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Engineering Evaluation/Cost Analysis for the EBR-II Final End State

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January 2010

Prepared for the
U.S. Department of Energy
DOE-ID Operations Office

ABSTRACT

This engineering evaluation/cost analysis was prepared to support determining the final end state of the Experimental Breeder Reactor II reactor building and reactor vessel. Performance of a Comprehensive Environmental Response, Compensation, and Liability Act non-time-critical removal action is the process to achieve this end state. The approach satisfies environmental review requirements and provides for stakeholder involvement while supplying a framework for selecting the removal action alternative to reach the end state.

This engineering evaluation/cost analysis evaluates four alternatives for the final end state of the reactor building and reactor vessel and recommends a preferred alternative.

The recommended alternative is Alternative 3: grouting the reactor vessel in place and removal of the reactor building. A concrete cover would be constructed over the Experimental Breeder Reactor II site and institutional controls would be maintained.

EXECUTIVE SUMMARY

The engineering evaluation/cost analysis process documented in this report supports determining the final end state of the Experimental Breeder Reactor II (EBR–II) reactor building and reactor vessel and recommends an alternative to reach that end state.

In keeping with the joint U.S. Department of Energy and U.S. Environmental Protection Agency *Policy on Decommissioning of Department of Energy Facilities Under the Comprehensive Environmental Response, Compensation, and Liability Act*, this engineering evaluation/cost analysis was developed in accordance with the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), as amended by Public Law 99-499, "Superfund Amendments and Reauthorization Act of 1986," and in accordance with the "National Oil and Hazardous Substances Pollution Contingency Plan." The removal action objectives of this non-time-critical removal action are consistent with the *Final Record of Decision Argonne National Laboratory-West, Operable Unit 9-04, Idaho National Engineering and Environmental Laboratory*. The objectives support the overall remediation goals established through the Federal Facility Agreement and Consent Order and the U.S. Department of Energy goal of reducing the "risk footprint" to as practicable extent as possible in consideration of as-low-as-reasonably-achievable (ALARA) principles governing radiological exposure to decommissioning personnel, safe engineering standards, Idaho CERCLA Disposal Facility (ICDF) waste acceptance criteria, and desired end states of CERCLA sites.

This engineering evaluation/cost analysis evaluates four non-time-critical removal action alternatives, listed below, for the final end state of the EBR-II reactor building and reactor vessel and recommends a preferred alternative. All alternatives would take place after the residual sodium metal and sodium-potassium alloy (NaK) remaining in the EBR-II systems are treated or removed, in compliance with the approved closure plan under the Hazardous Waste Management Act/Resource Conservation and Recovery Act closure plan. The four alternatives are as follows:

Alternative 1, No Action—Under the "no action" alternative, no decommissioning and demolition would be conducted at EBR-II and no further surveillance and maintenance would be performed at the facility. This alternative does not achieve remedial action objectives and does not reflect the U.S. Department of Energy mandate to monitor, maintain, and mitigate potential or actual releases from any facility or site to ensure protection of the public health or the environment.

Alternative 2, No Action: Continued Surveillance and Maintenance—Under Alternative 2, no action would be performed except surveillance and maintenance. This alternative offers no reduction in toxicity or volume of contaminants, but it does provide more protection from mobilization of the contaminants to the environment than Alternative 1.

Alternative 3, Grouting the EBR-II Reactor Vessel in Place—Under Alternative 3, systems and structures above the reactor building floor would be demolished and most of the remaining systems and structures below floor level, including the EBR-II reactor vessel, would be grouted in place. The end state of EBR-II under Alternative 3 is a concrete/grout monolith that extends approximately 8 ft above ground level.

Alternative 4, Removal of the EBR-II Reactor Vessel—Alternative 4 includes demolition of the EBR-II reactor building and removal and disposal of the EBR-II reactor vessel. The containment building would be demolished to ground level or below. Radioactive waste, including the reactor vessel and primary sodium tank components, would be removed from the site and disposed of at the ICDF in accordance with waste acceptance criteria. The end state for Alternative 4 is a below-ground-level concrete/grout monolith.

Alternative 3, grouting the reactor vessel in place, is the recommended alternative and meets the proposed removal action objectives regarding human health and environmental protectiveness, and it is the most cost-effective option between the action alternatives. It is also consistent with the remedial action objectives of the Operable Unit 9-04 Final Record of Decision; is compliant with applicable or relevant and appropriate requirements; and satisfies the U.S. Department of Energy goal of reducing the "risk footprint" to as practicable extent as possible in consideration of (a) ALARA principles governing radiological exposure to decommissioning personnel, (b) safe engineering standards, (c) waste acceptance criteria for the ICDF, and (d) desired CERCLA site end states.

CONTENTS

ABSTRACT iii

EXECUTIVE SUMMARY v

ACRONYMS xi

1. INTRODUCTION 1-1

1.1 Purpose 1-1

1.2 Scope 1-2

1.3 Document Organization 1-2

2. SITE CHARACTERIZATION 2-1

2.1 Site Description and Background 2-1

2.1.1 Idaho National Laboratory Site 2-1
2.1.2 Materials and Fuels Complex History 2-2
2.1.3 EBR-II Location 2-2
2.1.4 EBR-II History 2-2
2.1.5 EBR-II Systems and Component Descriptions 2-5
2.1.6 Reactor Vessel and Neutron Shield Description 2-6

2.2 Other Closure and Cleanup Activities at MFC 2-13

2.2.1 CERCLA Activities at MFC 2-13
2.2.2 Other Non-Time-Critical Removal Action Activities at MFC 2-13
2.2.3 Resource Conservation and Recovery Act Activities 2-13

3. IDENTIFICATION OF REMOVAL ACTION ALTERNATIVES FOR EBR-II 3-1

3.1 Alternative 1, No Action 3-1

3.2 Alternative 2, No Action: Continued Surveillance and Maintenance 3-2

3.3 Alternative 3, Grouting the EBR-II Reactor Vessel in Place 3-3

3.4 Alternative 4, Removal of the EBR-II Reactor Vessel 3-3

4. EBR-II RADIOLOGICAL AND NONRADIOLOGICAL INVENTORIES 4-1

4.1 Radiological Inventory 4-1

4.1.1 Inventory Above and Below Ground Level 4-2
4.1.2 Inventory 0 ft to 10 ft Below Ground Level 4-2
4.1.3 EBR-II Radiological Conditions 4-3

4.2 EBR-II Nonradiological Inventory 4-4

5. IDENTIFICATION OF REMOVAL ACTION OBJECTIVES 5-1

5.1 Removal Action Objectives 5-1

6. RISK ASSESSMENTS 6-1

6.1 Human Health Risk Assessment 6-1

6.1.1 Radiological Risk Analysis 6-1
6.1.2 Nonradiological Risk 6-3

6.2 Groundwater Risk Assessment 6-4

6.3 Ecological Risk Assessment 6-9

7. ALTERNATIVE ANALYSIS 7-1

7.1 Effectiveness of the Alternatives 7-1

7.1.1 Protectiveness 7-1

7.2 Implementability of the Alternatives 7-3

7.2.1 Technical and Administrative Feasibility 7-3
7.2.2 Availability of Equipment, Personnel, and Disposal Services 7-7

7.3 Cost of the Alternatives 7-7

7.4 Summary of Alternative Evaluation 7-8

8. RECOMMENDED REMOVAL ACTION ALTERNATIVE 8-1

8.1 Compliance with Environmental Regulations 8-1

8.2 Cultural Resources 8-7

8.3 Natural Resources 8-7

8.4 Compliance with Disposal Facility Waste Acceptance Criteria 8-7

8.4.1 ICDF Waste Acceptance Criteria 8-7
8.4.2 Achieving Removal Action Objectives 8-8

9. REFERENCES 9-1

Appendix A—EBR-II Facility Radiological Inventory (2009) A-1

Appendix B—EBR-II Nonradiolgical Inventory B-1

Appendix C—Ecological Screening Tables C-1

FIGURES

2-1. Aerial view of the Materials and Fuels Complex 2-3

2-2. Location of MFC at the INL Site 2-3

2-3. Location of the EBR-II at MFC 2-4

2-4. MFC-767 EBR-II reactor building 2-5

2-5. EBR-II cross-section drawing 2-7

2-6. EBR-II reactor building main floor 2-8

2-7. Primary coolant tank with cover being lowered into biological shield 2-8

2-8. Primary coolant tank support structure over the tank 2-9

2-9. Schematic of EBR-II reactor vessel and neutron shields 2-10

2-10. Photograph of EBR-II core and blanket area 2-11

2-11. EBR-II reactor cross section 2-11

2-12. EBR-II reactor vessel and storage basket shown inside primary coolant tank 2-12

3-1. Conceptual drawing of EBR-II Alternative 1, No Action, final end state 3-1

3-2. Conceptual drawing of EBR-II Alternative 2, No Action: Continued Surveillance and Maintenance, end state 3-2

3-3. Conceptual drawing of EBR-II Alternative 3, Grouting Reactor Vessel in Place, final end state 3-3

3-4. Conceptual drawing of EBR-II Alternative 4, Removal of the EBR-II Reactor Vessel, final end state 3-4

4-1. EBR-II radiological inventory intervals for risk assessments 4-3

4-2. Dose rates in EBR-II primary coolant tank taken November 2009 4-4

7-1. EBR-II primary coolant tank reactor vessel support components 7-4

8-1. Current photo of EBR-II at MFC 8-2

8-2. Conceptual end state of EBR-II under Alternative 3 8-2

TABLES

6-1. Risk in CY 2095 by radionuclide and pathway for EBR-II Alternative 1, No Action 6-2

6-2. Risk in CY 2095 by radionuclide and pathway for EBR-II Alternative 3, Grout Reactor Vessel in Place 6-2

6-3. Risk in CY 2095 by radionuclide and pathway for EBR-II Alternative 4, Removal of the EBR-II Reactor Vessel 6-3

6-4. Risk by metal and pathway for EBR-II Alternative 1, No Action 6-3

6-5. Risk by metal and pathway for EBR-II Alternative 3, Grouting Reactor Vessel in Place 6-3

6-6. Groundwater pathway risk assessment results for Alternative 1 radionuclide contaminants of concern 6-5

6-7. Groundwater pathway results for the Alternative 1 nonradiological chemicals of concern 6-6

6-8. Groundwater pathway risk assessment Alternative 1, 3, and 4 results 6-8

7-1. Cost estimates for the viable removal action alternatives in 2009 dollars 7-8

8-1. Summary of applicable or relevant and appropriate requirements for the EBR-II non-time-critical removal action 8-4

ACRONYMS

ALARA	as-low-as-reasonably-achievable
ARAR	applicable or relevant and appropriate requirement
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
COC	contaminant of concern
CY	calendar year
D&D	decommissioning and demolition
DEQ	Idaho Department of Environmental Quality
DOE	U.S. Department of Energy
DOE-ID	U.S. Department of Energy Idaho Operations Office
EBR	Experimental Breeder Reactor
EBSL	ecologically based screening level
EE/CA	engineering evaluation/cost analysis
EPA	U.S. Environmental Protection Agency
FFA/CO	Federal Facility Agreement and Consent Order
HWMA	Hazardous Waste Management Act
ICDF	Idaho CERCLA Disposal Facility
ICP	Idaho Cleanup Project
INEEL	Idaho National Engineering and Environmental Laboratory (now INL)
INL	Idaho National Laboratory
MCL	maximum contaminant level
MFC	Materials and Fuels Complex
NTCRA	non-time-critical removal action
OU	operable unit
PCB	polychlorinated biphenyl
RCRA	Resource Conservation and Recovery Act

ROD	record of decision
WAC	waste acceptance criteria
ZPPR	Zero Power Physics Reactor

Engineering Evaluation/Cost Analysis for the EBR-II Final End State

1. INTRODUCTION

1.1 Purpose

The engineering evaluation/cost analysis (EE/CA) process documented in this report supports determining the final end state of the Experimental Breeder Reactor II (EBR–II) reactor building and reactor vessel and recommends a non-time-critical removal action (NTCRA) alternative to reach that final end state.

On February 17, 2009, the American Recovery and Reinvestment Act of 2009 became effective (Public Law 111-5). The American Recovery and Reinvestment Act contains provisions to fund additional decommissioning activities at the Idaho Cleanup Project (ICP), including final disposition of the EBR-II vessel and building. The remedial action described in this EE/CA will be performed as part of the ICP. The EBR-II facility is located at the Materials and Fuels Complex (MFC), formerly Argonne National Laboratory–West, at the Idaho National Laboratory (INL) Site.

This EE/CA identifies proposed removal action alternatives and analyzes these alternatives for effectiveness, implementability, and cost. Following the issuance of this EE/CA for public comment and consideration of comments received, the U.S. Department of Energy (DOE) will issue, with concurrence from the Idaho Department of Environmental Quality (DEQ) and U.S. Environmental Protection Agency (EPA), an action memorandum documenting the selected alternative. This EE/CA is intended to satisfy environmental review requirements while providing a framework for selecting the final end state and satisfying Administrative Record requirements for documentation of the removal action. This EE/CA was developed in accordance with the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) as amended by the "Superfund Amendments and Reauthorization Act of 1986" (Public Law 99-499) and in accordance with the "National Oil and Hazardous Substances Pollution Contingency Plan" (40 CFR 300).

The removal action objectives of this NTCRA are consistent with the *Final Record of Decision, Argonne National Laboratory-West, Operable Unit 9-04 Idaho National Engineering and Environmental Laboratory (*DOE-ID 1998), which provides the basis for cleanup levels at MFC. The removal action objectives support the overall cleanup objectives established in the Federal Facility Agreement and Consent Order (FFA/CO) (DOE-ID 1991). These removal action objectives also support the DOE goal of reducing the "risk footprint" to as practicable extent as possible in consideration of as-low-as-reasonably-achievable (ALARA) principles governing radiological exposure to decommissioning personnel, safe engineering standards, Idaho CERCLA Disposal Facility (ICDF) waste acceptance criteria (WAC), and desired CERCLA site end states.

This NTCRA is also consistent with the joint DOE and EPA *Policy on Decommissioning of Department of Energy Facilities Under the Comprehensive Environmental Response, Compensation, and Liability Act* (DOE and EPA 1995), which established the CERCLA NTCRA process as the preferred approach for decommissioning surplus DOE facilities. Under this policy, an NTCRA may be executed if DOE determines that the action will *prevent, minimize, stabilize, or eliminate* a risk to human health or the environment. When DOE determines that the criteria for executing a CERCLA NTCRA have been met, DOE is thereby authorized to evaluate, select, and implement the removal action that DOE determines is most appropriate to address the potential risk posed by the release or threat of release. This

action is taken in accordance with applicable authorities and in conjunction with EPA and the State of Idaho pursuant to Section 5.3 of the FFA/CO (DOE-ID 1991).

Performance of this removal action will place EBR-II in a final end state that is protective of human health and the environment. As the lead agency, DOE has determined that a removal action is an appropriate means to accomplish the final end state and achieve environmental review requirements. Both DEQ and EPA concur that an NTCRA is warranted to place EBR-II in a final configuration that is protective of human health and the environment.

