranium (93.19%) foils in paper envelopes caught on fire in the ZPR-III Special Materials Vault as they were being removed from a sealed storage container. The fire was quickly extinguished. Alpha contamination was found on the outer clothing and person of [number redacted] workers. The workers were decontaminated and sent to Central Facilities for an *in-vivo* count. The results of the counts were less than detection limits. In addition, [sample type redacted] samples were also requested of the [number redacted] workers. A gamma count of the [sample type redacted] samples indicated detectable activity but not significant intakes. Clean-up operations included smears and air samples taken in the vault, Loading Room, Control Room, Airlock, Assembly Room, and corridors. Decontamination efforts were taken until the floors were restored to < 5 dpm/100cm^2 transferrable alpha contamination (Incident Report, 1966; Incident Summary, 1967).

5.4.1.12 Leaking Po-Be Source at ZPR-III (April 1967)

On April 14, 1967, the facility [job title redacted] suspected a possible leaking Po-Be source during a source exchange at ZPR-III. External contamination of 1800 dpm/100 cm^2 was discovered on a smear taken on the source. An air sampler in the Assembly Room (where the source was located) showed an "insignificant" amount of alpha activity. Analysis of the original smear indicated the contaminant to be Po-210. [Sample type redacted] and a [sample type redacted] were requested of the [job title redacted]. No activity was detected in either sample (Waste Discharges, 1958).

5.4.2 Incidents at the EBR-II Complex

ATTRIBUTION: Section 5.4.2 was completed by Jason Davis, Oak Ridge Associated Universities, and Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding the EBR-II Complex were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the EBR-II Complex are explained in the associated text.

ANL-W facilities and activities involved experimental reactor design and development, and irradiated fuel processing. ANL-W's monitoring and analytical programs were designed to initiate an investigation of any potential internal intake indicated by off-normal workplace indicators, such as personnel contamination or positive air sampling. NIOSH found extensive personal monitoring data for mixed fission and activation products (beta/gamma); therefore, only external exposures of regulatory significance, significant alpha-contamination incidents, and criticality incidents are presented below. Several other incidents are presented to illustrate the effectiveness of radiological controls in preventing the spread of contamination and minimizing the chance of a significant intake of radioactive material.

5.4.2.1 Perchloric Acid Explosion in L&O Chemistry Lab 40 (July 1962)

A perchloric acid explosion occurred in Chemistry Laboratory 40 on July 13, 1962. A glass beaker contained radioactive xenon and krypton gases absorbed on charcoal. The material was being dissolved in perchloric acid; the experiment was inadvertently allowed to go to dryness and an explosion occurred. Most of the debris was confined to the hood; however some glass fragments fell to the floor outside the hood. No activity was detected outside the hood; debris inside the hood was disposed of as dry active waste (Safety Monthly Report, Jul1962, PDF p. 143).

5.4.2.2 L&O Building Broken Specimen (March 1963)

On March 4, 1963, it was determined that some unirradiated, clad uranium-fissium alloy specimens had broken while being heated in an electric furnace. This triggered a survey of the entire Metallurgy Laboratory for both alpha and beta/gamma contamination using portable instruments and dry smears. No contamination was found. A special air sample indicated that no material was released into the air. The specimens in question were removed from the furnace, sealed in polyelthylene sleeves, and placed in dry active waste. A survey of the furnace after cool-down showed no contamination present on the interior surface (Safety Monthly Reports, 1963, PDF p. 130).

5.4.2.3 Hydrogen Explosion in EBR-II/FCF Transfer Lock (Dec. 1965)

On December 26, 1965, a hydrogen explosion occurred in the transfer lock between the EBR-II Reactor and FCF while the empty IBC was in the lock. This resulted in minor damage to the coffin and the lock. Specific requirements of the operating procedures limited the time an IBC containing a subassembly could be in the lock, but no such requirements were designated for an empty coffin. The empty IBC was placed in the transfer lock with the blower shut off. With the blower shut off, the battery charger was thus inadvertently placed on a high rate of charge, which caused hydrogen to be generated. It was surmised that, when the lock cover was actuated some hours later, a spark from a door limit switch ignited the hydrogen. The force of the resulting explosion blew the lid off the lock and bent the hinge; fortunately, no personnel were injured (Stevenson, 1987, PDF p. 72).

5.4.2.4 EBR-II Reactor Radioactive Sodium Fire (June 1967)

A small radioactive sodium fire occurred in the primary sodium-sampling room. Thereafter, access to the sampling room was restricted until the Na-24 radioactivity had decayed. Some Na-24 was transported out of the sampling room by the ventilation system, and caused minor air and surface contamination in the reactor building (Progress Report, Jun1967, PDF pp. 47-48).

5.4.2.5 FCF Contaminated Water Spill (June 1968)

In the FCF Reactor Passageway, a spill of subassembly wash water overflowed from an inter-building coffin (IBC) when the gate valve at the top of the IBC was not closed properly. The contaminated water spread on the floor of the Reactor Passageway and left dry residues emitting as high as 3E+04 dpm/100 cm^2. The floor was mopped to remove the residues (Stevenson, 1987, p. 242).

5.4.2.6 ZPPR-AFSR Dropped Blanket Drawer (January 1975)

On January 20, 1975, while spot-checking the loading for Assembly #5, a blanket drawer was dropped, striking eight fuel drawers that had been pulled out of the matrix about 3 inches. The cell was immediately evacuated via the work room. Radiological controls personnel were notified and the mound area placed in a state of isolation. Personnel surveys of the loading crew revealed no contamination and the crew was released from the work room.

Re-entry was made by a team of three men, one of whom was from Radiation Safety. Each man wore a self-contained breathing apparatus. Smears were taken. Direct surveys were found to be unreliable because the depleted uranium from the blanket drawer had spilled out and broken, the pieces and powder scattering over the fuel drawers. There were no high readings on the alpha meter even under this condition, the highest being around 300 cpm. One fuel plate had visible damage and one smear was taken on this plate. The drawers were bagged, taken to the work room and placed in the hoods for containment. No high alpha activity could be detected on the smears with an alpha meter.

Personnel surveys of the re-entry team revealed no contamination and they were released from the work room. The smears were counted and highest alpha activity noted was 150 dpm/100 cm^2. Three smears with activity between 120 to 150 dpm/100 cm^2 were selected for analysis. One of these was a smear taken on the visibly-damaged fuel plate. Two smears were sent to CF 690 for analysis, and one was kept at ZPPR for analysis. Recovery operations were halted until analysis results were in. The analysis results were identical on all three smears and indicated the presence of depleted uranium with no evidence of Pu.

Recovery began by removing the fuel from the drawers for inspection and contamination surveys. Smears samples were taken of each the 42 fuel plates. The highest activity recorded was 90 dpm/plate alpha. One more plate was found to be slightly dented. The two dented wafers were isolated in a sealed canister after they were decontaminated to <10 dpm/plate alpha. The rest of the fuel was isolated in sealed canisters and all canisters taken to the vault. Reflecting an overabundance of caution, this work was done with full-face respirators despite the work being performed under hoods. An air sample taken in the breathing zone decayed to 1.1 x 10^{-13} µCi/cc in 20 hours.

Decontamination of the cell and work room hoods began and continued throughout the next day. Activity was reduced to less than 15 dpm/100 cm^2 alpha and less than 100 dpm/100 cm^2 beta/gamma.

Air samples decayed to less than guidance levels for Pu after 24 hours of decay (Weekly Report, 1975, PDF pp. 5-6).

6.0 Summary of Available Monitoring Data for the Class Evaluated by NIOSH

The following subsections provide an overview of the state of the available internal and external monitoring data for the ANL-W class under evaluation.

6.1 Available ANL-W Internal Monitoring Data

ATTRIBUTION: Section 6.1 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

The Health and Safety Division of AEC's Idaho Operations furnished a comprehensive personnel metering program for radiation monitoring of all Idaho site personnel. The Analysis Branch of the division initially provided routine urinalysis of all personnel with sampling frequency varying between quarterly and yearly, depending on the expected potential for uptake (Annual Report, 1959).

Chemical and radiochemical urinalysis were run at three-month intervals on all people who had frequent contact with radioactive material and at six-month intervals on those people who had only occasional contact. Analyses were also made annually on about fifty NRTS personnel who were not exposed to radiation to provide a statistical control group (Health Physics Services, 1952).

Site-wide routine samples were initially analyzed only for gross alpha and beta, with isotopic analysis performed only if activity was discovered in the sample, although capabilities existed for analyses of specific isotopes, such as Pu-239, Sr-90, Co-60, and I-131 in body excreta, water, and air samples. The Analysis Branch also made analyses for materials related to good industrial hygiene practices, such as beryllium, lead, mercury, arsenic, fluorine, or any other chemical that may have been required (Annual Report, 1959).

Based on the loss of iodine due to volatilization during the wet-ashing process, as well as the small likelihood of a significant uptake of a pure beta-emitter without a correspondingly-large uptake of gamma-emitting fission products, the gross-beta procedure on urine samples was discontinued entirely in December 1960. However, in order to detect a slow build-up of the bone-seeking, beta-emitting strontium isotopes from low-level sources over long periods of time, specific strontium analyses were made for each individual who had worked in a reactor or chemical processing area for two years or longer (Murphy, 1966); these analyses were performed every two years and on termination of employment (Annual Report, 1961).

NIOSH has found only a small set of bioassay data for ANL-W workers for the period prior to 1958. During this evaluation, NIOSH learned that some monitoring data on early workers is stored at the INL records center and some at ANL in Chicago. In September 2015, NIOSH performed an

additional data capture at INL, partly to look for additional bioassay data for the period 1952 through 1960. In October 2015, NIOSH also performed a review of available data at ANL for ANL-W workers. NIOSH submitted to both INL and ANL a list of 50 former workers believed to have worked at ANL-W prior to 1961, requesting available internal and external monitoring data. Partial data was returned for 48 of the 50 requested workers. Not all of the 50 workers were employed at ANL-W for the entire period 1952 through 1960. Table 6-1 lists the number of workers with bioassay records of any kind, by year, available to NIOSH for these 50 workers. Bioassay data reported for 1952 represent baseline and incident monitoring.

Table 6-1: Bioassay Results for 50 Selected ANL-W Workers

Year	Subset of 50 Selected ANL-W Workers with Bioassay Results
1952	12
1953	None
1954	15
1955	24
1956	11
1957	5
1958	23
1959	6
1960	24

Both INL and ANL supplied bioassay monitoring data they each have available for the fifty former workers. However, data was not provided for some of the former workers, and for some of the years requested - varying by former worker. While there was some overlap of records supplied by INL and ANL, there is a clear trend in data reporting, by workers, as shown in Figure 6-1. Note that ANL reported bioassay data starting in 1952, but the number of workers for whom data was supplied trends downward. The trend for INL is reversed, which carries into the years after 1960. For 1952, INL provided data for only one former worker; ANL provided data for no former workers. For 1955, ANL provided bioassay data for two former workers; INL provided bioassay data for four former workers. Not each of the 50 former workers were employed at ANL-W each of the years from 1952 through 1960.

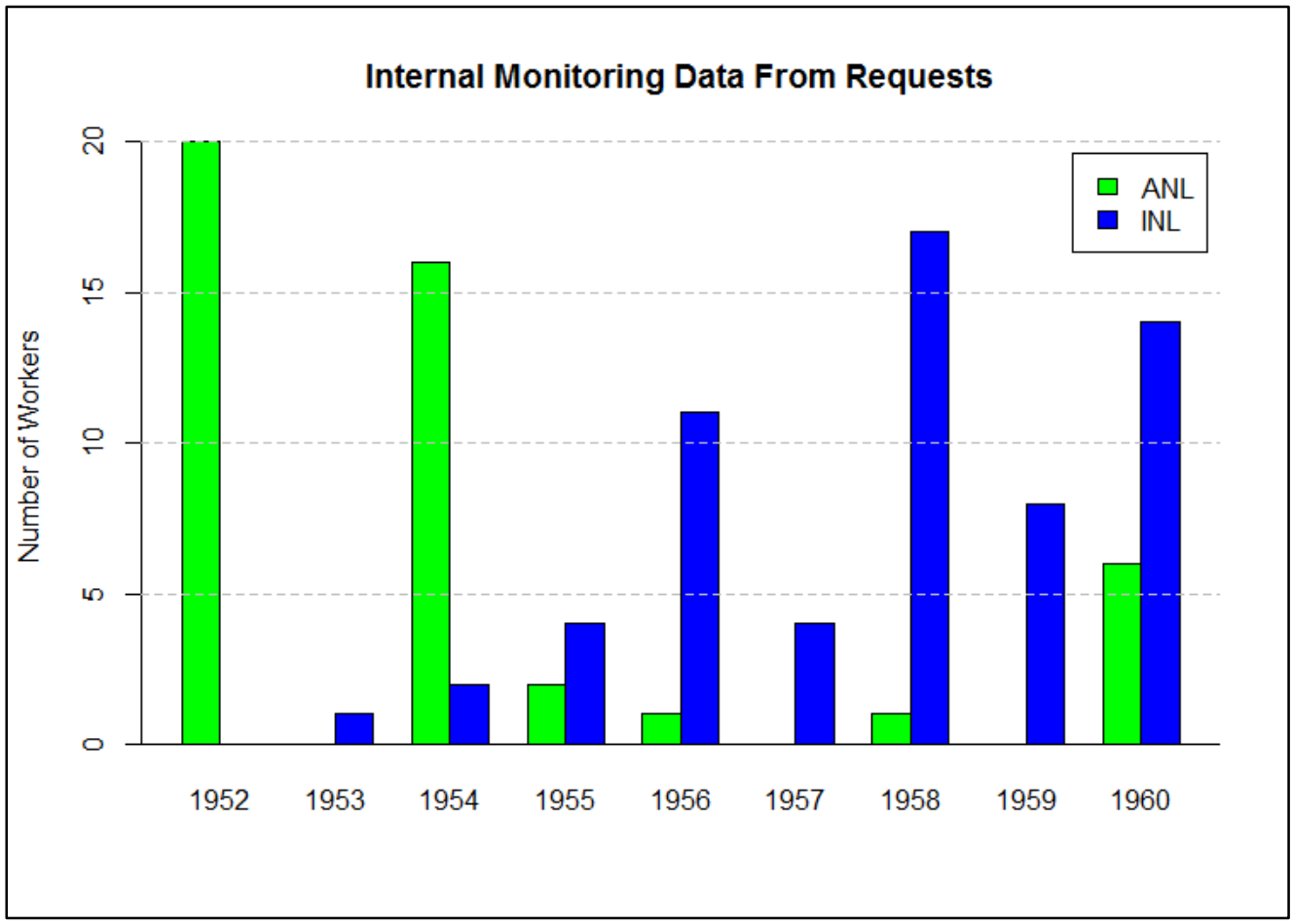
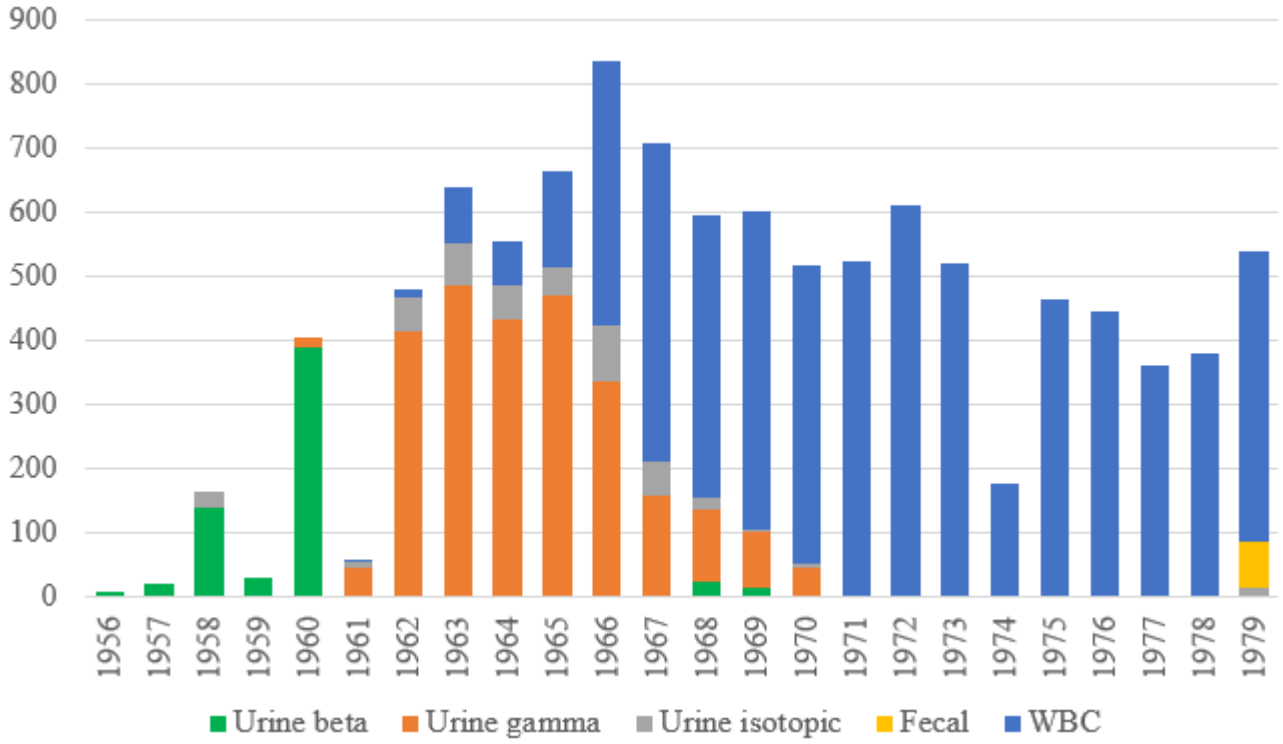


Figure 6-1: Bioassay Monitoring Data Trend

Whole-body counting (WBC) started at the Idaho site in January 1961. The convenience, sensitivity, and reliability of WBC made it the choice of personnel monitors for internal radioactivity from gamma-emitting nuclides. Consequently, urinalysis as a routine monitoring technique was gradually phased out by the laboratory. From that time, urine was analyzed only for iodine, uranium, and strontium or for elimination studies on specific isotopes (Annual Report, 1969). Early in the site's history, job duties and their corresponding exposure potentials were used to place employees into the following routine WBC frequencies: quarterly, yearly, or no count needed. Special analyses were requested by the relevant Health Physics sections when internal exposure was suspected due to events occurring in a plant; terminations that required a physical examination (as determined by the relevant Health & Safety sections) also required a whole-body count (see Figure 6-2) (Procedures, 1963).

A portable, unshielded whole-body counter was developed and constructed in 1963. A medical van containing a whole-body counter was designed and purchased in 1965. By 1969, WBC equipment available at the Analytical Chemistry Branch Laboratory included rotational counting, helical scanning, and a detector for determining Sr-90 in the skeleton by counting the Bremsstrahlung emission from the skull (Annual Report, 1969).



Source: Bioassay, 1956-1984; Bioassay, 1956-1967; Bioassay, 1958-1986

Figure 6-2: Total Internal Monitoring Performed at ANL-W (1956-1979)

In May 1967, the Health Services Laboratory was organized from six branches of the Idaho Operations' Health and Safety Division. Its primary function was to develop a technical staff in occupational medicine, health physics, radiological dosimetry, analytical chemistry, and instrumentation to support the Idaho site's occupational health programs (Annual Report, 1969). The Analytical Chemistry Branch had responsibility for estimating the dose received from internally-deposited radionuclides at the site until 1971. Sometime during 1971, the responsibility was transferred to a health physicist in the Environmental Sciences Branch (Methods, 1979).

ANL-W recognized the potential for plutonium uptake when handling plutonium-fuel-bearing assemblies and established a bioassay program to monitor those individuals known to be working with these materials. The program required that both a urine sample and a fecal sample be submitted annually for analysis. A total of 68 employees were included in this sampling plan in 1975: 32 from HFEF, 7 from TREAT, 21 from Radiation Safety, and 8 from ZPPR (Bioassay Instructions, 1975).

Details regarding the various analyses used and the associated minimum detectable activities are presented in the Technical Basis Document for INL - ANL-W - Occupational Internal Dose (ORAUT-TKBS-0007-5).

6.2 Available ANL-W External Monitoring Data

ATTRIBUTION: Section 6.2 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

Personal dosimeter data (i.e., film badge and/or TLD) are the primary data type used to reconstruct external doses for ANL-W workers. Those data are specifically used to reconstruct a worker's measured and missed external doses. For the period under evaluation, the ANL-W site monitored all workers for external photon and beta (electron) doses via their own dosimeters when inside the major radiological operating areas (Cipperley, 1958; Cipperley, 1968; ORAUT-TKBS-0007-6).

NIOSH does not possess a complete set of the ANL-W's dosimeter data; however, the DOE does have a complete set for the years 1958 and beyond. With two exceptions, the DOE provides the external dosimetry records for each energy employee who files an EEOICPA claim: (1) the dosimetry records for ANL-W workers are sometimes missing or incomplete for the years prior to 1958; and (2) external dosimetry records are currently not retrievable for certain workers who were only issued temporary dosimeters. Readily-retrievable permanent records were not created for some ANL-W workers (mostly visitors and subcontractors) who met the following criteria: (a) only had temporary dosimeters assigned; (b) only had dosimeter results below the dosimeter limits of detection (i.e., no positive external doses reported); and (c) did not have any bioassay records. The records of those temporary dosimeter results are available at the site; however, they have not been indexed by the DOE and are not currently in a readily-retrievable format. NIOSH is currently in the process of capturing the temporary dosimeter records and putting them in a retrievable format for the ANL-W worker EEOICPA claims.

A summary of the dosimetry data is provided later in Figures 6-4 and 6-5.

NIOSH submitted to both INL and ANL a list of fifty former workers believed to have worked at ANL-W prior to 1961, requesting available internal and external monitoring data. Not all of the fifty workers were employed at ANL-W for the entire period 1952 through 1960. Both INL and ANL supplied external monitoring data they each have available for the fifty former workers. However, data was not provided for some of the former workers, and for some of the years requested - varying by former worker. While there was some overlap of records supplied by INL and ANL there is a clear trend in reporting of data, by workers, as shown in Figure 6-3. Note that ANL reported external monitoring data for significantly more workers than did INL through 1955, after which ANL records drop off dramatically. For INL, the number of workers with reported external data steadily climbed from 1952 until 1960.

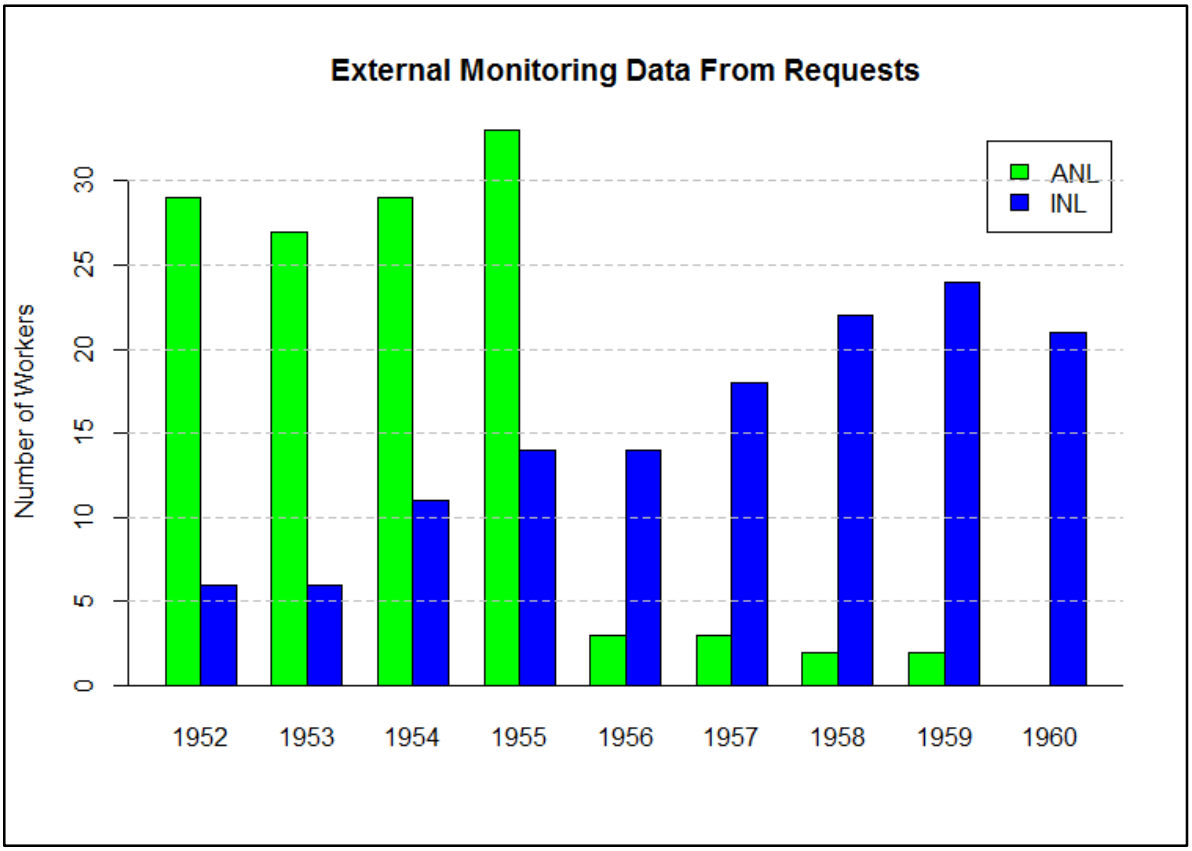


Figure 6-3: External Monitoring Data Trend

As early as 1955, ANL's entire Idaho Division was considered an active area, making the wearing of personal monitoring devices by all employees mandatory. The badges worn at this time contained sensitive film for measuring fast neutrons, in addition to the usual beta and gamma films (Personnel Exposure Reports, 1962, PDF p. 43).

In 1950, the Division of Medical Services was established and given the task of handling personnel monitoring for employees and contractors at the entire Idaho site. The Health Physics Division was established on June 30, 1951. This was the same year that EBR-I went "hot." The Personnel Metering Branch was established as one of seven branches of the AEC's Idaho Operations Office Health and Safety Division. The Personnel Metering Branch had the responsibility for determining external dose. The Branch furnished personnel metering services to 21 prime service contractors scattered throughout 18 separated areas within the Idaho site (Cipperley, 1959).

In the early days, dose was determined by film only encased in a film badge. Later, other devices were employed, such as pocket chambers (Pencils), finger rings, and wrist badges. These devices were used to detect all forms of radiation except alpha (i.e., beta, gamma, and neutrons). The Personnel Metering Branch maintained complete records was on all persons at the Idaho site. These records were cataloged by assigned badge number and were permanently filed according to date and alphabetically by surname. These records were summarized quarterly and statistically evaluated for each person (Puphal, 1996).

There were three types of film used in the badges serviced by the Personnel Metering Branch:

- The DuPont Beta-gamma Type 552 film packet was used in most of the area badges. Each film packet contained a piece of sensitive film and a piece of insensitive film along with a 1 mm cadmium filter. The sensitive film (502) had a threshold level of approximately 30 mR and became saturated at about 5 R. The insensitive film (510) had a threshold of approximately 150 mR and became saturated at about 30 R. If the sensitive film became saturated, the insensitive film was read (Wilson, 1961).

- The DuPont Type 558 film packet was sometimes referred to as a "Hanford-type" dosimeter. The packet's sensitive film (508) had a range of 10 mR to 10 R and was insensitive from 10 R to approximately 800 R. The insensitive film (1290) had a range of 10 R to 1,000 R (Wilson, 1961).

- Kodak Personal Neutron Monitoring film, Type A, was used in the badges when neutron coverage was requested, with one piece of neutron film placed in each packet. This film was sensitive to neutrons with energies greater than 0.5 MeV; the minimum detectable level was approximately 14 mrem. This film was worn in the badge with either of the types of beta-gamma film.

Over time, it became evident that better methods of exposure determination and more efficient record and reporting procedures were required. In March 1958, the Branch began the use of a modified version of the Hanford-type plastic film badge containing filters of 0.0393" cadmium, 0.005" silver, and 0.191" aluminum, providing absorber thicknesses of 950 mg/cm^2, 203 mg/cm^2, and 175 mg/cm^2 respectively (Cipperley, 1959).

Wrist badges were identical to the regular film badges except that the clips were replaced with expandable wrist bands similar to those found on many wrist watches.

Temporary badges were used primarily for visitors. A film packet was placed under a cadmium shield and secured by Scotch tape. The shield had a number perforated in it for identification when the film was exposed to X-rays. The cadmium shield covered only half of the film to show a difference between beta and gamma exposures. Any visitor, or any person using a temporary badge, was required to add his/her full name, "S" number (if an employee), business address and company represented. The badge was held by a clip like regular badges.

Finger rings were used in situations primarily involving radiation exposure to the hands. Originally, an aluminum ring with a silver filter was used, but this was replaced by a plastic ring with a cadmium filter. The rings were made up of three parts. The ring base and the finger band, a small filter, and the ring top. The film was placed in the ring base, covered by the filter, and the ring top was screwed on. Type 552 film was used in all rings (Puphal, 1996).

Film badge and finger ring services were provided to the EBR area on a weekly basis by the Personnel Metering Branch. Personnel exposure records were maintained by EBR staff (Personnel Metering, 1958). During the early years, monitoring summary data contained only the total number of badges issued during a monitoring period (see Figure 6-4). In December 1958, records began to include a tally of how many persons participated in the monitoring program (see Figure 6-5). By January 1967, the total number of badges issued was dropped from the summary reports entirely; only the number of individuals monitored was provided. Also in 1967, the site began transitioning from weekly and monthly film dosimetry exchanges to quarterly exchanges of thermoluminescent dosimeters (TLDs),

resulting in fewer individuals being recorded on each month's badge exchange listing (Annual Report, 1959; Annual Report, 1960; Annual Report, 1961; Annual Report, 1962; Exposure Resumes, 1966; Exposure Resumes, 1970; Exposure Records, 1972; Exposure Records, 1973; Exposure Records, 1974; Personnel Metering Tables, 1962; Personnel Metering Tables, 1963; Personnel Metering Tables, 1964; ZPPR Dosimetry Policy, 1971). Although ZPPR became operational in 1967, badges from that site were included in the EBR-II tally until November 1971, when they began to be reported separately (ZPPR Dosimetry Policy, 1971, PDF p. 8).

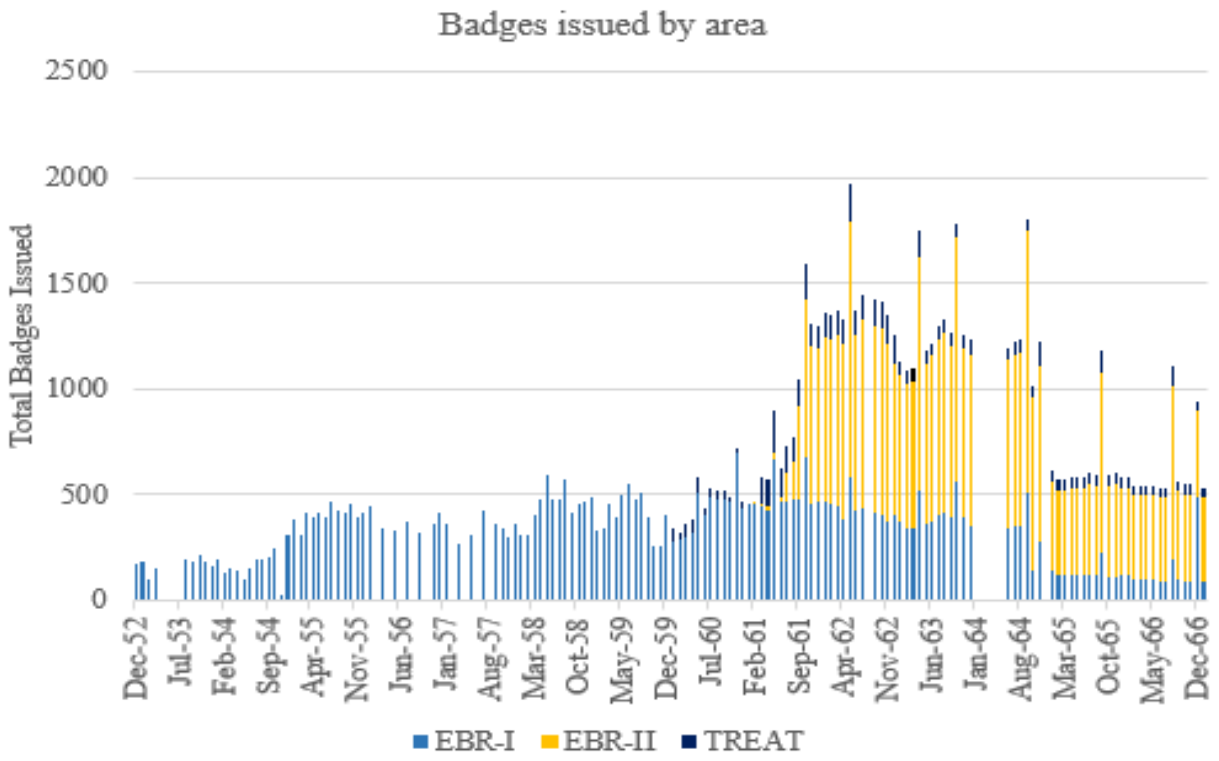


Figure 6-4: ANL-W Badges Issued by Area (1952-1966)

NOTE: Figure 6-4 was developed using Personnel Metering Monthly Reports; NIOSH has not secured access to all of these reports. Gaps in the data display (e.g., early 1964) do not necessarily indicate a lapse in monitoring; rather, such gaps indicate a time period for which NIOSH does not have access to the monthly reports.