This EE/CA will become part of the ICP CERCLA Administrative Record, which is part of the overall INL Site Administrative Record. Documentation supporting this EE/CA, such as the associated engineering design files (EDF-9359 and EDF-9374), will also be included in the Administrative Record. The INL Site Administrative Record is on the Internet at http://ar.inel.gov/ and is available to the public at the following locations:

Albertsons Library	INL Technical Library
Boise State University	DOE Public Reading Room
1910 University Drive	1776 Science Center Drive
Boise, ID 83725	Idaho Falls, ID 83415
(208) 426-1625	(208) 526-1185

1.2 Scope

This EE/CA evaluates four alternatives for the final end state of the EBR-II reactor building and reactor vessel and recommends a preferred alternative for the final end state of this facility.

1.3 Document Organization

The subsequent sections of this EE/CA are organized as follows:

- Section 2, "Site Characterization," describes the INL Site and the EBR-II facility; summarizes background information about the EBR-II reactor building, systems, and reactor vessel and MFC, the facility in which EBR-II is located; and summarizes remediation activities under way at MFC.

- Section 3, "Identification of Removal Action Alternatives for EBR-II," presents four NTCRA alternatives for the EBR-II facility and describes in detail these alternatives.

- Section 4, "EBR-II Radiological and Nonradiological Inventories," discusses the inventories (source terms) of radiological and nonradiological materials currently in the EBR-II reactor building.

- Section 5, "Identification of Removal Action Objectives," identifies the removal action objectives and goals for activities associated with this NTCRA.

- Section 6, "Risk Assessments," describes the soil pathways risk analysis for human health, groundwater risk analysis, and ecological risk assessment that were prepared for this EE/CA based on the radiological and nonradiological inventories discussed in Section 4 to evaluate the NTCRA alternatives.

- Section 7, “Alternative Analysis,” evaluates each of the NTCRA alternatives for effectiveness, implementability, and cost in accordance with EPA guidance (EPA 1993).

- Section 8, “Recommended Removal Action Alternative,” discusses in detail the NTCRA alternative recommended in this EE/CA, presents the applicable or relevant and appropriate requirements (ARARs) to which the removal action must conform, and discusses how the preferred alternative will meet the removal action objectives.

- Section 9 compiles references used throughout this document.

- The appended material includes inventories of the radiological and nonradiological constituents at the EBR-II facility and presents the initial ecological risk screening for EBR-II contaminants.

2. SITE CHARACTERIZATION

This section describes the INL Site and the EBR-II facility; summarizes background information about the EBR-II reactor building, systems, and reactor vessel and MFC, the facility in which EBR-II is located; and summarizes remediation activities under way at MFC.

2.1 Site Description and Background

2.1.1 Idaho National Laboratory Site

The INL Site, managed by DOE, is located 51 km (32 mi) west of Idaho Falls, Idaho. It occupies 2,305 km^2 (890 mi^2) of the northeastern portion of the Eastern Snake River Plain. In 1949, the U.S. Atomic Energy Commission established the INL Site, then called the National Reactor Testing Station. Its purpose was to conduct nuclear energy research and related activities. It was renamed the Idaho National Engineering Laboratory in 1974 and then the Idaho National Engineering and Environmental Laboratory (INEEL) in 1997. In 2003, the INEEL was restructured into two separate business units: one for laboratory research and development missions (i.e., INEEL) and one for environmental cleanup activities (i.e., Idaho Completion Project). In February 2005, the two business units came under the management of two separate contractors, Battelle Energy Alliance, LLC, for the laboratory mission, and CH2M-WG Idaho, LLC, for environmental remediation. In addition, the laboratory was designated as the lead DOE laboratory for U.S. nuclear energy research, the laboratory was renamed the INL in keeping with its mission realignment and multiple uses, and the physical Site area similarly is called the INL Site. The Idaho Completion Project was renamed the Idaho Cleanup Project (ICP), and its mission continues to focus on environmental remediation and cleaning up historic contamination at the INL Site.

The U.S. Department of Energy Idaho Operations Office (DOE-ID) controls all land within the INL Site. Public access is restricted to public highways, sponsored tours, special-use permits, and the EBR-I National Historic Landmark. In addition, DOE-ID is cognizant of the Shoshone-Bannock tribal members' need for access to areas on the INL Site for cultural and religious purposes.

The INL Site is located primarily in Butte County; however, it also occupies portions of Bingham, Bonneville, Clark, and Jefferson counties. The 2000 census indicated the following populations for cities in the region: Idaho Falls–50,730; Pocatello–51,466; Blackfoot–10,419; Arco–1,026; and Atomic City–25.

Surface water flows on the INL Site consist mainly of three streams draining intermountain valleys to the north and northwest of the INL Site: (1) the Big Lost River, (2) the Little Lost River, and (3) Birch Creek. All of the channels terminate on the INL Site. Flows from Birch Creek and the Little Lost River seldom reach the INL Site because of irrigation withdrawals upstream. The Big Lost River and Birch Creek may flow onto the INL Site before the irrigation season or during high-water years, but the terminal reaches are usually dry. In those few wetter years when the Big Lost River carries water to the end of its channel, the water sinks into the ground.

The physical characteristics, climate, flora and fauna, demography, and cultural resources of the INL Site and MFC are further described in the Operable Unit (OU) 9-04 Record of Decision (ROD) (DOE-CH 1998).

2.1.2 Materials and Fuels Complex History

Originally under the name of Argonne National Laboratory–West, MFC was established in 1949 and was operated by the University of Chicago under the direction of DOE's Chicago Operations Office. For the next 50 years, the primary function of the facility was to design and develop the next steps in nuclear reactor power stations. Some of the facilities at MFC that played a role in reactor design include the Transient Reactor Test facility, the Zero Power Physics Reactor (ZPPR), and EBR-II. The Transient Reactor Test facility was used for testing the performance of materials in a nuclear reactor, especially during power excursions. The ZPPR facility was used to test reactor design features for different fuel materials and configurations. In 1969, the ZPPR was put into operation; in 1989, it was put in standby; and in 2009, the reactor was dismantled. EBR-II operated for 30 years, providing power and serving as a point of research for reactor development and testing.

In 2005, Argonne National Laboratory–West was renamed the Materials and Fuels Complex (MFC) and began operating under the direction of DOE-ID. Two principal activities are being conducted at the complex today. The electrometallurgical treatment of sodium-bonded spent nuclear fuel from EBR-II takes place in shielded hot cells at the MFC-765, Fuel Conditioning Facility, and converts unstable spent nuclear fuel to stable and disposable waste forms. The other main activity is destructive and nondestructive examination of nuclear materials for government and private customers. This activity is performed in one of the most capable nuclear hot cells in the country, the Hot Fuel Examination Facility. Figure 2-1 is an aerial view of MFC.

2.1.3 EBR-II Location

EBR-II is located at the MFC in the southeast portion of the INL Site (see Figure 2-2). The EBR-II reactor building (MFC-767) is located approximately in the west-central portion of MFC (see Figure 2-3).

2.1.4 EBR-II History

While it was in active operation, EBR-II was an unmoderated, sodium-cooled reactor and power plant with a power output of 62.5 MW of heat. The energy of the reactor was converted to 20 MW of electricity through a conventional steam cycle. The reactor was initially fueled with U-235; however, it was later used as an irradiation facility for testing mixed oxide fuel pins for liquid-metal-cooled reactors. During operation of the EBR-II reactor, an integral fuel processing facility was contained in MFC-765 where the irradiated fuel was processed, fabricated, and assembled for return to the reactor.

EBR-II achieved initial "dry" (i.e., without sodium) criticality on September 30, 1961, and "wet" criticality (i.e., with the core submerged in liquid sodium coolant) on November 11, 1963. EBR-II went to full power (12 MW_e) on August 13, 1964. EBR-II was shut down in 1994 amid concerns about plutonium production. In that year, Argonne National Laboratory established the EBR-II Plant Closure Project and included three phases.

Phase I involved defueling the reactor and was performed from October 1994 to December 1996. During the defueling, all reactor core fueled subassemblies, neutron source, inner blanket, reflector, outer blanket, and control and experiment subassemblies were removed and replaced with unirradiated stainless steel dummy assemblies. The only irradiated components remaining in the reactor core/blanket regions following this activity were 12 control rod thimbles, eight control rod drive mechanisms, and a welded-in-place stainless steel dummy subassembly. All removed components are stored at the MFC with the exception of the fuel subassemblies, which are currently being transferred to dry storage at the INL Idaho Nuclear Technology and Engineering Center. Additional work activities, which reduced the pre-demolition source term of the facility, included the removal of the primary cold traps and cesium traps.

Figure 2-1. Aerial view of the Materials and Fuels Complex.

Figure 2-2. Location of MFC at the INL Site.

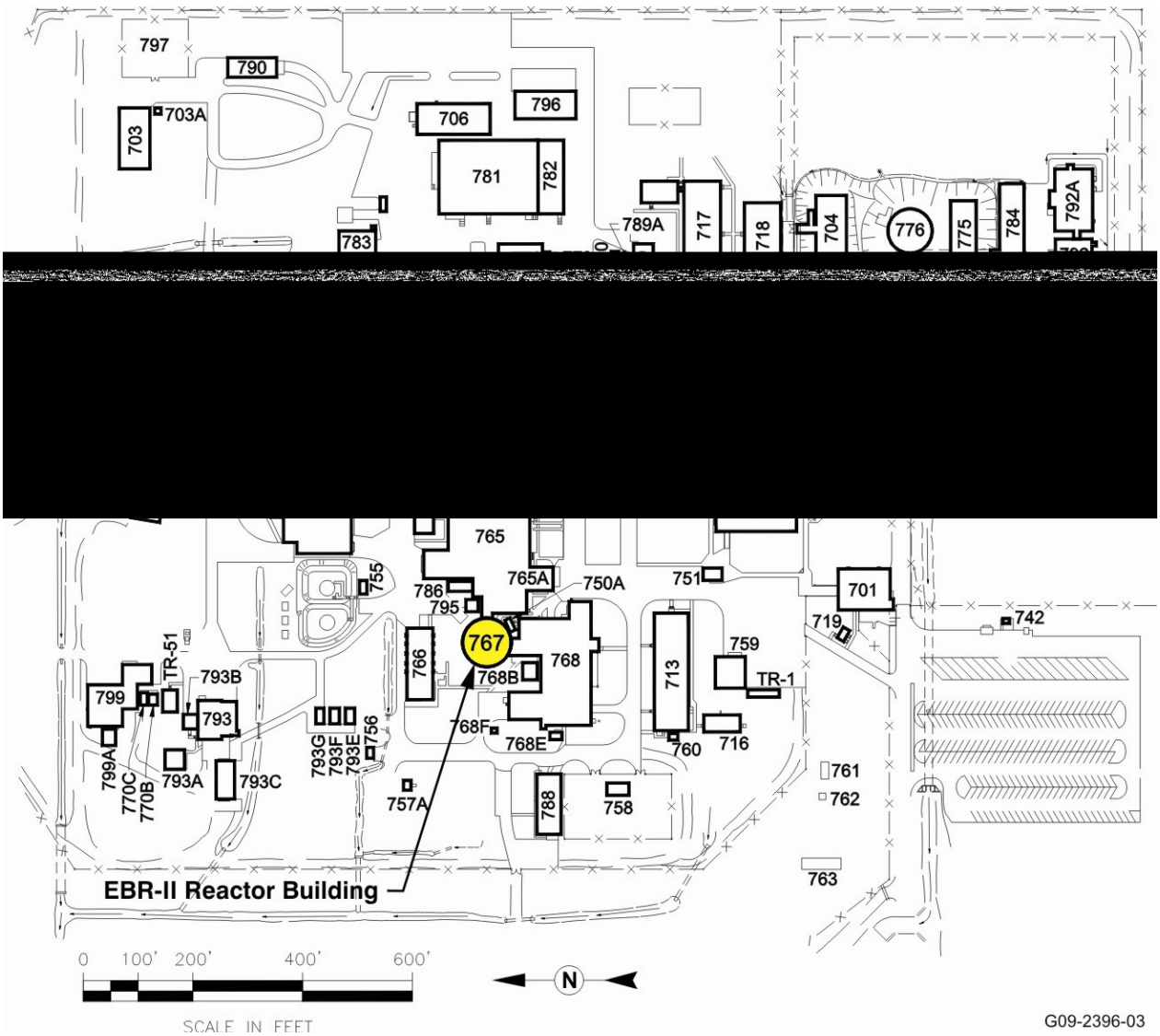

Figure 2-3. Location of the EBR-II at MFC.

Phase II involved removal of the bulk liquid sodium metal (the sodium was melted for removal) and some sodium-potassium alloy (NaK) from the primary and secondary systems and was completed in March 2001.

Phase III involved treating the EBR-II primary and secondary systems residual sodium volumes with CO_2 and water vapor in a process called carbonation. The purpose of carbonation was to cause the sodium to react with water to form a more stable material, such as sodium bicarbonate or sodium carbonate. Carbonation was stopped in March 2002, then restarted in May 2004 and continued through December 2005. Carbonation treatment was terminated in December 2005 because of diminishing reaction rates. Currently, the EBR-II reactor building has an estimated 850 gal of uncarbonated (elemental) sodium remaining on surfaces and in other ancillary equipment that will be treated or removed before the final end state of EBR-II is reached. There is also residual liquid NaK remaining inside pressure transducers and shutdown coolers located in the primary coolant tank.

2.1.5 EBR-II Systems and Component Descriptions

Descriptions of major EBR-II systems and components are provided in subsections below.

2.1.5.1 Reactor Building. Construction on the EBR-II reactor building (MFC-767) began in 1957 (see Figure 2-4). The building is 80 ft in diameter and approximately 139 ft in height, 47 ft below and 92 ft above ground level. The main floor of the facility sits approximately 7 ft above ground level. The structure is a concrete-lined, gas-tight cylindrical steel shell, constructed of a 1-in.-thick carbon steel plate, and is designed to withstand a static internal pressure of approximately 25 psig. Heat was removed from the reactor by the primary sodium coolant system and transferred to the secondary sodium system. The secondary coolant system transferred the heat out of the EBR-II reactor building to the adjacent Sodium Boiler Building, MFC-766, in which superheated steam was produced that drove a conventional turbine-generator that generated electricity.