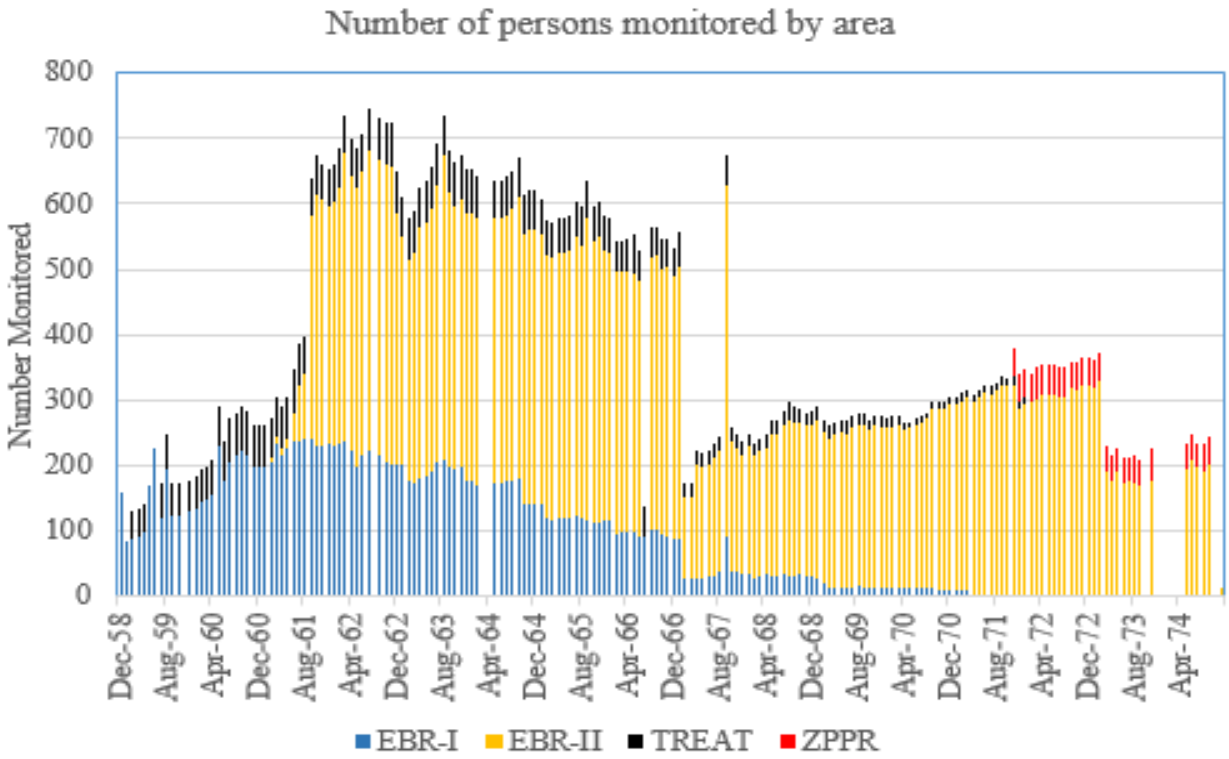


Figure 6-5: Number of Persons Monitored at ANL-W by Area (1958-1974)

NOTE: Figure 6-5 was developed using Personnel Metering Monthly Reports; NIOSH has not secured access to all of these reports. Gaps in the data display (e.g., early 1964) do not necessarily indicate a lapse in monitoring; rather, such gaps indicate a time period for which NIOSH does not have access to the monthly reports.

6.3 Available ANL-W Radiological Monitoring Data

Consideration of personnel exposures to radioactive material commenced before the start of radiological operations at ANL-W. The AEC-IDO took advantage of the practices and experiences of other AEC facilities established earlier, such as Oak Ridge National Laboratory and Hanford, to establish their health physics program. AEC-IDO policy regarding health physics responsibilities and services at INL was established in 1952 creating centralized services (Johnston, 1952), which included ANL-W even though ANL-W was under the AEC-Chicago Operations Office. ANL-W used air sampling and radiological contamination monitoring programs as qualitative indicators of internal exposures, with routine bioassay programs serving as verification programs that internal exposures had not occurred.

The contamination control limits for the detection and control of released activity beyond the control boundaries related to instrumentation capabilities and the basic philosophy of acceptance of detectable contamination. The contamination control limits for on-plant surfaces, and particularly personnel, were always close to the MDA, so that any detectable contamination was a signal for follow-up evaluations and actions.

The monitoring of radioactivity in the air in occupied areas was a fundamental component of the internal exposure control program. Beta/gamma CAMs were used from the beginning of all facility and program operations in routinely-occupied areas. The CAM systems provided real-time air-activity evaluations while fixed air samplers provided retrospective data and an average air concentration in an area or building.

In 1958, an informal study and information program was initiated within the Idaho Division Health Physics section. This resulted from a series of requests from ANL-Idaho staff for data and information on the detection and monitoring of plutonium aerosols under conditions of reactor operations. The material assembled in the course of the program was originally written in the form of a series of information memoranda circulated within the Idaho Division. These memoranda were revised, augmented, and updated to form a single report in 1963 that discusses the monitoring and sampling applications in the vicinity of operating nuclear reactors (Stoddart, 1963). The report describes the technical bases for the plutonium aerosol monitoring program within the ANL-Idaho facilities, as summarized in the following paragraphs.

Continuous monitoring and sampling was provided in all areas where plutonium was present. Minimum coverage was considered to be a continuous "self-monitoring" sampler (either a conventional alpha air monitor or a ratio detection monitor), supplemented by one or more conventional air samplers. Monitoring units were to be inspected daily for proper operation. All monitors were equipped with chart recorders and all charts were kept as permanent records. Filters were removed from monitoring units and routinely counted as a double check on the monitor's detection and recording system (Stoddart, 1963).

The conventional air-sampling units used were continuously-operating devices sampling at relatively low flow rates. Typical units sampled room air at 2 cfm on 2-in.-diameter HV-70 or Millipore filters. Samples were removed daily, Monday through Friday, and counted for alpha and beta-gamma activities (Stoddart, 1963).

In addition to continuous area monitoring, all operations involving the handling, transfer, examination, or inventory of plutonium were monitored for release of plutonium aerosols by moving one of the continuous monitoring devices to the vicinity of the operation. At other times, special-purpose monitors and samplers were used. Minimum coverage was considered to be provided by one continuous monitor supplemented by one or more conventional air samplers or special-purpose monitors (Stoddart, 1963).

The air samplers used for the purposes noted above would have been either low-flow-rate devices (as described above) or high-volume types. The latter were preferentially selected when a specific operation was of short duration because adequate volumes could be sampled in a brief period (Stoddart, 1963).

The low-flow-rate samplers used carbon-vane vacuum pumps and normally operated at flow rates of 1 to 2 cfm. The high-volume or high-flow-rate samplers were high-speed metal-vane or rotary positive-displacement devices. Sampling flow rates were dependent on the physical size of the filter, the type of filter, and the power rating of the unit. Depending on the unit and filter used, flow rates would have been on the order of 8 to 40 cfm (Stoddart, 1963).

One example of a special-purpose monitor is an experimental “sniffer.” This unit, built in the Idaho Division Health Physics Section's instrument shop, consisted of a portable alpha survey meter coupled with an air-sampling device to provide a portable unit capable of measuring alpha particles simultaneously with deposition on the paper filter. Air was drawn through the filter at a rate of 10 to 30 cfm. The unit was provided with a long “snout” to enable it to probe cavities, openings, and inaccessible areas (Stoddart, 1963).

By Idaho Division Health Physics Section policy, sample media from conventional air samplers, continuous monitors, and impaction samplers are routinely counted for both alpha and beta-gamma activities. Discriminating counters were used to count for gross alpha and for gross beta-gamma. Alpha counters were calibrated against plutonium standards. Beta-gamma counters were calibrated against Radium D, E, F (combined) standards. The standards used were secondary standards, prepared and calibrated by the background counting group of Argonne's Industrial Hygiene and Safety Division. All counters in regular use were calibrated daily for yield or geometry and for residual background (Stoddart, 1963).

Long-lived residual alpha activity (i.e., activity remaining after seven-day decay) was assumed to be plutonium unless positive isotopic or chemical identification was made. Operating procedures from the 1960s required isotopic identification on all samples having long-lived residual gross alpha activity greater than 50% of MPC(40) for plutonium after seven or more days of decay time. Identification was made by pulse-height analysis techniques (Stoddart, 1963).

6.3.1 Available ANL-W Radiological Monitoring Data for EBR-I Complex

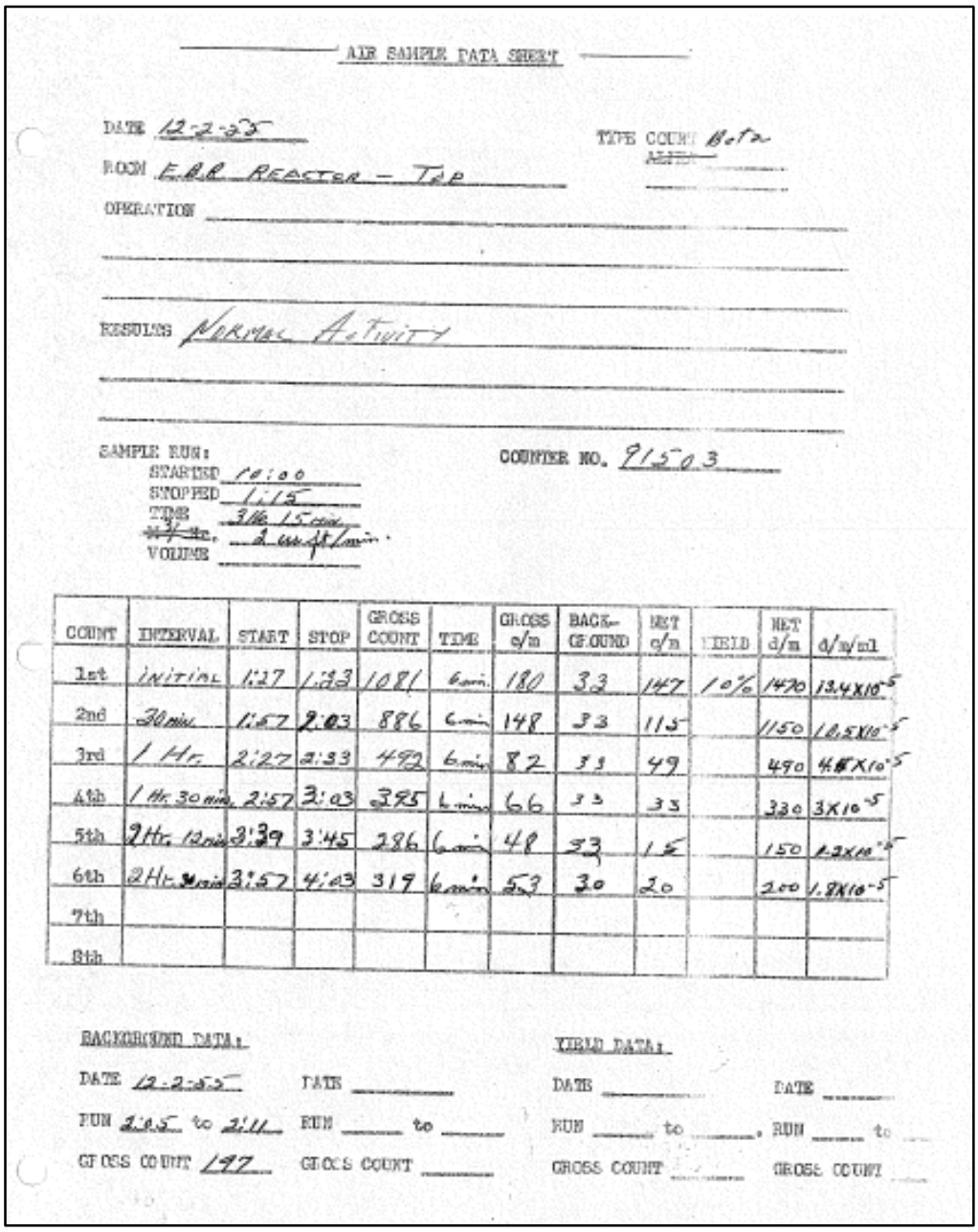
The radiological monitoring for EBR-I, ZPR-III, AFSR, and BORAX I-V was provided by a central Health Physics group located in the EBR-I Complex. The radiological monitoring practices and policies employed would be applicable to all of these facilities.

6.3.1.1 Available Radiological Monitoring Data for EBR-I

ATTRIBUTION: Section 6.3.1.1 was completed by Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding EBR-I were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to EBR-I are explained in the associated text.

Radiological monitoring in the form of routine survey, smear, and air monitoring was performed at EBR-I from the inception of operations. However, NIOSH found limited quantities and types of monitoring performed during the evaluation period. Many of the records found were located in the personal files of the EBR-I Complex Radiation Protection Manager; although the records were incomplete, they included data from important radiological events. Air monitoring data prior to 1955 were not found. The amount of air monitoring data from 1955 into the early 1960s is limited, after which it was determined to be much more complete and comprehensive. Air samples were counted for beta/gamma and alpha contamination. Contamination and radiation surveys prior to 1959 were limited, but surveys after 1960 are available. NIOSH collected a sampling of survey, smear, and air monitoring data for EBR-I with an emphasis on collecting as much alpha monitoring data as possible. Final comprehensive surveys of the EBR-I facility were performed after decontamination and decommissioning operations were completed in 1975 in preparation for turnover of the facility to the National Park Service (Keating, 1975).

Figure 6-6 shows an example of an air sampling data sheet for EBR-I recorded in 1955. Figure 6-7 shows an example of an alpha contamination smear sample data sheet for EBR-I recorded in 1961. These are just examples of the hundreds of air sampling and contamination smear surveys that were found for EBR-I.



AIR SAMPLE DATA SHEET

DATE 12-2-55

TYPE COUNT Beta

ROOM E.B.R. REACTOR - TAP

OPERATION

RESULTS Normal Activity

SAMPLE RUN:
STARTED 10:00
STOPPED 1:15
TIME 3 Hr. 15 min.
~~WT~~ Hr. 2 cu ft/min.
VOLUME

COUNTER NO. 91503

COUNT	INTERVAL	START	STOP	GROSS COUNT	TIME	GROSS c/m	BACK-GROUND	NET c/m	YIELD	NET d/m	d/m/ml
1st	Initial	1:27	1:33	1081	6 min.	180	33	147	10%	1470	13.4×10^{-5}
2nd	30 min.	1:57	2:03	886	6 min.	148	33	115		1150	10.5×10^{-5}
3rd	1 Hr.	2:27	2:33	492	6 min.	82	33	49		490	4.5×10^{-5}
4th	1 Hr. 30 min.	2:57	3:03	385	6 min.	66	33	33		330	3×10^{-5}
5th	2 Hr. 12 min.	3:39	3:45	286	6 min.	48	33	15		150	1.3×10^{-5}
6th	2 Hr. 57 min.	3:57	4:03	319	6 min.	53	30	20		200	1.8×10^{-5}
7th											
8th											

BACKGROUND DATA:

DATE 12-2-55 DATE

RUN 2:05 to 2:11 RUN to

GROSS COUNT 197 GROSS COUNT

YIELD DATA:

DATE DATE

RUN to , RUN to

GROSS COUNT GROSS COUNT

Source: Sample Data Sheets, 1955

Figure 6-6: Example of Air Sampling Data Sheet at EBR-I in 1955

SMEAR SAMPLE DATA

DATE	TIME	UNIT AREA SMEARED	REMARKS: Routine survey EBR-I ∝ counts only on this page. BY counts on page 2. See plot plan for point locations
12-12-61	1415	100 cm²	
BLDG. EBR-I	ROOM Routine survey	SMEAR MEDIA Whatman #2	
SAMPLED BY: PRA		☐ wet ☒ dry	
COUNTED BY: PRA		NOTES:	

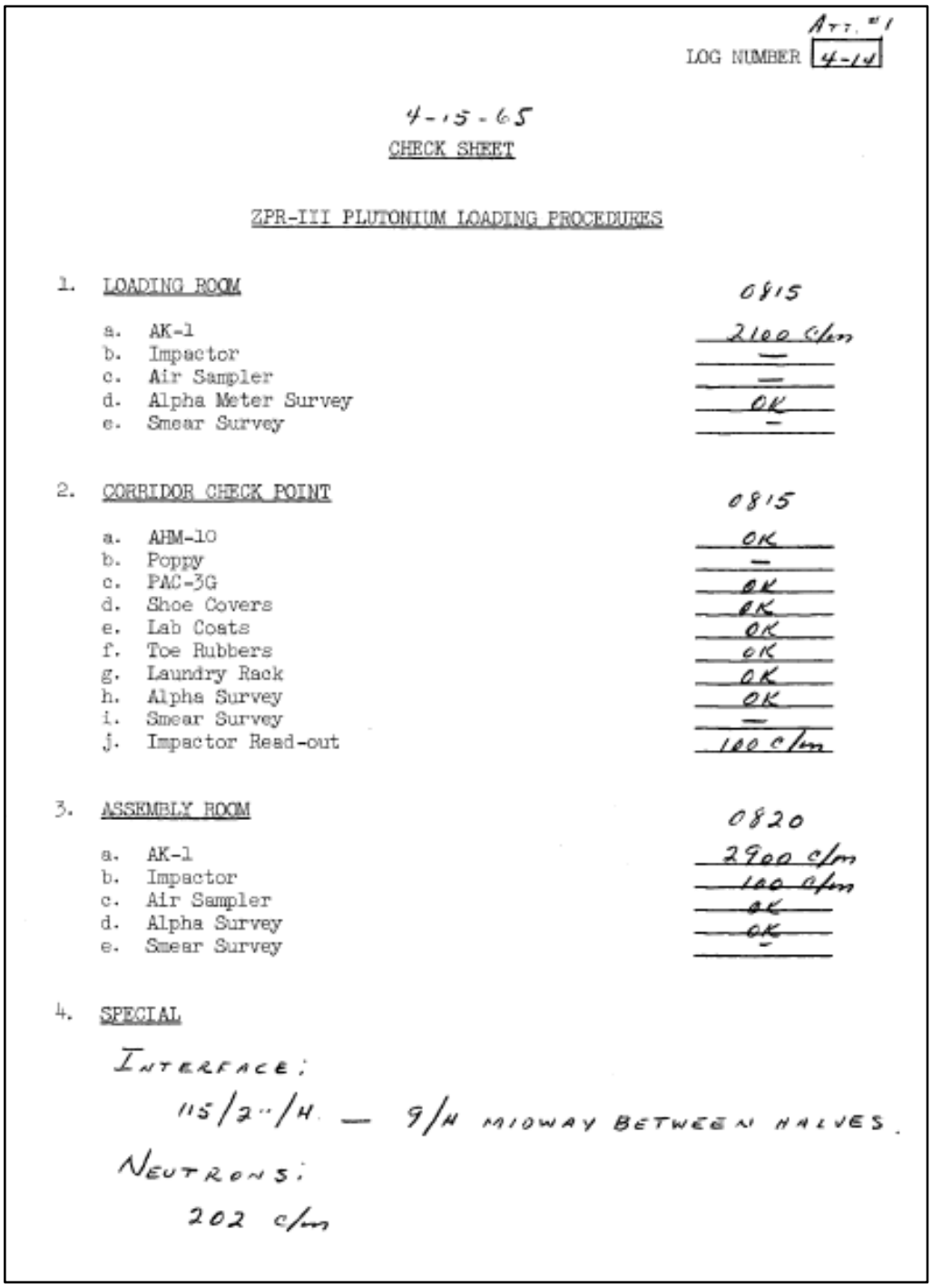
	Location Smeared	Date and Time Counted	Total Count	Count Time in Minutes	Counts per Minute Gross	Background Counts per Minute	Counts per Minute Net	Counter Yield	Disintegrations Per Minute Per Unit are	Unit Area	Type of Activity Counted	Initials
1.	Basement floor center	12-12-61 1430	1	1	1	2	0	.50	0	100 cm²	∝	PRA
2.	Basement Pu area	12-12-61 1432	0	1	0	2	0	.50	0	"	∝	PRA
3.	Basement west corridor	12-12-61 1434	1	1	1	2	0	.50	0	"	∝	PRA
4.	Machine shop floor	12-12-61 1437	1	1	1	2	0	.50	0	"	∝	PRA
5.	Hand rail reactor stairs	12-12-61 1440	0	1	0	2	0	.50	0	"	∝	PRA
6.	Reactor top	12-12-61 1441	0	1	0	2	0	.50	0	"	∝	PRA
7.	Control room table	12-12-61 1443	0	1	0	2	0	.50	0	"	∝	PRA
8.	Table lunch room	12-12-61 1445	0	1	0	2	0	.50	0	"	∝	PRA
9.	Floor 2nd Story corr.	12-12-61 1447	0	1	0	2	0	.50	0	"	∝	PRA
10.	Corridor by drinking fts	12-12-61 1450	0	1	0	2	0	.50	0	"	∝	PRA

Source: Sample Data Sheets, 1961

Figure 6-7: Example of Alpha Contamination Smear Survey at EBR-I in 1961

6.3.1.2 Available Radiological Monitoring Data for ZPR-III

Since the beginning of operations in 1955, radiological monitoring has been performed at ZPR-III. NIOSH was able to find examples of air monitoring from its initial years of operations (Air Sample Data Sheets, 1956; Air Sample Data Sheets, 1958). More comprehensive air monitoring records were found for later years. Radiation and contamination survey programs were employed from inception due to the handling of uranium fuel plates. This program became more rigorous with the introduction of plutonium fuel plates and the recognized potential for widespread uncontrolled contamination if a fuel plate rupture were to occur. The completion of a radiological monitoring "check sheet" was required each time plutonium fuel plates were handled. Figure 6-8 shows an example check sheet. One of the additional monitoring prerequisites was direct determination of surface alpha activity on all plutonium plates at routine intervals as well as before and after the handling of each plate. This procedure was called "piece counting." Figure 6-9 shows an example piece-counting sheet. NIOSH was able to collect many examples of "check sheets" and the data associated with the checklists.



Att. #1
LOG NUMBER 4-14

4-15-65

CHECK SHEET

ZPR-III PLUTONIUM LOADING PROCEDURES

1. LOADING ROOM — 0815

a.	AK-1	2100 c/m
b.	Impactor	—
c.	Air Sampler	—
d.	Alpha Meter Survey	OK
e.	Smear Survey	—

2. CORRIDOR CHECK POINT — 0815

a.	AHM-10	OK
b.	Poppy	—
c.	PAC-3G	OK
d.	Shoe Covers	OK
e.	Lab Coats	OK
f.	Toe Rubbers	OK
g.	Laundry Rack	OK
h.	Alpha Survey	OK
i.	Smear Survey	—
j.	Impactor Read-out	100 c/m

3. ASSEMBLY ROOM — 0820

a.	AK-1	2900 c/m
b.	Impactor	100 c/m
c.	Air Sampler	OK
d.	Alpha Survey	OK
e.	Smear Survey	—

4. SPECIAL

INTERFACE:

115/2"/H — 9/H MIDWAY BETWEEN HALVES.

NEUTRONS:

202 c/m

Source: Survey Reports, Jan1965

Figure 6-8: Example of Check Sheet Used for Plutonium Loading

July 1, 1963
Bkgd. 40 c/m Eff 36%

Size	Pot No.	Piece No.	Count Front	Count Back	Measured Center	Measured Edge	Comments	X-rayed
2x2x1/8	162	281284	4		108	120	Yes	
"	"	281285	10		108	120	"	
"	"	281286	4		107	118	"	
"	"	281287	50		108	119	No	Bulged front and rear
"	"	281288	4		108	119	Yes	
"	"	281289	22		108	119	"	
"	"	281290	32		108	118	"	
"	"	281293	18		107	120	"	
"	"	281294	12		108	119	"	
"	"	281295	6		108	119	"	
"	"	281329	14		107	118	"	
"	"	281331	44		107	119	"	
2x2x1/8	163	281296	24		108	120	Yes	
"	"	281297	76		107	120	"	
"	"	281298	26		108	120	"	
"	"	281299	14		108	119	"	
"	"	281300	16		107	119	"	
"	"	281301	10		108	119	"	
"	"	281302	34		108	118	"	
"	"	281303	18		108	119	"	
"	"	281304	32		107	118	"	
"	"	281305	0		108	119	"	
"	"	281306	26		107	119	"	
"	"	281307	14		108	120	"	

Source: Survey Reports, Jul1963

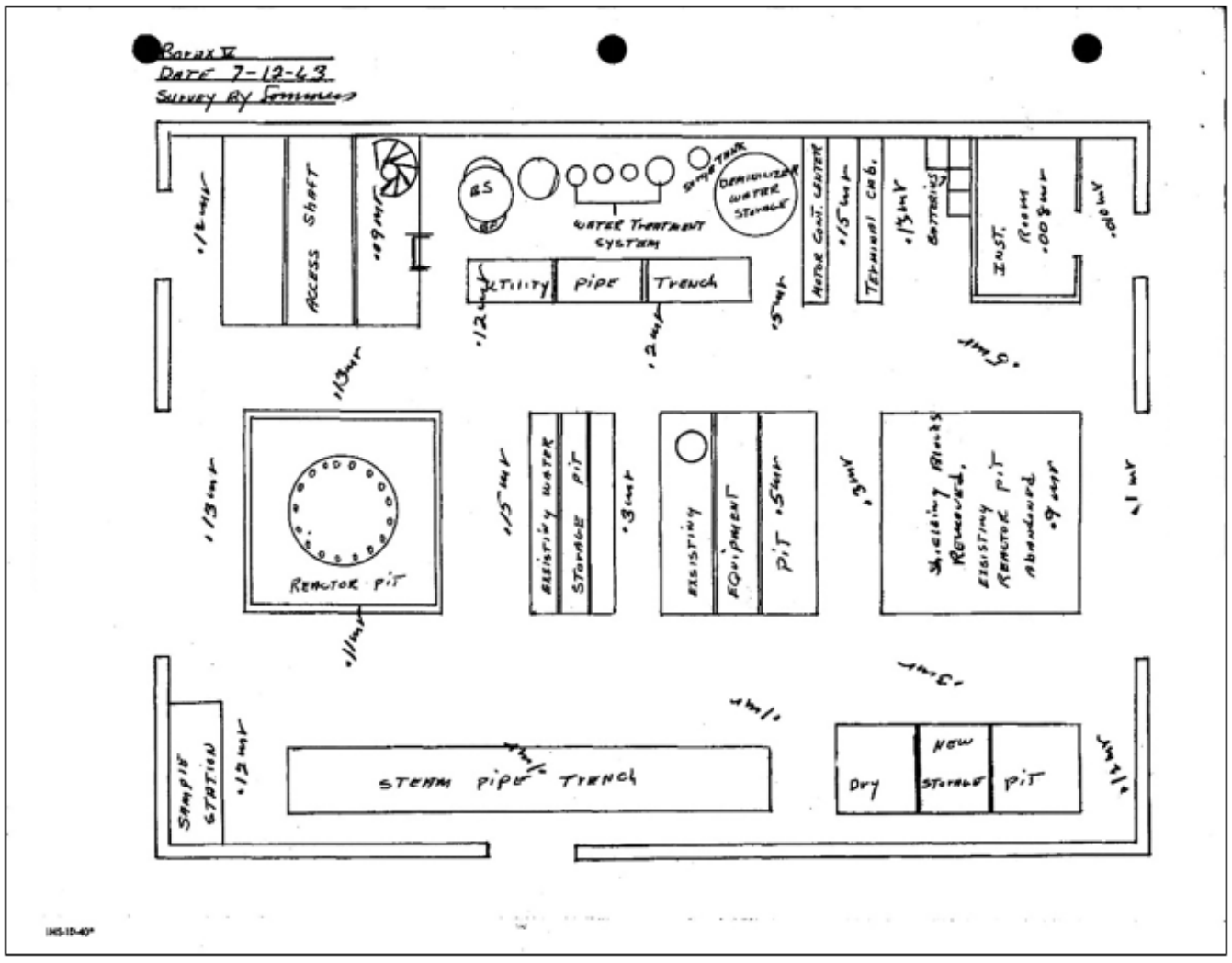
Figure 6-9: Example of "Piece Counting" of Plutonium Fuel Plates

6.3.1.3 Available Radiological Monitoring Data for BORAX

ATTRIBUTION: Section 6.3.1.3 was completed by Mike Mahathy, Oak Ridge Associated Universities. All conclusions drawn from the data regarding BORAX were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the BORAX are explained in the associated text.

From the onset of radiological operations through the end of the period under evaluation, health physics groups affiliated with ANL-W conducted air sampling and surveys for dose rate and contamination at the BORAX reactors. However, the degree and amount of radiological monitoring

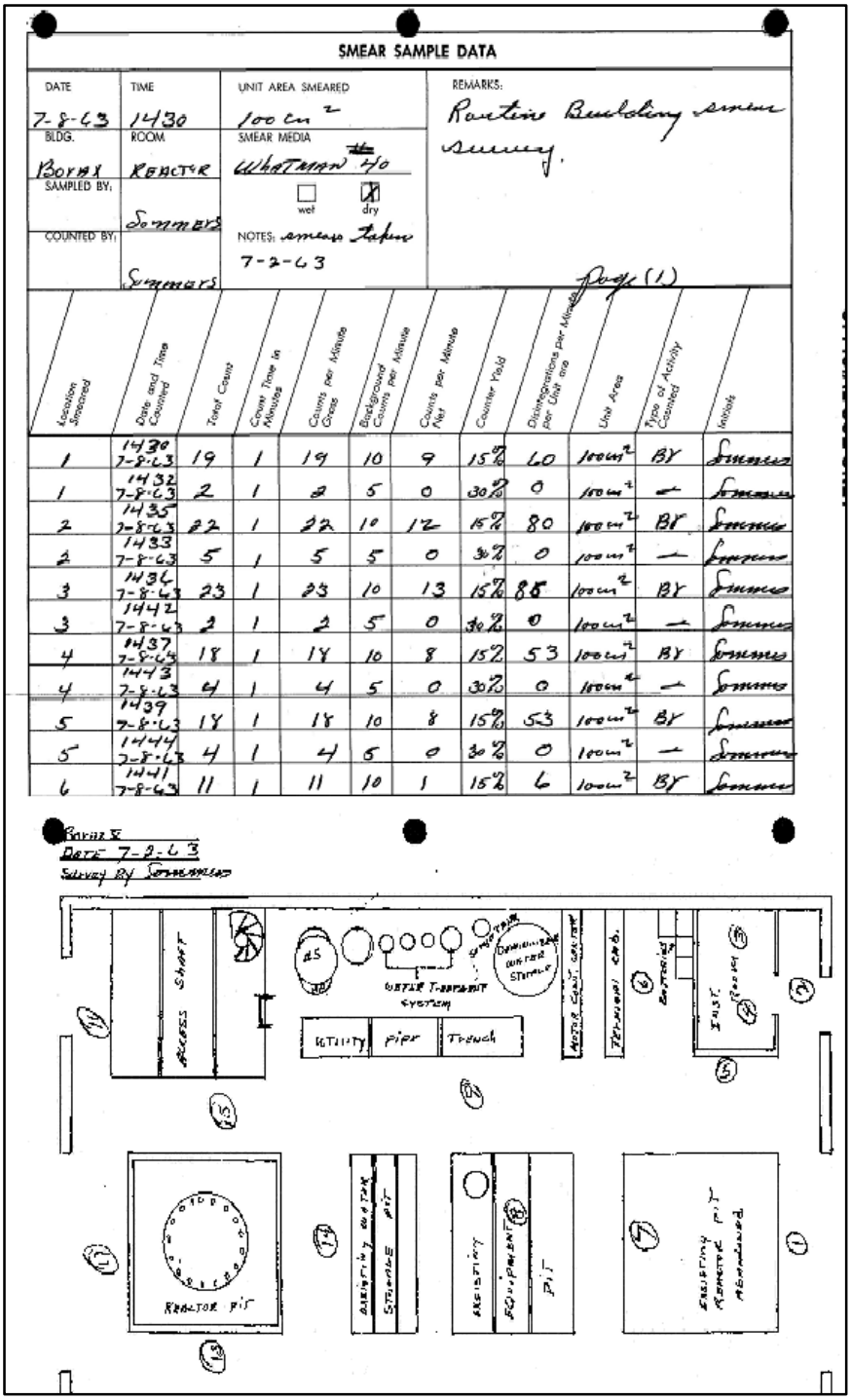
data available to NIOSH is very limited for BORAX-I through Borax-IV. NIOSH has collected a sampling of routine survey, smear, and air monitoring data for BORAX-V through the end of reactor operation and dismantling. Radiation surveys included measurements of beta/gamma and neutron dose rates. Smears were collected for beta/gamma and alpha contamination while air monitoring filters were counted for fission products and alpha releases. Figure 6-10 shows an example of a radiation survey (Survey Reports, Jun-Aug1963) from the hundreds of pages of radiation surveys available to NIOSH for BORAX-V.



Source Survey Reports, Jun-Aug1963

Figure 6-10: Example of Radiation Survey of BORAX-V Reactor Main Floor

Figure 6-11 shows a portion of another example routine survey of the reactor main floor with results and mapped locations (Survey Reports, Jun-Aug1963).



SMEAR SAMPLE DATA

DATE: 7-8-63 | TIME: 1430 | UNIT AREA SMEARED: 100 cm² | REMARKS: Routine Building smear survey.
BLDG.: BORAX | ROOM: REACTOR | SMEAR MEDIA: WHATMAN #40 | ☐ wet ☒ dry
SAMPLED BY: Summers
COUNTED BY: Summers
NOTES: smears taken 7-2-63

Page (1)

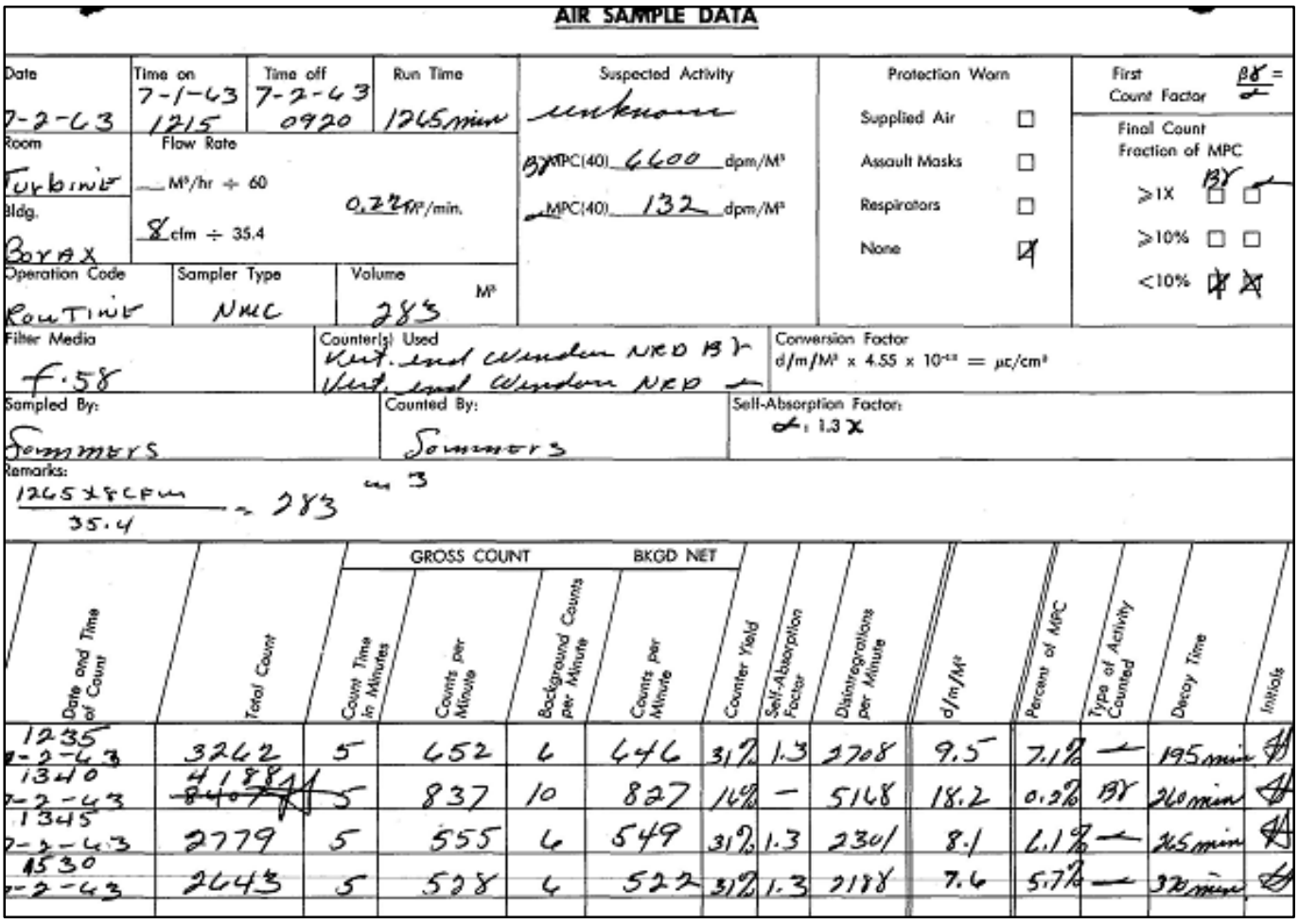
Location Smeared	Date and Time Counted	Total Count	Count Time in Minutes	Counts per Minute Gross	Background Counts per Minute	Counts per Minute Net	Counter Yield	Disintegrations per Minute per Unit area	Unit Area	Type of Activity Counted	Initials
1	1430 7-8-63	19	1	19	10	9	15%	60	100cm²	BY	Summers
1	1432 7-8-63	2	1	2	5	0	30%	0	100cm²	—	Summers
2	1435 7-8-63	22	1	22	10	12	15%	80	100cm²	BY	Summers
2	1433 7-8-63	5	1	5	5	0	30%	0	100cm²	—	Summers
3	1436 7-8-63	23	1	23	10	13	15%	85	100cm²	BY	Summers
3	1442 7-8-63	2	1	2	5	0	30%	0	100cm²	—	Summers
4	1437 7-8-63	18	1	14	10	8	15%	53	100cm²	BY	Summers
4	1443 7-8-63	4	1	4	5	0	30%	0	100cm²	—	Summers
5	1439 7-8-63	14	1	18	10	8	15%	53	100cm²	BY	Summers
5	1444 7-8-63	4	1	4	5	0	30%	0	100cm²	—	Summers
6	1441 7-8-63	11	1	11	10	1	15%	6	100cm²	BY	Summers

BORAX
DATE 7-2-63
Survey By Summers

Source: Survey Reports, Jun-Aug1963

Figure 6-11: Example of Smear Survey of BORAX Reactor Main Floor

Figure 6-12 shows an example of air monitoring results for the BORAX-V turbine building (Survey Reports, Jun-Aug1963) from routine air monitoring data available for the reactor over its operational life. Note that the filter was recounted for alpha contamination to account for short-lived alpha products.