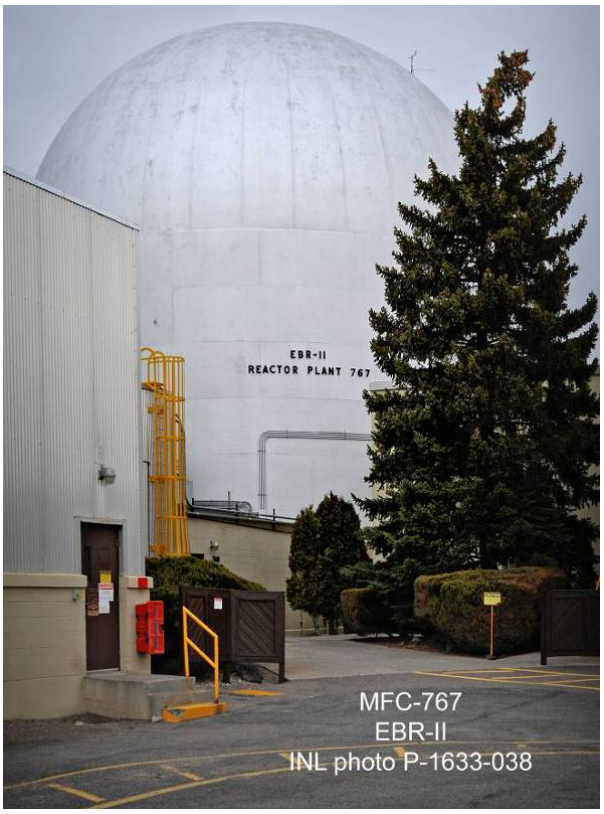

Figure 2-4. MFC-767 EBR-II reactor building.

The reactor building contains three main accessible working levels:

- The main floor contains the reactor subassembly movement components, cover gas cleanup system, and control rod drive tower, along with electrical distribution panels. A polar crane is situated near the top of the reactor building and was used for lifting equipment on the main floor and from lower levels of the building through hatches built in the main floor. The main floor is situated approximately 7 ft above ground level. Just below the main floor is a small space known as the depressed area. The depressed area floor is approximately at ground level and contains access to the emergency exit air lock and several support systems.

- The next level down is the basement area and contains the primary coolant purification system, Radioactive Sodium Chemistry Loop, primary coolant sampling system, ventilation ducting, and inert gas systems. Also located in the basement are two separate mezzanines that contained primarily electrical equipment.

- The lowest level of the building is known as the subbasement and houses primarily ventilation systems for the reactor shield coolant system and neutron detector thimble cooling.

Figure 2-5 is a cross section of EBR-II, and Figure 2-6 is a photo of the main reactor building floor.

The primary coolant tank makes up the balance of the space in the EBR-II reactor building and is described in the sections below.

2.1.5.2 EBR-II Primary Coolant Tank. The EBR-II reactor vessel and the primary coolant system, including the heat exchanger and sodium coolant pumps, are contained in a large tank that was operated completely submerged in the liquid sodium metal. The primary coolant tank is of double-wall construction to provide maximum reliability of sodium containment. The tank is constructed of Type 304 stainless steel. The inner tank has a 26-ft internal diameter and the outer tank wall has a 26-ft, 11-in. internal diameter. The inner side wall is constructed of 1/2-in.-thick plate. The outer side wall is constructed of 1/4-in.-thick plate. The bottoms of the tanks are 1-in.-thick plates and are stiffened with radial beams. The primary coolant tank support structure consists of a system of columns and beams that transmit the loads to the main internal building foundation. In combination with the biological shield, it forms a "pressure vessel" surrounding the primary tank. A ring of ordinary concrete measuring 6 ft thick forms the biological shield. The concrete is reinforced with continuous hoops of reinforcing rods to allow the concrete to resist an internal pressure of 75 psig.

The primary coolant tank cover is welded to the top of the inner and outer tank walls and provides numerous nozzles (e.g., access ports) for equipment access to the primary coolant tank. A large and a small rotating plug were fitted inside a large center access hole. The small plug rotated inside the large plug and the large plug rotated inside the primary coolant tank cover center hole to allow very accurate orientation over the reactor vessel below. Figure 2-7 is a photo during construction of the empty EBR-II primary coolant tank with the installed cover being lowered into the biological shield.

Once the primary coolant tank was in place in the biological shield, a six-spoke support structure was assembled over the tank, and the tank was then suspended from this structure within the biological shield. By suspending the coolant tank from the support structure, thermal expansion of the metals composing the tank could be accommodated without warping or damaging systems. Figure 2-8 is a photo of the primary coolant tank support structure assembled over the primary coolant tank.

2.1.6 Reactor Vessel and Neutron Shield Description

The reactor vessel is located in the primary coolant tank and was constructed of an outer shell, an inner shell, and thermal baffles, all of 304 stainless steel. The reactor vessel is considered to extend from the bottom grid plenum assembly to the reactor vessel cover on the top (see Figure 2-9).

During operation, the EBR-II reactor vessel contained a hexagonal central core, which, in turn, contained enriched uranium that was completely surrounded by radial and axial subassembly layers or "blankets." These blankets originally contained stainless steel or depleted uranium subassemblies that were intended to function both as a neutron reflector and as fertile material for the breeding of plutonium. In 1994, when the reactor was shut down, the radial blanket regions contained stainless steel reflector

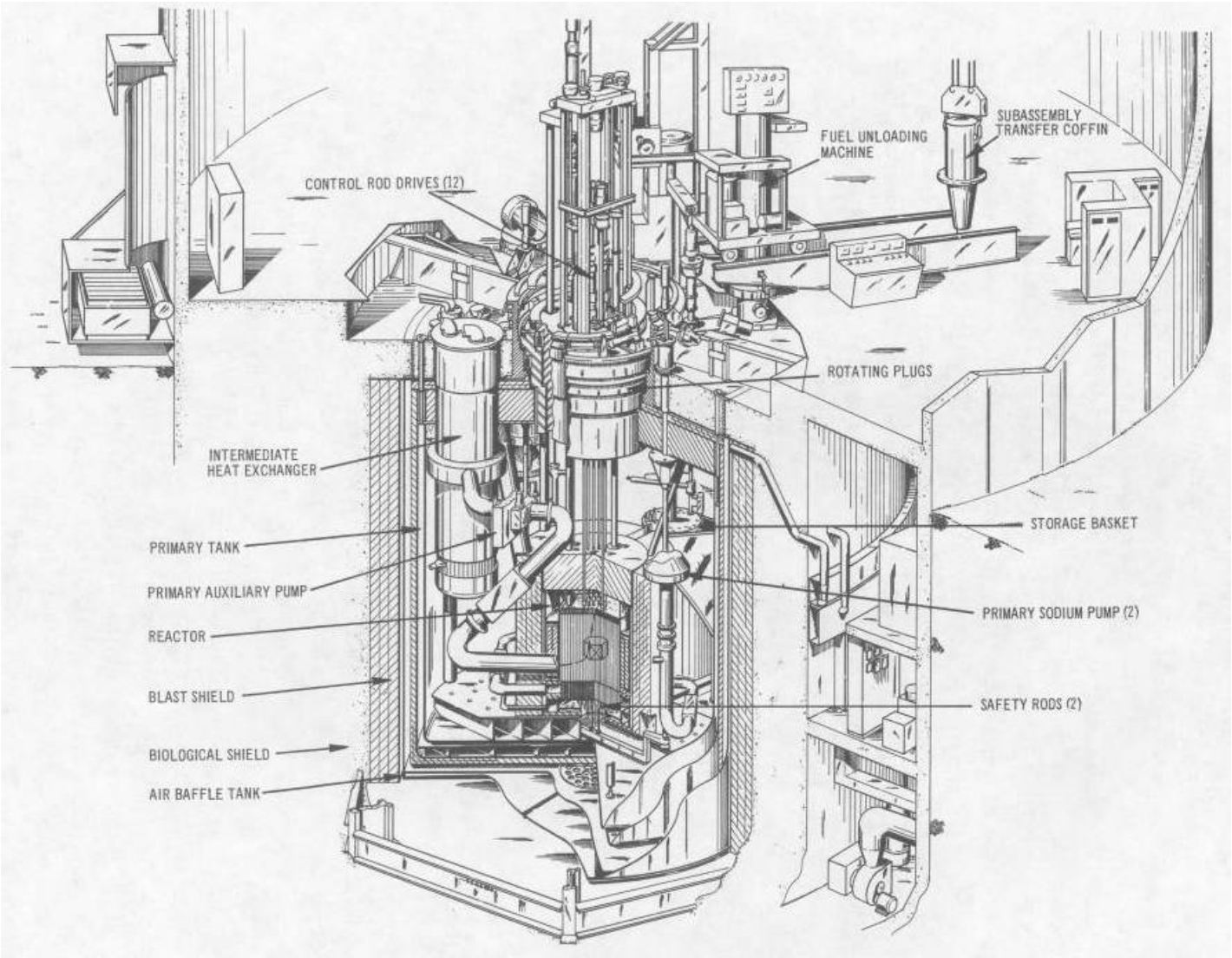

Figure 2-5. EBR-II cross-section drawing.

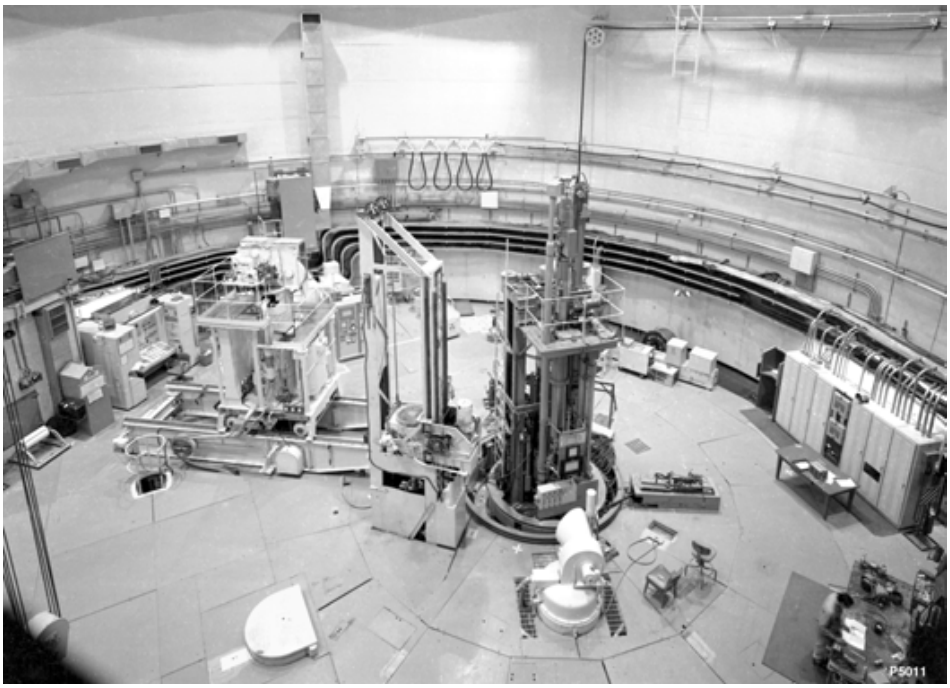

Figure 2-6. EBR-II reactor building main floor.

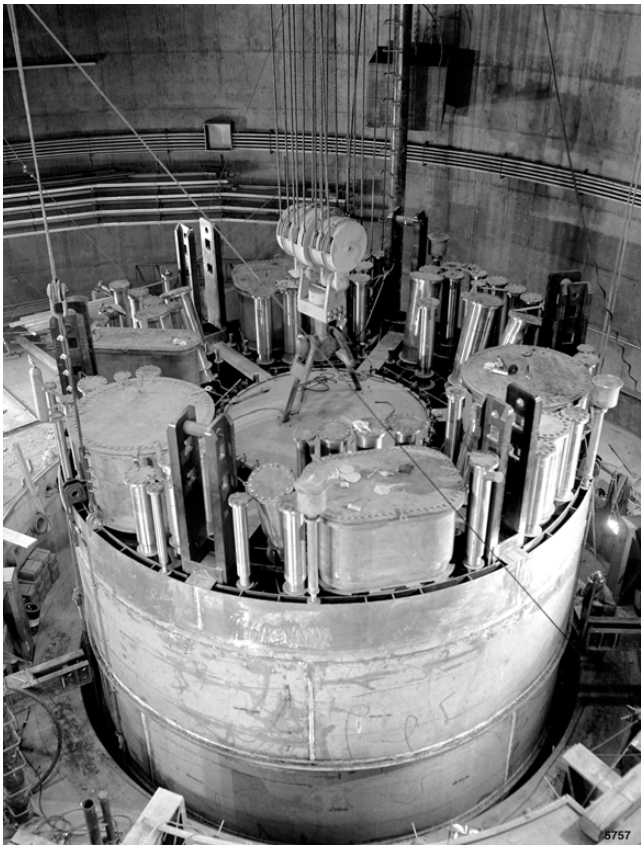

Figure 2-7. Primary coolant tank with cover being lowered into biological shield.

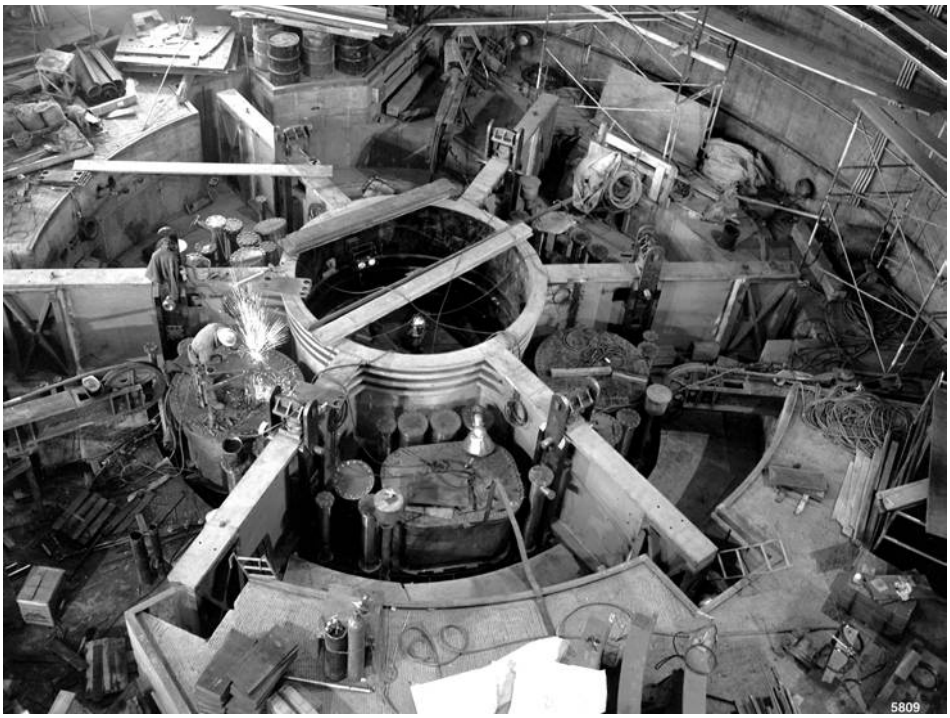

Figure 2-8. Primary coolant tank support structure over the tank.

pieces and depleted uranium, or stainless steel and nickel blanket pieces. The core and blanket region materials consisted of 636 free-standing and one welded-in-place subassemblies, which were contained in and supported by the reactor vessel. Figure 2-10 is a photograph of the core and blanket area of EBR-II. Figure 2-11 is a cross section of the EBR-II reactor vessel and neutron shields.