AIR SAMPLE DATA

Date	Time on	Time off	Run Time	Suspected Activity	Protection Worn	First Count Factor
7-2-63	7-1-63 1215	7-2-63 0920	1265 min	unknown	Supplied Air ☐	BY =
Room: Turbine	Flow Rate: ___ M³/hr ÷ 60			BY MPC(40) 6600 dpm/M³	Assault Masks ☐	Final Count Fraction of MPC
Bldg.: Borax	8 cfm ÷ 35.4	0.22 M³/min.		MPC(40) 132 dpm/M³	Respirators ☐	≥1X BY ☐ ☐
Operation Code: Routine	Sampler Type: NMC	Volume: 283 M³			None ☑	≥10% ☐ ☐
						<10% ☑ ☒

Filter Media: F.58 | Counter(s) Used: Vert. end Window NRD BY; Vert. end Window NRD | Conversion Factor: d/m/M³ x 4.55 x 10^{-13} = μc/cm³

Sampled By: Sommers | Counted By: Sommers | Self-Absorption Factor: α: 1.3 X

Remarks: $\frac{1265 \times 8 \text{ CFM}}{35.4} = 283$ m³

Date and Time of Count	Total Count	GROSS COUNT: Count Time in Minutes	GROSS COUNT: Counts per Minute	BKGD NET: Background Counts per Minute	BKGD NET: Counts per Minute	Counter Yield	Self-Absorption Factor	Disintegrations per Minute	d/m/M³	Percent of MPC	Type of Activity Counted	Decay Time	Initials
1235 7-2-63	3262	5	652	6	646	31%	1.3	2708	9.5	7.1%	—	195 min	
1340 7-2-63	~~4188~~ 840	5	837	10	827	14%	—	5168	18.2	0.2%	BY	260 min	
1345 7-2-63	2779	5	555	6	549	31%	1.3	2301	8.1	1.1%	—	265 min	
1530 7-2-63	2643	5	528	6	522	31%	1.3	2188	7.6	5.7%	—	320 min	

Source: Survey Reports, Jun-Aug1963

Figure 6-12: Example of Air Monitoring Results for BORAX-V Turbine Building

The report in Figure 6-12 is indicative of all air monitoring reports for the EBR-1 Complex. Reports listed the location of each sample, total sampling time, air sampler flow rate, and other data needed to convert counts per minute to dpm/cm^3, date and times of the initial counts, and results on the initial counts. Each report also indicates if sampling was routine or performed to monitor for suspected airborne activity from loss of containment. In general, air sample data reports would indicate the suspected alpha radionuclide, although for BORAX-V it was always uranium. Samples were recounted when results were near or greater than ten percent maximum permissible concentration (MPC) to account for decay of radon. The alpha sample from Figure 6-12 was counted three times.

6.3.1.4 Available Radiological Monitoring Data for AFSR

ATTRIBUTION: Section 6.3.1.4 was completed by Mike Mahathy, Oak Ridge Associated Universities. All conclusions drawn from the data regarding the AFSR were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the AFSR are explained in the associated text.

ANL-W Health Physics conducted surveys for dose rate and contamination at AFSR from the onset of radiological operations in 1959 (at the EBR-I Complex) and after AFSR's move to the EBR-II Complex in 1970. NIOSH has collected a representative routine survey and smear for AFSR as well as a vent fission product analysis performed in 1959. The early air monitoring demonstrated air activity involved short-lived radionuclides at well below ten percent MPC (Air Sample Data Sheet, 1959). ANL-W Health Physics installed a continuous air monitor in AFSR in October 1959 (Air Sample Data Sheets, 1965, PDF p. 173). Its continued use is documented in September 1960 (Safety Monthly Report, 1960, PDF p. 73) and March 1961 with an average 36 dpm/m^3 beta/gamma reported (Safety Monthly Report, 1961, PDF p. 217). However, NIOSH has not identified routine air monitoring data beyond 1959.

Smears were collected to assess possible beta/gamma and alpha contamination. Figure 6-13 shows an example of a radiation survey that reports smears by AFSR location (Survey Reports, Jan-Feb1963). Records of additional surveys are available. Radiation surveys included measurements of beta/gamma and neutron dose rates.

SMEAR SAMPLE DATA *Attachment #2* 2-6-L

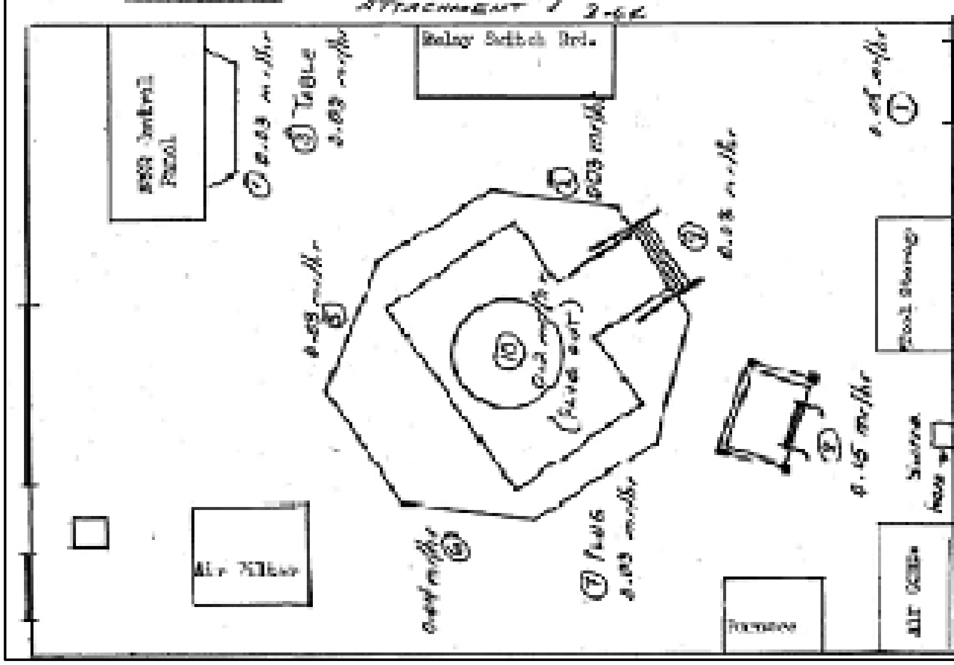
DATE	TIME	UNIT AREA SMEARED	REMARKS
2-26-63	1900	100 cm^2	Smears on AFSR Building Survey.
BLDG. AFSR	ROOM Reactor	SMEAR MEDIA Whatman #41 ☐ wet ☒ dry	
SAMPLED BY:	[signature]		
COUNTED BY:	[signature]	NOTES:	

# 1	2-26-63 1510	98	2	49	49	—	.15	—	100 cm^2	βγ	[illegible]
2	1512	94	2	47	49	—	.15	—	100 cm^2	βγ	[illegible]
3	1516	89	2	44.5	49	—	.15	—	100 cm^2	βγ	[illegible]
4	1518	97	2	48.5	49	—	.15	—	100 cm^2	βγ	[illegible]
5	1521	107	2	53.5	49	4.5	.15	30	100 cm^2	βγ	[illegible]
6	1523	87	2	43.5	49	—	.15	—	100 cm^2	βγ	[illegible]
7	1526	97	2	48.5	49	—	.15	—	100 cm^2	βγ	[illegible]
8	1529	92	2	46	49	—	.15	—	100 cm^2	βγ	[illegible]
9	1532	102	2	51	49	2.0	.15	13	100 cm^2	βγ	[illegible]
10	1534	86	2	43	49	—	.15	—	100 cm^2	βγ	[illegible]

AFSR-REACTOR SURVEY SHEET

BY [signature] DATE 2-26-63

Attachment #1 2-6-L



Source: Survey Reports, Jan-Feb1963

Figure 6-13: Example of AFSR Radiation Survey with Smears

The potential for neutron exposure was the primary concern of Health Physics at AFSR. ANL-W Health Physics monitored irradiations and reactor power tests for thermal and fast neutrons. Neutrons were measured by instruments that included the Rudolph Meter (a proton-recoil proportional detector for fast neutrons in the 0.2 to 10 MeV energy range) and the Fairport detector (a proton-recoil proportional detector with energy range 0.1 to 15 Mev). Neutron flux was reported with some neutron surveys (Air Sample Data Sheet, 1963, PDF p. 2; Neutron Survey Sheet, 1966, PDF p. 68). Figure 6-14 shows an example of a neutron survey report. Additional reports are available.

NEUTRON SURVEY SHEET

DATE: 6-24-64

AREA: AFSR

SURVEYOR: SEVERN

ATT. #1, LOG 6-31

LOCATION	NEUTRONS				GAMMA		TOTAL DOSE All Radiations	NOTES
	THERMAL		FAST					
	$n_{th}/cm^2/sec$	mrem/hr	~~n/cm²/sec~~ mrep/hr	mrem/hr	Gamma	mrem/hr	mrem/hr	
RADIATION ZONE BOUNDARY	2000	~7.5	1.2	12	15	15	34.5	NO PERSONNEL IN THIS AREA
WORK AREA	250	~1.0	0.5	5	5	5	11.0	PERSONNEL WORKED IN THIS AREA ~ 10 Min.

REMARKS: REACTOR POWER 1000 WATTS — ALL READINGS TAKEN IN THE VICINITY OF GRAPHITE HOLE.

Source: Neutron Survey Sheet, 1966, PDF p. 68

Figure 6-14: Example of ASFR Neutron Survey

6.3.2 Available ANL-W Radiological Monitoring Data for EBR-II Complex

ATTRIBUTION: Section 6.3.2 was completed by Jason Davis, Oak Ridge Associated Universities, and Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding the EBR-II Complex were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the EBR-II Complex are explained in the associated text.

The radiological monitoring at the EBR-II Complex was provided by a central Health Physics Group so that the radiological monitoring practices and policies would be consistently applied to all of the facilities. The Health Physics Group had satellite offices in several of the major facilities in the EBR-II Complex. The satellite offices typically served more than one facility and health physics technicians often rotated through the various satellite offices. As indicated by a former Radiological Controls manager, this rotation was intentionally scheduled to allow technicians to develop a familiarity with each of the ANL-W facilities (Personal Communication, 2015q).

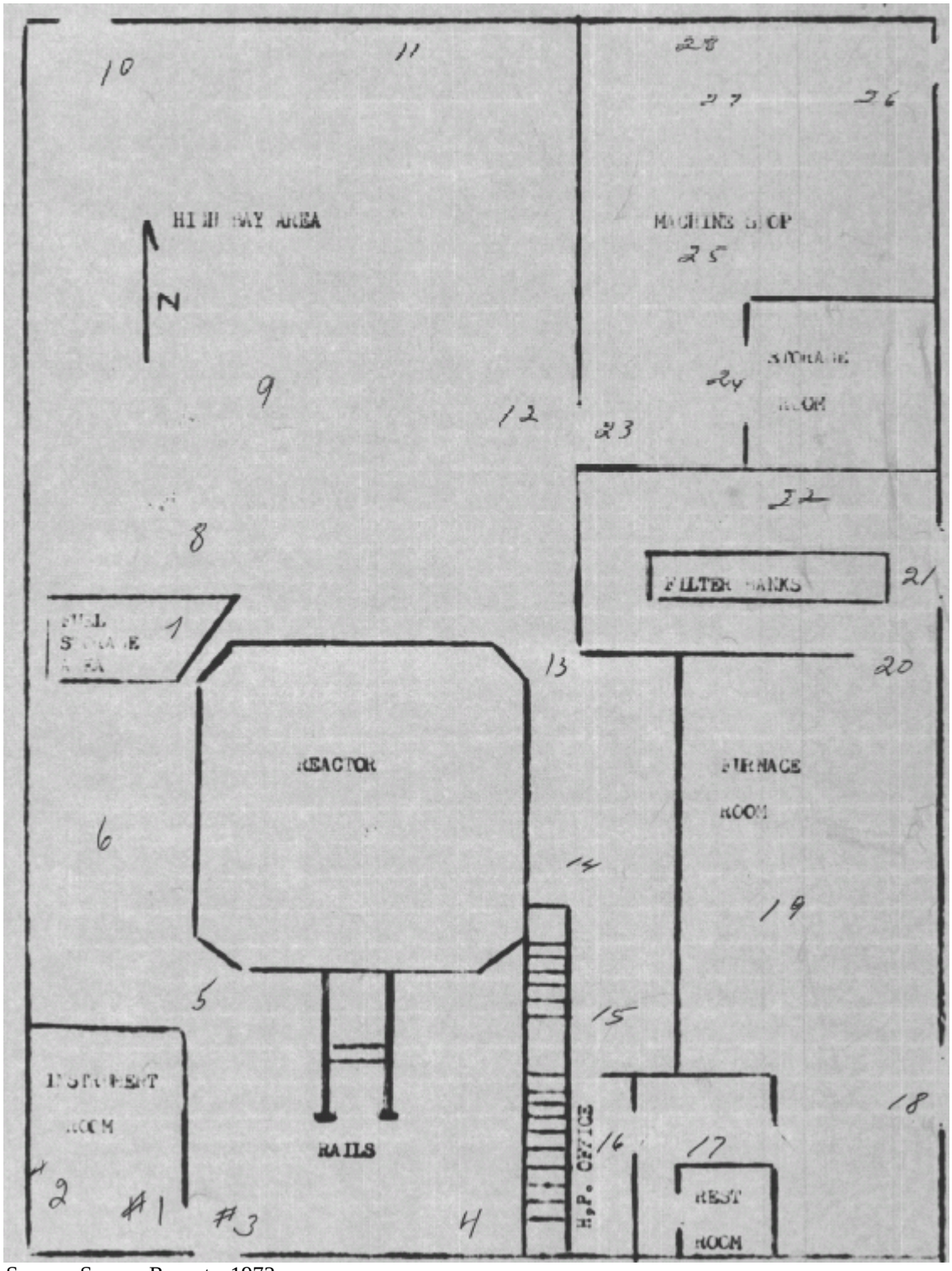
From the onset of radiological operations through the end of the period under evaluation, ANL-W Health Physics performed routine and non-routine dose rate surveys, contamination surveys, and air monitoring at the radiological facilities within the EBR-II Complex. NIOSH has collected numerous samples of the available dose rate survey, contamination survey, and air monitoring data for the EBR-II Complex facilities for each year through the end of the period under evaluation. In areas where alpha contamination levels were potentially significant relative to the beta-gamma contamination levels, contamination smear samples were typically counted for both gross beta/gamma and gross alpha radioactivity. Contamination smear samples collected from areas where alpha contamination levels were not significant relative to the beta-gamma contamination levels were typically only counted for gross beta/gamma radioactivity. Air monitoring filters were counted for gross beta/gamma and gross alpha radioactivity, and sometimes received a second or third decay count for gross alpha radioactivity.

6.3.2.1 Available Radiological Monitoring Data for TREAT

Airborne activity in the TREAT Reactor Building was monitored with a cart-mounted continuous-sampling unit with an integral recorder and an alarm that automatically reset. The detector’s range was 50-50,000 counts/min. A remote alarm was installed in the Control Building.

A criticality monitor was located at the north end of the reactor building. Normally, the monitor's green light would be on, but a radiation level of approximately l R would activate the alarm. This would cause the red light and the horn to oscillate on and off until the radiation level fell below the trip point. Malfunction of the monitor would also cause the alarm light and horn to be activated. Two Nuclear Accident Dosimeters (NAD) were also located in the reactor building.

Figure 6-15 shows an example of a radiation survey with smears performed at the TREAT reactor room. Figure 6-16 shows a TREAT radiation survey report that corresponds to the survey displayed in Figure 6-15.



Source: Survey Reports, 1972

Figure 6-15: Example of TREAT Reactor Room Radiation Survey with Smears

RADIATION SURVEY REPORT

4-15
LOG NUMBER

LOCATION: TREAT RA - Main Floor
DATE: 4-24-72
TIME: 1300
REQUESTOR: Routine
SURVEYOR: D. Nield
REQUEST: Routine Survey RA Main Floor

RESULTS

Max Smear < 100 d/m βγ

No α Detected

General work area 0.5 mR/hr

See attch. for Smear area

INSTRUMENTS USED:

- [ ] SMEAR
- [ ] JUNO
- [ ] α METER
- [ ] GM
- [ ] OTHER
- [ ] SCINTILATOR

PROTECTIVE CLOTHING WORN:

- [ ] FULL ANTI-C
- [ ] GLOVES
- [ ] LAB COAT
- [ ] SHOE COVERS
- [ ] EYE SHIELD
- [ ] OTHER

AIR SAMPLE TAKEN: YES [ ] NO [ ]

RESULTS:

WAS RESPIRATORY EQUIMENT USED? YES [ ] NO [ ]

TYPE: MASK — ACME [ ], SCOTT [ ], COMFO [ ]; CANISTER — 502 R [ ], ALL PURPOSE [ ]

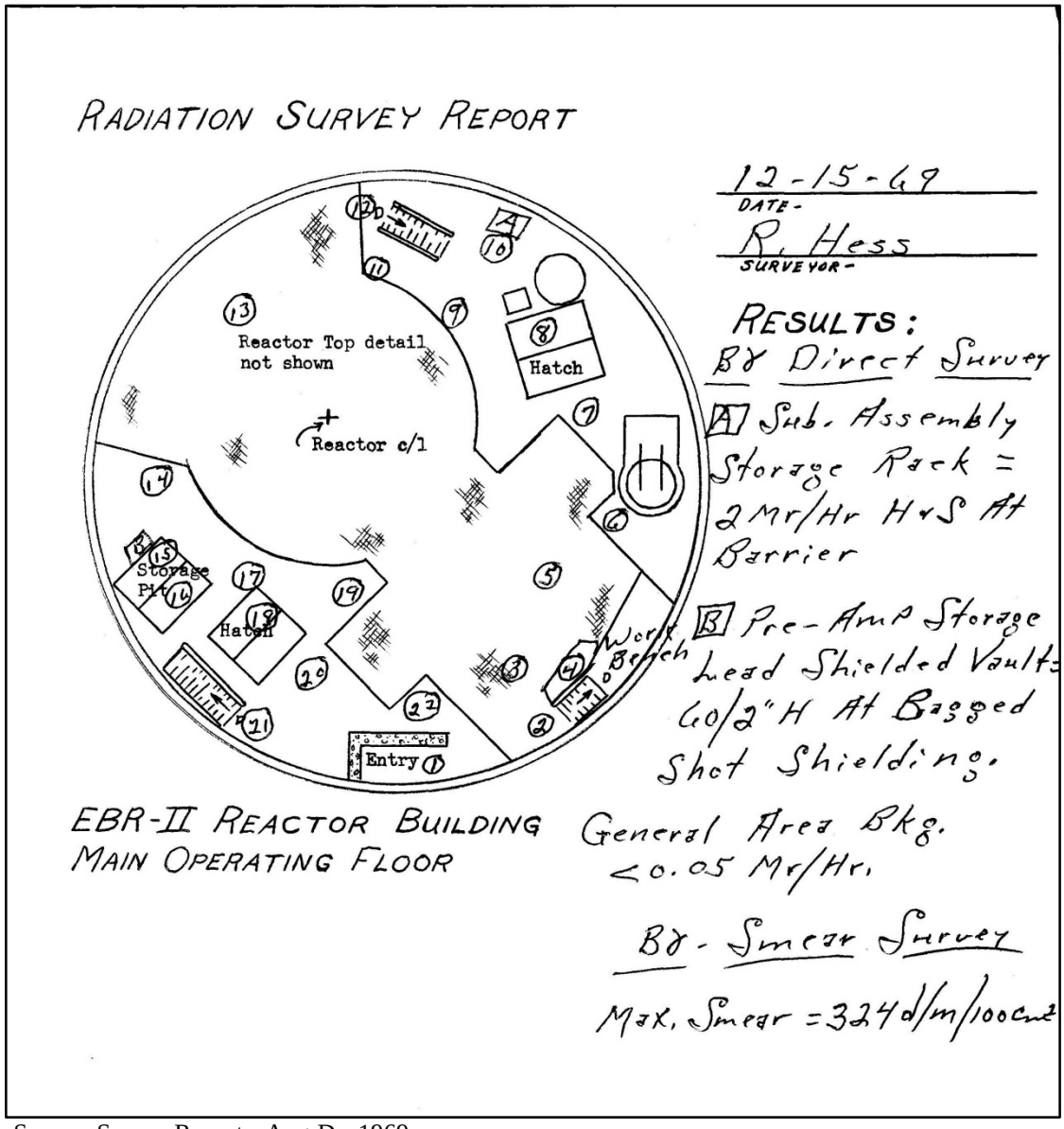
INS-10-10
10-69

Source: Survey Reports, 1972

Figure 6-16: TREAT Radiation Survey Report Corresponding to Figure 6-15

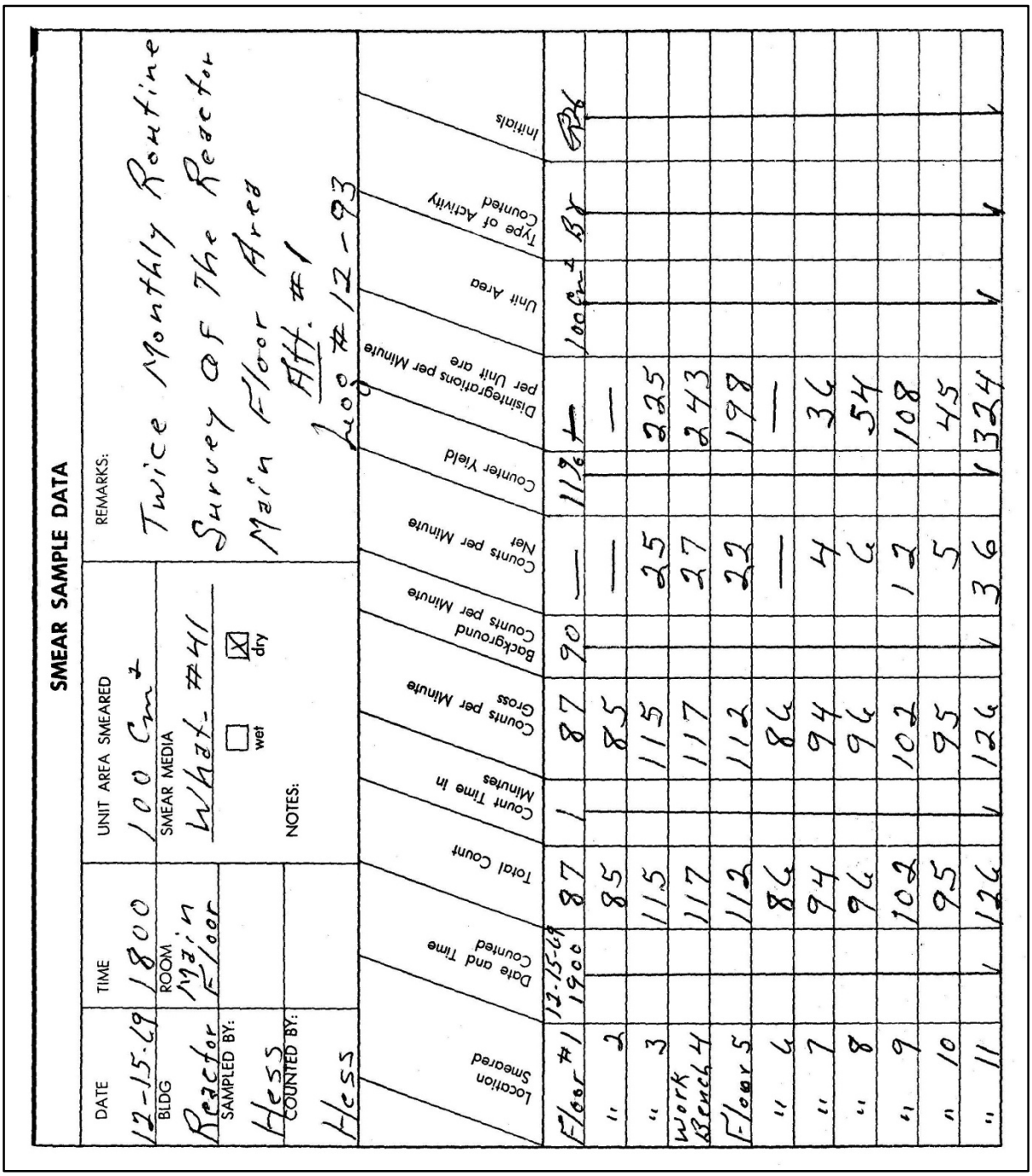
6.3.2.2 Available Radiological Monitoring Data for EBR-II Reactor

Figure 6-17 shows an example of a survey report for the EBR-II Reactor recorded in 1969. Figure 6-18 shows an example of a contamination smear sample data sheet for the EBR-II Reactor recorded in 1969. Figure 6-19 shows an example of an air sample data sheet for the EBR-II Reactor recorded in 1969.



Source: Survey Reports, Aug-Dec1969

Figure 6-17: Example of EBR-II Reactor Survey Report from 1969



SMEAR SAMPLE DATA

DATE: 12-15-69 | TIME: 1800 | UNIT AREA SMEARED: 100 Cm²

BLDG: Reactor | ROOM: Main Floor | SMEAR MEDIA: Whatman #41

SAMPLED BY: Hess | ☐ wet ☒ dry

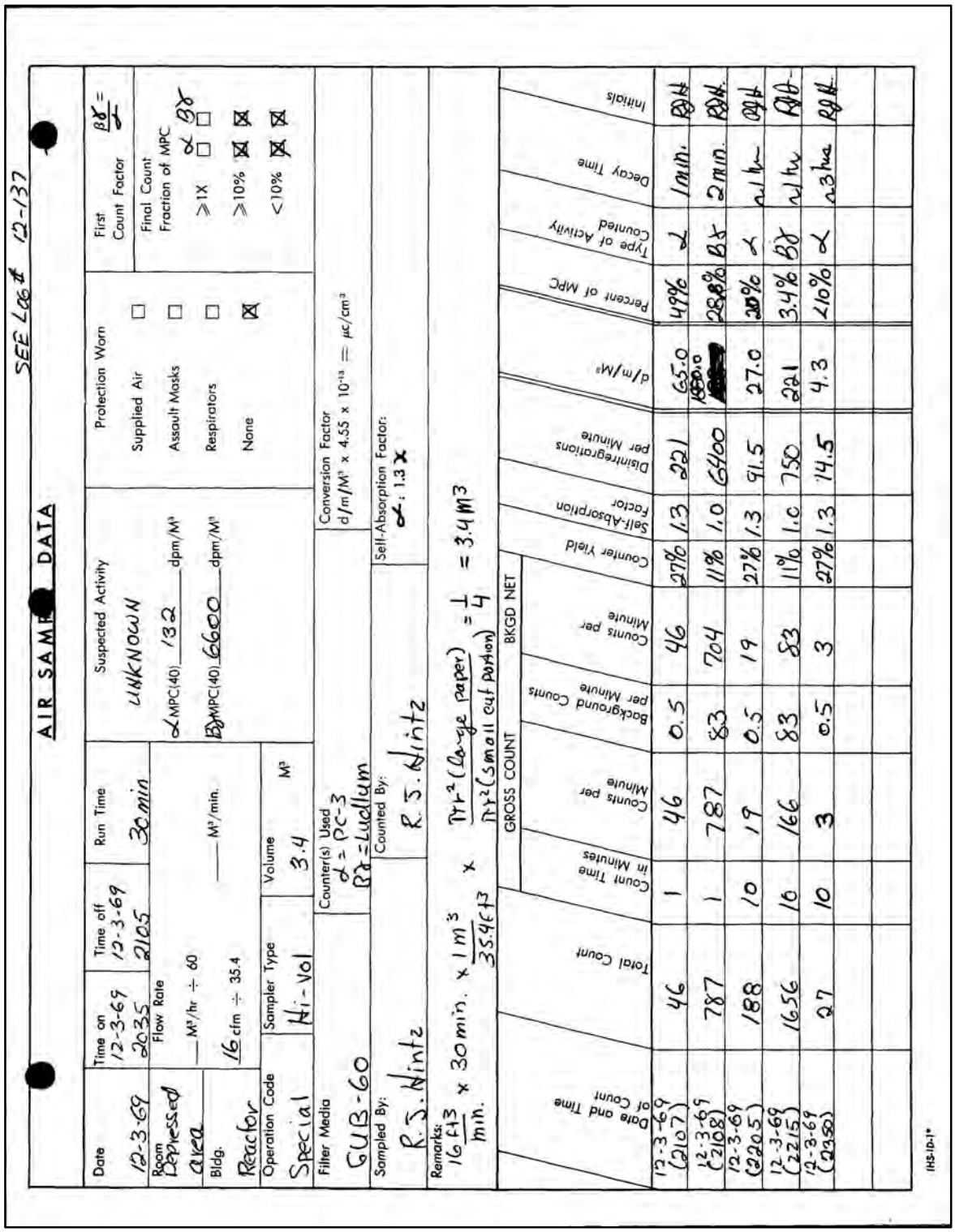
COUNTED BY: Hess | NOTES:

REMARKS: Twice Monthly Routine Survey Of The Reactor Main Floor Area Aft. #1 100 #12-93

Location Smeared	Date and Time Counted	Total Count	Count Time In Minutes	Counts per Minute Gross	Background Counts per Minute	Counts per Minute Net	Counter Yield	Disintegrations per Minute per Unit area	Unit Area	Type of Activity Counted	Initials
Floor #1	12-15-69 1900	87	1	87	90	—	11%	—	100 Cm²	BK	
" 2		85		85		—		—			
" 3		115		115		25		225			
Work Bench 4		117		117		27		243			
Floor 5		112		112		22		198			
" 6		86		86		—		—			
" 7		94		94		4		36			
" 8		96		96		6		54			
" 9		102		102		12		108			
" 10		95		95		5		45			
" 11		126	1	126	1	36	1	1324	1	1	1

Source: Survey Reports, Aug-Dec1969

Figure 6-18: Example of EBR-II Reactor Smear Sample Data Sheet from 1969

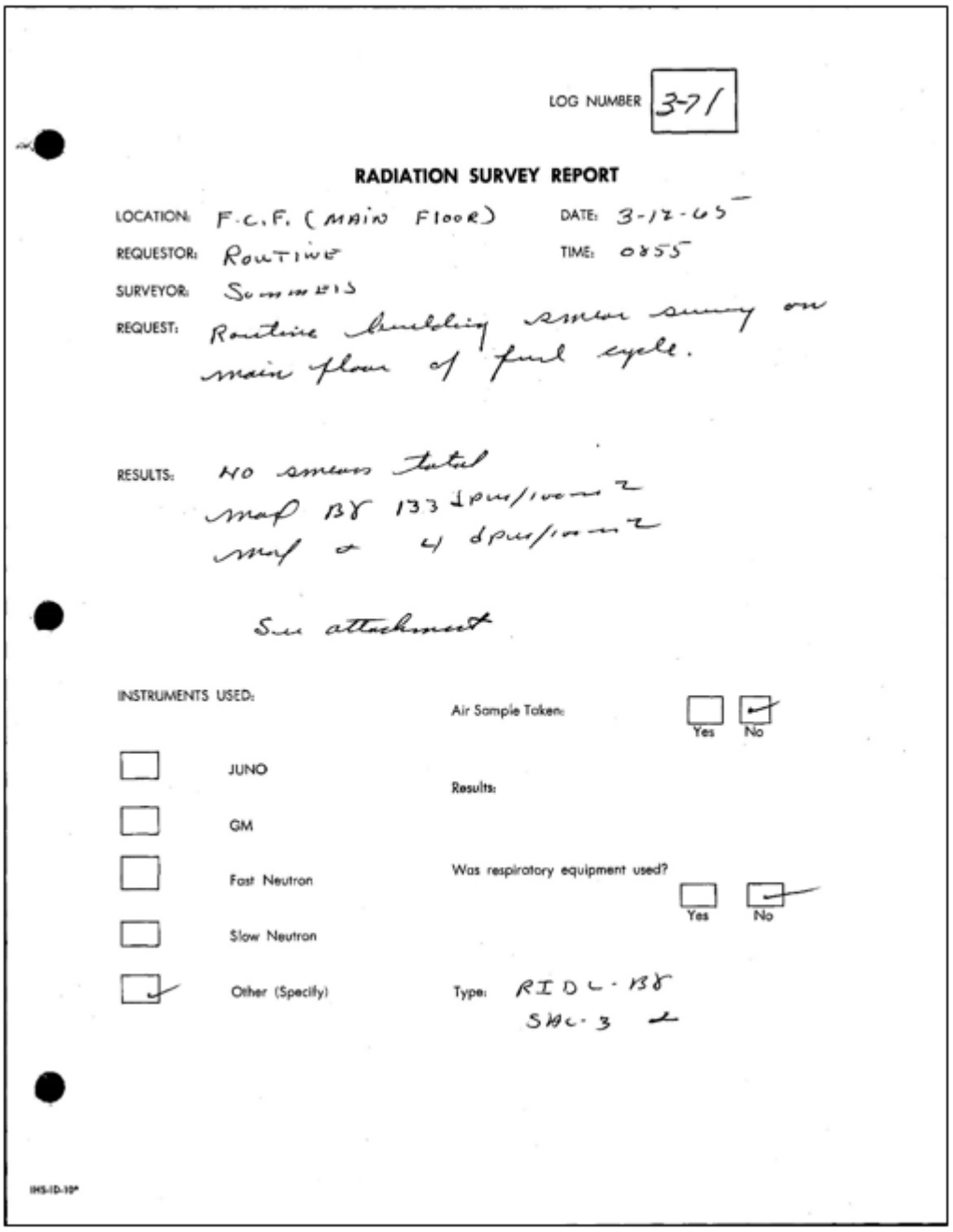


Source: Survey Reports, Aug-Dec1969

Figure 6-19: Example of EBR-II Reactor Air Sample Data Sheet from 1969

6.3.2.3 Available Radiological Monitoring Data for FCF

Figure 6-20 shows an example of a survey report for the FCF main floor recorded in 1965. Figure 6-21 shows an example of an air sample data sheet for the FCF recorded in 1965. Figure 6-22 shows an example of a contamination smear sample data sheet for the FCF recorded in 1965.



LOG NUMBER 371

RADIATION SURVEY REPORT

LOCATION: F.C.F. (MAIN FLOOR) DATE: 3-12-65

REQUESTOR: ROUTINE TIME: 0855

SURVEYOR: Sommers

REQUEST: Routine shielding smear survey on main floor of fuel cycle.

RESULTS: NO smears total
map BY 133 dpm/100 cm²
map A 4 dpm/100 cm²

See attachment

INSTRUMENTS USED:

- ☐ JUNO
- ☐ GM
- ☐ Fast Neutron
- ☐ Slow Neutron
- ☑ Other (Specify)

Air Sample Taken: ☐ Yes ☑ No

Results:

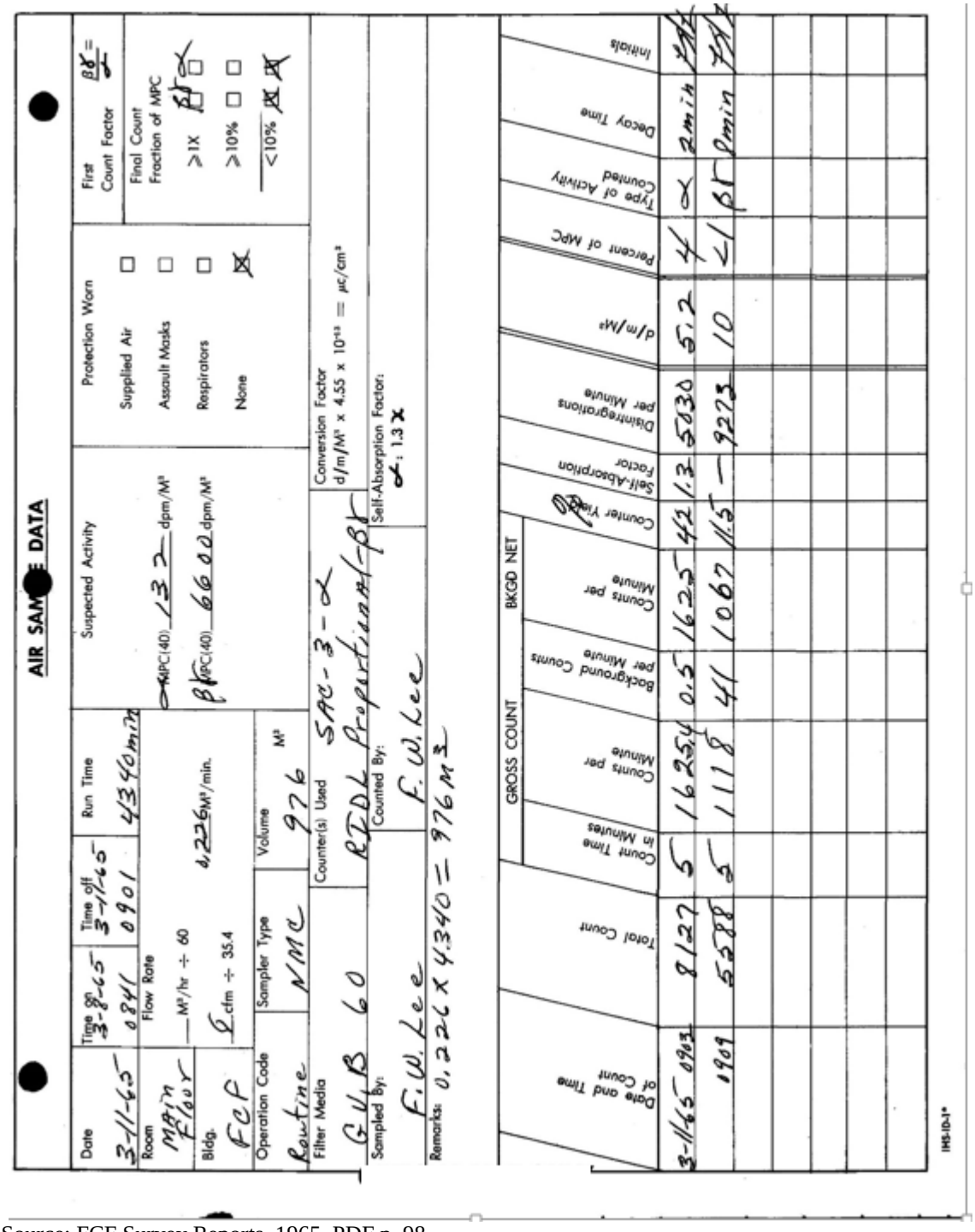
Was respiratory equipment used? ☐ Yes ☑ No

Type: RIDL-138
SAC-3

IHS-ID-10*

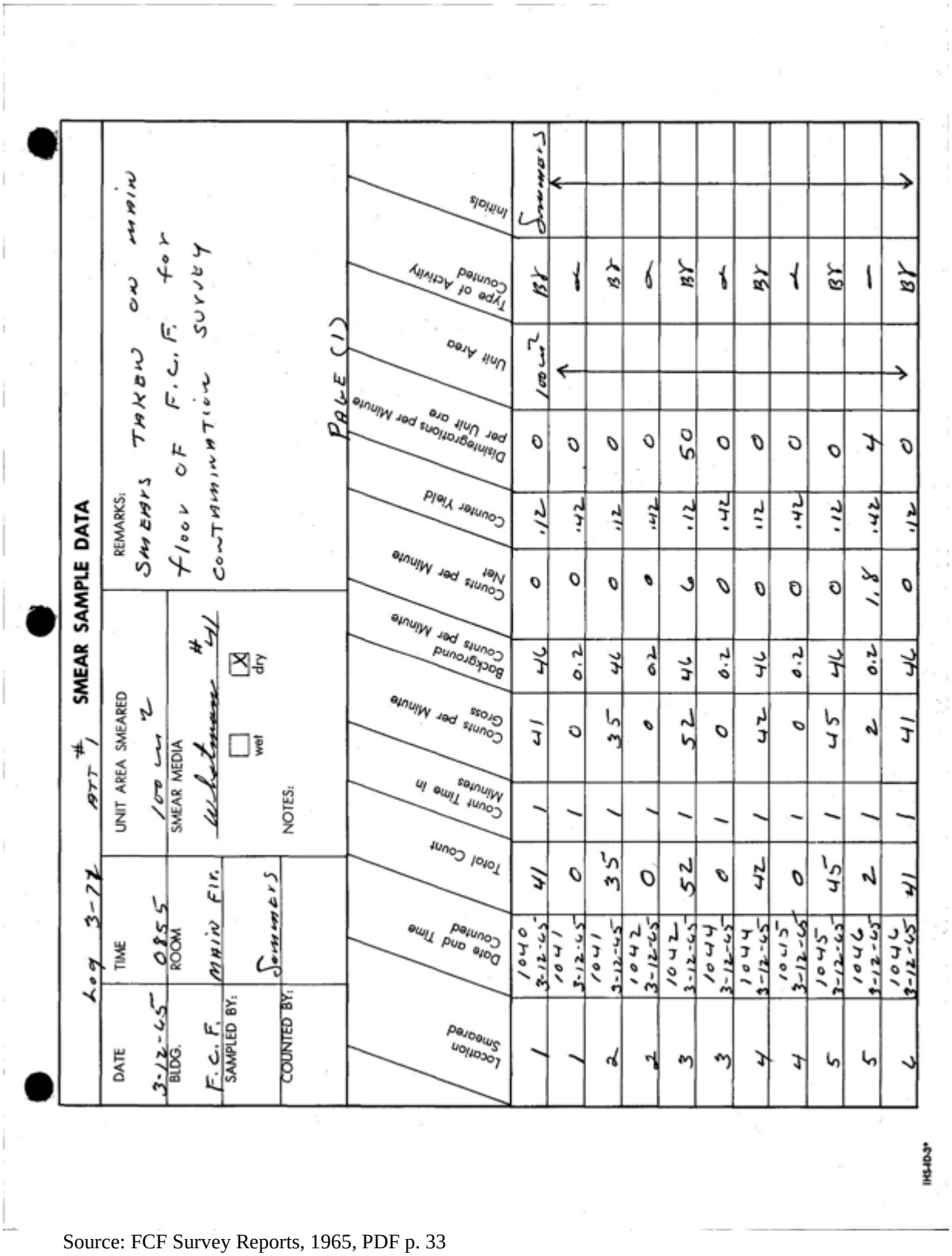
Source: FCF Survey Reports, 1965, PDF p. 32

Figure 6-20: Example of FCF Survey Report from 1965



Source: FCF Survey Reports, 1965, PDF p. 98

Figure 6-21: Example of FCF Air Sample Data Sheet from 1965



Log 3-72 PRT #1 SMEAR SAMPLE DATA

DATE 3-12-65 TIME 0855
BLDG. F.C.F. ROOM Main Flr.
SAMPLED BY: Summers
COUNTED BY:
UNIT AREA SMEARED 100 cm²
SMEAR MEDIA Whatman #41
☐ wet ☒ dry
NOTES:
REMARKS: Smears taken on main floor of F.C.F. for contamination survey

PAGE (1)

Location Smeared	Date and Time Counted	Total Count	Count Time in Minutes	Counts per Minute Gross	Background Counts per Minute	Counts per Minute Net	Counter Yield	Disintegrations per Minute per Unit area	Unit Area	Type of Activity Counted	Initials
1	1040 3-12-65	41	1	41	46	0	.12	0	100 cm²	BY	Summers
1	1041 3-12-65	0	1	0	0.2	0	.42	0	↑	—	
2	1041 3-12-65	35	1	35	46	0	.12	0		BY	
2	1042 3-12-65	0	1	0	0.2	0	.42	0		—	
3	1042 3-12-65	52	1	52	46	6	.12	50		BY	
3	1044 3-12-65	0	1	0	0.2	0	.42	0		—	
4	1044 3-12-65	42	1	42	46	0	.12	0		BY	
4	1045 3-12-65	0	1	0	0.2	0	.42	0		—	
5	1045 3-12-65	45	1	45	46	0	.12	0		BY	
5	1046 3-12-65	2	1	2	0.2	1.8	.42	4		—	
6	1046 3-12-65	41	1	41	46	0	.12	0	↓	BY	↓

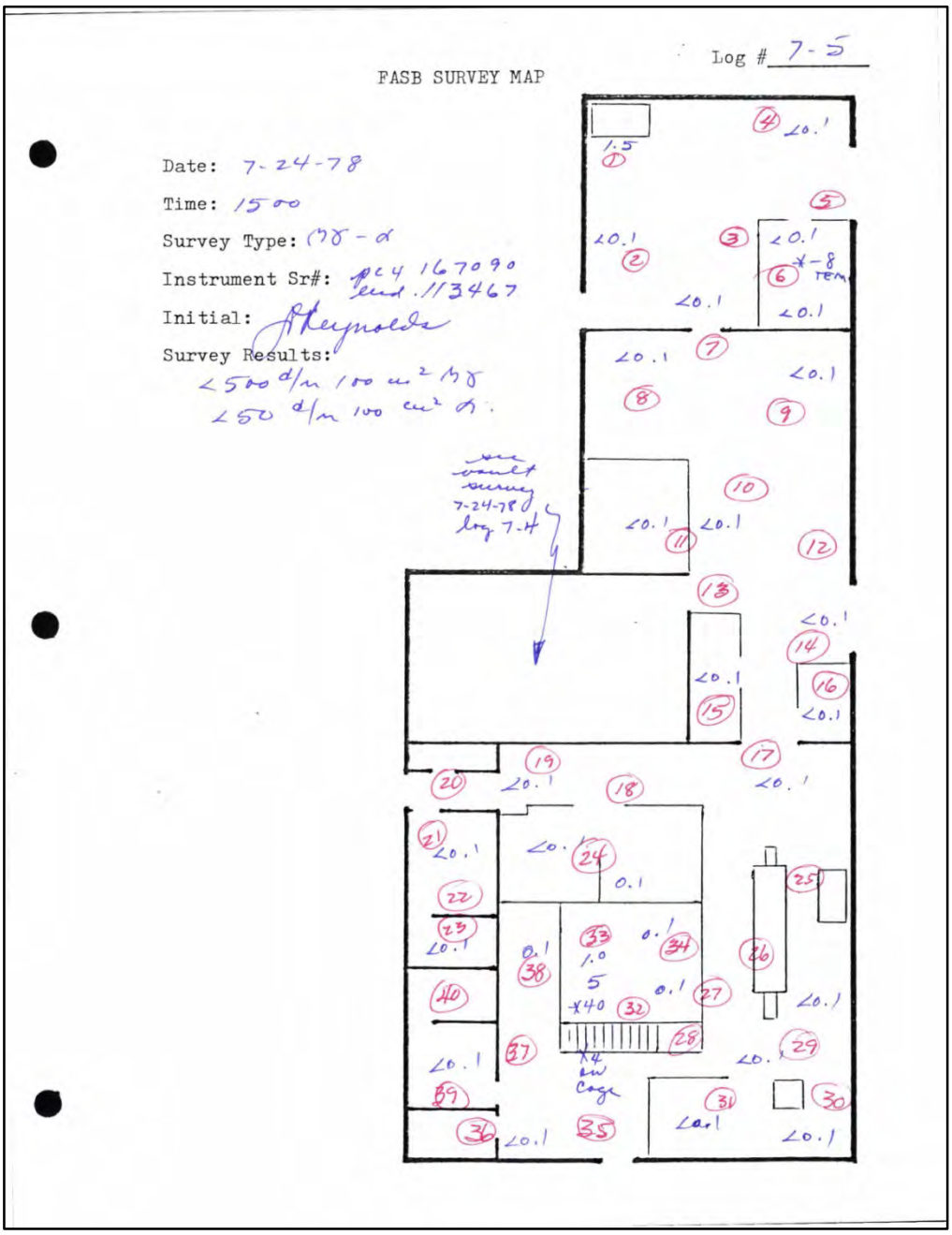
IHS-HDS*

Source: FCF Survey Reports, 1965, PDF p. 33

Figure 6-22: Example of FCF Smear Sample Data Sheet from 1965

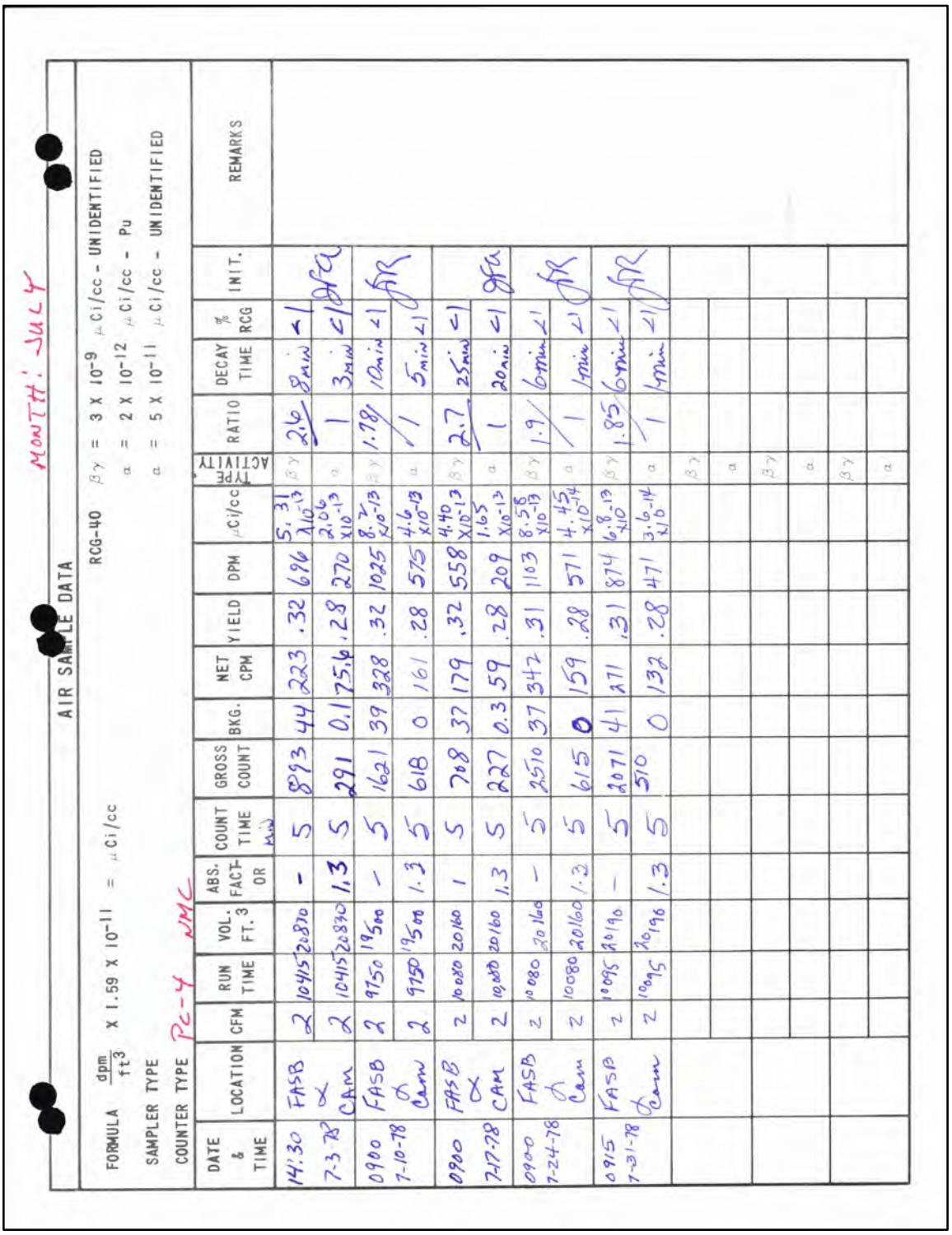
6.3.2.4 Available Radiological Monitoring Data for FASB

Figure 6-23 shows an example of a survey report for the FASB recorded in 1978. Figure 6-24 shows an example of an air sample data sheet for the FASB recorded in 1978. Figure 6-25 shows an example of a contamination smear sample data sheet for the FASB recorded in 1973.



Source: Survey Reports, 1978

Figure 6-23: Example of FASB Survey Report from 1978



AIR SAMPLE DATA

MONTH: JULY

FORMULA $\frac{dpm}{ft^3}$ X 1.59 X 10^{-11} = μCi/cc

RCG-40 $\beta\gamma$ = 3 X 10^{-9} μCi/cc - UNIDENTIFIED
α = 2 X 10^{-12} μCi/cc - Pu
α = 5 X 10^{-11} μCi/cc - UNIDENTIFIED

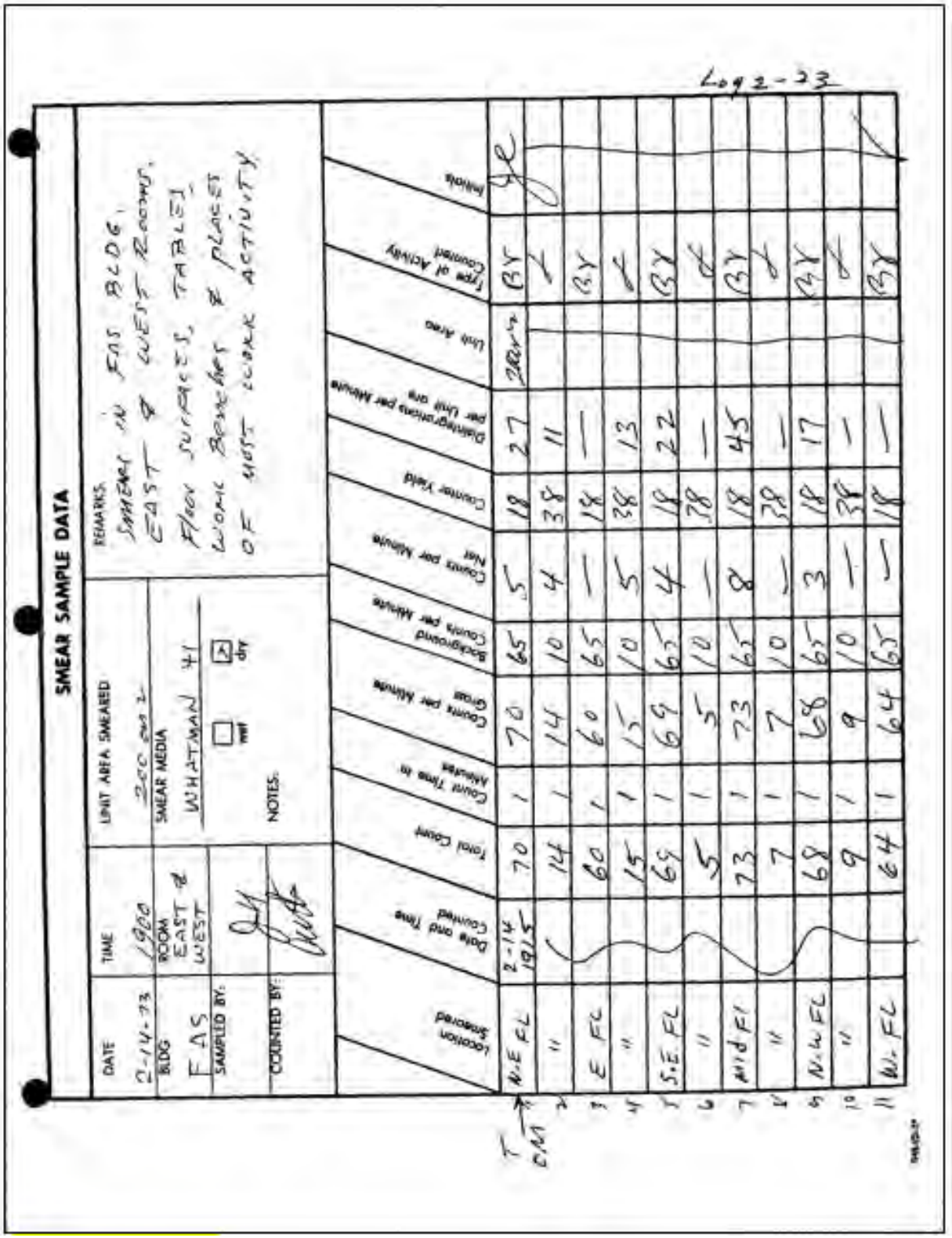
SAMPLER TYPE PC-4 VMC

COUNTER TYPE

DATE & TIME	LOCATION	CFM	RUN TIME	VOL. FT.^{3}	ABS. FACTOR	COUNT TIME (min)	GROSS COUNT	BKG.	NET CPM	YIELD	DPM	μCi/cc	TYPE ACTIVITY	RATIO	DECAY TIME	% RCG	INIT.	REMARKS
1430 7-3-78	FASB & CAM	2	10415	20870	-	5	893	44	223	.32	696	5.31×10^{-13}	$\beta\gamma$	2.6	8min	<1		
		2	10415	20870	1.3	5	291	0.1	175.6	.28	270	2.06×10^{-13}	α	1	3min	<1	AFA	
0900 7-10-78	FASB & CAM	2	9750	19500	-	5	1621	39	328	.32	1025	8.75×10^{-13}	$\beta\gamma$	1.78	10min	<1		
		2	9750	19500	1.3	5	618	0	161	.28	575	4.6×10^{-13}	α	1	5min	<1	AFR	
0900 7-17-78	FASB & CAM	2	10080	20160	1	5	708	37	179	.32	558	4.40×10^{-13}	$\beta\gamma$	2.7	25min	<1		
		2	10080	20160	1.3	5	227	0.3	59	.28	209	1.65×10^{-13}	α	1	20min	<1	AFA	
0900 7-24-78	FASB & CAM	2	10080	20160	-	5	2510	37	342	.31	1103	8.58×10^{-13}	$\beta\gamma$	1.9	6min	<1		
		2	10080	20160	1.3	5	615	0	159	.28	571	4.45×10^{-14}	α	1	1min	<1	AFR	
0915 7-31-78	FASB & CAM	2	10095	20190	-	5	2071	41	271	.31	874	6.8×10^{-13}	$\beta\gamma$	1.85	6min	<1		
		2	10095	20190	1.3	5	510	0	132	.28	471	3.6×10^{-14}	α	1	1min	<1	AFR	

Source: Survey Reports, 1978

Figure 6-24: Example of FASB Air Sample Data Sheet from 1978

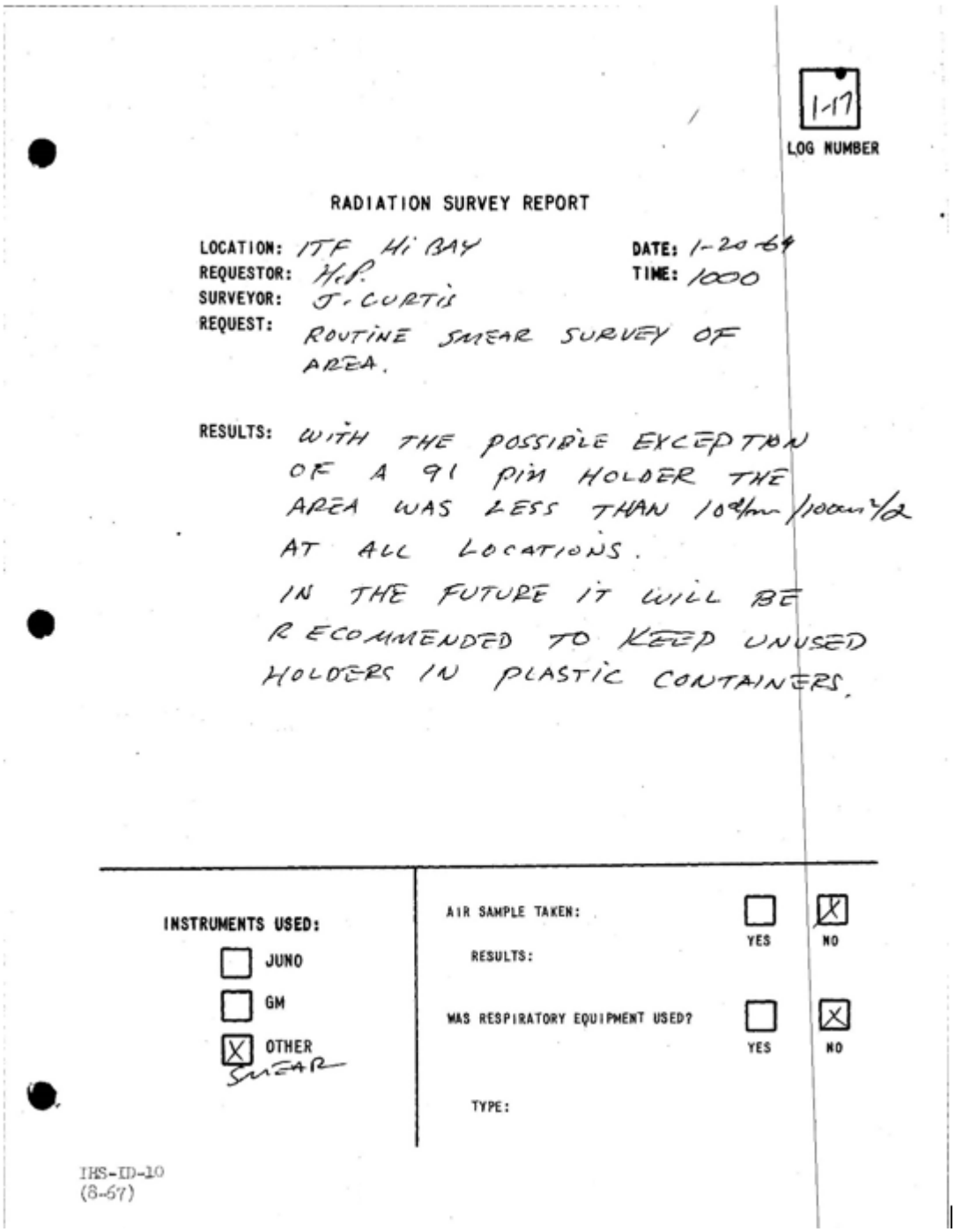


Source: Survey Reports, 1973

Figure 6-25: Example of FASB Smear Sample Data Sheet from 1973

6.3.2.5 Available Radiological Monitoring Data for ITF

Figure 6-26 shows an example of a survey report for the ITF recorded in 1969. Figure 6-27 shows an example of an air sample data sheet for the ITF recorded in 1969. Figure 6-28 shows an example of a contamination smear sample data sheet for the ITF recorded in 1969.



1-17
LOG NUMBER

RADIATION SURVEY REPORT

LOCATION: ITF Hi BAY
REQUESTOR: H.P.
SURVEYOR: J. CURTIS
REQUEST: ROUTINE SMEAR SURVEY OF AREA.

DATE: 1-20-69
TIME: 1000

RESULTS: WITH THE POSSIBLE EXCEPTION OF A 91 PIN HOLDER THE AREA WAS LESS THAN 10 dpm/100cm²/2 AT ALL LOCATIONS.
IN THE FUTURE IT WILL BE RECOMMENDED TO KEEP UNUSED HOLDERS IN PLASTIC CONTAINERS.