The radial neutron shield reduced neutron leakage from the reactor to minimize activation of the secondary sodium coolant and components exterior to the reactor. It also served as a neutron reflector, conserving a relatively small number of neutrons that would have otherwise leaked from the reactor. The entire radial shield consists of an inner shield, located between the inner and outer shells of the reactor vessel, and an outer shield located outside of the vessel. A total of 108 "cans" form the two layers of inner neutron shield and 948 cans form the five layers of outer neutron shield. Most cans are 4.3 in. square and 52.5 in. long. The cans contain either graphite or borated graphite or were empty and open at the top to allow liquid sodium to fill the can. The cans that contained sodium were allowed to drain when the sodium was drained from the primary coolant tank because of a small hole in the bottom of each can.

The reactor vessel assembly includes the double-walled reactor vessel, seven rows of neutron shielding cans, coolant plenums, grid plates, the reactor vessel cover, and the 637 stainless steel subassemblies. The reactor vessel assembly is approximately 12 ft in diameter and 13 ft high and weighs approximately 170 tons.

2.1.6.1 Primary Coolant Tank Components. The primary coolant tank held approximately 86,000 gal of molten sodium metal during reactor operations. The operating temperature of the EBR-II reactor coolant system was between 700 and 883°F. During operations, sodium was continuously circulated through the reactor vessel to maintain the desired core temperature. Sodium was pumped by two primary coolant pumps that were suspended from the primary coolant tank cover into the sodium. Coolant piping ran from the pumps to the plenums below the reactor vessel and collected the thermally "hot" coolant at the top of the reactor vessel and transported it to the intermediate heat exchanger. The

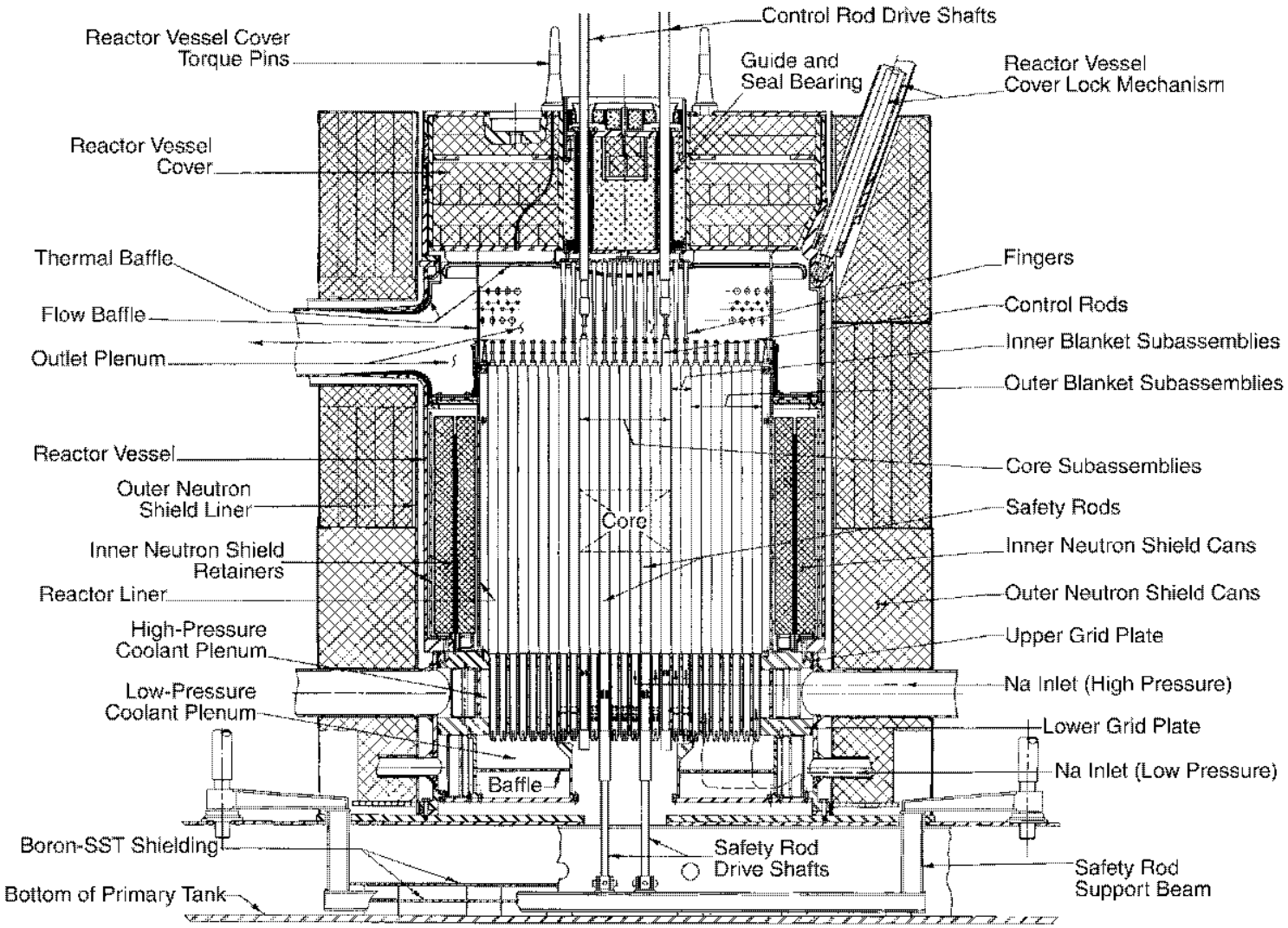

Figure 2-9. Schematic of EBR-II reactor vessel and neutron shields.

Figure 2-10. Photograph of EBR-II core and blanket area.

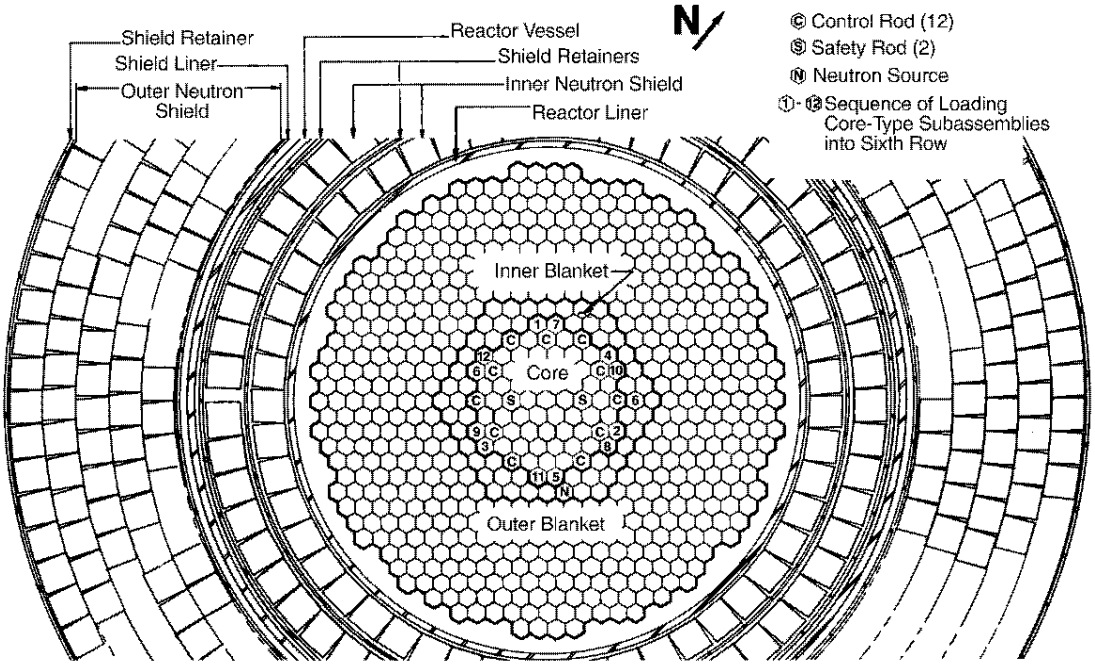

Figure 2-11. EBR-II reactor cross section.

primary sodium was then cooled by transferring its thermal energy to the secondary sodium loops present in the heat exchanger. Like the primary pumps, the intermediate heat exchanger was suspended into the primary coolant tank from the tank cover. Primary coolant was released back into the primary coolant tank after it passed through the intermediate heat exchanger. The secondary sodium left the building though piping that ran to the adjacent sodium boiler building where this sodium was in turn cooled by transferring thermal energy to a water loop that produced steam, which was then used to generate electricity in a turbine generator.

The melting point of sodium is 208°F. Six submersible heaters maintained the sodium in the reactor at approximately 700°F when it was not in operation. Two shutdown cooler columns in the primary coolant tank transported excess heat to heat exchangers on the outside of the reactor building from the primary coolant if the primary pumps failed.

The primary coolant tank also held the subassembly handling system and storage basket. Subassemblies were loaded and unloaded from the primary coolant tank and placed in the reactor vessel using the transfer arm. Subassemblies removed from the reactor core could be placed in the storage basket to allow short half-lived radionuclides to decay to a lower activity before being removed from the primary coolant tank. The foreground in Figure 2-12 shows the top of the reactor vessel with the control rods descending into the reactor vessel cover, and the storage basket is shown on the far side of the reactor vessel. This photo was taken during construction of components in the primary coolant tank.

The primary coolant tank also contained many thermocouples for measuring temperatures and pressure transducers. Numerous hollow tubes called “thimbles” extend from the primary coolant tank cover into the tank to allow instrumentation such as fission chambers for measuring neutron flux. Overall, the primary coolant tank is very crowded with systems, structures, and components.

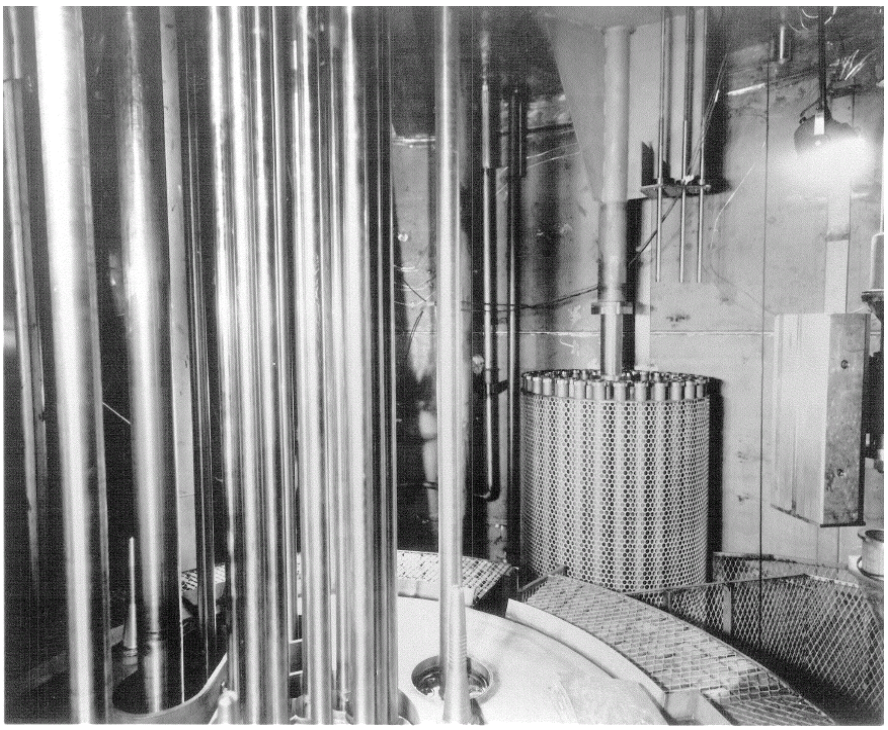

Figure 2-12. EBR-II reactor vessel and storage basket shown inside primary coolant tank.

2.2 Other Closure and Cleanup Activities at MFC

Closure and cleanup activities have taken place and will continue at MFC under several programs and regulatory authorities. The following sections briefly describe those activities.

2.2.1 CERCLA Activities at MFC

The MFC FFA/CO remedial action objectives for Waste Area Group 9 are documented in the OU 9-04 ROD (DOE-CH 1998), the *Final Explanation of Significant Differences to the Record of Decision for Argonne National Laboratory-West Operable Unit 9-04, Idaho National Engineering and Environmental Laboratory* (DOE-CH 2000), and the *Explanation of Significant Difference Argonne National Laboratory-West, Operable Unit 9-04* (DOE-CH 2004). Past release sites requiring further actions are documented in the OU 9-04 ROD. Remedial actions are complete for seven of the release sites identified, and continued institutional controls to prevent inadvertent access to three of the release sites are in place.

The OU 9-04 ROD states that groundwater monitoring will continue until 2018. The MFC groundwater monitoring program consists of one upgradient well and three downgradient wells. In addition, one production well is sampled from within the MFC security area. All wells are sampled twice annually—typically in April and October.

2.2.2 Other Non-Time-Critical Removal Action Activities at MFC

Decommissioning and demolition (D&D) of the ZPPR is nearing completion. D&D of the reactor portion of the ZPPR facility was conducted in accordance with a CERCLA NTCRA as allowed by the *Action Memorandum for General Decommissioning Activities under the Idaho Cleanup Project* (DOE-ID 2009a).

Other facilities at MFC are undergoing D&D or are scheduled for D&D under the Action Memorandum for General Decommissioning. These facilities include MFC-766, Sodium Boiler Building; MFC-750A, EBR-II Experimental Building; MFC-757A, EBR-II Cooling Tower System Building; MFC-793A, Alcohol Recovery Facilities Storage Pad; MFC-793B, Alcohol Recovery Facility; and MFC-795, EBR-II Cover Gas Cleanup System. D&D of these facilities should be completed by the end of Calendar Year (CY) 2011. D&D has been completed on MFC-793E and F, Sodium Storage Buildings, and MFC-752, Cooling Tower Pad.

2.2.3 Resource Conservation and Recovery Act Activities

Residual sodium and sodium-potassium (NaK) alloy remaining in the EBR-II facilities are being managed outside of the NTCRA and in accordance with the *HWMA/RCRA Storage and Treatment Permit for the Experimental Breeder Reactor-II and Buildings MFC-793E and MFC-793F located at the Materials and Fuels Complex on the Idaho National Laboratory* (DEQ 2009). This Resource Conservation and Recovery Act (RCRA) (42 USC 6901 et seq.) permit and the associated closure plan will guide the handling, storage, and treatment of sodium and NaK until the closure performance standards provided in the closure plan are satisfied. Sodium and NaK treatment is expected to take place before the initiation of activities under this NTCRA, but some closure activities may indeed occur concurrently with this NTCRA. There currently is estimated to be 500 gal of elemental sodium in the primary coolant tank and 350 gal in the ancillary systems (e.g., Radioactive Sodium Chemistry Loop). Much of the sodium in the primary tank is coated with a sodium bicarbonate layer that is a result of previous treatment (carbonation). Moist carbon dioxide was vented into the primary coolant tank and some of the ancillary equipment to slowly react the sodium. This treatment resulted in the sodium

bicarbonate layer and will likely have to be dissolved in future treatment regimes to allow treatment of the underlying untreated sodium metal.