INSTRUMENTS USED:
☐ JUNO
☐ GM
☒ OTHER SMEAR

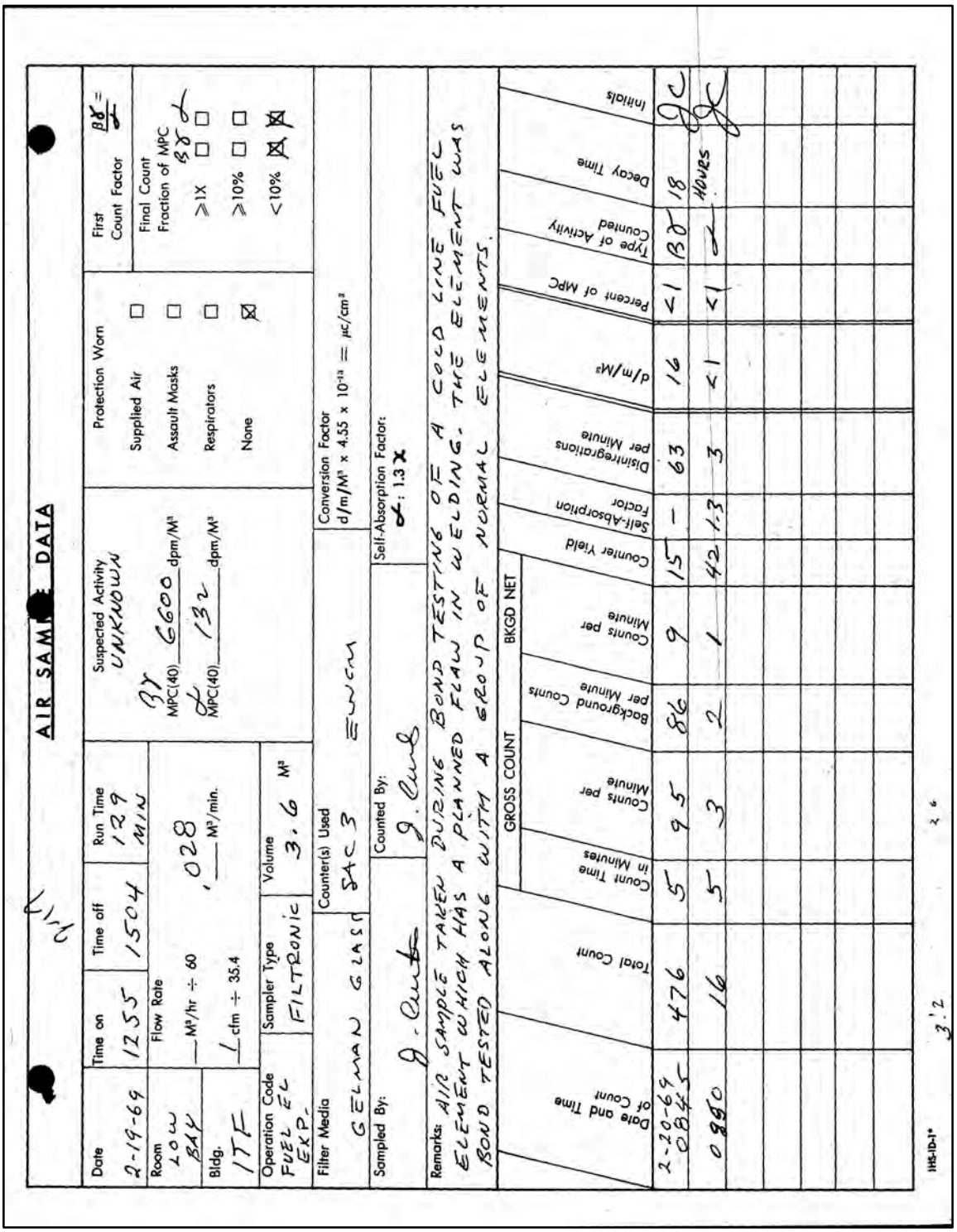
AIR SAMPLE TAKEN: ☐ YES ☒ NO
RESULTS:

WAS RESPIRATORY EQUIPMENT USED? ☐ YES ☒ NO
TYPE:

IHS-ID-10
(8-67)

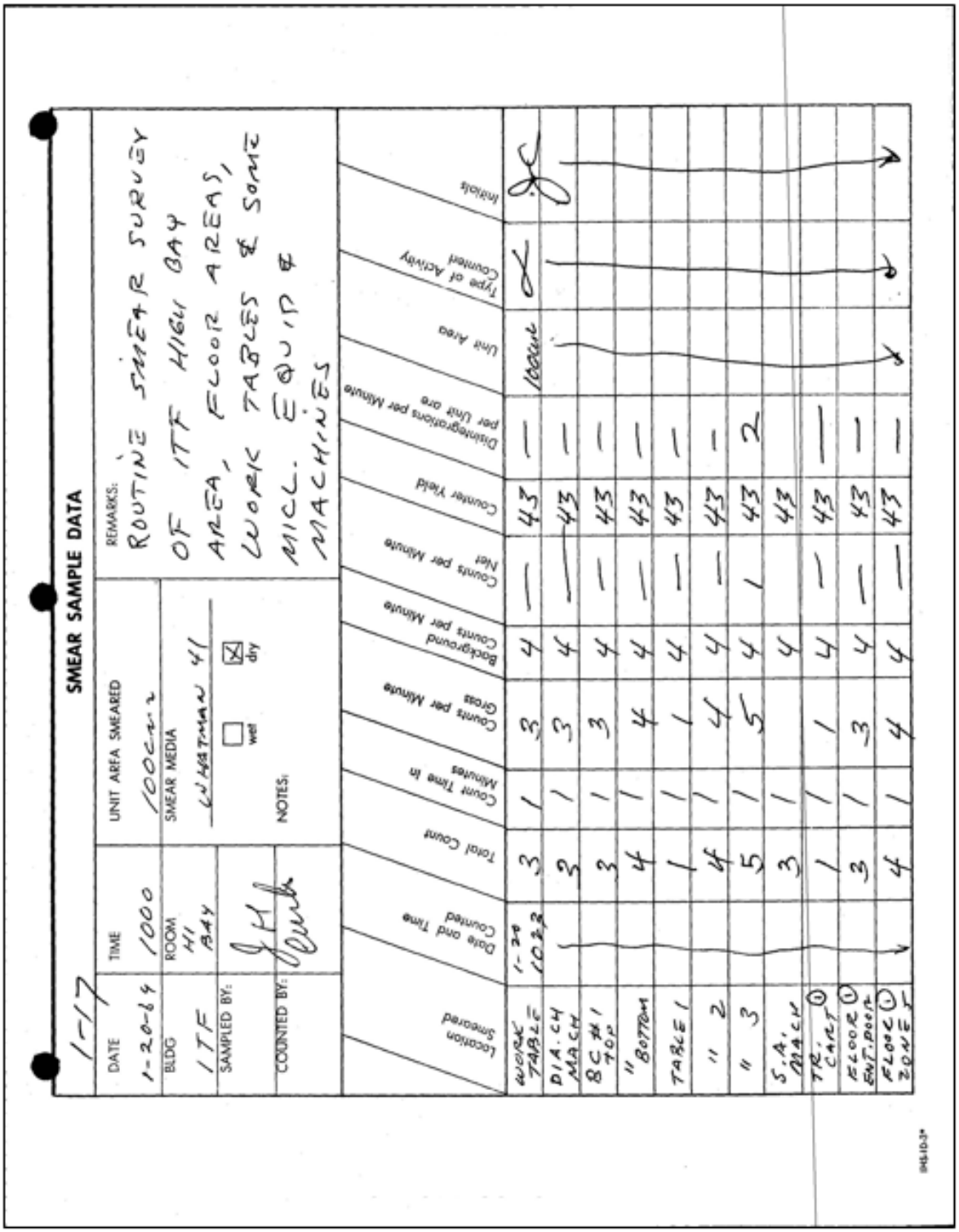
Source: ITF Survey Data, 1967-1969

Figure 6-26: Example of ITF Survey Report from 1969



Source: Survey Reports, Feb1969

Figure 6-27: Example of ITF Air Sample Data Sheet from 1969

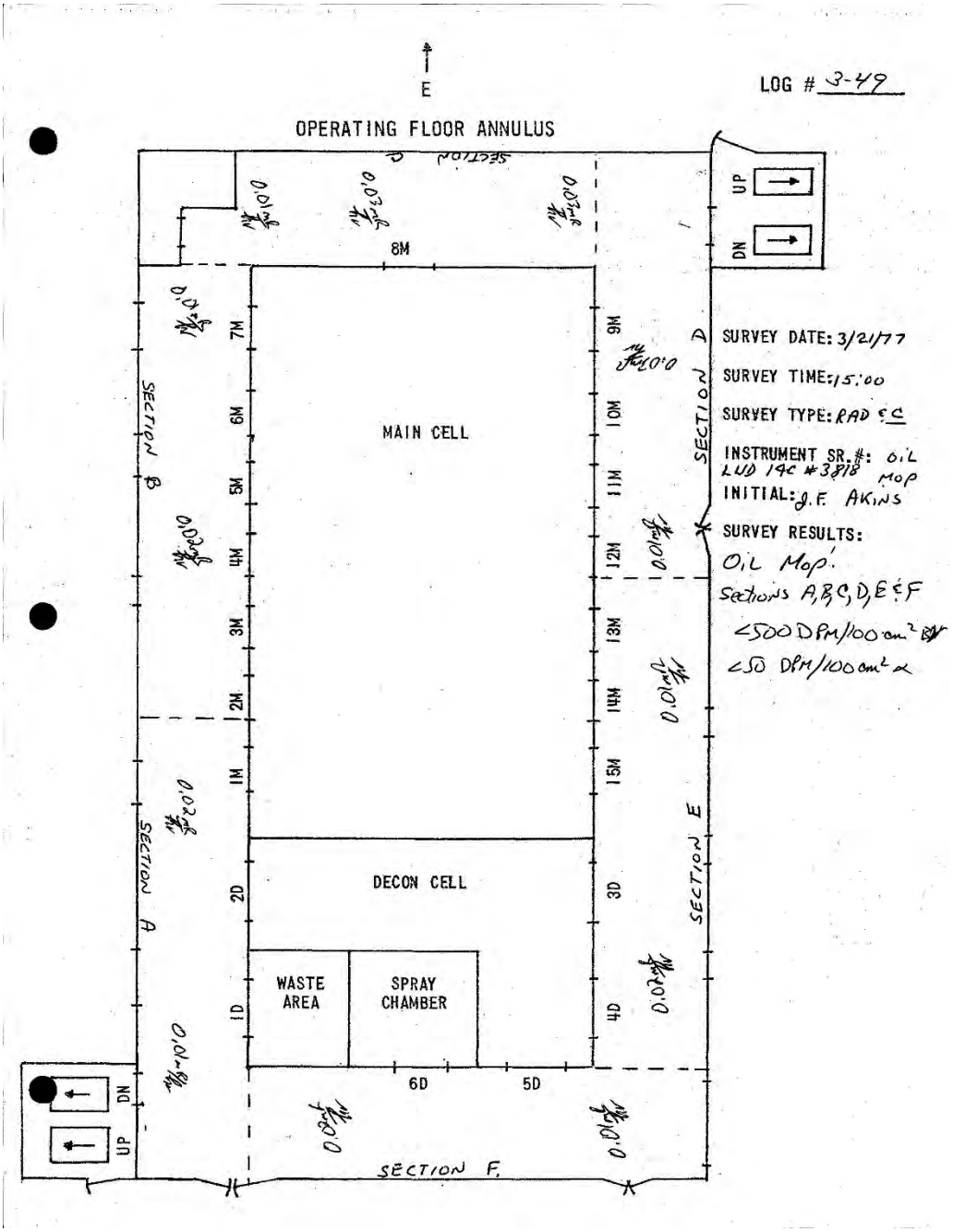


Source: ITF Survey Data, 1967-1969

Figure 6-28: Example of ITF Smear Sample Data Sheet from 1969

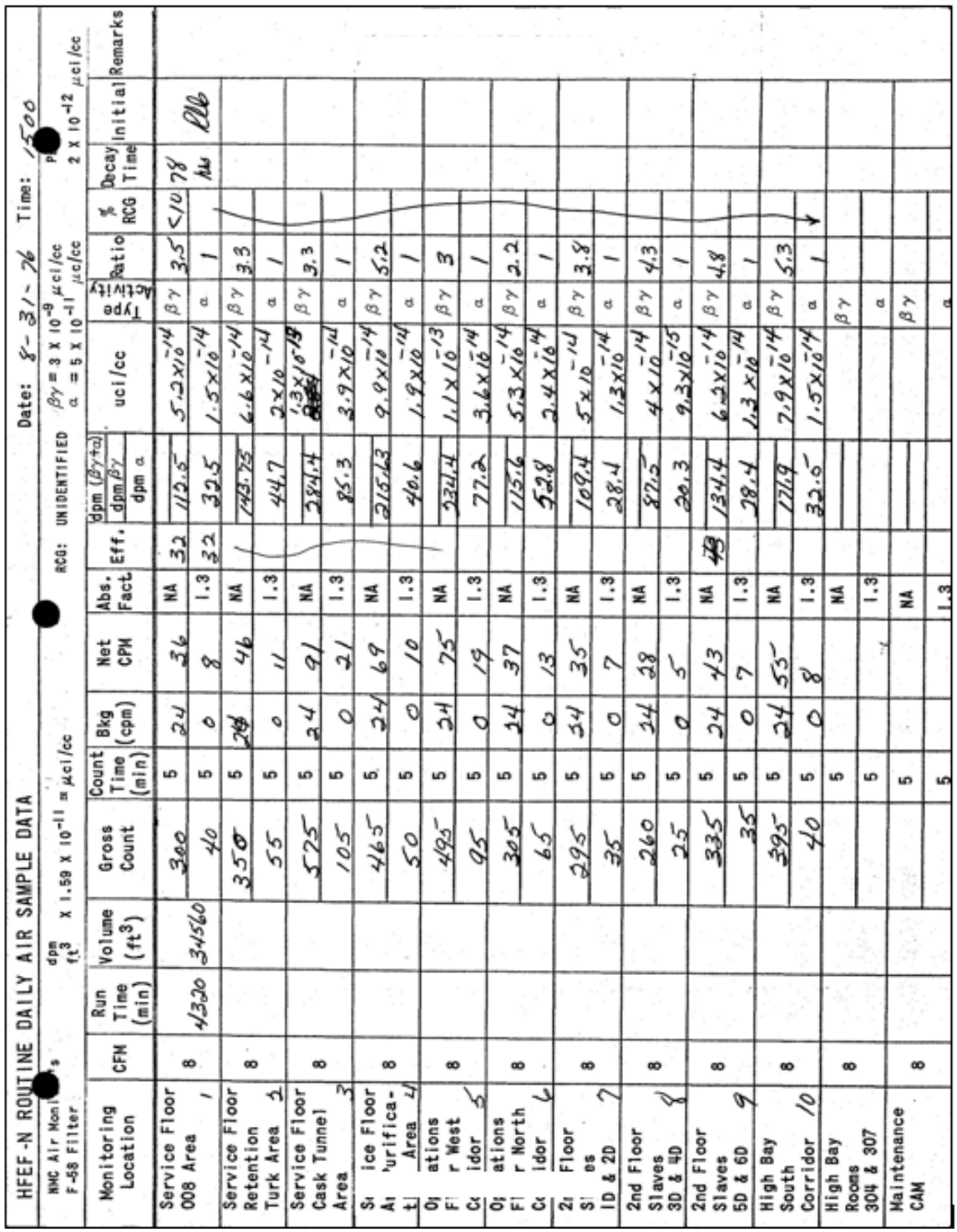
6.3.2.6 Available Radiological Monitoring Data for HFEF-N

Figure 6-29 shows an example of a survey report for the HFEF-N recorded in 1977. Figure 6-30 shows an example of an air sample data sheet for the HFEF-N recorded in 1976. Figure 6-31 shows an example of a contamination smear sample data sheet for the HFEF-N recorded in 1976.



Source: HFEF Survey Data, 1977, PDF p. 180

Figure 6-29: Example of HFEF-N Survey Report from 1977



HFEF-N ROUTINE DAILY AIR SAMPLE DATA

Date: 8-31-76 Time: 1500

NMC Air Monitors F-58 Filter

dpm/ft³ x 1.59 x 10⁻¹¹ = μci/cc

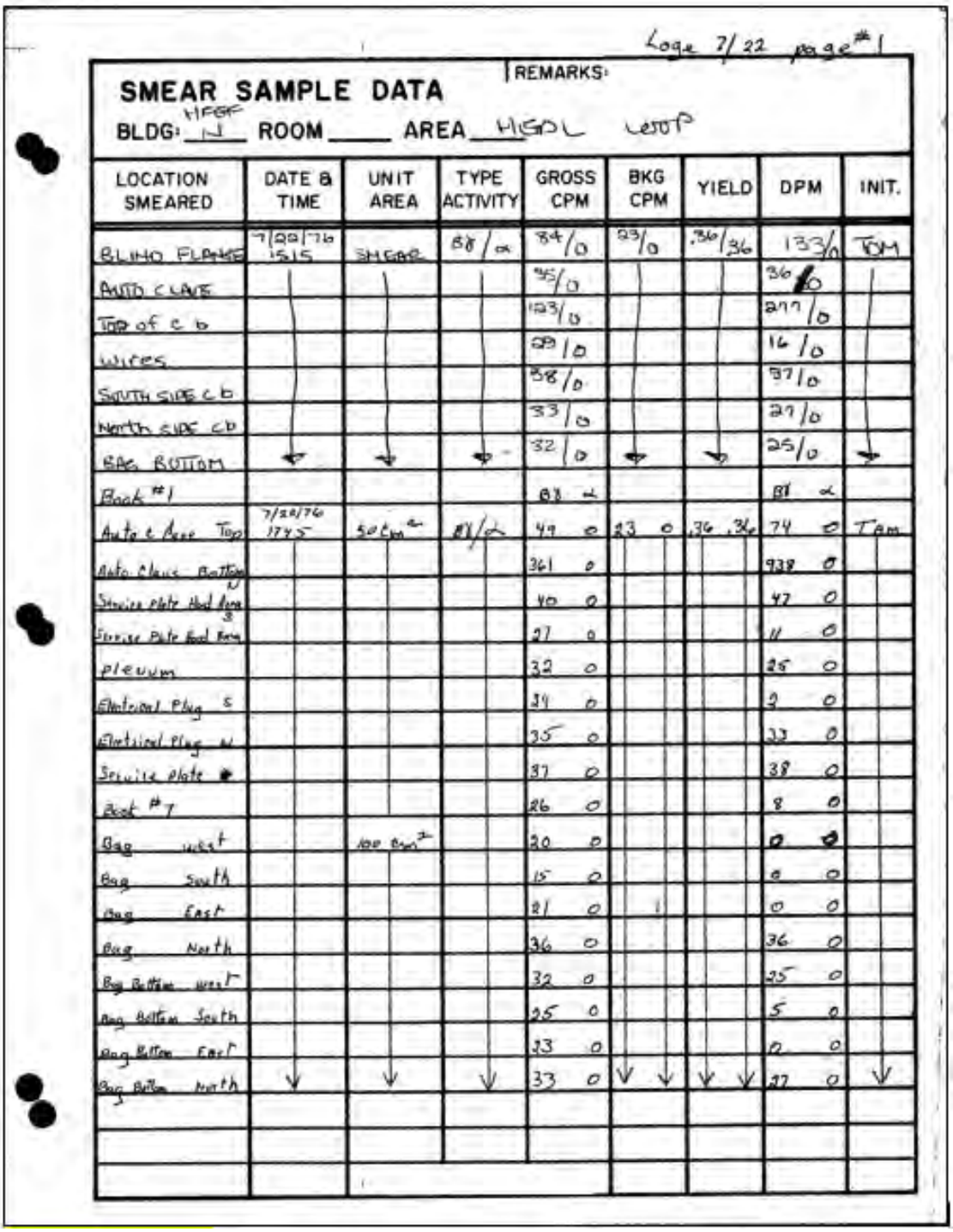
RCG: UNIDENTIFIED βγ = 3 x 10⁻⁹ μci/cc α = 5 x 10⁻¹¹ μci/cc

2 x 10⁻¹² μci/cc

Monitoring Location	CFM	Run Time (min)	Volume (ft³)	Gross Count	Count Time (min)	Bkg (cpm)	Net CPM	Abs. Fact.	Eff.	dpm (βγ+α) dpm βγ dpm α	uci/cc	Activity Type	Ratio	% RCG	Decay Time	Initial	Remarks
Service Floor 008 Area 1	8	4320	34560	300	5	24	36	NA	32	112.5	5.2x10⁻¹⁴	βγ	3.5	<10	78 hrs	DLB	
				40	5	0	8	1.3	32	32.5	1.5x10⁻¹⁴	α	1				
Service Floor Retention Turk Area 2	8			350	5	24	46	NA		143.75	6.6x10⁻¹⁴	βγ	3.3				
				55	5	0	11	1.3		44.7	2x10⁻¹⁴	α	1				
Service Floor Cask Tunnel Area 3	8			525	5	24	91	NA		284.4	1.3x10⁻¹³	βγ	3.3				
				105	5	0	21	1.3		85.3	3.9x10⁻¹⁴	α	1				
Service Floor Air Purifica- tions Area 4	8			465	5	24	69	NA		215.63	9.9x10⁻¹⁴	βγ	5.2				
				50	5	0	10	1.3		40.6	1.9x10⁻¹⁴	α	1				
Operations Floor West Corridor 5	8			495	5	24	75	NA		234.4	1.1x10⁻¹³	βγ	3				
				95	5	0	19	1.3		77.2	3.6x10⁻¹⁴	α	1				
Operations Floor North Corridor 6	8			305	5	24	37	NA		115.6	5.3x10⁻¹⁴	βγ	2.2				
				65	5	0	13	1.3		52.8	2.4x10⁻¹⁴	α	1				
2nd Floor Slaves 1D & 2D 7	8			295	5	24	35	NA		109.4	5x10⁻¹⁴	βγ	3.8				
				35	5	0	7	1.3		28.4	1.3x10⁻¹⁴	α	1				
2nd Floor Slaves 3D & 4D 8	8			260	5	24	28	NA		87.5	4x10⁻¹⁴	βγ	4.3				
				25	5	0	5	1.3		20.3	9.3x10⁻¹⁵	α	1				
2nd Floor Slaves 5D & 6D 9	8			335	5	24	43	NA	45	134.4	6.2x10⁻¹⁴	βγ	4.8				
				35	5	0	7	1.3		28.4	1.3x10⁻¹⁴	α	1				
High Bay South Corridor 10	8			395	5	24	55	NA		171.9	7.9x10⁻¹⁴	βγ	5.3				
				40	5	0	8	1.3		32.5	1.5x10⁻¹⁴	α	1				
High Bay Rooms 304 & 307	8				5			NA				βγ					
					5			1.3				α					
Maintenance CAM	8				5			NA				βγ					
					5			1.3				α					

Source: Survey Reports, 1976

Figure 6-30: Example of HFEF-N Air Sample Data Sheet from 1976



Loge 7/22 page #1

SMEAR SAMPLE DATA REMARKS:

BLDG: HFEF N ROOM ____ AREA HIGH LEVEL

LOCATION SMEARED	DATE & TIME	UNIT AREA	TYPE ACTIVITY	GROSS CPM	BKG CPM	YIELD	DPM	INIT.
BLIND FLANGE	7/22/76 1515	SMEAR	BB/α	84/0	23/0	.36/.36	133/0	TAM
ANTE CHAMBER				95/0			36/0	
Top of C b				123/0			277/0	
wires				29/0			16/0	
SOUTH SIDE C b				58/0			97/0	
North SIDE CB				33/0			27/0	
BAG BOTTOM				32/0			25/0	
Book #1				81 α			81 α	
Ante Chamber Top	7/22/76 1745	50 cm²	81/α	49 0	23 0	.36 .36	74 0	TAM
Ante Chamber Bottom				361 0			938 0	
Service plate Head Area 3				40 0			47 0	
Service Plate Head Area				27 0			11 0	
plenum				32 0			25 0	
Electrical Plug S				39 0			9 0	
Electrical Plug W				35 0			23 0	
Service plate #				37 0			38 0	
Book #7				26 0			8 0	
Bag West		100 cm²		30 0			0 0	
Bag South				15 0			0 0	
Bag East				21 0			0 0	
Bag North				36 0			36 0	
Bag Bottom West				32 0			25 0	
Bag Bottom South				25 0			5 0	
Bag Bottom East				23 0			0 0	
Bag Bottom North				33 0			27 0	

Source: Survey Reports, 1976

Figure 6-31: Example of HFEF-N Smear Sample Data Sheet from 1976

6.3.2.7 Available Radiological Monitoring Data for ZPPR

Five alpha air monitors were installed in ZPPR to provide for personnel protection. Two of these were located in the reactor-air-cooling system (one for each reactor half). These were intended to warn personnel of any airborne plutonium particulates from the fuel in the core. The other three alpha air monitors were portable and could be moved to optimum locations for personnel protection. One monitor was located in-cell to monitor cell air activity. Another was positioned in the workroom. A spare portable alpha air monitor was available for use as conditions warranted. Three beta-gamma air-particulate monitors were also provided for detection of airborne contamination. One was in the reactor cell and the other in the workroom. The third monitor was a spare and could be moved as needed (ZPPR FSAR, 1972).

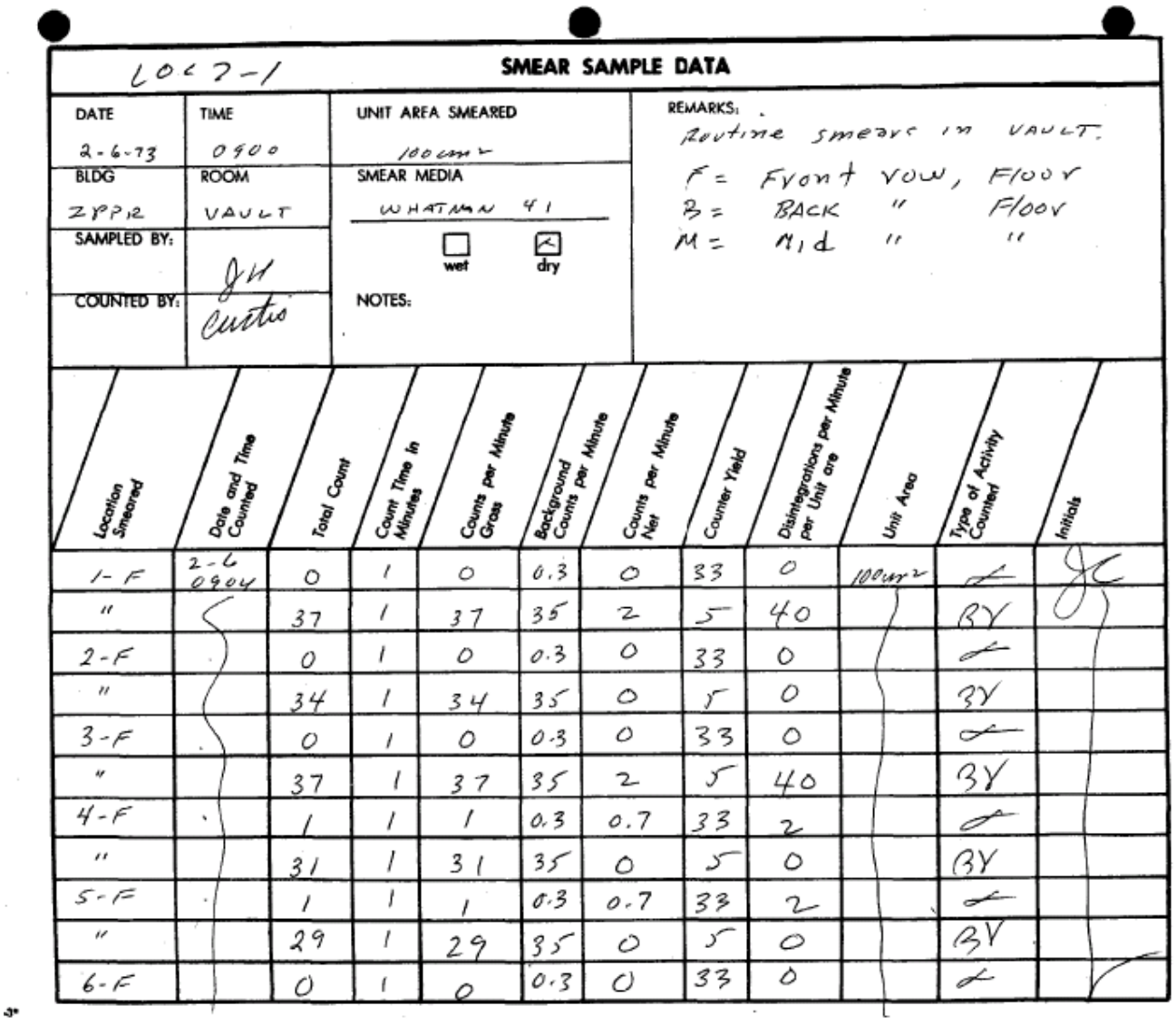
Table 6-2 shows the trip settings for all particulate air monitors within the ZPPR facility (ZPPR Trip Settings, 1969).

Table 6-2: Trip Settings for ZPPR Particulate Air Monitors

Monitor Type	Workroom or Cell	Upstream Equipment Room	Downstream Equipment Room
Alpha	N/A	2.0×10^{-8} µCi/cm^3	6.0×10^{-12} µCi/cm^3
Beta/Gamma	3500 cpm	2.0×10^{-7} µCi/cm^3	8.0×10^{-11} µCi/cm^3

Seven area-gamma monitors were dispersed in the ZPPR Facility. Two gamma monitors were located in the reactor cell, two in the inside equipment room, one in the workroom, one in the core coating room, and one at the suspect filters in the fan loft. Three gamma-criticality monitors were provided in the mound (reactor cell) area. One was located on top of the loading hood in the workroom, one in the storage vault, and one in the core-coating room. All three of these monitors had local and remote (ZPPR control room) alarms (ZPPR FSAR, 1972).

Daily surveys were conducted of the ZPPR lunch room, RadCon office, and coffee area. ZPPR cells, workroom, control room, counting room, storage areas, office area hallways, and hot and cold change areas were surveyed weekly. The vault, AFSR, basement, mock-up building, TLD lab, annex office, and annex basement were surveyed monthly (ZPPR Surveys, 1976). Figure 6-32 shows a data sheet for a smear sample survey conducted in ZPPR in 1973.



LOC 7-1

SMEAR SAMPLE DATA

DATE	TIME	UNIT AREA SMEARED
2-6-73	0900	100 cm²
BLDG	ROOM	SMEAR MEDIA
ZPPR	VAULT	WHATMAN 41
SAMPLED BY:	JH	☐ wet ☒ dry
COUNTED BY:	Curtis	NOTES:

REMARKS: Routine smears in VAULT.
F = Front row, floor
B = BACK " "
M = Mid " "

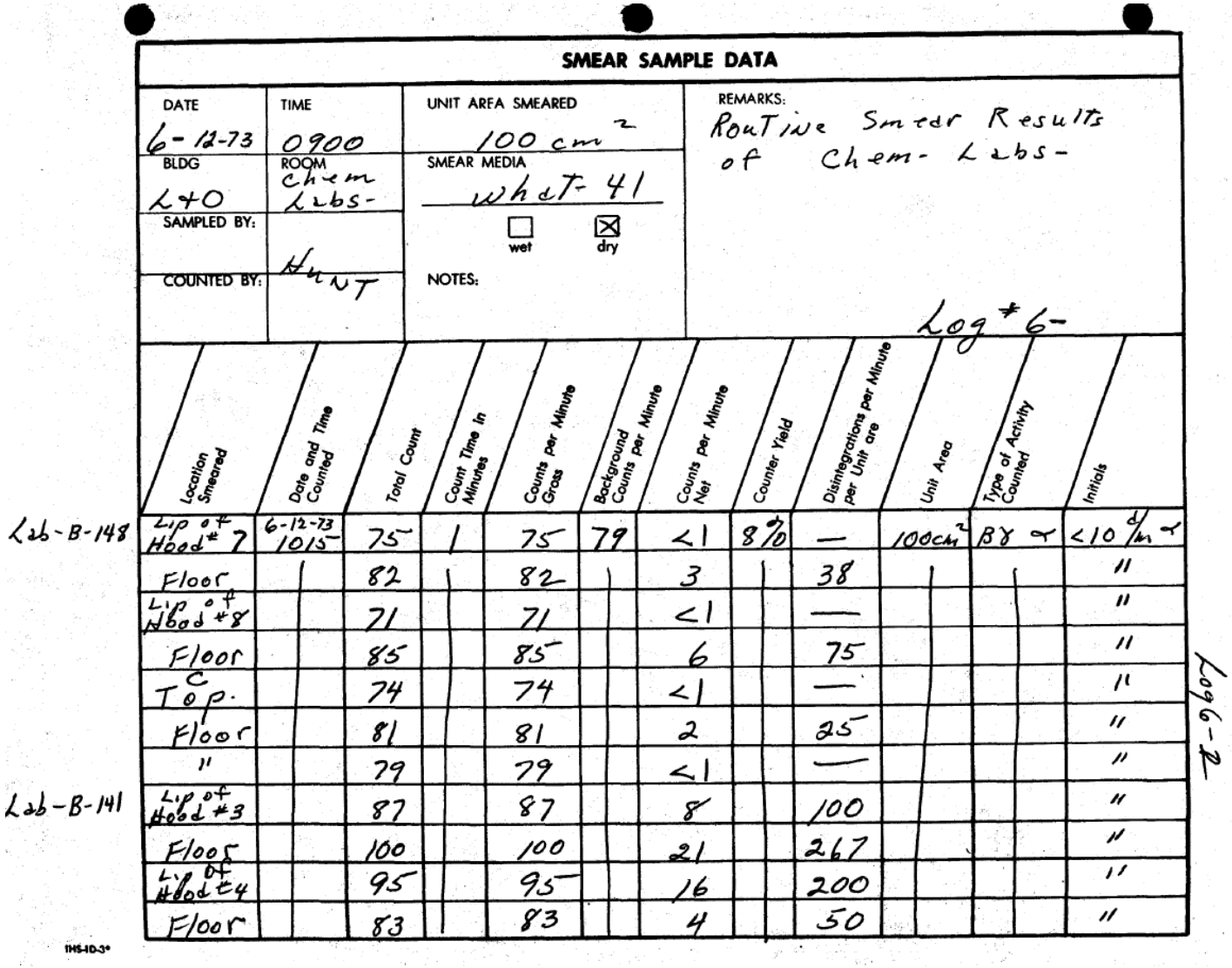
Location Smeared	Date and Time Counted	Total Count	Count Time in Minutes	Counts per Minute Gross	Background Counts per Minute	Counts per Minute Net	Counter Yield	Disintegrations per Minute per Unit are	Unit Area	Type of Activity Counted	Initials
1-F	2-6 0904	0	1	0	0.3	0	33	0	100 cm²	α	JC
"		37	1	37	35	2	5	40		BY	
2-F		0	1	0	0.3	0	33	0		α	
"		34	1	34	35	0	5	0		BY	
3-F		0	1	0	0.3	0	33	0		α	
"		37	1	37	35	2	5	40		BY	
4-F		1	1	1	0.3	0.7	33	2		α	
"		31	1	31	35	0	5	0		BY	
5-F		1	1	1	0.3	0.7	33	2		α	
"		29	1	29	35	0	5	0		BY	
6-F		0	1	0	0.3	0	33	0		α	

Source: ZPPR, 1973

Figure 6-32: Example ZPPR Smear Sample Data Sheet from 1973

6.3.2.8 Available Radiological Monitoring Data for the L&O Building

Daily air samples were taken in the area of the Junior Caves, and daily surveys were taken of any carpeted areas within the building. Survey frequencies for other areas varied depending on the primary function of the area. Chemistry labs, loading docks, and the rear access area and receiving areas for the Junior Caves were all surveyed weekly. Twice-monthly surveys were conducted in the locker room, cafeteria, Metallurgy Lab, Health Physics office, Junior Cave shop area, and L&O office area corridors. All other areas, including interior stairwells, mass spectrometry shop, machine shop, and library were surveyed monthly (L&O Surveys, 1969). Figure 6-33 shows a routine smear survey of the L&O Building Chemistry Labs. Figure 6-34 shows an example survey map for the L&O Building Junior Cave Area.



SMEAR SAMPLE DATA

DATE: 6-12-73 | TIME: 0900 | UNIT AREA SMEARED: 100 cm² | REMARKS: Routine Smear Results of Chem-Labs-

BLDG: L40 | ROOM: Chem Labs- | SMEAR MEDIA: what 41 ☐ wet ☒ dry

SAMPLED BY: | COUNTED BY: Hunt | NOTES:

Log # 6-

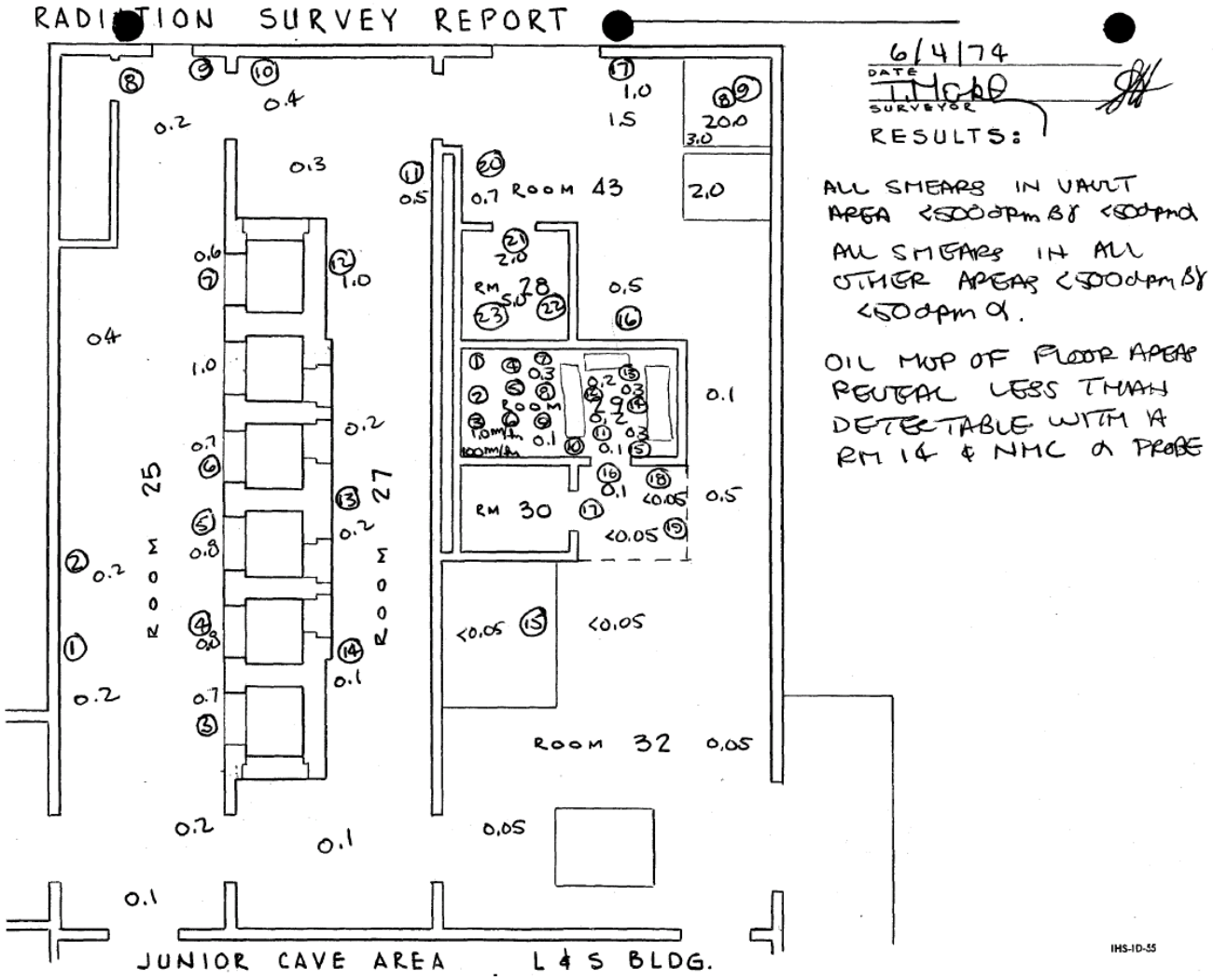
	Location Smeared	Date and Time Counted	Total Count	Count Time in Minutes	Counts per Minute Gross	Background Counts per Minute	Counts per Minute Net	Counter Yield	Disintegrations per Minute per Unit are	Unit Area	Type of Activity Counted	Initials
Lab-B-148	Lip of Hood # 7	6-12-73 1015	75	1	75	79	<1	8%	—	100cm²	Bx α	<10 d/m α
	Floor		82		82		3		38			"
	Lip of Hood #8		71		71		<1		—			"
	Floor		85		85		6		75			"
	C Top.		74		74		<1		—			"
	Floor		81		81		2		25			"
	"		79		79		<1		—			"
Lab-B-141	Lip of Hood #3		87		87		8		100			"
	Floor		100		100		21		267			"
	Lip of Hood #4		95		95		16		200			"
	Floor		83		83		4		50			"

Log # 6-2

IHS-10-3*

Source: L&O Surveys, 1973

Figure 6-33: Example L&O Building Routine Surveys – 1973



Source: L&O Surveys, 1974

Figure 6-34: Example L&O Building Junior Cave Area Survey Map

7.0 Feasibility of Dose Reconstruction for the Class Evaluated by NIOSH

The feasibility determinations for the class of employees under evaluation in this report are governed by both EEOICPA and 42 C.F.R. § 83.13(c)(1). Under that Act and rule, NIOSH must establish whether or not it has access to sufficient information either to estimate the maximum radiation dose for every type of cancer for which radiation doses are reconstructed that could have been incurred under plausible circumstances by any member of the class, or to estimate the radiation doses to members of the class more precisely than a maximum dose estimate. If NIOSH has access to sufficient information for either case, NIOSH would then determine that it would be feasible to conduct dose reconstructions.