Actions are currently ongoing in MFC-767 to remove shielding lead and lead-containing components, mercury switches, silver solder, circuit boards, and florescent lamps to ensure compliance with Hazardous Waste Management Act (HWMA) (IC 1983)/RCRA regulations. Also, asbestos is being removed throughout the facility in accordance with National Emission Standards for Hazardous Air Pollutants regulations. While the intent of this NTCRA is to remove accessible lead and lead shielding, it must be recognized that not all lead will be removed under this NTCRA. Some lead wall anchors would remain in place under all the alternatives. Additional lead that will not be removed is found in the painted surfaces of the buildings that have had lead-containing paint applied at various times over the lifetime of this facility.

3. IDENTIFICATION OF REMOVAL ACTION ALTERNATIVES FOR EBR-II

This section proposes and discusses four alternatives for evaluation for the EBR-II reactor vessel disposition and containment building end state.

3.1 Alternative 1, No Action

Under the "no action" alternative, no D&D would be conducted at EBR-II and no further surveillance and maintenance would be conducted at the facility. The No Action alternative offers no reduction in toxicity, mobility, or volume of contaminants.

For risk analysis, the No Action alternative is a hypothetical, conservative, baseline assumption in that the sum of all identified radiological or chemical contamination, if not properly contained or controlled, may be released to the environment causing an unacceptable risk to potential receptors. These assumptions are for comparative purposes only and do not reflect the DOE requirement to comply with federal and state laws that require DOE to monitor, maintain, and mitigate potential or actual hazardous or radiological constituent releases to the public or the environment from any facility or site. Currently, administrative and physical controls are in place to prevent unacceptable exposures of ionizing radiation and other chemical hazards from the contaminated materials contained in EBR-II; however, implementation of the No Action alternative would remove these controls. Therefore, Alternative 1 is not considered a viable alternative and is presented only for comparative purposes.

Under the No Action alternative, the EBR-II containment building is assumed to degrade over the next 85 years to the point where it crumbles to the ground and contamination becomes available for uptake for the hypothetical future resident. All the rubble and contamination from the reactor building are assumed to be mixed with the soil in the interval at the top 10 ft below ground surface. Figure 3-1 is a conceptual drawing of the final end state of Alternative 1.

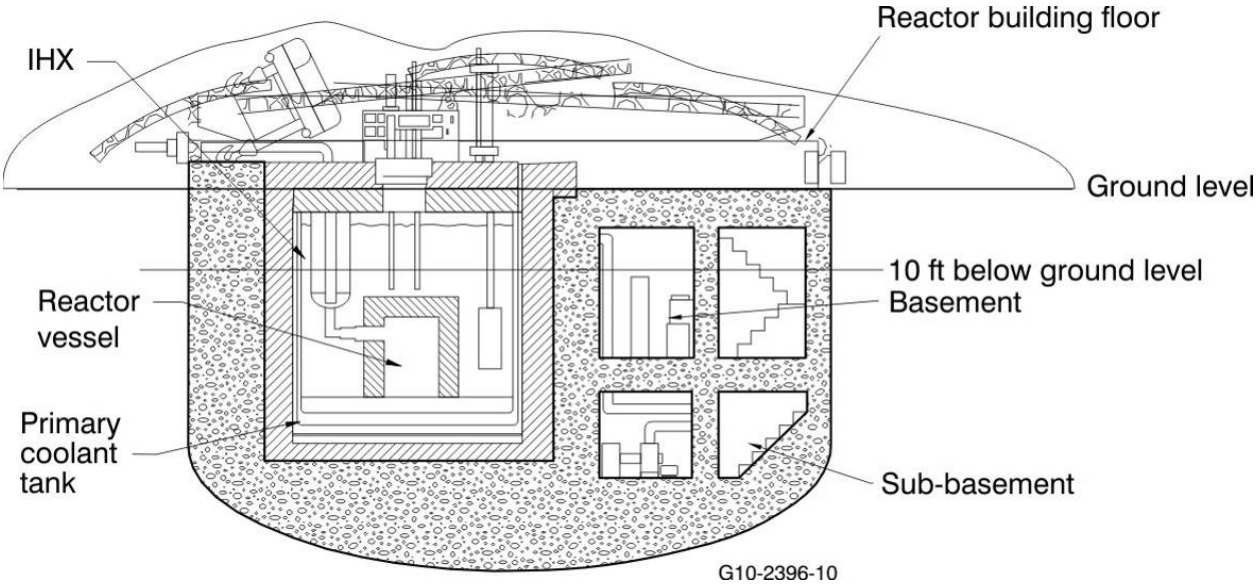

Figure 3-1. Conceptual drawing of EBR-II Alternative 1, No Action, final end state.

3.2 Alternative 2, No Action: Continued Surveillance and Maintenance

Alternative 2 includes no action except surveillance and maintenance. This alternative also offers no reduction in toxicity or volume of contaminants, but it does provide more protection from mobilization of the contaminants to the environment than Alternative 1.

Maintenance includes maintaining the MFC-767 EBR-II containment building and ancillary support systems that provide power and ventilation to the reactor building. Surveillance includes periodic facility inspections to ensure building integrity and systems operability. Various systems, such as ventilation systems, will need to remain operable to prevent the potential release of radiological or chemical constituents to the environment from causing an unacceptable risk to the public and nearby workers. This assumption is not consistent with the DOE goal to reduce surveillance and maintenance costs of legacy buildings and structures. Figure 3-2 is a conceptual drawing representing Alternative 2.

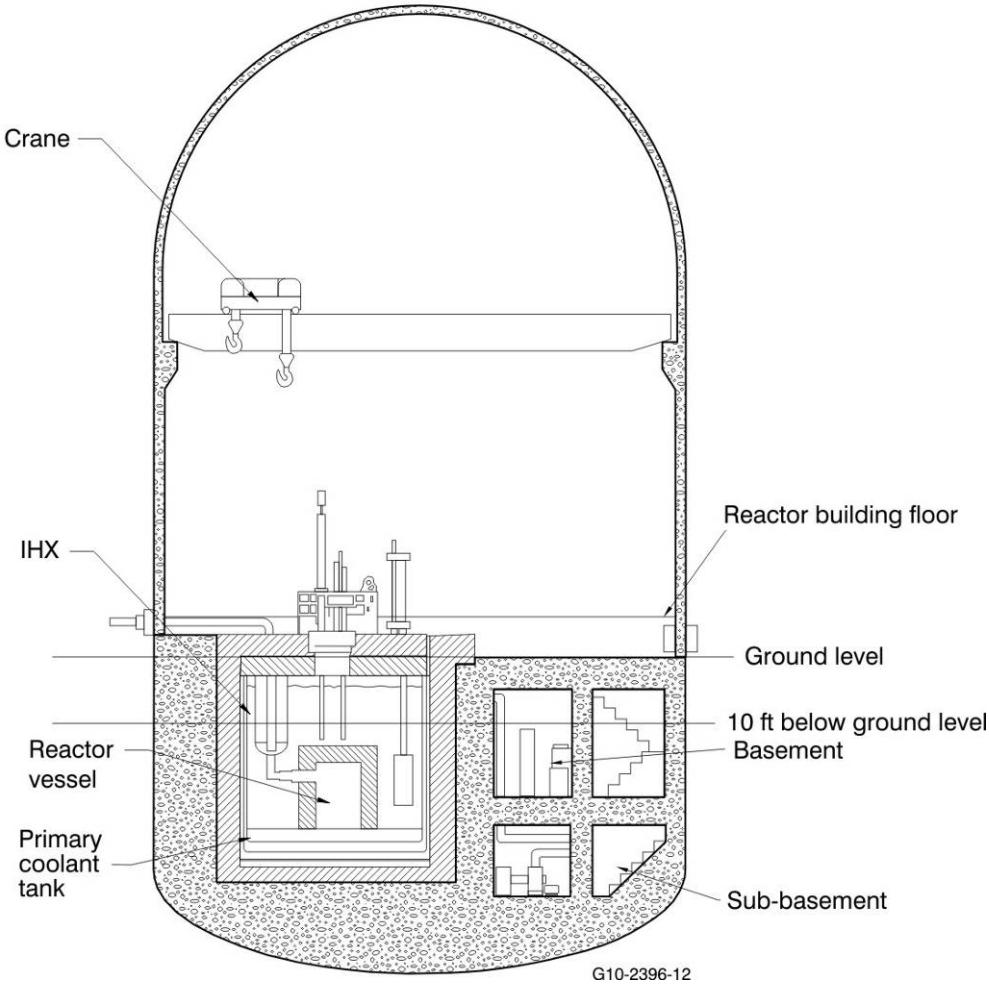

Figure 3-2. Conceptual drawing of EBR-II Alternative 2, No Action: Continued Surveillance and Maintenance, end state.

3.3 Alternative 3, Grouting the EBR-II Reactor Vessel in Place

Under Alternative 3, systems and structures above the reactor building floor would be demolished and most of the remaining systems and structures below floor level, including the EBR-II reactor vessel, would be grouted in place. The final end state of EBR-II under Alternative 3 is a concrete/grout monolith that contains the EBR-II primary coolant tank with internal components, including the reactor vessel. Void spaces remaining would be grouted as practicable, including the basement, sub-basement, and interior of the primary coolant tank, resulting in encapsulation of the reactor vessel. The concrete/grout monolith will extend approximately 8 ft above ground level and will be finished with a concrete cover to facilitate drainage away from the EBR-II site. Residual radioactive materials at EBR-II remaining after D&D activities are completed would stay in place and be managed under the Long-term Management and Control Program. Alternative 3 would take place after the residual sodium metal and sodium-potassium alloy (NaK) remaining in the EBR-II systems are treated or removed. Systems that will be removed include the Radioactive Sodium Chemistry Loop and the Sodium Purification System since both of these systems contain sodium and are shielded by lead. Treatment or removal will be in compliance with the approved HWMA/RCRA closure plan. Figure 3-3 is a conceptual drawing of the Alternative 3 final end state.

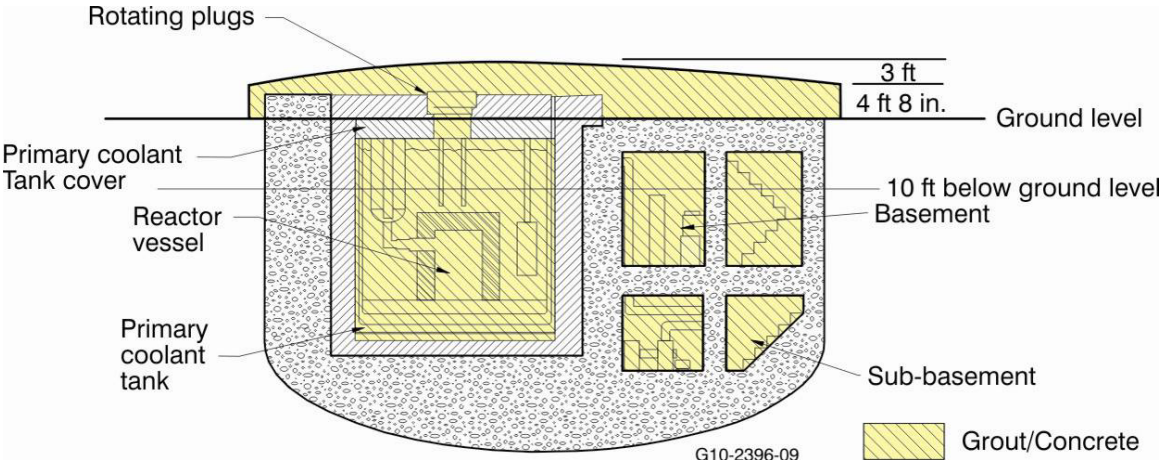

Figure 3-3. Conceptual drawing of EBR-II Alternative 3, Grouting Reactor Vessel in Place, final end state.

3.4 Alternative 4, Removal of the EBR-II Reactor Vessel

Alternative 4 includes demolition of the EBR-II reactor building and removal and disposal of the EBR-II reactor vessel. To allow removal of the reactor vessel, the primary coolant tank, primary coolant tank cover, tank support structure, heat exchanger, coolant pumps, and much of the coolant lines would be removed. The containment building would be demolished to ground level or below. Radioactive waste, including the reactor vessel and primary sodium tank components, would be removed from the EBR-II site and disposed of at ICDF in accordance with ICDF WAC. Void spaces would be grouted as practicable, including the void left by removal of the primary coolant tank. Residual radioactive materials at EBR-II remaining after D&D activities are completed would be left in place and managed under the Long-Term Management and Control Program. Alternative 4 would take place after the residual sodium metal and sodium-potassium alloy (NaK) remaining in the EBR-II systems are treated or removed. Treatment or removal will be in compliance with the approved HWMA/RCRA closure plan. Figure 3-4 is a conceptual drawing of the Alternative 4 final end state.

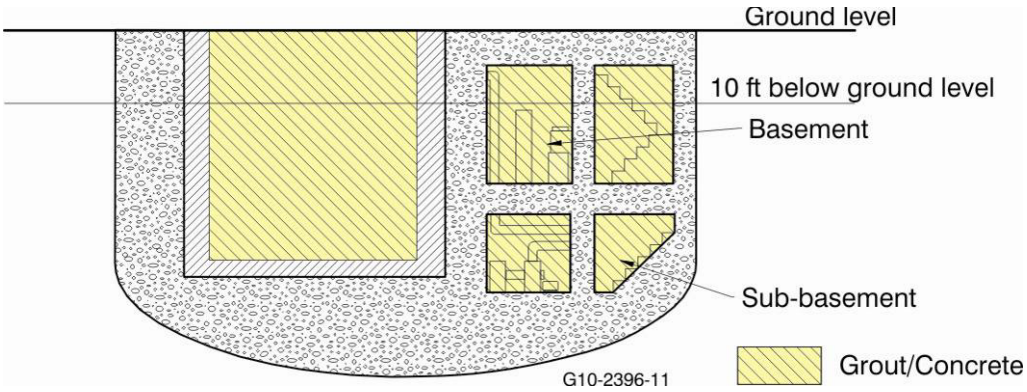

Figure 3-4. Conceptual drawing of EBR-II Alternative 4, Removal of the EBR-II Reactor Vessel, final end state.

4. EBR-II RADIOLOGICAL AND NONRADIOLOGICAL INVENTORIES

This section discusses the inventories (source terms) of radiological and nonradiological materials currently within the EBR-II reactor building. The distribution of these inventories in the EBR-II reactor building is also discussed and summarized tabularly in Appendixes A and B to support the risk assessments that are provided in Section 6.