In determining feasibility, NIOSH begins by evaluating whether current or completed NIOSH dose reconstructions demonstrate the feasibility of estimating with sufficient accuracy the potential radiation exposures of the class. If the conclusion is one of infeasibility, NIOSH systematically evaluates the sufficiency of different types of monitoring data, process and source or source term data, which together or individually might assure that NIOSH can estimate either the maximum doses that members of the class might have incurred, or more precise quantities that reflect the variability of exposures experienced by groups or individual members of the class as summarized in Section 7.6. This approach is discussed in DCAS's SEC Petition Evaluation Internal Procedures which are available at http://www.cdc.gov/niosh/ocas. The next four major subsections of this Evaluation Report examine:

- The sufficiency and reliability of the available data. (Section 7.1)
- The feasibility of reconstructing internal radiation doses. (Section 7.2)
- The feasibility of reconstructing external radiation doses. (Section 7.3)
- The bases for petition SEC-00224 as submitted by the petitioner. (Section 7.4)

7.1 Pedigree of ANL-W Data

This subsection answers questions that need to be asked before performing a feasibility evaluation. Data Pedigree addresses the background, history, and origin of the data. It requires looking at site methodologies that may have changed over time; primary versus secondary data sources and whether they match; and whether data are internally consistent. All these issues form the bedrock of the researcher's confidence and later conclusions about the data's quality, credibility, reliability, representativeness, and sufficiency for determining the feasibility of dose reconstruction. The feasibility evaluation presupposes that data pedigree issues have been settled.

7.1.1 Internal Monitoring Data Pedigree Review

ATTRIBUTION: Section 7.1.1 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

Upon request, ANL-W provides database printouts and copies of original hardcopy bioassay results, as well as whole-body-count/chest-count records. NIOSH has obtained both bioassay results and a limited number of air monitoring results for the period under evaluation. Bioassay sample results were obtained starting in 1953. The bioassay results are comprised of urinalyses starting in 1953, and fecal results starting in 1961. NIOSH has also obtained whole-body-counting results starting in 1961. During the evaluation period, the bioassay records were determined to be incomplete prior to 1958 due to missing records. Many of the early bioassay records were discovered in the personal files of the EBR-I Complex Radiation Protection Manager; although the records were incomplete, they included data from important radiological events.

A large amount of raw bioassay data exists at ANL-W. Many of these records have been compiled into a single database, which also contains bioassay data for INL, with a data validation and comment field added to each record. These data are in the process of being cross-verified with scanned copies of 32,951 pages of internal dosimetry records (Bioassay, 1956-1984; Bioassay, 1956-1967; Bioassay, 1958-1986). Duplicate records and unusable records (e.g., gross errors) are not included in this assessment.

7.1.2 External Monitoring Data Pedigree Review

ATTRIBUTION: Section 7.1.2 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

Upon request, ANL-W provides database printouts and copies of original hard-copy external dosimetry results. NIOSH has obtained personnel dosimetry information and area monitoring data for much of the period under evaluation. External dosimetry results were obtained starting in 1952, with some breaks in 1955 through 1957. This break appears to be attributable to a change in the site responsible for maintaining the external dosimetry results. From a review of external dosimetry records of early ANL-W workers, from both ANL-E (Lemont, IL) and INL, the responsibility clearly rested with ANL-E through March 1955. After that, there appears to have been a transition of responsibility to INL with much more complete records after 1957. A large amount of personnel dosimetry data exists at ANL-W, including duplicate records and unusable records (e.g., gross errors). NIOSH has determined that all the workers in ANL-W's radiological areas were monitored for external dose starting in 1958 and the dose values exist in original hard-copy form as well as database form that are (and have been) provided by DOE for each worker. In addition to the dosimetry data that are considered a primary resource for supporting external dose reconstructions, NIOSH also has access to ANL-W area monitoring data (including radiation survey data and source-term information)

as well as onsite ambient (environmental) data that are also in original hard-copy form. These data support the assignment of onsite ambient doses for all monitored ANL workers to account for any radiation dose that may have been inappropriately subtracted from their dosimeter results as background radiation, as well as the assignment of dose to unmonitored individuals.

7.1.3 Area Monitoring Data Pedigree Review

ATTRIBUTION: Section 7.1.3 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

Data from the time period under evaluation consist of area survey reports, contamination smear survey reports, and air sampling data. These surveys are copies of original reports and are therefore considered primary data sources. However, during the evaluation period it was determined that there are known gaps in the area monitoring data prior to 1958.

7.2 Evaluation of Bounding Internal Radiation Doses at ANL-W

ATTRIBUTION: Section 7.2 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

The principal source of internal radiation doses for members of the class under evaluation would have been inhalation and ingestion of radiological contamination during the following operations:

- Destructive and non-destructive nuclear reactor tests
- Irradiation of experimental nuclear reactor fuels
- Irradiation of various materials to transform them by neutron bombardment and/or to study their behavior within a nuclear reactor environment
- Uranium fuel production, which involved work with uranium and thoria
- Waste management operations supporting all site radiological activities

The potential internal sources would have been dependent on the operational area and activities. The major intake sources, as indicated in the bioassay records and incident reports, have been mixed fission products (often limited to radioactive iodines and noble gases), activation products, uranium, and plutonium. However, the records list a wide spectrum of radionuclides that were monitored and an even longer list of codes used to identify the radionuclides, groups of radionuclides, specific measurement techniques, or combinations of radionuclides and techniques. Many of the radionuclides apply to a small set of workers on a research project, or to workers whose tasks could have exposed them to many different sources (e.g., radiation monitoring technicians) (ORAUT-TKBS-0006-5).

Given the large variety of facilities and the ever-changing cast of contractors, the AEC established and ran a Health and Safety (H&S) Laboratory at the Idaho site to provide technical support and consistency. The organization went through a series of name changes: Health Services Laboratory (HSL), Health and Safety Division (HSD), Idaho Center for Radiological and Environmental Sciences, and Radiological and Environmental Sciences Laboratory (RESL). During the period under evaluation, *in-vitro* bioassay samples collected throughout the Idaho site were analyzed by the HSL in Building CF-646. Beginning in 1963, *in-vitro* bioassay and WBC facilities were both housed in Building CF-690, where a newly-designed counter was used (INEL Bioassay History, 1994). Eventually, a mobile whole-body counter was built and periodically brought to the ANL-W facilities, reducing the issue of lost work time for WBCs.

For ANL-W EEOICPA claims, ambient environmental internal doses are only assessed for certain unmonitored workers, as discussed in the guidance in ORAUT-TKBS-0007-6, Section 5.6. ORAUT-TKBS-0007-4 and its supplement (Peterson, 2004) provide the historical background, rationale, and environmental intake data for reconstructing occupational environmental internal doses at the ANL-W site. In summary, ORAUT-TKBS-0007-4 provides annual environmental intakes of the significant radionuclides for each major operating area on the ANL-W site. Because environmental exposures are accounted for in personnel occupational exposures and do not impact the ability to reconstruct dose with sufficient accuracy, further assessment of environmental internal exposure is not included in this report.

The following subsections address the ability to bound internal doses, methods for bounding doses, and the feasibility of internal dose reconstruction.

7.2.1 Evaluation of Bounding Internal Radiation Doses for the EBR-I Complex

ATTRIBUTION: Section 7.2.1 was completed by Mike Mahathy, Oak Ridge Associated Universities, and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the EBR-I Complex were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the EBR-I Complex are explained in the associated text.

The philosophy of ANL-W's internal dose monitoring program established a routine monitoring and control program based on mixed fission products as the controlling radionuclides. That is, if the mixed fission products were controlled to permissible levels, the alpha hazards would be adequately controlled. However, potential exceptions to this were noted in Section 5.2.2.

ANL-W workers participated in bioassay programs for potential intakes of mixed fission products by either urinalysis or whole-body counts, either routinely or when workplace indicators (e.g., elevated airborne radioactivity, contamination event, positive nasal smear) indicated that an intake may have occurred. When workplace indicators indicated that an intake may have occurred, "special" (non-routine) bioassay would be requested by ANL-W Health Physics.

As indicated in Section 5.2.1, the categories of radionuclides at the EBR-I Complex included: mixed fission and activation products, radioiodines, plutonium, uranium, thorium, and other radionuclides.

7.2.1.1 Evaluation of Bounding Process-Related Internal Doses for the EBR-I Complex

NIOSH obtained routine bioassay data for EBR-I Complex workers for some of the period under evaluation. NIOSH obtained routine urinalysis data starting in 1958 and *in-vivo* data starting in 1962. NIOSH obtained some bioassay results (both urinalysis and fecal) for the period 1952 through 1957, but those bioassay results appear to be incomplete and possibly such monitoring was not based on routine sampling. Incident-driven *in-vitro* bioassay following radiological events of concern were discovered during the evaluation period. For example, there was extensive bioassay for workers after the discovery of a ruptured blanket brick in April 1955, and after the partial core meltdown in November 1955.

The principal sources of internal radiation doses for members of the class working at the EBR-I, ZPR-III, AFSR, and the BORAX-I, -II, -III, -IV, and -V reactors include uranium and mixed fission products; thorium was a potential source of exposure with BORAX-IV as well. The following subsections address the ability to bound internal doses, methods for bounding doses, and the feasibility of internal dose reconstruction.

BORAX-I – BORAX-V and AFSR

Bioassay Information and Available Data

As previously discussed in Section 6.1, there is significant programmatic information about the ANL-W's bioassay program, and after 1957, there are extensive *in-vitro* and *in-vivo* bioassay data that NIOSH believes are sufficient for bounding internal radiation doses from EBR-I Complex radionuclides. The majority of the available bioassay data include: gross beta radioactivity in urine, gross gamma radioactivity in urine, Sr-90 in urine, and whole-body count results.

Not all of the 50 workers referenced in Table 6-1 were employed at ANL-W for the entire period from 1952 through 1960. Table 7-1 presents the internal monitoring data available, by analysis and radiation type, for 12 of the 50 workers known to have worked at ANL-W for the entire nine-year period.

NOTE: Data for nine of the twelve workers are available for 1952; however, those results are not included as it is not clear whether or not they represented baseline, routine, or incident sampling; thus, the analysis below covers the eight years from 1953 through 1960.

Table 7-1 was compiled using data obtained directly from INL and ANL in response to record requests, and data identified during data captures (including bioassay lab results cards). In 1953, none of the workers was monitored. In 1954, five workers were monitored for uranium intakes. In 1955, all 12 were monitored for intakes of uranium and/or plutonium. NIOSH has found bioassay data for only two of the 12 workers for 1956 and no monitoring data for 1957. In 1958, 11 workers were monitored for intakes of beta-emitting radiation and two were monitored for uranium intakes. For 1959, NIOSH has found bioassay data for only two of the 12 workers. For 1960, NIOSH has found bioassay data for 11 of the workers; gross beta results are available for all 11 while gross gamma is available for one. The occupation of worker No. 7 was [job category redacted].

NIOSH believes bioassay data available for the period 1953 through 1957 are inadequate to bound radionuclide intakes. NIOSH has found a small number of bioassay results for 1959. NIOSH intends

to use bioassay data available for 1958 and 1960 to interpolate intakes and doses for 1959. All bioassay data obtained for 1955 resulted from incident follow-ups, not routine monitoring.

Table 7-1: Internal Monitoring Data from All Sources for 12 Selected Workers (1953-1960)

Worker	1953	1954	1955	1956	1957	1958	1959	1960
1	No data provided	urine U-235, fecal U-235	urine U-235, urine Pu	urine beta	No data provided	urine beta	No data provided	urine beta
2	No data provided	No data provided	urine U-235, urine Pu	urine beta	No data provided	urine beta, urine U-235	No data provided	urine beta
3	No data provided	urine U-235	urine U-235, urine Pu	No data provided	No data provided	urine beta	urine beta	urine beta, gamma
4	No data provided	No data provided	urine U-235	No data provided	No data provided	urine beta	No data provided	urine beta
5	No data provided	fecal U-235	urine U-235, urine Pu	No data provided	No data provided	urine beta, urine U-235	No data provided	urine beta
6	No data provided	urine U-235	urine U-235	No data provided	No data provided	urine beta	No data provided	urine beta
7	No data provided	No data provided	urine U-235	No data provided	No data provided	No data provided	No data provided	No data provided
8	No data provided	No data provided	urine U-235	No data provided	No data provided	urine beta	No data provided	urine beta
9	No data provided	urine beta, urine U-235	urine U-235, urine Pu	No data provided	No data provided	urine beta, urine U-235	No data provided	urine beta
10	No data provided	No data provided	urine U-235	No data provided	No data provided	urine beta	No data provided	urine beta
11	No data provided	urine beta, urine U-235	urine U-235, urine Pu	No data provided	No data provided	urine beta	urine beta	urine beta
12	No data provided	No data provided	urine U-235	No data provided	No data provided	urine beta	No data provided	urine beta

Green indicates alpha radiation monitoring.
Blue indicates beta/gamma radiation monitoring.

Only data received directly from INL and ANL in response to records requests were used to populate Table 7-2. Had additional data not been located during data capture, NIOSH would not know workers were monitored for intakes in 1955 and some in 1958, 1959, and 1960.

Table 7-2: Internal Monitoring Data Provided by Sites for 12 Selected Workers (1953-1960)

Worker	1953	1954	1955	1956	1957	1958	1959	1960
1	No data provided	urine U-235, fecal U-235	No data provided	urine beta	No data provided	urine beta	No data provided	urine beta
2	No data provided	No data provided	No data provided	urine beta	No data provided	No data provided	No data provided	No data provided
3	No data provided	urine U-235	No data provided	No data provided	No data provided	urine beta	urine beta	urine beta, gamma
4	No data provided	No data provided	No data provided	No data provided	No data provided	urine beta	No data provided	urine beta
5	No data provided	fecal U-235	No data provided	No data provided	No data provided	urine beta, urine U-235	No data provided	urine beta
6	No data provided	urine U-235	No data provided	No data provided	No data provided	urine beta	No data provided	No data provided
7	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided
8	No data provided	No data provided	No data provided	No data provided	No data provided	urine beta	No data provided	urine beta
9	No data provided	urine beta, urine U-235	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided
10	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided
11	No data provided	urine beta, urine U-235	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided
12	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided	No data provided

Green indicates alpha radiation monitoring.
Blue indicates beta/gamma radiation monitoring.

Using external dosimetry reports, NIOSH assembled a roster of workers at AFSR from 1961 through 1970. NIOSH researched bioassay results available for each worker. Table 7-3 presents the number of AFSR workers monitored by method each year and the total number of workers with no bioassay results. Only two workers were not monitored in 1961 and one each in 1968 and 1969. In each instance, the occupation of said workers was [job category redacted]; each of the four workers was monitored in the subsequent year. Table 7-4 presents similar information for BORAX-V workers through the operation of that reactor from 1961 through 1964. Results are available for each of the BORAX-V workers, indicating they were all routinely monitored.

Table 7-3: Monitoring of AFSR Workers

Worker Status	1961	1962	1963	1964	1965	1966	1967	1968	1969	1970
Total Workers	9	10	9	8	8	9	9	9	7	6
Workers with *In-Vitro* Results	7	10	8	8	8	6	2	2	0	0
Workers with *In-Vivo* Results	0	0	6	2	1	6	9	7	6	6
Workers with No Results	2	0	0	0	0	0	0	1	1	0

Table 7-4: Monitoring of BORAX-V Workers

Worker Status	1961	1962	1963	1964
Total Workers	28	28	34	28
Workers with *In-Vitro* Results	28	28	34	28
Workers with *In-Vivo* Results	0	2	17	5
Workers Not Monitored	0	0	0	0

Routine bioassay for BORAX-V and AFSR workers is available starting in 1961, as shown in Tables 7-3 and 7-4. The earliest *in-vivo* data for BORAX-V workers in the available data are for 1962. Starting with BORAX-V in 1962, NIOSH has obtained urinalysis and/or *in-vivo* results for all workers on external monitoring rosters for the reactor.

From 1958 through 1961, urinalysis results will be used to bound doses received from alpha, beta and X-ray, and gamma-emitting radiation. While such doses would be very small, NIOSH will use a ratio of uranium to fission products to derive doses received from uranium in 1958 when BORAX-IV conducted cladding failure experiments (Borax-IV Test, 1959). Starting with 1963, the *in-vivo* data will be used with urinalysis results to bound doses received from X-ray and gamma-emitting radiation.

Air Monitoring Information and Available Data

As previously discussed in Section 6.3, there is significant programmatic information about the ANL-W's air monitoring programs, and there are extensive air sampling data that NIOSH believes are sufficient for bounding internal radiation doses from the radionuclides if they could not be bounded using the available bioassay data (e.g., thorium, uranium without fission products). However, the air monitoring data prior to 1958 are incomplete. While records exist indicating that air monitoring was performed and the data collected and analyzed, the data are limited at best. The primary source of pre-1958 air monitoring data was from the personal files of the EBR-I Complex Radiological Protection Manager. ANL-W did not routinely monitor for airborne radioactivity at AFSR due to the solid structure of the small lab-scale reactor. The majority of the available air sampling data includes both gross beta-gamma radioactivity and gross alpha radioactivity results for each air sample.

Alternative Data Sources for Bounding Internal Dose

NIOSH has found little bioassay data for 1959 for the EBR-I complex but results for 1959 will be interpolated using results obtained for 1958 and 1960.

As previously discussed in Section 6.3, there is significant programmatic information about the ANL-W's contamination monitoring programs, and there are extensive contamination smear data that could be used to support bounding potential doses to certain radionuclides. Although available, NIOSH does not have any current plans to use these data.

7.2.1.2 Methods for Bounding Operational Period Internal Doses for the EBR-I Complex

Fission and Activation Products

Beginning in 1958, potential intakes of mixed fission and activation products for monitored and unmonitored EBR-I Complex workers can be estimated using the approaches described in ORAUT-TKBS-0007-5. For urine samples analyzed for gross beta, gross gamma, and/or strontium radioactivity, NIOSH will assess Sr-90 and/or Cs-137 intakes in accordance with ORAUT-TKBS-0007-5 and ORAUT-OTIB-0060. For whole body counts, NIOSH will assess Cs-137 intakes in accordance with ORAUT-TKBS-0007-5 and ORAUT-OTIB-0060. In accordance with the guidance in ORAUT-OTIB-0054, NIOSH will use Sr-90 and/or Cs-137 intakes to estimate the intakes of the other mixed fission and activation products that may have been present, unless a bioassay result was available for a specific radionuclide. In the event that a bioassay result was available for a specific radionuclide, NIOSH will use that result in lieu of the ORAUT-OTIB-0054 approach.

Based on the procedural information and the data on-hand, NIOSH finds that it has adequate monitoring data to allow for sufficiently accurate estimation of internal doses attributable to mixed fission and activation products for EBR-I Complex workers at the EBR-I Complex from 1958 forward. Such doses cannot be bounded from 1952 through 1957 due to the lack of adequate monitoring data, as demonstrated in Table 7-1.

Radioiodines

Based on the availability of gross-gamma urine-sample data and whole-body count data for EBR-I Complex workers, NIOSH finds that it has adequate monitoring data to allow for sufficiently accurate estimation of internal doses attributable to radioiodines from 1958 forward. Such doses cannot be bounded from 1952 through 1957 due to the lack of adequate monitoring data, as demonstrated in Table 7-1.

The potential intakes of radioiodines without other mixed fission products present can be estimated for monitored EBR-I Complex workers using the bioassay data and guidance provided in Section 5.5.4 of ORAUT-TKBS-0007-5.

When radioiodines are present with other mixed fission products, the potential intakes of radioiodines for monitored EBR-I Complex workers can be accounted for using the bioassay data and guidance provided in Section 5.5.1 of ORAUT-TKBS-0007-5, which employs the use of ORAUT-OTIB-0054.

Based on the available information, the presence of airborne radioiodine without other mixed fission products present was uncommon at the EBR-I Complex, and unmonitored intakes of radioiodines likely only occurred when other mixed fission products were present. Therefore, potential intakes of radioiodines for unmonitored EBR-I Complex workers can be estimated using the guidance provided in Section 5.6 of ORAUT-TKBS-0007-5.

Radioactive Noble Gases

ANL-W was not involved in the mining, milling, or separation of uranium or thorium from ores. Fuels arrived at the site pre-formed and extruded and did not age to the point where radon or thoron would have been a significant concern. Therefore, radon and thoron are not considered a significant internal exposure concern. Because fission-produced radioactive noble gases do not represent a significant internal exposure concern, NIOSH finds there is no need to assess the internal doses attributable to radioactive noble gases.

Actinides with Mixed Fission Products Present

With a few exceptions, the majority of the actinides at the EBR-I Complex were present with mixed fission products. The potential intakes of actinides when mixed fission products were present can be estimated using the approach described in Section 5.5.2 of ORAUT-TKBS-0007-5. Based on that approach, actinide intakes are estimated by multiplying the assessed Sr-90 and/or Cs-137 intakes by actinide-to-Sr-90 and/or actinide-to-Cs-137 ratios.

Based on the procedural information and the data on hand, NIOSH finds that it has adequate monitoring data to allow for sufficiently accurate estimation of internal doses attributable to actinides when mixed fission products were present for EBR-I Complex workers during the entire period that radiological operations took place at the EBR-I Complex when sufficient monitoring data are available.

Thorium was a concern only at the BORAX-IV reactor although thoria was sealed in fuel elements in the reactor core. There was the potential for very small intakes of thorium resulting from release of radioactivity with the BORAX-IV fuel failure testing conducted in 1958; however, no alpha radiation was detected from monitoring conducted during the events and post-events experiments. Potential thorium intakes during the testing conducted in 1958 can be bounded using thorium-to-fission-product ratios (Borax-IV Test, 1959).

Uranium without Mixed Fission Products Present

The potential for uranium exposures was minimal because encapsulated fuel sources were used at the EBR-I Complex reactors. EBR-I Health Physics used air monitoring data and/or contamination smear data to control potential uranium exposure at these reactors. There was potential for very small intakes of uranium resulting from release of radioactivity during the BORAX-IV fuel-failure testing conducted in 1958; however, no alpha radiation was detected from monitoring conducted during the events and post-events experiments. Uranium intakes during the 1958 testing can be bounded using uranium equivalence of fission-gas release ratios derived from fuel analysis given in reference: Borax-IV Test, 1959. A small potential existed for exposure to depleted uranium used as blanket material in the ZPR-III mock-ups. However, the low-specific-activity of the material, small contamination levels when discovered, and rigorous contamination monitoring practices at ZPR-III

make those exposures highly unlikely. Because there are inadequate uranium bioassay data to estimate the potential intakes of uranium without mixed fission products at the EBR-I Complex after 1957, NIOSH will bound the potential intakes of uranium by assigning uranium intakes based on 10% of the maximum permissible concentration (MPC) that ANL-W was using. A review of the available air monitoring data indicated that the airborne alpha radioactivity at the EBR-I Complex was typically below 10% of the MPC. Because it cannot be determined when and where the workers within the EBR-I Complex worked, these uranium intakes will be assessed for all workers any time they were within the EBR-I Complex after 1957.

Plutonium without Mixed Fission Products Present

ZPR-III plutonium fuel plates and the Mark III core of EBR-I represent the major sources of plutonium during operations at the EBR-I complex. With the exception of contamination surveys of the initial plutonium fuel plates for ZPR-III, there is no evidence of plutonium contamination while plutonium fuel plates were used. Likewise, the third core at EBR-I was a plutonium core but functioned without a breach in fuel rod cladding throughout its use. At the EBR-I Complex, there are no known exposure scenarios to plutonium without mixed fission products being present.

7.2.2 Evaluation of Bounding Internal Radiation Doses for the EBR-II Complex

ATTRIBUTION: Section 7.2.2 was completed by Jason Davis, Oak Ridge Associated Universities, and Brian Gleckler, Dade Moeller, Inc. All conclusions drawn from the data regarding the EBR-II Complex were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the EBR-II Complex are explained in the associated text.

The philosophy of ANL-W's internal dose monitoring program established a routine monitoring and control program based on mixed fission products as the controlling radionuclides. That is, if the mixed fission products were controlled to permissible levels, the alpha hazards would be adequately controlled. However, potential exceptions to this were noted in Section 5.2.2

ANL-W workers participated in bioassay programs for potential intakes of mixed fission products by either urinalysis or whole-body counts, either routinely or when workplace indicators (e.g., elevated airborne radioactivity, contamination event, positive nasal smear) indicated that an intake may have occurred. When workplace indicators indicated that an intake may have occurred, "special" (non-routine) bioassay would be requested by ANL-W Health Physics.

7.2.2.1 Evaluation of Bounding Process-Related Internal Doses for the EBR-II Complex

The principal sources of internal radiation doses for members of the evaluated class working at the TREAT, L&O Building, and ZPPR facilities include mixed fission and activation products, radioiodines, radioactive noble gases, plutonium, uranium, thorium, other actinides, and other radionuclides. The following subsections address the ability to bound internal doses, methods for bounding doses, and the feasibility of internal dose reconstruction.

Bioassay Information and Available Data

As previously discussed in Section 6.1, there is significant programmatic information about the ANL-W's bioassay program, and after 1957, there are extensive *in-vitro* and *in-vivo* bioassay data that NIOSH believes are sufficient for bounding internal radiation doses from most EBR-II Complex radionuclides. The majority of the available bioassay data include: gross beta radioactivity in urine, gross gamma radioactivity in urine, Sr-90 in urine, and whole-body count results.

Air Monitoring Information and Available Data

As previously discussed in Section 6.3, there is significant programmatic information about the ANL-W's air monitoring programs, and there are extensive air sampling data that NIOSH believes are sufficient for bounding internal radiation doses from the radionuclides that could not be bounded using the available bioassay data (e.g., thorium, uranium without fission products, etc.). The majority of the available air sampling data includes both gross beta-gamma radioactivity and gross alpha radioactivity results for each air sample. In addition, a significant percentage of the air samples were counted more than once for gross alpha radioactivity, and the subsequent results were often lower, suggesting that the bulk of the detected activity was due to short-lived, naturally-occurring radon decay products.

Alternative Data Sources for Bounding Internal Dose

As previously discussed in Section 6.3, there is significant programmatic information about the ANL-W's contamination monitoring programs, and there are extensive contamination smear data that could be used to support bounding potential doses to certain radionuclides. Although available, NIOSH does not have any current plans to use these data.

7.2.2.2 Methods for Bounding Operational Period Internal Doses for the EBR-II Complex

Fission and Activation Products

The potential intakes of mixed fission and activation products for monitored and unmonitored EBR-II Complex workers can be estimated using the approaches described in ORAUT-TKBS-0007-5. For urine samples analyzed for gross beta, gross gamma, and/or strontium radioactivity, NIOSH will assess Sr-90 and/or Cs-137 intakes in accordance with ORAUT-TKBS-0007-5 and ORAUT-OTIB-0060. For whole-body counts, NIOSH will assess Cs-137 intakes in accordance with ORAUT-TKBS-0007-5 and ORAUT-OTIB-0060. In accordance with the guidance in ORAUT-OTIB-0054, NIOSH will use Sr-90 and/or Cs-137 intakes to estimate the intakes of the other mixed fission and activation products that may have been present, unless a bioassay result is available for a specific radionuclide. In the event that a bioassay result is available for a specific radionuclide, NIOSH will use that result in lieu of the ORAUT-OTIB-0054 approach.

Based on the procedural information and the data on hand, NIOSH finds that it has adequate monitoring data to allow for sufficiently accurate estimation of internal doses attributable to mixed fission and activation products for EBR-II Complex workers during the entire period that radiological operations took place at the EBR-II Complex.

Radioiodines

Based on the availability of the gross gamma urine sample data and whole-body count data for EBR-II Complex workers, NIOSH finds that it has adequate monitoring data to allow for sufficiently accurate estimation of internal doses attributable to radioiodines for all periods at the EBR-II Complex.

The potential intakes of radioiodines without the other mixed fission products present can be estimated for monitored EBR-II Complex workers using the bioassay data and guidance provided in Section 5.5.4 of ORAUT-TKBS-0007-5. When the radioiodines are present with the other mixed fission products, the potential intakes of radioiodines for monitored EBR-II Complex workers can be accounted for using the bioassay data and guidance provided in Section 5.5.1 of ORAUT-TKBS-0007-5, which employs the use of ORAUT-OTIB-0054.

Based on the available information, the presence of airborne radioiodine without the other mixed fission products present was uncommon at the EBR-II Complex, and unmonitored intakes of radioiodines likely only occurred when the other mixed fission products were present. Therefore, potential intakes of radioiodines are estimated for unmonitored EBR-II Complex workers using the guidance provided in Section 5.6 of ORAUT-TKBS-0007-5.

Radioactive Noble Gases

ANL-W was not involved in the mining, milling, or separation of uranium or thorium from ores. Fuels arrived at the side pre-formed and extruded and did not age to the point where radon or thoron would have been a significant concern. Therefore, radon and thoron are not considered a significant internal exposure concern. Because radioactive noble gases do not represent a significant internal exposure concern, NIOSH finds there is no need to assess the internal doses attributable to radioactive noble gases.

Actinides with Mixed Fission Products Present

With a few exceptions, the majority of the actinides at the EBR-II Complex were present with mixed fission products. The potential intakes of actinides when mixed fission products were present can be estimated using the approach described in Section 5.5.2 of ORAUT-TKBS-0007-5. Based on that approach, actinide intakes are estimated by multiplying the assessed Sr-90 and/or Cs-137 intakes by actinide-to-Sr-90 and/or actinide-to-Cs-137 ratios.

Based on the procedural information and the data on hand, NIOSH finds that it has adequate monitoring data to allow for sufficiently accurate estimation of internal doses attributable to actinides when mixed fission products were present for EBR-II Complex workers during the entire period that radiological operations took place at the EBR-II Complex.

Uranium without Mixed Fission Products Present

The potential for uranium exposures within ZPPR was minimal because encapsulated and coated fuel sources were used (SRDB 138139). ZPPR Health Physics used air monitoring data and/or contamination smear data to control potential uranium exposure at this reactor during normal operations. Loading and unloading procedures were carefully monitored to detect any loose

contamination on fuel plates. Exposures to uranium without mixed fission products can be bound using 10% of MPC (maximum permissible concentration) from available air monitoring data.

Because of FCF Hot-Line start-up activities, FCF machine shop activities, and Cold-Line fuel production at the FCF, ITF, and FASB, more-than-incidental intakes of depleted and enriched uranium without mixed fission products present could have occurred from August 1967 to as late as 1994. For the period of August 1967 through June 1983, no uranium bioassay data could be found for ANL-W workers. Beginning in July 1983, there is a significant increase in uranium bioassays for ANL-W workers, and the bioassay data can be used to estimate the intakes and doses from these uranium exposures.

Because there are inadequate uranium bioassay data to estimate the potential intakes of depleted and enriched uranium without mixed fission products during the period of August 1967–June 1983, NIOSH will estimate those intakes using the gross alpha radioactivity air sampling results for those areas. In the instances where the air samples were counted for alpha radioactivity more than once due to the presence of short-lived alpha-emitting radionuclides, the latest result for gross alpha radioactivity will be used for this approach, since U-234, U-235, and U-238 are long-lived alpha-emitting radionuclides. Because it cannot be determined when and where the workers within the EBR-II Complex worked, these uranium intakes will be assessed for all workers any time they were within the EBR-II Complex during the period of August 1967–June 1983. For periods that workers were only assigned EBR-I Complex dosimeters, no additional uranium intakes will be assessed using this approach.

As a simplification and because the potential exposures to depleted uranium and enriched uranium were intermixed, all alpha radioactivity in the air sample results, along with the resulting inhalation intake estimates, will be assessed as 100% U-234. For the inhalation rate, a rate of 1.2 m^3/hr (9.6 m^3 per 8 hr workday) will be assumed (ICRP 66, 1994). Because air sampling data can only be used to estimate potential intakes due to inhalation, potential intakes due to ingestion will be estimated based on the recommendations in the Technical Information Bulletin: Estimation of Ingestion Intakes (OCAS-TIB-009). Because the uranium that was processed at the EBR-II Complex may have been recycled uranium, and because the Y-12 Plant was the likely source of the high-enriched uranium metal feedstock being used to produce the EBR-II Reactor fuel, intakes of the potential radioactive impurities in the uranium will be calculated by multiplying the inhalation and ingestion uranium intakes by the values in Table 7-5.

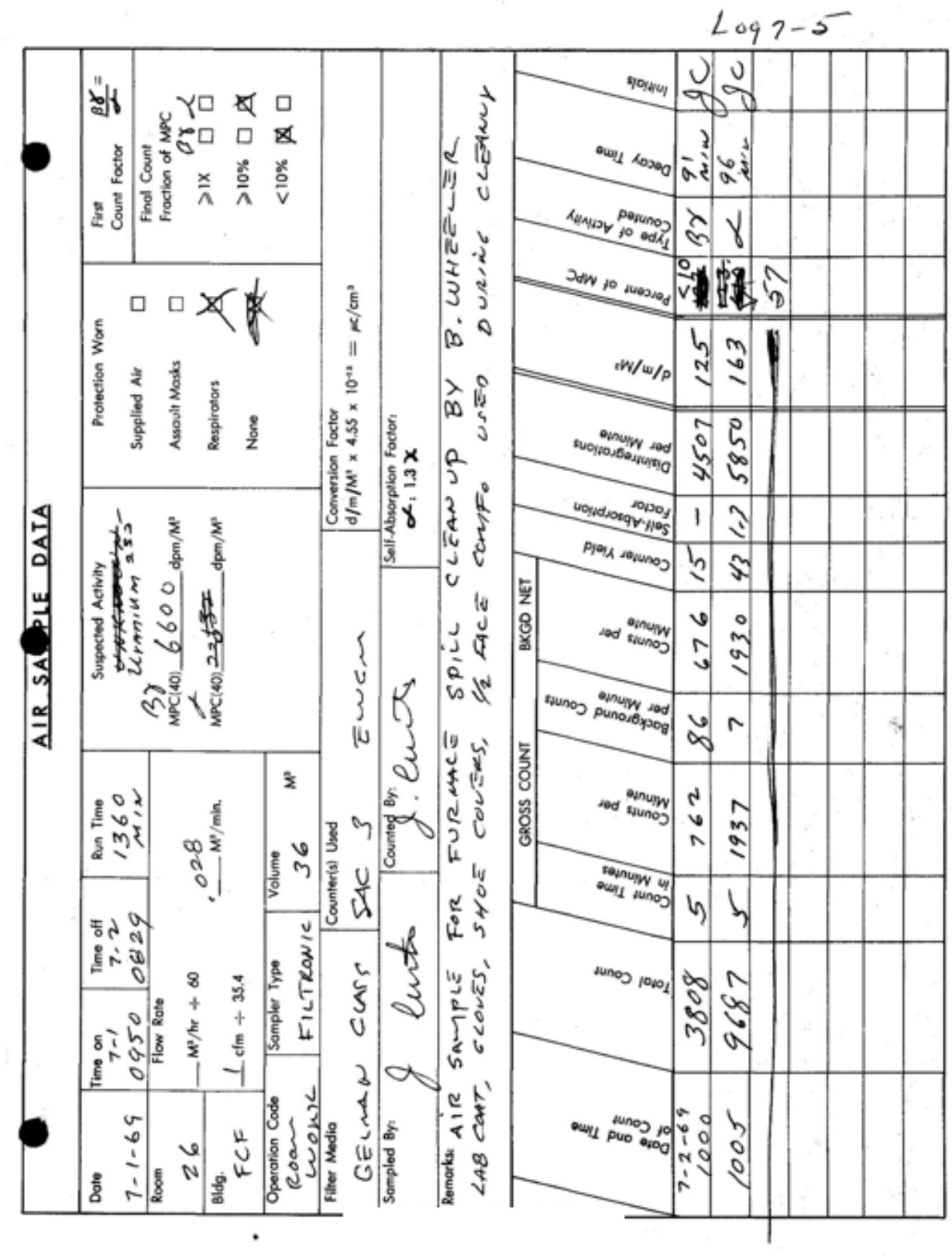
Table 7-5: Default Ratios for Recycled Uranium Intakes

Radionuclide	Default ratio (pCi/pCi total U)
Tc-99	1.3E-02
Th-228	5E-04
Np-237	6E-04
Pu-238	2E-04

Note: The values in this table are based on the values provided in Table 5-8 of the Y-12 Plant Occupational Internal Dose TBD (ORAUT-TKBS-0014-5). The values have also been adjusted in accordance with that TBD for use on all types of cases, including best estimate cases.

Figure 7-1 provides an example of a 1969 air sample result for the FCF Cold-Line. During the collection period for this sample, some uranium was spilled from the melt-refining process and cleaned up in Room 26 (Survey Reports, July1969).

NOTE: The potential uranium exposures and available Cold-Line air monitoring for the period of January 1970 – January 1972 were lower because no Cold-Line fuel pin and fuel element production took place at ANL-W during that period. At the end of December 1969, the Cold-Line fuel pin and fuel element production equipment in FCF Rooms 20 and 26 was put on stand-by (Progress Report, Dec1969, PDF p. 88). However, the Cold-Line continued to fabricate unirradiated subassemblies from vendor-produced fuel elements (at ITF) and to perform impact-bonding of the unbonded vendor fuel elements (in FCF Room 23) (Progress Report, Dec1969, PDF p. 88). The FCF's Cold-Line fuel pin and fuel element production was likely shut down due to a large backlog of completed fuel elements, and given that ANL-W was beginning to receive large numbers of acceptable vendor- produced fuel elements. In January 1970, there were a total of 23,766 completed fuel elements ready to be put into EBR-II Reactor subassemblies (Progress Report, Jan1970, PDF p. 78). In the mid-1970s, the FCF's Cold-Line fuel pin and fuel element production equipment was deactivated (Batte, 1986, PDF p. 9). Fuel pin fabrication in FCF Room 20 was not re-initiated until 1982 (Batte, 1986, PDF p. 9). Given that Cold-Line fuel production in FASB did not begin until February 1972, no Cold-Line fuel pin and fuel element production took place at ANL-W during the period of January 1970 – January 1972.



Source: Survey Reports, July1969

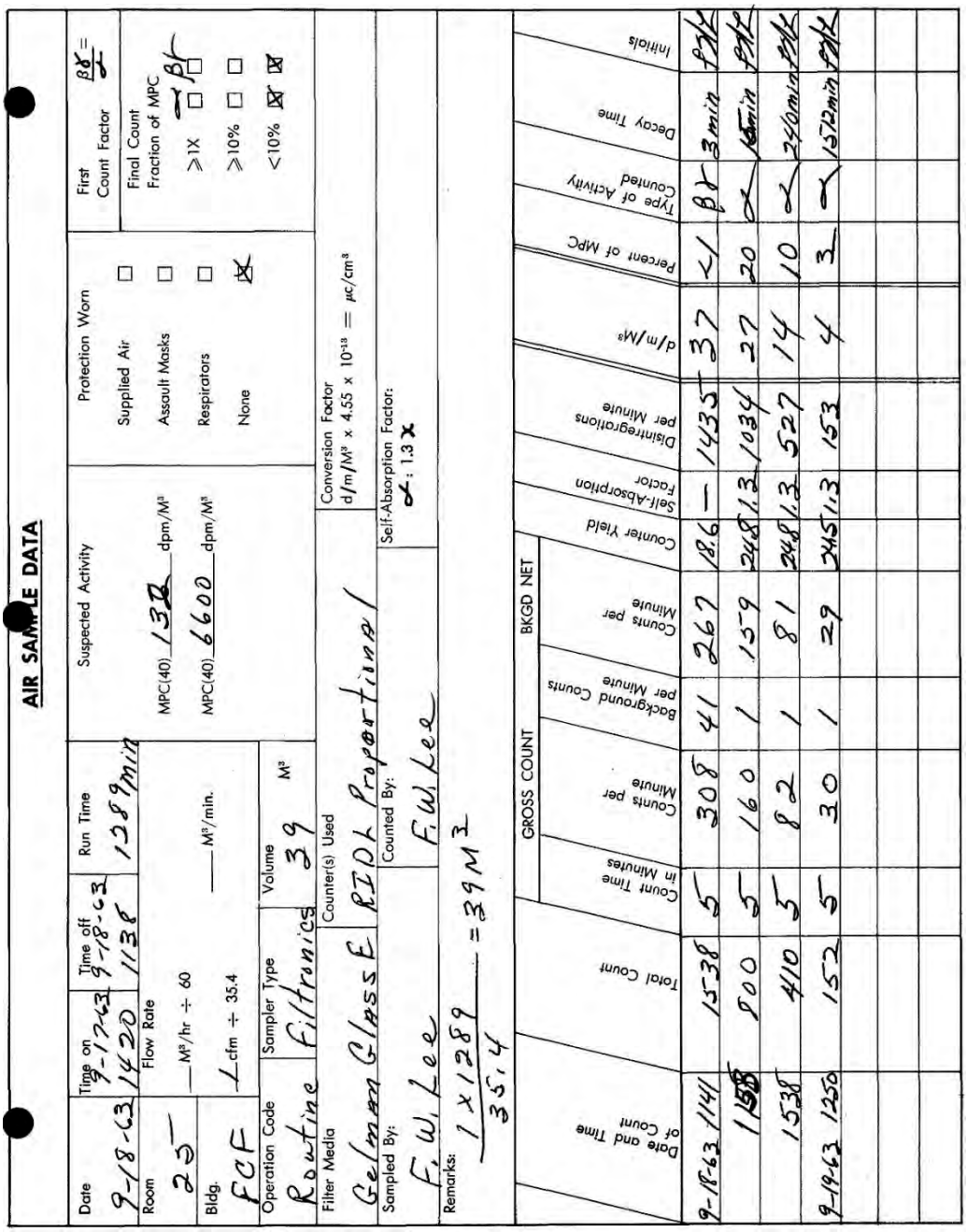
Figure 7-1: Example FCF Cold-Line Air Sample Result from 1969

Thorium without Mixed Fission Products Present

During the period of FCF Hot-Line fuel production (August 1963–November 1967), more than incidental intakes of thorium without mixed fission products present could have occurred in FCF Room 25 due to the use of thoria to coat crucibles and molds for the FCF's Hot-Line. By December 1967, the use of thoria as a refractory coating was discontinued, which eliminated the potential for thoria intakes after November 1967. Because there are inadequate thorium bioassay data to estimate the potential intakes of thorium without mixed fission products during the period of August 1963–November 1967, NIOSH will bound the potential intakes of thorium by assigning thorium intakes based on 10% of the maximum permissible concentration (MPC) that ANL-W was using. A review of the available air monitoring data indicated that the airborne alpha radioactivity in FCF Room 25 was typically below 10% of the MPC. Because it cannot be determined when and where the workers within the EBR-II Complex worked, these thorium intakes will be assessed for all workers any time they were within the EBR-II Complex during the period of August 1963–November 1967.

Because nearly all thorium deposits contain some uranium, Th-230 from the uranium series will be extracted with the other thorium isotopes. Therefore, any thorium will consist of the isotopes of Th-232, Th-228, and Th-230. Between Th-232, Th-228, and Th-230, Th-232 has the highest internal dose coefficient for all organs. Therefore, 100% of the gross alpha radioactivity in the air sample results will be attributed to Th-232 for the internal dose calculations. Inhalation intakes will be calculated using an inhalation rate of 1.2 m^3/hr (9.6 m^3 per 8 hr workday) (ICRP 66, 1994). Because air sampling data can only be used to estimate potential intakes due to inhalation, potential intakes due to ingestion will be estimated based on the recommendations in the Technical Information Bulletin: Estimation of Ingestion Intakes (OCAS-TIB-009).

Figure 7-2 provides an example of a 1963 air sample result for FCF Room 25. During the collection period for this sample, some thoria was spilled on the floor from one of the hoods in Room 25 (FCF Survey Reports, 1963).



AIR SAMPLE DATA

Date: 9-18-63 | Time on: 9-17-63 1420 | Time off: 9-18-63 1138 | Run Time: 1389 min

Room: 25 | Flow Rate: ___ M³/hr ÷ 60 ___ M³/min. | 1 cfm ÷ 35.4

Bldg.: FCF | Operation Code: Routine | Sampler Type: Filtrones | Volume: 39 M³

Filter Media: Gelman Glass E | Counter(s) Used: RIDL Proportional

Sampled By: F. W. Lee | Counted By: F. W. Lee | Self-Absorption Factor: α: 1.3 X

Remarks: 1 x 1289 = 39M³ 35.4

Suspected Activity: MPC(40) 138 dpm/M³ | MPC(40) 6600 dpm/M³

Protection Worn: Supplied Air ☐ | Assault Masks ☐ | Respirators ☐ | None ☒

Conversion Factor d/m/M³ x 4.55 x 10⁻¹³ = μc/cm³

First Count Factor: BK =

Final Count Fraction of MPC: ≥1X ☐ BK | ≥10% ☐ | <10% ☒

Date and Time of Count	Total Count	Count Time in Minutes	GROSS COUNT Counts per Minute	BKGD NET Background Counts per Minute	Counts per Minute	Counter Yield	Self-Absorption Factor	Disintegrations per Minute	d/m/M³	Percent of MPC	Type of Activity Counted	Decay Time	Initials
9-18-63 1141	1538	5	308	41	267	18.6	—	1435	37	7	β	3 min	PBL
1558	800	5	160	1	159	248	1.3	1034	27	20	✓	15 min	PBL
1538	410	5	82	1	81	248	1.3	527	14	10	✓	24 min	PBL
9-19-63 1250	152	5	30	1	29	245	1.3	153	4	3	✓	1512 min	PBL

Source: FCF Survey Reports, 1963

Figure 7-2: Example FCF Room 25 Air Sample Result from 1963

Plutonium without Mixed Fission Products Present

The potential for plutonium exposures within ZPPR was minimal because encapsulated and coated fuel sources were used (SRDB 138139). ZPPR Health Physics used air monitoring data and/or contamination smear data to control potential plutonium exposure at this reactor during normal operations. Loading and unloading procedures were carefully monitored to detect any loose

contamination on fuel plates. Exposures to plutonium without mixed fission products can be bound using 10% of MPC (maximum permissible concentration) from available air monitoring data.

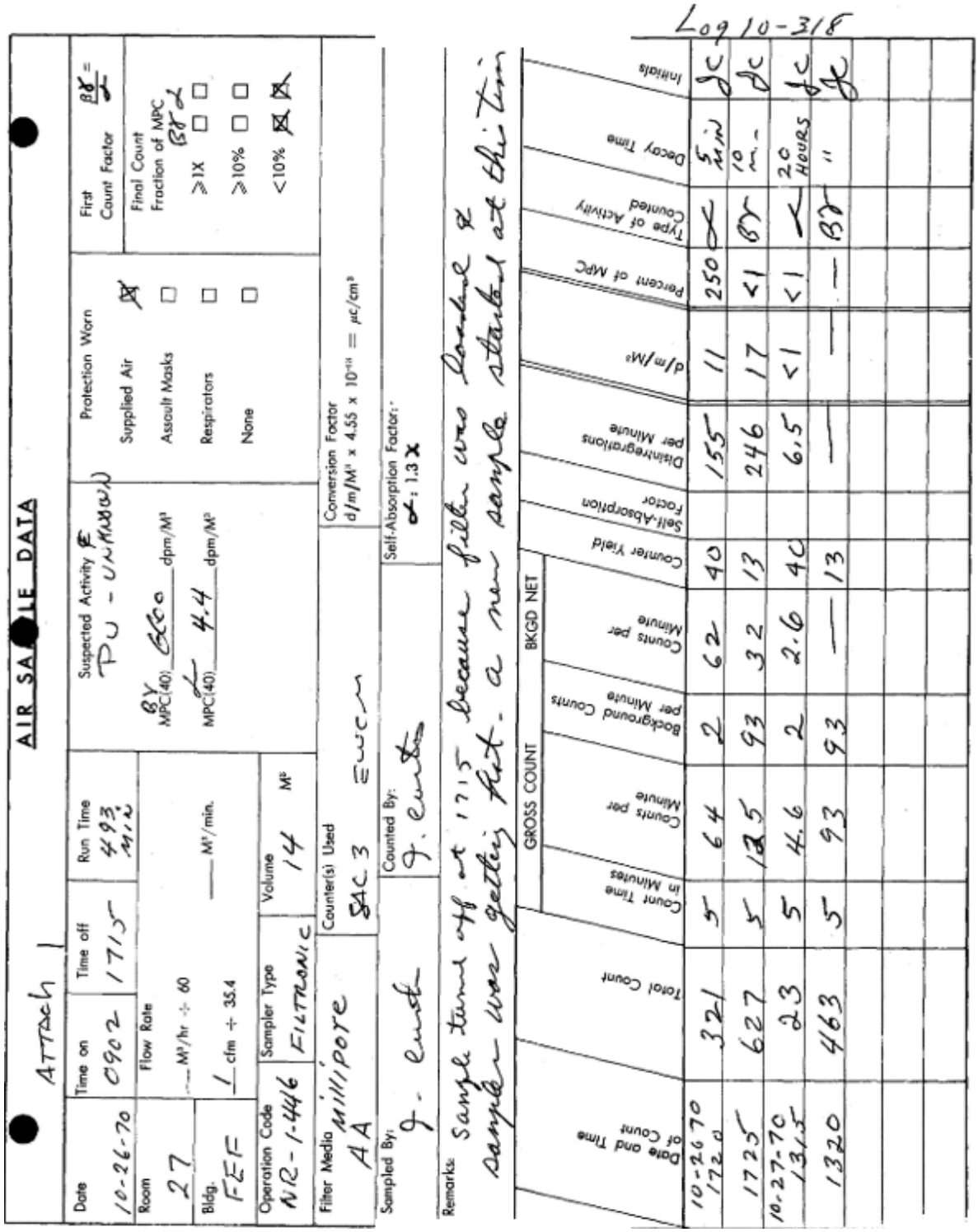
Intermittent work associated with the Mark-II type RAS-TREAT sodium-loop experiments at the FCF could have resulted in potential intermittent exposures to plutonium without mixed fission products present when that work included the removal of the test sections and when that work was performed outside of the Air and Argon Cells. As indicated in Section 5.1.2.3, the test section removal work took place outside of the FCF hot cells during the period of April 1970–December 1972, and was performed mostly in FCF Rooms 22 and 27. After 1972, RAS-TREAT sodium-loop experiment work appears to have been performed exclusively in the Argon Cell, and the experiments were irradiated to much higher levels. Therefore, potential plutonium exposures after 1972 would have occurred with mixed fission products being present (Cook, 1975).

Because there isn't adequate plutonium bioassay data to estimate the potential intakes of plutonium without mixed fission products during the period of April 1970–December 1972, NIOSH will estimate those intakes using the gross alpha radioactivity air sampling results that were collected during RAS-TREAT sodium-loop experiment work in FCF Rooms 22 and 27. In the instances where the air samples were counted for alpha radioactivity more than once due to the presence of short-lived alpha-emitting radionuclides, the latest result for gross alpha radioactivity will be used for this approach, since isotopes of plutonium are long-lived alpha-emitting radionuclides. Because it cannot be determined when and where the workers within the EBR-II Complex worked, these plutonium intakes will be assessed for all workers any time they were within the EBR-II Complex during the period of April 1970–December 1972.

The plutonium-bearing Mark-II type loops that were handled at the FCF contained experimental mixed oxide (MOX) fuels that were mostly enriched UO_2 with a lesser quantity of PuO_2 (FCF Rad Survey, 1970, PDF pp. 208-211; Cook, 1975; Dickerman, 1979). The plutonium in these fuels was predominately Pu-239. Because Pu-239 has a significantly shorter half-life than the isotopes in enriched uranium, the alpha radioactivity in these experimental uranium-plutonium fuels would have been dominated by plutonium, even though more uranium was present in terms of mass. Given that, and because the exact composition of each of the experimental fuels is uncertain, 100% of the gross alpha activity in the air samples will be assumed to be Pu-239. The assumption of 100% of the alpha radioactivity in the air samples being attributed to Pu-239 would result in a bounding dose for exposures to these experimental fuels.

Using the records already available to NIOSH, if it can be determined when each plutonium-bearing loop experiment test section was handled in FCF Rooms 22 and 27, acute plutonium intakes for each specific period of that work will be estimated from the available air sample data. Otherwise, chronic intakes will be assessed as a bounding approach. Inhalation intakes will be calculated using an inhalation rate of 1.2 m^3/hr (9.6 m^3 per 8 hr workday) (ICRP 66, 1994). Because air sampling data can only be used to estimate potential intakes due to inhalation, potential intakes due to ingestion will be estimated based on the recommendations in the Technical Information Bulletin: Estimation of Ingestion Intakes (OCAS-TIB-009).

Figure 7-3 provides an example of a 1970 air sample result for FCF Room 27 that was collected during some RAS-TREAT sodium-loop experiment work (Survey Reports, Aug1970).



Source: Survey Reports, Aug1970, PDF p. 111

Figure 7-3: Example Air Sample Result for FCF RAS-TREAT Sodium-Loop Work from 1970

Other Radionuclides

Even though EBR-II Complex workers rarely received bioassay measurements for H-3 and/or C-14 during the period of 1949–1979, these two radionuclides were likely only present where mixed fission and activation products were present. Furthermore, the source terms for H-3 and C-14 and their ability to contribute to the workers' internal doses were likely insignificant compared to the workers' exposures to mixed fission and activation products.

The potential intakes of Na-24 can be estimated for monitored and unmonitored EBR-II Complex workers using the guidance provided in Section 5.5.8 of ORAUT-TKBS-0007-5. Urine samples analyzed for gross gamma radioactivity and whole-body counts can be used to assess any potential intakes of Na-24. Based on the availability of the gross beta urine sample data, gross gamma urine sample data, and whole-body count data for EBR-II Complex workers, NIOSH finds that it has adequate monitoring data to allow for sufficiently accurate estimation of internal doses attributable to Na-24 for EBR-II Complex workers the entire period that radiological operations took place at the EBR-II Complex.

7.2.3 Internal Dose Reconstruction Feasibility Conclusion

ATTRIBUTION: Section 7.2.3 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

NIOSH has come to the following conclusions regarding the feasibility of internal dose reconstruction for the EBR-I Complex facilities and EBR-II Complex facilities under evaluation for this report:

- NIOSH concludes that, due to missing records, the available bioassay data are such that internal radiation doses received from intakes of fission products and actinides cannot be completely reconstructed with sufficient accuracy from April 10, 1951 through December 31, 1957.

- NIOSH concludes that the available bioassay data are such that internal radiation doses received from intakes of fission products and actinides can be completely reconstructed with sufficient accuracy from January 1, 1958 through December 31, 1979.

Although NIOSH found that it is not possible to completely reconstruct internal radiation doses for the period from April 10, 1951 through December 31, 1957, NIOSH intends to use any internal monitoring data that may become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures). Dose reconstructions for individuals employed at ANL-W during the period from April 10, 1951 through December 31, 1957, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate.

7.3 Evaluation of Bounding External Radiation Doses at ANL-W

The principal source of external radiation doses for members of the evaluated class was a combination of direct exposures from nuclear reactors, and exposures to the various radionuclide source terms at the site, predominately mixed fission products, activation products, and uranium (ORAUT-TKBS-0007-6). See Section 5.0 for a more detailed discussion of the radionuclides that are known to have been present.

The following subsections address the ability to bound external doses, methods for bounding doses, and the feasibility of external dose reconstruction.

7.3.1 Evaluation of Bounding Process-Related External Doses

ATTRIBUTION: Section 7.3.1 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

The following subsections summarize the extent and limitations of information available for reconstructing the process-related external doses of members of the class under evaluation.

Personnel Dosimetry Data

NIOSH has captured external monitoring data on ANL-W workers for the entire period of this evaluation. However, external monitoring data available from 1952 through 1957 appears to be incomplete. NIOSH reviewed external monitoring data supplied for the 50 former workers discussed in Section 6.1 for the period 1951 through 1960. Table 7-6 shows the results of this review. NOTE: Not all 50 workers were employed during the entire period from 1952 through 1960. Table 7-7 provides the external monitoring data for 12 workers who were known to be at ANL-W for all years from 1952 through 1960, and thus, represents a better assessment of the completeness of records by year.

Table 7-6: Available External Monitoring Data for 50 Selected ANL-W Workers

Year	Subset of 50 Selected ANL-W Workers with Available External Monitoring Data
1951	18
1952	32
1953	32
1954	32
1955	39
1956	16
1957	18
1958	27
1959	28
1960	20

External monitoring results available for the same twelve former workers used for Table 7-3 are given in Table 7-7 where "partial" denotes that results for only some of the year are available to NIOSH.

Table 7-7: Available External Monitoring Data for 12 Selected Workers (1952-1960)

Worker	1952	1953	1954	1955	1956	1957	1958	1959	1960
1	Partial	complete	complete	partial	none	none	complete	complete	complete
2	Partial	complete	complete	partial	none	none	none	none	none
3	Partial	complete	complete	partial	none	none	complete	complete	complete
4	Partial	complete	complete	partial	partial	partial	complete	complete	complete
5	Partial	complete	complete	partial	partial	partial	complete	complete	complete
6	Partial	complete	complete	partial	partial	partial	complete	complete	complete
7	Partial	complete	complete	partial	partial	partial	complete	complete	complete
8	Partial	complete	complete	partial	partial	partial	complete	complete	complete
9	Partial	complete	complete	partial	partial	partial	complete	complete	complete
10	Partial	complete	complete	partial	partial	partial	partial	complete	complete
11	Partial	complete	complete	partial	partial	partial	complete	complete	complete
12	Partial	complete	complete	partial	partial	partial	complete	complete	complete

Electron and Photon Dosimetry Data

Because all workers in ANL-W's radiological areas were monitored for exposures to photon radiation, and because the DOE provides workers' dosimeter data to NIOSH for dose reconstructions, the external electron and photon doses for the monitored workers after 1957 can be reconstructed using the guidance in ORAUT-TKBS-0007-6. Because complete sets of dosimetry records are not always available for the years prior to 1958, not all of the workers' doses for the years of 1951–1957 can be reconstructed. Because the control dosimeters at ANL-W were likely exposed to elevated levels of onsite ambient (environmental) external radiation due to being located at the entrances of the major operating areas, onsite ambient external doses also need to be assigned to all monitored ANL-W workers to account for any radiation dose that may have been inappropriately subtracted from their dosimeter results as background radiation (ORAUT-TKBS-0007-6).

For unmonitored workers, onsite ambient (environmental) external doses may be assigned. The onsite ambient external doses are limited to photon doses because the unmonitored workers would not have been close enough to the radioactive source terms to receive an external electron dose.

Neutron Dosimetry Data

The majority of the ANL-W workers were not monitored for neutron dose. However, the available information indicates that neutron doses were unlikely and that neutron radiation fields are specific to a few facilities (ORAUT-TKBS-0007-6). During the period under evaluation, ORAUT-TKBS-0007-6 indicates that the only location where neutron exposures were possible were at EBR-I, TREAT, ZPPR, and HFEF-N's NRAD Facility (ORAUT-TKBS-0007-6). Recently-captured information, such as the references for Section 5.0, indicates that neutron calibration/check sources (e.g., Ra-Be, Po-Be, Pu-Be, Am-Be, Cf-252) were present at most of the major facilities at one time or another. Therefore,

anyone working near those sources while they were unshielded would have also been exposed to neutron radiation.

Informal observations/reviews of the ANL-W's neutron dosimetry data, conducted while performing dose reconstructions and while capturing records at the ANL-W site, indicate that the neutron doses for the monitored workers are typically less than the dosimeters' limits of detection. An exception may have been EBR-I, due to the limitations in the sensitivity of NTA (nuclear track emulsion Type A). In the ANL-W claims, career neutron doses of 500 mrem or greater are rare. The available information also indicates that ANL-W investigated neutron exposures to unmonitored workers and estimated doses for those workers. Given that the ANL-W likely monitored the workers with the highest potential to receive neutron doses, and given that the monitored workers' neutron doses were typically less than the dosimeters' limits of detection, it is unlikely that an unmonitored ANL-W worker received more than an incidental exposure to neutron radiation.

Because unmonitored neutron doses were unlikely at the ANL-W, and because DOE provides the workers' dosimeter data to NIOSH for dose reconstructions, neutron doses for the workers can be reconstructed using the guidance in ORAUT-TKBS-0007-6.

7.3.2 Evaluation of Bounding Ambient Environmental External Doses

ATTRIBUTION: Section 7.3.2 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

Table 4-13 of ORAUT-TKBS-0007-4 provides the onsite ambient (environmental) external doses from 1952 through the end of the period under evaluation. For the year 1951, the onsite ambient doses can be bounded by assigning the onsite ambient doses being assigned for the years 1952–1972 because radiological operations prior to 1952 were minimal compared to 1952 and later (ORAUT-TKBS-0007-2).

Because all workers in the ANL-W's radiological areas were monitored for external dose, and because DOE provides workers' dosimeter data to NIOSH for dose reconstructions, the only doses that need to be assigned for unmonitored workers at ANL-W are onsite ambient external doses. Because the control dosimeters at ANL-W were likely exposed to elevated levels of onsite ambient radiation due to being located at the entrances of the major operating areas, onsite ambient doses are also assigned to all monitored ANL-W workers to account for any radiation dose that may have been inappropriately subtracted from their dosimeter results as background radiation (ORAUT-TKBS-0007-6).

7.3.3 ANL-W Occupational X-Ray Examinations

NIOSH has sufficient information through documents in the SRDB and ORAUT-TKBS-0007-3 to support the ability to bound the occupational medical doses for the evaluated class.

7.3.4 Methods for Bounding External Dose at ANL-W

There is an established protocol for assessing external exposure when performing dose reconstructions (these protocol steps are discussed in the following subsections):

- Electron and Photon Doses
- Neutron Dose
- Occupational Medical Dose
- Ambient Environmental External Dose

7.3.4.1 Methods for Bounding Operational Period External Dose

Electron and Photon Doses

While the primary method is the use of individual external dosimetry records provided by DOE, it was determined that external doses prior to 1958 could not be adequately reconstructed due to missing records. However, dosimeter exchange frequencies and thresholds are known and documented. Missed dose may be applied using that information after 1958. The necessary biases/corrections that need to be made to the worker's electron and photon dosimetry results are provided in ORAUT-TKBS-0007-6. To account for unmonitored radiation exposures, onsite ambient external doses can be assigned in accordance with the guidance in ORAUT-TKBS-0007-4 and ORAUT-PROC-0060.

Neutron Dose

While the primary method is the use of individual external dosimetry records provided by DOE, it was determined that external doses prior to 1958 could not be adequately reconstructed due to missing records. However, dosimeter exchange frequencies and thresholds are known and documented. Missed neutron dose may be applied using that information after 1958. The necessary biases/corrections that need to be made to the worker's neutron dosimetry results are provided in ORAUT-TKBS-0007-6. Because unmonitored neutron doses were unlikely at ANL-W, unmonitored neutron doses do not need to be assigned.

Occupational Medical Dose

NIOSH has determined that occupational medical doses can be reconstructed. NIOSH has sufficient information in documents in the SRDB and ORAUT-TKBS-0007-3 to support the ability to bound occupational medical doses for the evaluated class.

7.3.4.2 Methods for Bounding Ambient Environmental External Doses

NIOSH has determined that ambient environmental external doses can be reconstructed. The ambient environmental external doses can be reconstructed using the guidance in ORAUT-TKBS-0007-4 and ORAUT-PROC-0060.

7.3.5 External Dose Reconstruction Feasibility Conclusion

ATTRIBUTION: Section 7.3.5 was completed by Mike Mahathy and Jason Davis, Oak Ridge Associated Universities, Brian Gleckler, Dade Moeller, Inc., and Mitch Findley, MJW Corporation. All conclusions drawn from the data regarding the specified facilities were peer-reviewed by the individuals listed on the cover page. The rationale for all conclusions in subsequent sections of this document devoted to the specified facilities are explained in the associated text.

NIOSH has reached the following conclusions regarding the feasibility of external dose reconstruction:

- External dose for external electron and photon exposures cannot be reconstructed for ANL-W workers from April 10, 1951 through December 31, 1957. External dose from electron and photon exposures can be reconstructed with sufficient accuracy from January 1, 1958 through December 31, 1979.

- External dose for neutron exposures cannot be reconstructed for ANL-W workers from April 10, 1951 through December 31, 1957. External dose from neutron exposures can be reconstructed with sufficient accuracy from January 1, 1958 through December 31, 1979.

- Occupational medical dose for occupational X-ray exposures can be reconstructed for all members of the evaluated ANL-W class for the entire period under evaluation.

- Ambient environmental external dose for environmental external exposures can be reconstructed for all applicable members of the evaluated ANL-W class during the entire period under evaluation.

Although NIOSH found that it is not possible to completely reconstruct external radiation doses for the period from April 10, 1951 through December 31, 1957, NIOSH intends to use any external monitoring data that may become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures). Dose reconstructions for individuals employed at ANL-W during the period from April 10, 1951 through December 31, 1957, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate.

7.4 Evaluation of Petition Basis for SEC-00224

The following assertion was made on behalf of petition SEC-00224 for ANL-W:

SEC-00224: *There was no internal monitoring for plutonium, neptunium, or fission products at ANL-W between 1949 and 1979.*

NIOSH has determined that internal monitoring of personnel at ANL-W focused on fission products from the beginning of radiological operations at the site. *In-vitro* bioassay began in 1952 with gross beta and gross beta/gamma analyses and a single uranium analysis. For ANL-W, 1961 marked the beginning of *in-vivo* bioassay in the form of whole-body counting for gamma-emitting radionuclides that are common in fission products. *In-vitro* and *in-vivo* measurements by year are presented in Section 6.1.

During the course of this evaluation, it was determined that internal monitoring for fission products was adequately performed, and that internal monitoring for plutonium and neptunium, while limited, was also adequately performed. Based on the information available regarding ANL-W operations, it is apparent that Np was only present in relatively small quantities within the irradiated reactor fuels and was also present with other actinides and fission products. No bioassay monitoring data specific to Np exposures has been identified during the research performed for this ER. Based on the available information, neptunium (and other actinides) were never present at the EBR-II Complex without mixed fission products also being present. Potential dose from neptunium will be accounted for by using the actinide ratios in the INL – ANL-W Occupational Internal Dose TBD (ORAUT-TKBS-0007-5).

For the years prior to 1958, not all of the external dosimetry and bioassay records are available. Due to inadequate availability of the early external dosimetry and bioassay records, NIOSH is recommending the period from April 10, 1951 through December 31, 1957 at ANL-W for inclusion in the SEC. After 1957, the ANL-W bioassay records being provided by the INL appear to be complete. Even though bioassay monitoring for exposures to actinides other than plutonium and neptunium (e.g., uranium and thorium) was not adequately performed, those intakes can be estimated using other types of monitoring data (e.g., air monitoring data, contamination surveys, etc.).

7.5 Other Potential SEC Issues Relevant to the Petition Identified During the Evaluation

During the feasibility evaluation for SEC-00224, an issue was identified that needs further analysis and resolution. The issue and its current status is:

ISSUE: Verification and validation (V&V) of bioassay data.

RESPONSE: It is firmly believed that the *in-vivo* and *in-vitro* bioassay datasets for ANL-W/INL are complete. However, a V&V has not been finished at this time and will be performed as part of the development of a future INL internal co-worker study.

7.6 Summary of Feasibility Findings for Petition SEC-00224

This report evaluates the feasibility for completing dose reconstructions for employees at the ANL-W from April 1951 through December 1979. NIOSH found that the available monitoring records, process descriptions and source term data available are not sufficient to complete dose reconstructions for the evaluated class of employees.

Table 7-8 summarizes the results of the feasibility findings at ANL-W for each exposure source during the time period April 10, 1951 through December 31, 1957 and from January 1, 1958 through December 31, 1979.

Table 7-8: Summary of Feasibility Findings for SEC-00224
April 10, 1951 through December 31, 1957
January 1, 1958 through December 31, 1979

Source of Exposure	April 10, 1951 through December 1, 1957 Reconstruction Feasible (Yes or No)	January 1, 1958 through December 31, 1979 Reconstruction Feasible (Yes or No)
Internal[1]	**No[1]**	**Yes**
- U	No	Yes
- Pu	No	Yes
- Th	No	Yes
- Mixed fission products	No	Yes
- Radioiodines	No	Yes
- Other transuranic radionuclides	No	Yes
- Other radionuclides	No	Yes
External	**No[2]**	Yes
- Gamma	No	Yes
- Beta	No	Yes
- Neutron	No	Yes
- Occupational Medical X-ray	Yes	Yes

PARTIAL DOSE RECONSTRUCTION INFORMATION:

[1] INTERNAL: For the purpose of partial dose reconstruction for members of the proposed SEC class who worked at ANL-W from 1951 through 1957, partial DRs could include dose from any radionuclides for which specific personal monitoring data exist that can be assessed by methods defined in approved NIOSH dose reconstruction procedures or processes.