4.1 Radiological Inventory

In the below sections, radiological inventories are discussed in the context of activity, or the rate at which a radioactive material decays, measured in curies (1 curie = 3.7E+10 decays per second), which is directly related to the quantity (mass) of the radioactive material. Items making a significant contribution to radiological inventories within the MFC-767 facility include the following items:

- Activated items remaining in the core/blanket region of the reactor vessel, including the 12 control rod thimbles, a welded-in dummy subassembly, and the remaining eight control rod drive rods
- Activated components within the primary tank but outside of the core/blanket
- Activation activity associated with the test facility extension tubes stored within the pentagon area
- The detectors for the wide range of nuclear instruments and delayed neutron detectors associated within in the primary coolant tank thimbles and shielded deep pit and pentagon area
- Residual primary sodium within the primary tank
- Residual primary sodium within the primary ancillary systems
- Secondary sodium remaining in the intermediate heat exchanger
- Residual surface contamination on components with the primary tank
- Surface contamination in the shielded deep pit and pentagon area.

While areas containing surface contamination are posted throughout the facility, the level of contamination and the actual contaminated area are both very small and thus do not significantly contribute to the facility inventory. Appendix A lists the current (CY 2009) radiological inventory of EBR-II.

The total radiological inventory in the EBR-II building is 1.50E+04 Ci. Of the 1.50E+04 Ci remaining, 1.44E+04 Ci (95.84%) are contained within the primary coolant tank. Of this primary coolant tank activity, a total of 1.437E+04 Ci (99.92%) is from activated materials remaining in the core/blanket regions and activated materials forming the reactor vessel structure. The remainder of the primary coolant tank activity is from the following sources: primary system surface contamination—1.12E+01 Ci, residual primary sodium metal and NaK—7.04E-02 Ci, residual secondary sodium metal remaining in the intermediate heat exchanger—6.14E-04 Ci, and U-235 activity remaining in the neutron detectors—2.96E-05 Ci.

Of the 6.24E+02 Ci in the remainder of the EBR-II reactor building, the vast majority of this activity is from activated material stored in the storage pits within the facility and accounts for 6.236E+02 Ci (99.9963%). Other significant contributors to the inventory outside of the primary tank comprise the following: primary sodium remaining in ancillary equipment—2.08E-02 Ci and surface contamination in the shielded pentagon area—2.14E-03 Ci. For determining radiological inventories, the sodium inventories are assumed to remain in the primary and secondary reactor systems after RCRA closure is complete either as sodium bicarbonate or some other nonreactive form of sodium. The activated

material in the storage pits is stored below the 10-ft below-ground-level interval and, therefore, does not contribute to the radiological inventory in the interval that is used in the calculation of Alternative 3 risks (10 ft below to 7 ft above ground level). The radiological inventory below ground level, including the primary coolant tank and reactor vessel, was used to determine the long-term groundwater risk.

As a fast breeder reactor, large neutron flux fields were generated in the core of the reactor that had the potential to transmutate the impurities in EBR-II materials of construction into long half-lived transuranic isotopes. The transuranic activity of various components in the reactor vessel was calculated, and the inner shield wall of the reactor vessel had the highest transuranic activity of 0.77 nCi/g.

4.1.1 Inventory Above and Below Ground Level

Radiological inventories were calculated for the aboveground and belowground inventories to support the groundwater risk assessment presented in Section 6.2. Also, the ground-level to 10-ft below-ground-level inventories were calculated to support the human health and ecological risk assessments presented in Sections 6.1 and 6.3, respectively.

The only two significant contributors to the EBR-II above-ground-level inventory are (1) the controlled temperature profile condenser in the cover gas cleanup system and (2) the pentagon surfaces. Both of these contributors are above the main reactor floor (7 ft or more above ground level) and will be removed under either Alternative 3 or Alternative 4. The belowgrade source term was determined by subtracting the abovegrade source term from the total facility source term. The results of this calculation are presented in Appendix A.

4.1.2 Inventory 0 ft to 10 ft Below Ground Level

The human health and ecological risk assessments presented in Sections 6.1 and 6.3, respectively, evaluate the contamination in the interval from ground level to 10 ft below ground level. Sources of radiological activity contributing to the current (2009) inventory in the region from ground level to 10 ft below grade include the following:

- Approximately one-half of the contaminated surface area of the intermediate heat exchanger.
- Two primary pump impellers.
- Approximately one-fourth of the contaminated surface area of the shutdown cooler plugs.
- Condensed sodium vapor on the primary tank cover. Because the amount of this condensed vapor was never quantified, it is assumed to be 1% of the total residual primary sodium.
- The majority of the sodium purification and sampling systems is positioned at a depth greater than 10 ft below ground level. Small-diameter lines lead to these systems to and from the primary tank. Within the 0 to 10-ft belowgrade region, 1% of the activity associated with the ancillary equipment sodium is assumed to be contained within these lines.
- Approximately one-fourth of the contaminated NaK remaining in the shutdown cooler plugs.

The assumed final end state of the facility under Alternative 3 will include approximately 7 ft of the existing above-ground-level structures and components; therefore, the ground-level to 7 ft above-ground-level inventory was considered. This area contains mostly concrete and solid and/or sealed metal components that, through field survey and modeling, are shown not to contain significant radiological inventory. The radiological inventory in the 0 to 10-ft interval is considered bounding for any contamination that may exist in this interval; therefore, the inventory in the 0 to 10-ft below-round-level

inventory was used in the Alternative 3 human health risk assessment. The radiological inventory below ground level, including the primary coolant tank and reactor vessel, was used to determine the long-term groundwater risk. Figure 4-1 depicts a cutaway drawing of the EBR-II building and the radiological curies of each interval that was used in the risk assessments.

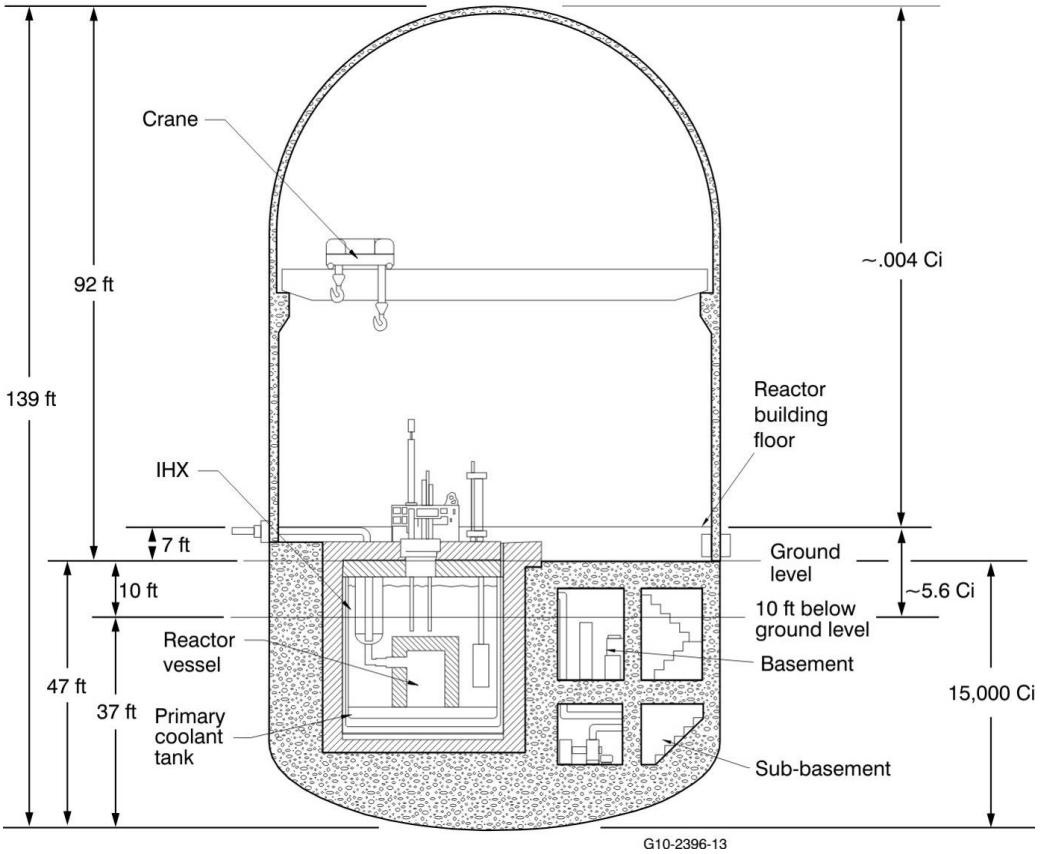

Figure 4-1. EBR-II radiological inventory intervals for risk assessments.

4.1.3 EBR-II Radiological Conditions

In this section, radiological conditions are discussed in terms of radiation fields that a worker would be exposed to if working in proximity to the inventories listed above. Radiation means ionizing radiation, or capable of producing ions, which, at EBR-II, is primarily gamma rays. Gamma ray exposure is expressed as roentgen (R) and the rate of exposure as R/hr or mR/hr (an mR is a one-thousandth of an R).

Dose rates in the primary tank down the O2 thimble (a tube that extends through the primary tank cover just outside of the neutron shield that surrounds the reactor vessel) showed measured exposure rates of approximately 14 R/hr near the core vertical centerline to as high as 233 R/hr near the open reactor top. The dose rates, measured in 1-ft intervals in the primary tank, are presented in Figure 4-2. The modeled exposure rate inside the reactor vessel above the core/blanket region is approximately 3,500 R/hr, and the modeled maximum exposure rate on the outside of the primary coolant tank is 2 R/hr. The general area dose rates in the accessible areas throughout the facility are generally low (less than 0.5 mR/hr), though higher dose rates exist in areas containing residual sodium, such as the Radioactive Sodium Chemistry Loop cubicles and the shielded pentagon area.

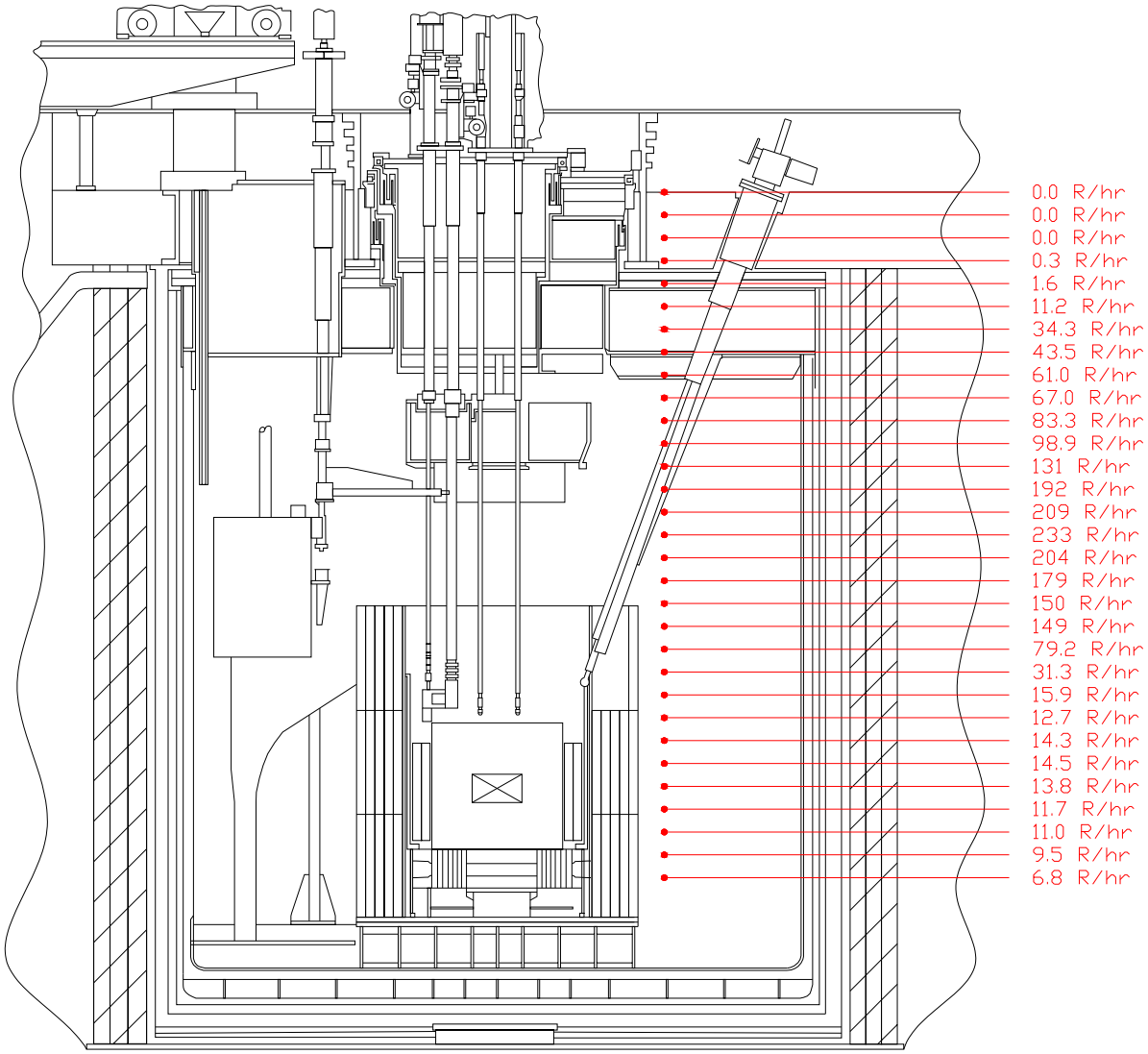

Figure 4-2. Dose rates in EBR-II primary coolant tank taken November 2009.

4.2 EBR-II Nonradiological Inventory

The EBR-II nonradiological inventory was determined by the inspection of drawings, system description documents, and measurements and order of magnitude estimates taken of the EBR-II reactor building. Building materials, such as carbon steel and stainless steel, were further broken down into their constituent chemicals, like chromium, manganese, and nickel. Constituents were evaluated in the risk assessments presented in Section 6. Appendix B lists the total nonradiological inventory for each alternative. Inventories do not include materials such as lead shielding, mercury, and asbestos. These materials are being removed to satisfy HWMA/RCRA and other applicable laws.

EBR-II also included a relatively small amount of depleted uranium that was used as shielding for fuel handling components and in the air purification system. Approximately 265 lb of this material has been included in the Appendix B, Table B-1, nonradiological source term for the facility. The radiological source term in Table A-1 also includes the curie content of the depleted uranium.

5. IDENTIFICATION OF REMOVAL ACTION OBJECTIVES

This section identifies the removal action objectives and goals (EPA 2009a) for the activities associated with this NTCRA.

5.1 Removal Action Objectives

The removal action objectives for this NTCRA are to achieve a final end state of EBR-II consistent with, or more conservative than, the remedial action objectives listed in the OU 9-04 ROD (DOE-CH 1998). The removal action objectives for this NTCRA should achieve the following:

- Inhibit direct exposure to radionuclide contaminants of concern (COCs) that would result in a total excess cancer risk greater than 1 in 10,000 (1E-04) to 1 in 1,000,000 (1E-06) for current and future workers and future residents

- Inhibit ingestion of radionuclide and nonradiological COCs by all affected exposure routes (including groundwater, soil, and homegrown produce ingestion) that would result in a total excess cancer risk greater than 1 in 10,000 (1E-04) to 1 in 1,000,000 (1E-06) or a hazard index of 1 or greater for current and future workers and future residents

- Prevent unacceptable internal exposure of biota that would result in the lack of maintenance or recovery of healthy local populations/communities of ecological receptors that are or should be present at or near the site.