[2] EXTERNAL: For the purpose of partial dose reconstruction for members of the proposed SEC class who worked at ANL-W from 1951 through 1957, partial DRs should include external dose based on the specific personal monitoring data that exist that can be assessed by methods defined in approved NIOSH dose reconstruction procedures or processes

As of January 15, 2016, a total of 273 claims have been submitted to NIOSH for individuals who worked at ANL-W during the period under evaluation in this report. Dose reconstructions have been completed for 237 individuals (~87%).

Although NIOSH found that it is not possible to completely reconstruct radiation doses for the proposed class, NIOSH intends to use any internal and external monitoring data that may become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures). Therefore, dose reconstructions for individuals employed at ANL-W during the period from April 10, 1951 through December 31, 1957, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate.

8.0 Evaluation of Health Endangerment for Petition SEC-00224

The health endangerment determination for the class of employees covered by this evaluation report is governed by both EEOICPA and 42 C.F.R. § 83.13(c)(3). Under these requirements, if it is not feasible to estimate with sufficient accuracy radiation doses for members of the class, NIOSH must also determine that there is a reasonable likelihood that such radiation doses may have endangered the health of members of the class. Section 83.13 requires NIOSH to assume that any duration of unprotected exposure may have endangered the health of members of a class when it has been established that the class may have been exposed to radiation during a discrete incident likely to have involved levels of exposure similarly high to those occurring during nuclear criticality incidents. If the occurrence of such an exceptionally high-level exposure has not been established, then NIOSH is required to specify that health was endangered for those workers who were employed for a number of work days aggregating at least 250 work days within the parameters established for the class or in combination with work days within the parameters established for one or more other classes of employees in the SEC.

Based on the sum of information available from available resources, NIOSH's evaluation determined that it is not feasible to estimate radiation dose with sufficient accuracy for members of the NIOSH-evaluated class for the time period from April 10, 1951 through December 31, 1957. NIOSH has determined that no ANL-W workers were directly exposed to any of the criticality experiments or incidents/accidents that occurred at ANL-W. Therefore, the resulting NIOSH-proposed SEC class must include a minimum required employment period of 250 days as a basis for specifying that health was endangered for this time period. NIOSH further determined that it is feasible to estimate radiation dose with sufficient accuracy for members of the NIOSH-evaluated class for the time period from January 1, 1958 through December 31, 1979. Therefore, a health endangerment determination is not required for this time period.

9.0 Class Conclusion for Petition SEC-00224

Based on its full research of the class under evaluation, NIOSH has defined a single class of employees for which NIOSH cannot estimate radiation doses with sufficient accuracy. The NIOSH-proposed class to be added to the SEC includes all employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Argonne National Laboratory-West between April 10, 1951 and December 31, 1957 for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort. NIOSH determined that it is feasible to estimate radiation dose with sufficient accuracy for members of the NIOSH-evaluated class for the time period from January 1, 1958 through December 31, 1979.

NIOSH has carefully reviewed all material sent in by the petitioner, including the specific assertions stated in the petition, and has responded herein (see Section 7.4). NIOSH has also reviewed available technical resources and many other references, including the Site Research Database (SRDB), for information relevant to SEC-00224. In addition, NIOSH reviewed its NOCTS dose reconstruction database to identify EEOICPA-related dose reconstructions that might provide information relevant to the petition evaluation.

These actions are based on existing, approved NIOSH processes used in dose reconstruction for claims under EEOICPA. NIOSH's guiding principle in conducting these dose reconstructions is to ensure that the assumptions used are fair, consistent, and well-grounded in the best available science. Simultaneously, uncertainties in the science and data must be handled to the advantage, rather than to the detriment, of the petitioners. When adequate personal dose monitoring information is not available, or is very limited, NIOSH may use the highest reasonably possible radiation dose, based on reliable science, documented experience, and relevant data to determine the feasibility of reconstructing the dose of an SEC petition class. NIOSH contends that it has complied with these standards of performance in determining the feasibility or infeasibility of reconstructing dose for the class under evaluation.

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10.0 References

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Attachment 1: Data Capture Synopsis

Table A1-1: Data Capture Synopsis for Argonne National Laboratory-West

Data Capture Information	General Description of Documents Captured	Date Completed	Uploaded to SRDB
Primary Site/Company Name: Argonne National Laboratory-West (ANL-W) DOE 1949-2005 Physical Size of the Site: Approximately 259 acres with 52 major buildings. Site Population: In 1993 there were approximately 860 employees on site. ANL-W facilities include Experimental Breeder Reactor (EBR) I and II, Zero Power Reactor (ZPR), Zero Power Physics Reactor (ZPPR), the Boiling Water Reactor Experiment Reactors (BORAX), Hot Fuel Examination Facility (HFEF), Transient Reactor Test Facility (TREAT), Argonne Fast Source Reactor (AFSR), and the Neutron Radiography Facility (NRAD).	The following documents originated at ANL-W and are, in fact, ANL-W documents. However, since ANL-W was integrated into Idaho National Laboratory (INL) in 2005 they were identified in the collections at INL: Incident reports and summaries, air sample results including actinide concentrations, whole body count results, fecal and urine bioassay results, facility routine survey schedules, ZPPR neutron measurements and dose assignments, radiation work permits, radiological surveys, environmental reports, tritium contamination at HFEF, environmental air samples, operational, safety, and Health Physics monthly reports, air dust survey results, material transfers, safe work permits, Health Physics logs, respiratory protection, effluent releases, facility plot and floor plans, neutron sources, neutron radiography, decontamination and decommissioning of EBR-I, EBR-II daily radiation safety logs, plutonium reactor core loadings, photographs, waste management and forecasts, 1968 EBR-II Sodium Boiler fire, non-routine sample processing, smear sample data, cell entry exposure logs, temporary badge reports, alpha contamination survey reports, interviews with site experts, surveys of EBR-II, TREAT, and ZPPR, reactor safety inspections, a photo of the first plutonium produced at EBR, EBR external dosimetry files, contamination reports, operations and maintenance manuals, Reactor Development Program progress reports, EBR film reports, exposure summaries, EBR-II fuel studies, the Fuel Cycle Facility description, an addendum to the HFEF/N facility safety report, a documented communication, laboratory and office building survey logs and air samples, radiation dose histories for a group of 50 early ANL-W workers, accumulated personnel exposure summaries, 1951-1962, sample record sheets for selected ANL-W workers, a 1962 Burial Ground survey log sheet, and IDO H&S sample record sheets for EBR.	11/05/2015	2,239
State Contacted: NA	Contacting the state was not considered necessary since Argonne National Laboratory-West, now known as the Materials and Fuels Complex of Idaho National Laboratory, is an active DOE site and cooperates with relevant data collection.	08/03/2015	0

Data Capture Information	General Description of Documents Captured	Date Completed	Uploaded to SRDB
Albany Research Center (ARC)	History of remediation at ARC, where surveys were performed by ANL.	11/01/2012	1
Argonne National Laboratory East (ANL-E)	1955 Industrial Hygiene and Safety Division monthly reports, monthly Operation Clean Sweep status reports, BORAX-I design calculations, ZPR-III operating instructions, EBR-II technical specifications, the 1971 TREAT manual, the first plutonium fuel loading of EBR-I, Radiation Safety Section monthly reports, air sample data, ZPR-III hazards report, Health Physics reports, 1958-1961 bioassay reports, an EBR-II loss of water incident, a 1965 explosion at EBR-II, plutonium production at EBR-I, EBR-I shielding measurements, radiation dose histories for 11 selected workers, and radiation dose histories for a group of 50 early ANL-W workers.	10/23/2015	111
Atlantic Richfield	ZPPR external exposure reports and study, 1967-1968.	09/12/2005	3
Battelle Memorial Institute - King Avenue	A listing of DOE research papers 1949-1983, annual audits of the Plutonium Laboratory, and a status report on the decontamination and decommissioning of the Plutonium Laboratory.	01/08/2013	3
Brookhaven National Laboratory	A 1974 comparison of external exposures at AEC facilities and a 1949 description of the AEC Four Point Cancer Program.	11/18/2008	2
BWX Technologies, Inc.	Documentation of ZPPR fuel element production at the NUMEC Parks Township Facility.	05/24/2005	1
Claimant Provided	A matrix showing Argonne's part in nuclear weapon production.	04/18/2005	1
Curtiss-Wright, Cheswick, PA	Material transfer records with package and fuel pin surveys.	04/26/2009	3
Dade Moeller	The Leonard Koch book describing EBR-II.	05/26/2015	1
DOE Albuquerque	The 1984 hazard classification of ANL-W nonreactor nuclear facilities.	04/15/2010	1
DOE Carlsbad	A report on the measurement of the 10-B(n,He) and 6-Li(n,He) reaction rates in ZPPR-13C core.	08/10/2010	1
DOE Germantown	Material receipts at the FMPC from Argonne, the DOE Oak Ridge Operations Records Holding Area search request procedure, and the 2010 Argonne EEOICPA Reasonable Search Procedures.	03/07/2011	3
DOE Legacy Management - Grand Junction Office	Material transfers, February 1955 accountability station codes, early production progress reports showing work at Argonne, the disposition of CANEL equipment, and a 1959 index to reactor fuel data sheets.	08/25/2011	21
DOE Legacy Management - Morgantown	An environmental, safety, and health needs site summary, June 1955 accountability station codes, EPA proposed standards for airborne radionuclide releases, and recycled uranium project data.	03/03/2011	6

Data Capture Information	General Description of Documents Captured	Date Completed	Uploaded to SRDB
DOE Legacy Management - MoundView (Fernald Holdings, includes Fernald Legal Database)	Material transfers, Materials Testing Reactor (MTR) meeting notes, 1983 semiannual index of DOE unusual occurrence reports, a 1987 DOE precious metal report, 1982 and 1984 ANL-W effluent discharge reports, material transfers and a report on the incineration of solid wastes.	05/13/2010	13
DOE Legacy Management - Westminster	1969 and 1982 Rocky Flats reports on processing ZPPR fuel fabrication residues.	10/23/2014	2
DOE Oak Ridge Operations Office	A report on neutron and gamma exposures from radiation accidents, internal audit criteria for industrial hygiene programs, a 1945 discussion on research and production processes, and planning for future use and disposition of U-233.	07/09/2012	4
DOE Oak Ridge Operations Office - Records Holding Task Group (RHTG)	A monthly production report, resumes of activities, and uranium production statistics with a thorium inventory.	04/08/2011	4
DOE Office of Scientific and Technical Information (OSTI)	Surveys of irradiation facilities, performance testing of personnel dosimetry services, material transfers, breeder fuel development reports, EBR-II operational Health Physics problems, a power reactor fuel symposium, and a 1953 Reactor Engineering Division quarterly report.	05/22/2015	16
DOL - Paragon	Boron-10 project reports, 1952 operational reports, a requested material transfer, and a low-level mixed waste assessment in support of the DOE programmatic waste management environmental impact statement.	01/23/2012	6
Energy Technology Engineering Center (ETEC)	An exposure summary for the de-cladding of an EBR-II fuel element.	12/01/2005	1
Federal Records Center (FRC) - Denver	A 1967 review of criticality accidents and the inclusion of ANL-W in the study of the recovery of Pu-244 from Mark 18A targets.	01/31/2012	2
Federal Records Center (FRC) - Kansas City	1983-1985 nuclear fuel cycle and waste management field work packages.	08/15/2008	1
Federal Records Center (FRC) - San Bruno	LBNL Health Physics quarterly reports, selected nuclear materials management and safeguard systems samples, 1960 operations documents, and isotopes received 1949-1960.	08/02/2012	9
General Atomics	Material transfer reports 1964-1970.	11/02/2005	1
General Electric - Evendale	A brief history of GE at INL.	11/12/2010	1
General Electric - Vallecitos	A 1968 progress report on the recovery of plutonium from ZPPR wastes.	02/02/2007	1
Hagley Museum & Library	Trip reports to ANL-W.	09/29/2010	3
Hanford	Hanford periodic operational and radiation protection reports mentioning ANL-W, trip reports, disposition of EBR-II test fuel, annual inventories by mass and activity for 1983-1991, and test program correspondence.	10/21/2014	37

Data Capture Information	General Description of Documents Captured	Date Completed	Uploaded to SRDB
Interlibrary Loan	An Argonne history, neutron tomography of fuel bundles, irradiated fuel examination at HFEF, proceedings of the 1993 incineration conference, proceedings of a 1964 symposium on surface contamination, and the operation of the EBR-II Fuel Cycle Facility.	08/31/2015	7
Internet - Defense Technical Information Center (DTIC)	A bibliography of occupational dose reduction reports, the final proposed environmental statement for liquid metal fast breeder reactors, EBR-II reports, corrosion of metal waste forms and metallic uranium, and the 1998 Defense Nuclear Facilities Safety Board report to Congress.	10/04/2013	8
Internet - DOE	The DOE Standard: Guide of Good Practices for Occupational Radiological Protection in Plutonium Facilities.	05/11/2007	1
Internet - DOE Environmental Management	Linking Legacies Chapter 3: Wastes.	10/28/2007	1
Internet - DOE Legacy Management	Idaho National Laboratory decontamination and decommissioning logic diagrams and the April 1993 DOE interim mixed waste inventory report, volume 2.	08/27/2014	3
Internet - DOE Legacy Management Considered Sites	The April 1993 DOE interim mixed waste inventory report, volume 1.	04/24/2012	1
Internet - DOE Oak Ridge Operations Office	The Interim Action Proposed Plan for Fuel and Flush Salt Removal from the Molten Salt Reactor Experiment, Oak Ridge National Laboratory, Oak Ridge, Tennessee.	01/14/2014	1
Internet - DOE OpenNet	Linking Legacies Appendix B, summaries of purchase order status reports, a plutonium balance report, a SNAPTRAN monitoring report, Argonne history, radiation protection standards, history of the Manhattan District Research Division, the cross section of carbon, 1951 stable isotopes shipments and requests, 1965 Idaho Falls Weather Bureau Research Station Program and follow-up of survivors of serious radiation accidents.	09/17/2015	18
Internet - DOE OSTI	Chemical Engineering Division periodic reports, EBR-1 progress report for 1951-1953, and the Tiger Team Assessment of INEL.	09/16/2015	13
Internet - DOE OSTI / SC&A	The 1945-1955 summary of radiation accidents and incidents.	02/21/2007	1
Internet - DOE OSTI Energy Citations	Fuel development reports, material transfers, examinations of irradiated fuel specimens, overview of the Hot Fuel Examination Facility, capabilities of ZPPR to support active interrogation research of special nuclear material, EBR-II rotating plug maintenance, and the chemical processing of Pu-238.	03/28/2013	23

Data Capture Information	General Description of Documents Captured	Date Completed	Uploaded to SRDB
Internet - DOE OSTI Information Bridge	Periodic radioactive waste reports, environmental reports and plans, plutonium contamination monitoring in the HFEF hot cells, the sodium process facility, fast neutron radiography research, neutron monitoring, glove box usage and decontamination reports, ZPPR progress reports, spent fuel treatment, transuranic waste reports, fast reactor safety, safety analysis of the use of plutonium in ZPR-6 and -9, EBR-II operational and safety reports, and the confirmatory radiological survey of the BORAX-V turbine building.	03/29/2013	167
Internet - DOE OSTI Information Bridge / SC&A / INL	Health Physics aspects of the 07/22/1954 BORAX experiment.	07/06/2007	1
Internet - DOE OSTI SciTech Connect	Reactor physics, development, and engineering progress reports, radioactive waste management, fuels and materials periodic reports, the shutdown of EBR-II, Nevada National Security Site waste acceptance procedures, environmental reports, a hot cell window shielding analysis, the remote treatment facility for transuranic waste, metallurgy division periodic reports, a report on BORAX-V, Reactor Development Program progress reports, EBR-II summaries of operating experience, a nuclear waste semiannual progress report, failed fuel monitoring and surveillance techniques, and the manufacture of fuel and blanket elements for BORAX-IV.	11/23/2015	208
Internet - DOE NNSA Nevada Site Office	No relevant documents identified.	02/13/2015	0
Internet - Energy Employees Claimant Assistance Project (EECAP)	Soils characterization and remediation reports.	03/24/2014	3
Internet - Google	Environmental reports and plans, change notice 2 to the DOE Standard Guide of Good Practices for Occupational Radiological Protection in Plutonium Facilities, DOE occupational radiation exposure reports, the EBR-I designation as a national historic landmark, engineering evaluations, groundwater evaluations, records of decision for ANL-W facilities, hazards summary reports, transuranic waste reports, 2011 plutonium contamination in ZPPR, a waste disposal inventory, material transfers, incident reports, radiological effluent reports, Materials Testing Reactor design, safety and health assessments, a history and status of fast breeder reactor programs, the EBR-II containment building, EBR-II fuels and materials, Reactor Development Program Progress reports, and EBR-II test and operating experience.	12/05/2015	250
Internet - Hanford DDRS	Hanford periodic operational reports mentioning ANL-W, a trip report, production of plutonium for ZPPR, and dissolution of spent ZPPR fuel in nitric acid.	03/25/2015	26

Data Capture Information	General Description of Documents Captured	Date Completed	Uploaded to SRDB
Internet - Hathitrust	The compendium of radiation safety information for plutonium.	10/25/2012	1
Internet - Health Physics Journal	Environmental iodine monitoring, the high level waste tank closure project, the transuranic waste absorbent addition project, and testing hot cell shielding in the Fuel Conditioning Facility (FCF).	05/19/2015	4
Internet - IAEA INIS	EBR-II advanced driver fuels and removal of a perchlorate contaminated hot cell acid fume system.	02/13/2015	2
Internet - Journal of Occupational and Environmental Hygiene	No relevant documents identified.	02/13/2015	0
Internet - National Academies Press (NAP)	A nuclear technologies timeline and electrometallurgical techniques for spent fuel treatment.	02/12/2015	2
Internet - National Institute for Occupational Safety and Health (NIOSH)	The 2004 and 2006 reports on residual radioactive and beryllium contamination.	01/25/2007	2
Internet - NRC Agencywide Document Access and Management (ADAMS)	Spent fuel and radioactive waste inventories, waste management papers and reports, environmental impact statements for spent fuel and high-level waste storage and disposition, surplus plutonium disposition, natural phenomena hazards design and evaluation criteria, and the examination of spent fuel rods after 15 years in dry storage.	12/11/2014	54
Internet - Oak Ridge National Laboratory (ORNL) Library	ORNL periodic operational and research reports mentioning ANL-W, the effect of various materials on unfired Vycor glass, secondary gamma ray production from californium fission spectrum neutrons, and the interim assessment of the denatured U-233 fuel cycle feasibility.	04/08/2013	30
Internet - University of Hawaii, Manoa	The 1965 airborne gamma survey of exposure rates in the National Reactor Testing Station Area.	11/08/2007	1
Internet - University of North Texas	ANL 1957 and 1958 annual reports, periodic Reactor Division reports, the facility for photographing meltdown experiments in TREAT, a hazard summary addendum for EBR-II, and the preliminary design and hazards report for BORAX-V.	06/25/2015	11
Internet - US Army Corps of Engineers (USACE)	No relevant documents identified.	10/28/2014	0
Internet - US Environmental Protection Agency NEPIS	A 1997 Idaho Superfund progress report, Records of Decision, and a 1987 mixed energy waste study.	12/19/2014	4
Internet - US Transuranium and Uranium Registries	No relevant documents identified.	10/28/2014	0
Kansas City Plant	The rework of cesium pellets and April 1991 daily operations reports for the Secretary of Energy.	10/20/2014	2
Karam, P. Personal Library	Selected pages from the Chemistry of the Actinide Elements, Second Edition Volume 1.	04/11/2007	1
Lawrence Livermore National Laboratory	Heavy Element Program for HUTCH and Program Need for Extended Intense Steady State Irradiations at the ICT Accelerator.	04/28/2015	2

Data Capture Information	General Description of Documents Captured	Date Completed	Uploaded to SRDB
Los Alamos National Laboratory	An accelerator characterization report of DOE facilities, a request for Np-237, and an Np-237 project report.	08/15/2014	3
Los Alamos National Laboratory - LAHDRA	A 1952 survey of AEC facilities, quantities and characteristics of contact-handled low-level mixed waste, a 1981 monthly operational Health Physics report, a history of critical experiments, and the environmental impact statement for storage and disposition of weapons-usable fissile materials.	12/06/2007	5
Metals and Controls Corporation	Confirmation that Metals and Controls supplied fuel elements for ANL-W.	08/24/2004	1
Missouri Department of Natural Resources	Plutonium working group reports.	10/01/2008	2
Mound Museum	The radiometric determination of uranium in soil and air.	07/11/2008	1
National Archives and Records Administration (NARA) - Atlanta	A 1947 report on AEC research and development projects, correspondence on UF6 to UF4 processing, volumes 1 and 2 of the DOE indoor radon study, and documentation pertaining to the U-233 - thorium project.	05/20/2008	4
National Archives and Records Administration (NARA) - Chicago	Periodic reports, Reactor Engineering Division work programs, EBR-I work schedules, material transfers, the 1951 redesign of air sampling equipment to meet a lower plutonium maximum permissible concentration, personnel access lists, EBR-I shielding measurements, 1954 Industrial Hygiene and Safety Division equipment requirements, schedule of EBR operations, EBR plutonium loading, and EBR-I operating instructions.	06/12/2015	88
National Archives and Records Administration (NARA) - College Park	A requisition for graphite for the Materials Test Reactor and monthly SF material inventories and progress reports.	09/12/2013	2
National Archives and Records Administration (NARA) - Seattle	Incident reports, facility Health Physics reports, employee thyroid counts, a 1966 Burial Ground report, neutron dosimetry data, radioactive waste reports, material shipment records, and a contamination survey of product bottles.	12/18/2014	30
National Institute for Occupational Safety and Health (NIOSH)	Annual and semiannual reports to Congress, the rolling of billets at Bethlehem Steel, Rocky Flats waste buried in the subsurface disposal area, worker outreach meeting minutes and confirmation, the 1963 plutonium aerosol monitoring program, EBR plutonium bioassay results, ZPPR radiation survey reports, researcher notes, a report on aquifer contamination, a 1953 Chemical Engineering Division summary report, and a DOE classification release letter.	12/07/2015	27
National Institute for Occupational Safety and Health (NIOSH) / SC&A	The needs assessment for the former worker medical surveillance program at INEL and highly enriched uranium working group reports.	02/16/2006	6

Data Capture Information	General Description of Documents Captured	Date Completed	Uploaded to SRDB
Nevada Test Site	A summary of the volume of ANL-W waste buried at the Nevada Test Site.	10/01/2003	1
Nuclear Metals, Inc.	Notations regarding the fabrication of EBR-I fuel rods.	05/24/2012	1
Nuclear Materials and Equipment Corporation (NUMEC)	ZPPR fuel contract documents, ZPPR fuel shipping data, periodic operations reports, and the 1969 report of a compliance meeting close-out.	04/07/2015	17
Nuclear Regulatory Commission Non-Public Holdings	1979 requests for authorization as shipper.	06/04/2012	2
Nuclear Regulatory Commission Public Document Room	NUMEC AEC license documents pertaining to the manufacture of ZPPR fuel elements, the fabrication of EBR-II fuel elements, and the analysis of West Valley terminal waste forms including ANL waste.	12/19/2014	7
Nuclear Testing Archive (NTA)	A cutaway photograph of ZPPR.	05/12/2015	1
Oak Ridge Library for Dose Reconstruction	A list of AEC incidents, a material transfer from ORNL to ANL-W, and a report on methods of solid waste management.	04/27/2011	3
Oak Ridge National Laboratory (ORNL)	Periodic ORNL reports that mention ANL-W, abstracts of reports added to the ORNL libraries, and dose measurements at the Bulk Shielding Facility.	02/07/2013	21
Ohio Department of Health	An environmental and waste management report.	11/03/2008	1
ORAU Team	Technical basis documents, documented communications, dose reconstruction procedures, the evaluation of incomplete internal dosimetry records, compiled internal dosimetry spreadsheets, and a 2004 Argonne directory.	07/22/2015	57
Pantex Plant	Records transfer control documents indexed from 1952-1990.	02/06/2008	1
Rocky Flats Environmental Technology Site (RFETS)	Radiation exposure of Rocky Flats employees during the manufacture of ZPPR fuel elements.	05/15/2006	1
SAIC	1967 and 1971 radiation exposure summaries.	09/02/2004	2
Santa Susana Field Laboratory (SSFL)	The proposed work plan for the examination of EBR-II sodium cold traps and a user authorization for the pulsed neutron calibration of flowmeters.	08/10/2009	2
Savannah River Site (SRS)	Periodic reports mentioning ANL-W, a temporary procedure for receiving and packaging EBR-I fuel, thorium slug failures, and a trip report to the neptunium information meeting.	02/08/2012	15
S. Cohen & Associates (SC&A)	Inventory and manufacturing statements, an MTR progress report, Chemical Processing Plant (CPP) monthly reports, and a citation for an evaluation of the ZPPR roof filter system.	06/13/2011	66

Data Capture Information	General Description of Documents Captured	Date Completed	Uploaded to SRDB
SC&A / INL	Analysis of EBR-II and ZPPR stack effluents, BORAX surveys and sample analyses, decontamination and decommissioning of EBR-I, environmental reports, studies of the ANL-W industrial waste ponds, groundwater reports, long-lived gross beta air concentrations at EBR-II, radioactive waste management reports, radio ecological effects on vicinity populations, Burial Ground air samples, inventory and manufacturing statements, CPP monthly reports, effluent analyses, environmental monitoring data and reports, meeting minutes, EBR-II fact sheet and photographs, and episodic releases.	06/24/2010	215
SC&A / Internet - DOE OpenNet	The 1944-1994 history of plutonium production, acquisition, and utilization.	10/28/2014	1
SC&A / Internet - Google	Savannah River Site 50 year report.	04/22/2008	1
SC&A / Internet - University of North Texas	The BORAX V design and hazards summary report.	10/13/2014	1
SC&A / Pinellas Plant	A 1993 waste generation and minimization report.	06/24/2010	1
SC&A / SSFL	ZPPR fuel specifications, the de-cladding of gas tagged EBR-II fuel elements, draft letter proposing Atomics International as the supplier of ZPPR fuel plates, and the follow-up to the meeting on supplying ZPPR fuel plates.	06/24/2010	4
Senator John Heinz History Center	A history of Westinghouse and atomic power.	12/20/2007	1
Southern Illinois University, Edwardsville, IL	Transcript of day two of the Fifty-Sixth Meeting of the Advisory Board on Radiation and Worker Health.	10/29/2008	1
Unknown	A 2000 review of criticality accidents, hazard summary of EBR-II, design summary of TREAT, 1958 BORAX atmospheric releases, analysis for the decommissioning of BORAX-V, functional and operational design requirements for decontamination and decommissioning of the EBR-I Mark-II NAK, DOE occupational exposure reports, dose estimation from a plutonium contaminated hand wound, ANL-W skin contaminations, the 1989 ANL-W routine bioassay program, radiation exposure summaries, proceedings of the United Nations 1955 Geneva conference, and occupational experience with BORAX and ZPR-III.	02/10/2011	73
Unknown / SC&A	A 1957 report of core changes from the EBR-I meltdown and information for use in controlling radiation emergencies.	08/05/2003	2
Washington State University	EBR-II reports and delayed neutron detection in the sodium loop safety facility.	05/29/2015	4
TOTAL	**N/A**	**N/A**	**4,017**

Table A1-2: Databases Searched for Argonne National Laboratory-West

NOTE: Database search terms employed for each of the databases listed below are available in the Excel file called "ANL-W Rev 02, (83.13) 01-11-16"

Database/Source	Keywords/Phrases	Hits	Selected
Defense Technical Information Center (DTIC) https://www.dtic.mil/ COMPLETED 04/21/2015	See Note above	286	0
DOE CEDR https://www.orau.gov/cedr COMPLETED 02/13/2015	See Note above	0	0
DOE Hanford DDRS http://reading-room.labworks.org/Catalog/Search.aspx COMPLETED 05/13/2015	See Note above	0	0
DOE Legacy Management Considered Sites http://www.lm.doe.gov/considered_Sites/ COMPLETED 10/29/2014	See Note above	264	0
DOE NNSA - Nevada Site Office http://nnsa.energy.gov COMPLETED 02/13/2015	See Note above	2	0
DOE OpenNet http://www.osti.gov/opennet/advancedsearch.jsp COMPLETED 10/28/2014	See Note above	88	2
DOE OSTI Energy Citations http://www.osti.gov/energycitations/ COMPLETED 07/12/2010	See Note above	1,081	1
DOE OSTI Information Bridge http://www.osti.gov/bridge/advancedsearch.jsp COMPLETED 07/12/2010	See Note above	971	1
DOE OSTI SciTech Connect http://www.osti.gov/scitech COMPLETED 10/28/2014	See Note above	315,622	14
Energy Employees Claimant Assistance Project (EECAP) http://www.eecap.org COMPLETED 02/13/2015	See Note above	0	0
Google http://www.google.com COMPLETED 10/29/2014	See Note above	189,587	48

Database/Source	Keywords/Phrases	Hits	Selected
HP Journal http://journals.lww.com/health-physics/pages/default.aspx COMPLETED 04/21/2015	See Note above	3,212	0
Journal of Occupational and Environmental Hygiene http://www.tandfonline.com/loi/uoeh20#.VtS2f00o670 COMPLETED 02/13/2015	See Note above	14	0
National Academies Press http://www.nap.edu/ COMPLETED 10/29/2014	See Note above	11,990	2
NEPIS http://nepis.epa.gov/ COMPLETED 10/29/2014	See Note above	4	0
NRC ADAMS Reading Room http://www.nrc.gov/reading-rm/adams.html#web-based-adams COMPLETED 07/12/2015	See Note above	890	2
United States Army Corps of Engineers (USACE) http://www.usace.army.mil/ COMPLETED 10/28/2014	See Note above	3	0
U.S. Transuranium & Uranium Registries http://www.ustur.wsu.edu/ COMPLETED 10/28/2014	See Note above	0	0

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Attachment 2: Acronyms and Abbreviations

a.k.a.	also known as
ADAMS	Agency-wide Documents Access and Management
AEC-IDO	U.S. Atomic Energy Commission-Idaho Operations Office
AFSR	Argonne Fast Source Reactor
AL	Analytical Laboratory
ALIPs	annular linear-induction pumps
ANL	Argonne National Laboratory
ANL-E	Argonne National Laboratory-East
ANL-PIO	Argonne National Laboratory –Public Information Office
ANL-W	Argonne National Laboratory–West
Board	Advisory Board on Radiation and Worker Health
BORAX	Boiling Water Reactor Experiments
C	Celsius
CAM	continuous air monitor
cc	cubic centimeter
CEDR	Comprehensive Epidemiologic Data Resource
CFA	Central Facilities Area (INL)
cfm	cubic feet per minute
Ci	Curie
cm	centimeter
cpm	counts per minute
CPP	Chemical Processing Plant
CRBRP	Clinch River Breeder Reactor Plant
D&D	decontamination and decommissioning
DCAS	Division of Compensation Analysis and Support
DOE	U.S. Department of Energy
dpm	disintegrations per minute
DTIC	Defense Technical Information Center (database)
DU	depleted uranium
EBR-I	Experimental Breeder Reactor-I
EBR-II	Experimental Breeder Reactor-II

EDMS	Electronic Document Management System
EECAP	Energy Employees Claimant Assistance Project
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ETR	Engineering Test Reactor
FASB	Fuel Assembly & Storage Building
FASS	Fixed air-sampling system
FCF	Fuel Cycle Facility
FEF	Fuels and Examination Facility
FFTF	Fast Flux Test Facility
FSAR	Final Safety Analysis Report
ft.	foot
g	gram
H&S Laboratory	Health and Safety Laboratory
HEPA	high-efficiency particulate air (filter)
HFEF	Hot Fuel Examination Facility
HFEF-N	Hot Fuel Examination Facility-North
HFEF-S	Hot Fuel Examination Facility-South
HHS	U.S. Department of Health and Human Services
hr	hour
HS	Health and Safety
HSD	Health and Safety Division
HSL	Health Services Laboratory
IBC	Inter-Building Coffin
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
ID	interior diameter
IFR	Integral Fast Reactor
in.	inch
INEEL	Idaho National Engineering and Environmental Laboratory
INL	Idaho National Laboratory
IOO	Idaho Operations Office (AEC)
ITF	Inspection & Test Facility
Kel-F	Teflon-based protective coating
keV	kiloelectron-volt, 1,000 electron-volts

kg	kilogram
kW	kilowatt
L&O Building	Laboratory & Operations Building
lb	pound
LMFBR	Liquid Metal Fast Breeder Reactor
LMR	Liquid Metal Reactor
LOD	Limits of detection
LOF	loss of flow
M	meter
MED	Manhattan Engineer District
MeV	megavolt-electron, 1 million electron-volts
MFC	Materials and Fuels Complex (new name for ANL-W)
MFP/MAP	mixed fission products/mixed activation products
min	minute
MOX	mixed oxide (fuel)
MPC	maximum permissible concentration
mR	milliroentgen
mr/hr	milliroentgen/hour
mrem	millirem
MSD	Materials Science Division
MTR	Materials Test Reactor
MW	megawatt
MWe	Megawatt electric (power plant output)
$n/cm^2/sec$	neutrons per cubic centimeter per second
NAD	Nuclear Accident Dosimeter
NARA	National Archives and Records Administration
NDA Laboratory	non-destructive analysis
NIOSH	National Institute for Occupational Safety and Health
NOCTS	NIOSH OCAS Claims Tracking System
N-RAD	Neutron Radiography Facility
NRAD	Neutron Radiography Facility
NRC	Nuclear Regulatory Commission
NRTS	National Reactor Testing Station
NTA Film	Nuclear Track A film

OCAS	Office of Compensation Analysis and Support
OD	outer diameter
ORAU	Oak Ridge Associated Universities
ORAUT	Oak Ridge Associated Universities Team
ORT	operational reliability testing
OSTI	Office of Scientific and Technical Information
OTIB	ORAUT Technical Information Bulletin
PBF	Power Burst Facility
PDF	portable document format
PRNC	Puerto Rico Nuclear Center
QA	quality assurance
RAS	Reactor Analysis and Safety Division
RAS-TREAT	Reactor Analysis and Safety Division-Transient Reactor and Experiment Test
RBCB	run-beyond-cladding-breach
RESL	Radiological and Environmental Sciences Laboratory
RSWF	Radioactive Scrap and Waste Facility
RWMC	Radioactive Waste Management Complex
SEC	Special Exposure Cohort
SEFOR	Southwest Experimental Fast Oxide Reactor
SPM	Special Material
SRDB	Site Research Database
TBD	Technical Basis Document
TLD	thermoluminescent dosimeter
TOP	transient overpower
TREAT Facility	Transient Reactor Experiment and Test Facility
TRIGA®	Training, Research, Isotopes, General Atomics
µCi	microcurie
µCi/cc	microcurie
w/o	without
WBC	whole-body counting
ZPR	Zero Power Reactor