The acceptable risk level at the INL Site established for CERCLA removal actions objectives is 1E-04, based on (1) conservative approaches in risk assessment that tend to overestimate the risk, (2) remote location of the INL Site, and (3) use of 1E-04 in previous risk management decisions across the INL Site.

In addition to the remediation objectives, the selected alternative should incorporate the DOE goal of reducing the “risk footprint” to as practicable extent as possible in consideration of ALARA principles governing radiological exposure to decommissioning personnel, safe engineering standards, ICDF WAC, and desired CERCLA site end states.

6. RISK ASSESSMENTS

This section describes the three risk assessments prepared for this EE/CA to evaluate the removal action alternatives. The risk assessments were based on the radiological and nonradiological inventories discussed in Section 4. Section 6.1 discusses the methodology and summarizes the results of the soil pathways risk analysis for human health. Section 6.2 describes the results of the groundwater risk analysis, and Section 6.3 provides the results of the ecological risk assessment.

Risk assessments were performed for Alternatives 1, 3, and 4. For these assessments, the end states, including the Alternative 3 and 4 end-state concrete/grout monoliths, were conservatively assumed to deteriorate to rubble over the next 85 years (to CY 2095, the date used by the ICP CERCLA program to establish remediation goals) and radiological and chemical constituents would be available for uptake by human and ecological receptors. Risk assessments were not performed for Alternative 2 because the assumption is that the EBR-II building would be maintained to prevent releases of contaminants to the environment for uptake by a human or ecological receptor.

As identified in Section 5, the acceptable risk level at the INL Site for CERCLA removal actions objectives is 1E-04 (one excess cancer risk in 10,000 people) based on (1) conservative approaches in risk assessment that tend to overestimate the risk, (2) remote location of the INL Site, and (3) use of 1E-04 in previous risk management decisions across the INL Site.

6.1 Human Health Risk Assessment

To ensure that the human health risk assessment is conservative and overestimates or bounds the risk, a very conservative scenario was assumed for contaminant uptake. The assumption used is that a family would build a house with a basement (10 ft below ground level) on the EBR-II rubble, live there for 30 years, including 6 years of childhood, and be exposed to hazardous chemicals and radiation via external exposure, soil ingestion, and inhalation of fugitive dust. Detailed descriptions of the modeling methodology, COCs screening, and results are presented in EDF-9359, "EBR-II Proposed Removal Action: Human and Ecological Risk Assessment."

6.1.1 Radiological Risk Analysis

Human health risk assessment is a two-step process. The first step is screening the soil contamination source term against soil screening levels. For a given radionuclide, a soil concentration at or below the screening level indicates that risk from that radionuclide is less than 1E-06. The original (2009 pre-decommissioning) radiological source term used for this screening is presented in Appendix A. The source term was decayed to CY 2095 (85 years), and the daughter products of the radionuclide decay were included in the screening calculations. The applicable source term for each of the alternatives evaluated is assumed to be spread throughout a volume of soil 10 ft thick and the 80-ft-diameter area of the EBR-II reactor building. A soil density of 93.5 lb/ft^3 was used to yield a contaminated soil mass of approximately 4.7E+05 lb. Calculated soil concentrations are quite low for many radionuclides. Previous experience has shown that minimum soil screening levels for the soil ingestion, inhalation, and external exposure pathways are no less than 1E-01 pCi/g. To be conservative, soil concentrations were screened so that all radionuclides with soil concentrations greater than 1E-03 pCi/g were retained for the risk assessment.

The second step of the risk assessment is to perform risk calculations on those radionuclides not screened out in the first step. The EPA's Risk Calculator on the Risk Assessment Information System web site (http://rais.ornl.gov/cgi-bin/prg/RISK_search?select=rad) (EPA 2009b) was used to calculate risk. The calculator uses equations from the *Risk Assessment Guidance for Superfund: Volume I, Human Health Evaluation Manual (Part A, B, C, D, E & F)* (EPA 1989) and *Guidance for Conducting Remedial*

Investigations and Feasibility Studies under CERCLA, Interim Final (EPA 1988). The calculated risk represents the incremental probability of an individual developing cancer over a lifetime. For calculations of exposure to soil, the Boise climatic zone and a site area of 0.5 acres were selected. For all other parameters, default values were used.

The results of the soil pathways radiological risk analysis are shown in Tables 6-1 through 6-3 for Alternatives 1, 3, and 4, respectively. For Alternative 1, the acceptable carcinogenic risk of 1E-04 is exceeded (1.9E-04) for the sum of all exposure pathways, with Sr-90 representing the largest contribution to the risk (1.69E-04).

The radiological risk from Alternative 3 (see Table 6-2) meets the acceptable risk level of 1E-04 at 9.1E-05.

The radiological risk from Alternative 4 at 6.7E-06 also would meet the acceptable risk level of 1E-04.

Table 6-1. Risk in CY 2095 by radionuclide and pathway for EBR-II Alternative 1, No Action.

Radionuclide	Concentration in Soil (pCi/g)	Ingestion of Soil Risk	Inhalation of Soil Risk	External Exposure to Soil Risk	Total Soil Risk
Cs-137+D	4.76E-02	1.87E-09	1.97E-14	7.85E-07	7.87E-07
H-3	2.59E-02	3.48E-12	1.20E-16	0	3.48E-12
Ni-59	4.02E-03	3.71E-12	9.02E-17	0.00E+00	3.71E-12
Ni-63	2.53E-01	5.13E-10	1.80E-14	0.00E+00	5.13E-10
Pa-234	2.62E-02	8.54E-15	6.77E-20	7.52E-11	7.53E-11
Pa-234m	1.64E+01	0	0	1.08E-12	1.08E-12
Sr-90+D	6.62E+02	8.58E-05	2.57E-09	8.31E-05	1.69E-04
Th-234	1.64E+01	4.40E-09	7.70E-14	7.61E-09	1.20E-08
U-234	3.94E-03	7.84E-10	2.16E-12	8.90E-12	7.95E-10
U-238+D	1.64E+01	4.34E-06	7.38E-09	1.68E-05	2.11E-05
Total risk		9.02E-05	9.96E-09	1.01E-04	1.9E-04

Table 6-2. Risk in CY 2095 by radionuclide and pathway for EBR-II Alternative 3, Grout Reactor Vessel in Place.

Radionuclide	Concentration in Soil (pCi/g)	Ingestion of Soil Risk	Inhalation of Soil Risk	External Exposure to Soil Risk	Total Soil Risk
Cs-137+D	1.13E-02	4.45E-10	4.67E-15	1.86E-07	1.87E-07
H-3	1.01E-02	1.36E-12	4.69E-17	0	1.36E-12
Pa-234	8.23E-03	2.68E-15	2.13E-20	2.36E-11	2.36E-11
Pa-234m	5.20E+00	0	0	3.43E-13	3.43E-13
Sr-90+D	3.31E+02	4.29E-05	1.29E-09	4.16E-05	8.45E-05
Th-234	5.20E+00	1.39E-09	2.44E-14	2.41E-09	3.81E-09
U-234	1.25E-03	2.49E-10	6.86E-13	2.82E-12	2.52E-10
U-238+D	5.20E+00	1.38E-06	2.34E-09	5.31E-06	6.69E-06
Total risk		4.43E-05	3.63E-09	4.71E-05	9.1E-05

Table 6-3. Risk in CY 2095 by radionuclide and pathway for EBR-II Alternative 4, Removal of the EBR-II Reactor Vessel.

Radionuclide	Concentration in Soil (pCi/g)	Ingestion of Soil Risk	Inhalation of Soil Risk	External Exposure to Soil Risk	Total Soil Risk
Cs-137+D	2.83E-03	1.11E-10	1.17E-15	4.67E-08	4.68E-08
Pa-234	8.23E-03	2.68E-15	2.13E-20	2.36E-11	2.36E-11
Pa-234m	5.20E+00	0	0	3.43E-13	3.43E-13
Th-234	5.20E+00	1.39E-09	2.44E-14	2.41E-09	3.81E-09
U-234	1.25E-03	2.49E-10	6.86E-13	2.82E-12	2.52E-10
U-238+D	5.20E+00	1.38E-06	2.34E-09	5.31E-06	6.69E-06
Total risk		1.38E-06	2.34E-09	5.36E-06	6.7E-06

6.1.2 Nonradiological Risk

Nonradiological risk is presented for only those metals with soil concentrations exceeding EPA regional screening levels.

Nonradiological risk from Alternative 1, No Action, is summarized in Table 6-4. The hazard index of 2 exceeds the target index of 1, with most of the risk resulting from nickel. Smaller contributions to the risk come from copper and manganese. The soil ingestion exposure route dominates the risk for both adult and child future residents.

Nonradiological risk from Alternative 3 is summarized in Table 6-5. The total hazard index is less than 1 (and is from nickel alone), indicating it is unlikely for even sensitive subpopulations to experience adverse health effects.

All metal soil concentrations for Alternative 4 were below EPA regional screening levels; therefore, no risk numbers were calculated.

Table 6-4. Risk by metal and pathway for EBR-II Alternative 1, No Action.

	Soil Concentration (mg/kg)	Adult Ingestion of Soil Hazard Index	Child Ingestion of Soil Hazard Index	Inhalation of Soil Particulates and Volatiles Hazard Index	Dermal Soil Hazard Index	Total Soil Hazard Quotient
Copper	1.15E+04	3.94E-01	3.68E+00	0	5.24E-03	0.399
Manganese	1.07E+04	3.19E-01	2.97E+00	5.23E-02	3.18E-02	0.403
Nickel	1.78E+04	1.22E+00	1.14E+01	0	1.80E-02	1.24
Hazard index		1.93E+00	1.80E+01	5.23E-02	5.50E-02	2.0

Table 6-5. Risk by metal and pathway for EBR-II Alternative 3, Grouting Reactor Vessel in Place.

	Soil Concentration (mg/kg)	Adult Ingestion of Soil Hazard Index	Child Ingestion of Soil Hazard Index	Inhalation of Soil Particulates and Volatiles Hazard Index	Dermal Soil Hazard Index	Total Soil Hazard Quotient
Nickel	4.13E+03	2.83E-01	2.64E+00	0	4.18E-03	0.287
Hazard index		2.83E-01	2.64E+00	0.00E+00	4.18E-03	0.3

6.2 Groundwater Risk Assessment

This section describes the results of the groundwater pathway risk assessment for the estimated radionuclide and nonradiological inventory to be left in place after the D&D of the EBR-II building. The risk assessment evaluates potential adverse health effects to human receptors via the groundwater pathway. Detailed descriptions of the modeling methodology, COC screening, and results are presented in EDF-9374, "Groundwater Pathway Risk Assessment for EBR-II Closure."

The primary conservative assumption is that all metals have already corroded and the contaminants in the metals are available for transport. In reality, the vast majority of the inventory is contained in stainless steel and carbon steel. The steel will eventually corrode and the contaminants will be slowly released within the EBR-II building and be made available for transport through the subsurface to the aquifer.

Groundwater pathway risk results are shown in Table 6-6 for the radionuclide COCs from the EBR-II building for Alternative 1, No Action. The first three columns of the table list the EBR-II inventory decayed to CY 2095 and the maximum containment levels (MCLs) for the radionuclides. The next four columns show the groundwater model simulation results. The results include the maximum predicted groundwater concentrations, the risk that would result from the predicted maximum concentration, and the predicted maximum concentration as a percent of the MCL. In addition, the time of the predicted maximum concentration is given as a calendar year. As previously discussed, it is assumed that the contaminants will enter the subsurface and begin transport to the aquifer starting in CY 2095. For I-129 (for example), the conservative transport model estimates that the maximum aquifer concentration will occur 90 years after transport through the subsurface begins in CY 2095. Therefore, the predicted maximum aquifer concentration will occur in CY 2185.

As shown in Table 6-6, the risk for each radionuclide is below or within the EPA acceptable target risk range of 1E-06 to 1E-04. The radionuclides predicted to have the largest risks are C-14 (6E-05), Cl-36 (4E-06), Tc-99 (1E-06), and U-238 (1E-06). For all other radionuclides, the predicted risk is below 1E-06. The predicted maximum concentration of C-14 is approximately equal to the C-14 MCL of 2,000 pCi/L. All other radionuclide predicted maximum groundwater concentrations are 10% or less than MCLs (EPA 2009b). Radionuclides with a 0.00E+00 value indicate that the radionuclide has decayed away before it reaches the aquifer.

Groundwater pathway concentration and hazard quotient results are shown in Table 6-7, for the chemical COCs from the EBR-II building, for Alternative 1 (No Action). The first three columns of the table list the EBR-II inventory and the MCLs for the chemicals. The next four columns show the groundwater model simulation results. The results include the maximum predicted groundwater concentrations, the chemicals' hazard quotient that would result from the predicted maximum concentration, and the predicted maximum concentration as a percent of the MCL. In addition, the time of the predicted maximum concentration is given as a calendar year as described above for radionuclides.

As shown in Table 6-7, all predicted maximum groundwater concentrations are less than the chemical's respective MCL. The chemicals with predicted maximum aquifer concentrations closest to the MCLs are copper and uranium. The predicted maximum copper concentration is 7.4% of the MCL and is predicted to occur after CY 10,000. The predicted maximum uranium concentration is 5.1% of the MCL and is predicted to occur after CY 4500. Each of the other chemical COCs has predicted maximum concentrations that are less than the MCL by more than two orders of magnitude. The hazard quotients are also at least two orders of magnitude below the target index of 1.0."

Table 6-6. Groundwater pathway risk assessment results for Alternative 1 radionuclide contaminants of concern.

Radionuclide	Inventory (2095) (Ci)	MCL [a,b] (pCi/L)	Model-Predicted Maximum Contaminant Concentrations and the Time of the Predicted Maximums			
			Time (calendar year)	Groundwater Concentration (pCi/L)	Risk	% MCL
Ag-108m	2.83E-02	NA	7,468	3.58E-09	6E-16	NA
Am-241[c]	2.17E-07	Total alpha < 15	5,390	2.10E-06	3E-12	0.00001%
C-14	8.29E+00	2,000	2,224	2.00E+03	6E-05	100%
Cl-36	1.72E-01	700	2,185	6.10E+01	4E-06	8.7%
Co-60	9.10E-02	100	19,838	0.00E+00	0E+00	0%
Cs-137	8.66E-03	200	863,605	0.00E+00	0E+00	0%
Eu-152	5.30E-02	200	587,615	0.00E+00	0E+00	0%
H-3	1.38E-01	20,000	2,137	1.88E+00	4E-09	0.01%
Hf-178m	1.53E-02	NA	777,695	0.00E+00	0E+00	NA
Ho-166m	3.45E-02	NA	17,257	2.14E-09	4E-16	NA
I-129	1.97E-08	1	2,185	6.99E-06	2E-11	0.0007%
Mo-93	1.32E-01	NA	5,485	5.41E-01	4E-08	NA
Nb-94	1.33E-01	NA	31,226	3.33E-02	5E-09	NA
Ni-59	5.37E+01	300	38,220	3.02E+01	2E-07	10%
Ni-63	3.32E+03	50	176,725	0.00E+00	0E+00	0%
Np-237	1.44E-08	Total alpha < 15	5,390	1.39E-07	2E-13	0.000001%
Pu-238 [d]	4.94E-08	Chemical MCL	4,584	6.28E-07	9E-13	NA
Pu-239	1.22E-02	Total alpha < 15	13,373	9.12E-03	3E-08	0.06%
Pu-240	2.72E-04	Total alpha < 15	10,446	9.73E-05	3E-10	0.0006%
Pu-241 [c]	1.01E-07	Total alpha < 15	5,390	9.78E-07	1E-12	0.00001%
Sr-90	1.41E+00	8	23,568	0.00E+00	0E+00	0%
Tc-99	1.33E-01	900	2,265	2.50E+01	1E-06	2.8%
U-233	4.19E-04	Chemical MCL	4,581	5.31E-03	9E-09	NA
U-234	2.20E-07	Chemical MCL	4,584	2.80E-06	4E-12	NA
U-235	2.96E-05	Chemical MCL	4,589	3.79E-04	6E-10	NA
U-238	4.04E-02	Chemical MCL	4,589	5.17E-01	1E-06	NA

a. The MCL for the uranium is covered as a chemical concentration rather than an activity concentration.

b. MCL values are found in http://www.epa.gov/safewater/radionuclides/pdfs/guide_radionuclides_table-betaphotonemitters.pdf (EPA 2009c) and http://www.epa.gov/safewater/contaminants/index.html#7 (EPA 2009d).

c. Am-241 and Pu-241 are relatively short-lived radionuclides and relatively immobile. Risk in the groundwater from disposal of Am-241 and Pu-241 will result from the transport of their primary progeny, Np-237. Therefore, Am-241 and Pu-241 are assumed to decay to Np-237 in the simulations and transport of Np-237 is predicted. Details are provided in EDF-9374.

d. Pu-238 is a relatively short-lived radionuclide and relatively immobile. Risk in the groundwater from disposal of Pu-248 will result from the transport its primary progeny, U-234. Therefore, Pu-248 is assumed to decay to U-234 in the simulations and transport of U-234 is predicted. Details are provided in EDF-9374.

MCL maximum containment level
NA not applicable

Table 6-7. Groundwater pathway results for the Alternative 1 nonradiological chemicals of concern.

Chemical	Inventory (kg)	MCL or TT (mg/L)	Model-Predicted Maximum Contaminant Concentrations and the Time of the Predicted Maximums			
			Calendar Time (year)	Groundwater Concentration (pCi/L)	Hazard Quotient	% MCL
Aluminum	6.85E+03	No MCL	102,375	2.18E-03	6.0E-05	NA
Antimony	1.00E+00	6.0E-03	22,222	1.59E-06	1.1E-04	0.03%
Boron	4.08E+02	No MCL	4,188	6.23E-03	8.5E-04	NA
Chromium	5.47E+04	1.0E-01	a	1.61E-04	2.9E-06	0.16%
Copper (and compounds) [b]	2.45E+04	1.3E+00	10,200	9.65E-02	6.6E-02	7.4%
Lead [b]	1.40E+01	1.5E-02	42,260	1.11E-05	NA	0.07%
Manganese (and compounds)	2.28E+04	No MCL	22,222	3.62E-02	4.1E-02	NA
Nickel	3.78E+04	No MCL	42,260	3.01E-02	4.1E-02	NA
Uranium	1.20E+02	3.0E-02	4,589	1.54E-03	1.4E-02	5.1%
Zinc	1.25E+03	No MCL	8,597	6.15E-03	5.6E-04	NA

a. The chromium concentration reaches its peak in the aquifer within about 300 years and then stays at that concentration for many thousands of years until all of the chromium has been transported to the aquifer. Therefore, no time to peak is provided.

b. Lead and copper are regulated by a Treatment Technique that requires systems to control the corrosiveness of their water. If more than 10% of tap water samples exceed the action level, water systems must take additional steps.

MCL maximum contaminant level
NA not applicable
TT Treatment Technique – A required process intended to reduce the level of a contaminant in drinking water

For Alternative 3, the EBR-II reactor would be grouted in place and, for Alternative 4, the reactor would be removed and then the building would be grouted. The risk assessment analysis differences between the evaluation of Alternative 1 and Alternatives 3 and 4 are in the inventories and the assumptions of contaminant mobility in the building after closure. Contaminant mobility is different for Alternative 1 and Alternatives 3 and 4 because in Alternative 1, the contaminants in the EBR-II building are assumed to be in soils and in Alternatives 3 and 4 the contaminants in the EBR-II building are assumed to be in concrete (a grouted environment). The mobility differences in the source term (within the EBR-II building at its final end state) are simulated in the model by applying K_d (distribution coefficient) values for a soil environment in Alternative 1 and K_d values for a grouted environment for Alternatives 3 and 4.

A grouted closure (such as proposed for Alternatives 3 and 4) will tend to decrease contaminant mobility (increase the K_d), slowing the contaminants' release to the subsurface and decreasing the predicted risk in the aquifer. Therefore, in general, the contaminant risk for Alternatives 3 and 4 will be smaller than the risks for Alternative 1 (EDF-9374). However, there are two exceptions to this rule—strontium from the radionuclide COCs and chromium from the chemical COCs:

- Strontium in a grouted environment is more mobile than in the natural INL Site environment. For the Alternative 1 Sr-90 simulations, a K_d of 12 mL/g was used and the resulting predicted concentration was zero. However, for a more mobile form of strontium, the predicted aquifer concentration will not necessarily be zero. The mobility of strontium in grout has been evaluated for a number of projects often related to the ROD for High-Level Waste Environmental Impact Statement (DOE 2005). The strontium K_d value in a grout environment is estimated at 1 mL/g.

Therefore, Sr-90 should be evaluated for Alternatives 3 and 4 using this smaller K_d value in the source area. The decreased K_d for Sr-90 in grout will result in a much higher flux of Sr-90 from the EBR-II building source area than under Alternative 1.

- Chromium in a grouted environment is more mobile than in the natural INL Site environment. As the pH increases, the predominant valence state of chromium changes from trivalent (relatively immobile) to hexavalent (relatively mobile) chromium. Therefore, in the grouted source area (high pH), the solubility limit for chromium (hexavalent) would be relatively high and the K_d value would be relatively low. However, as the chromium leaves the vicinity of the EBR-II building and the grout environment, the predominant valence state will change to the trivalent valence and to a low solubility and relatively high K_d value. For Alternatives 3 and 4, chromium within the EBR-II building (source term) could be simulated with a zero K_d value. However, since the chromium will change from hexavalent to trivalent during transport through the vadose zone to the aquifer, the chromium solubility limit of 5.2E-02 mg/L will still dominate the deep vadose zone transport concentrations. In other words, increased solubility and mobility in the source (EBR-II building) will not impact the predicted aquifer concentrations of chromium. There is no need to simulate chromium for Alternatives 3 and 4 since it has been simulated for Alternative 1, and Alternative 3 and 4 simulations will not change the predicted aquifer concentrations.

Alternatives 3 and 4 predicted that aquifer concentrations are generally less than the Alternative 1 concentrations, estimated risks, and hazard quotients. To illustrate the impact of a grouted environment on subsurface transport and decreased inventories for Alternatives 3 and 4, some key COCs were chosen to evaluate for Alternatives 3 and 4 and compared with Alternative 1 results. The contaminants chosen for the comparison are the primary risk drivers or the contaminants with predicted maximum aquifer concentrations closest to the MCL (C-14, copper, and uranium). In addition, because Alternative 1 is not conservative for Sr-90, it was evaluated as well. For all other COCs, the predicted maximum risk, hazard quotients, and concentrations for Alternatives 3 and 4 will be significantly less than the Alternative 1 results, which are less than the groundwater performance criteria. Therefore, no analysis is necessary.

Table 6-8 shows the comparison of the C-14, Sr-90, copper, and uranium (nonradiological chemical) predicted maximum concentrations, risk (for C-14 and Sr-90), and percentage of MCL (for copper and uranium). The table shows that inventories and the K_d values in the source change with the alternative. All other parameters are unchanged. For copper, the K_d in a grouted source term is unknown. Therefore, the source term K_d values are assumed to be the same for Alternatives 1, 3, and 4.

The impact of reduced inventories and the change of environment from soil to grout were evaluated and are shown in Table 6-8. With the exception of Sr-90, the predicted maximum concentrations, risks, and hazard quotients for Alternatives 3 and 4 are significantly smaller than the Alternative 1 results. The Sr-90 results increase for Alternative 3 and 4 relative to Alternative 1, but the results are still very small (essentially zero).

This conservative risk assessment demonstrates that, for Alternative 1, the inventory to be left in place after the D&D of EBR-II building will meet the CERCLA groundwater performance criteria of contaminant concentrations in the Snake River Plain Aquifer of less than or equal to a cumulative carcinogenic risk of 1E-04. For Alternative 1, the predicted maximum concentration of C-14 is approximately equal to the C-14 MCL of 2,000 pCi/L, and all other predicted maximum groundwater concentrations are 10% or less than MCLs. Based on this conservative risk assessment, for Alternatives 3 and 4, the contamination at the EBR-II building is not expected to result in groundwater concentrations that exceed the performance criteria for any COCs.

Table 6-8. Groundwater pathway risk assessment Alternative 1, 3, and 4 results.

Contaminant of Interest	MCL		Inventory		K_d (mL/g)		Model-Predicted Maximum Contaminant Concentrations and the Time of the Predicted Maximums				
					Source[a]	Vadose Zone	Time (calendar year)	Maximum Contaminant Concentration[b]		Risk	%MCL
Alternative 1 (No Action)											
C-14	2,000	pCi/L	8.29	Ci	0.1	0.1	2,224	2,004	pCi/L	6.48E-05	100%
Sr-90	8	pCi/L	1.41	Ci	12	12	23,568	0	pCi/L	0.00E+00	0.00%
Copper	1.3	mg/L	24,470	kg	20	20	10,200	9.65E-02	mg/L	NA	7.4%
Uranium	0.03	mg/L	120	kg	6	6	4,589	1.54E-03	mg/L	NA	5.1%
Alternative 3 (smaller inventory and/or grouted source term)											
C-14	2,000	pCi/L	8.29	Ci	1,000	0.1	2,433	1.58	pCi/L	5.14E-08	0.08%
Sr-90	8	pCi/L	1.41	Ci	1	12	2,610	9.18E-09	pCi/L	1.42E-14	0.00%
Copper	1.3	mg/L	15,650	kg	20	20	10,200	6.17E-02	mg/L	NA	4.7%
Uranium	0.03	mg/L	83	kg	5,000	6	10,852	3.30E-06	mg/L	NA	0.01%
Alternative 4 (smaller inventory and/or grouted source term)											
C-14	2,000	pCi/L	0.075 [c]	Ci	1,000	0.1	2,433	0.014	pCi/L	4.63E-10	0.00%
Sr-90	8	pCi/L	6.5E-10 [c]	Ci	1	12	2,610	4.23E-18	pCi/L	6.55E-24	0.00%
Copper	1.3	mg/L	5,693	kg	20	20	10,200	2.24E-02	mg/L	NA	1.7%
Uranium	0.03	mg/L	83	kg	5,000	6	10,852	3.30E-06	mg/L	NA	0.01%

a. No reference is available for the copper K_d in a grout environment. Therefore, the copper has been simulated as if the source term is soil for all three scenarios.

b. The risk for radionuclides is calculated based on the peak 30-year average concentration. This table only has the peak concentrations shown. As can be seen in this table, the peak 30-year average concentrations are slightly smaller than the peak concentrations.

c. The Alternative 4 inventory for C-14 and Sr-90 is the Alternative 4 inventory in Table A-1 decayed until the year 2095.

MCL maximum containment level
NA not applicable

6.3 Ecological Risk Assessment

The screening ecological risk assessment followed the approach presented in the *Guidance Manual for Conducting Screening Level Ecological Risk Assessments at the INEL* (VanHorn, Hampton, and Morris 1995) and documented in the *Comprehensive Remedial Investigation/Feasibility Study for Waste Area Groups 6 and 10 Operable Unit 10-04* (DOE-ID 2001) and the *Risk-Based Screening and Assessment Report for Waste Area Group 1 Soils* (Van Horn and Stacey 2004). Contaminants of potential concern that exceeded screening were further evaluated.

The initial screening process includes three major steps:

1. Step 1 of the ecological screening process is to distinguish potential contamination associated with the site from naturally occurring background conditions. The comparison is primarily conducted using the composite background values from the *Background Dose Equivalent Rates and Surficial Soil Metal and Radionuclide Concentrations for the Idaho National Engineering Laboratory* (Rood, Harris, and White 1996) or from other sources, as identified.

2. Step 2 of the ecological screening process is an essential nutrient analysis. Site chemicals that are considered essential nutrients are not evaluated further unless the concentration is much greater (10 times) than the background value. The six metals routinely eliminated by this screening step are aluminum, calcium, iron, magnesium, potassium, and sodium (Cirone 1991).

3. For the remaining chemicals, Step 3 of the ecological chemical screening process is to compare potential contaminants associated with the site with ecologically based screening levels (EBSLs) or EPA ecological soil screening levels, as noted. If the maximum concentration for a given chemical is greater than or equal to the most conservative EBSL or other screening level, then the chemical is retained for further evaluation. The screening levels used for the screening are consistent with the INL-wide screening levels. Details for INL-specific EBSL development and values are documented in Appendix D2 of the *Work Plan for Waste Area Groups 6 and 10 Operable Unit 10-04 Comprehensive Remedial Investigation/Feasibility Study* (DOE-ID 1999). New ecological soil screening levels provided in the new or updated chemical-specific documents from EPA (EPA 2008) were included, as available.

The results of the screening (Appendix C) indicated that no radionuclide concentrations exceed EBSLs for Alternatives 1, 3, and 4; however, some of the metal concentrations for all three alternatives exceed the EBSLs. Further evaluation of the seven metals that did not screen for Alternative 1 indicates that boron, chromium, copper, manganese, and nickel concentrations have hazard quotients exceeding 10. Of the five metals that did not screen and required further assessment for Alternative 3, chromium, manganese, and nickel concentrations had hazard quotients greater than 10. None of the concentrations of metals that required further assessment (copper, nickel, and uranium) for Alternative 4 resulted in hazard quotients greater than 10.

It is important to remember how the metal inventories associated with EBR-II were developed. They indicate that belowground structures and systems (including piping, wiring, stainless steel, and utility systems) were included in the source-term development. As discussed, the inclusion of these items in the overall soil concentration is a highly conservative assumption. In the environment at this site, items such as wiring, piping, and large stainless steel equipment pieces will not degrade to a bioavailable form uniformly throughout this soil, as was modeled. Therefore, any concentration of concern would be highly localized in or near the 80-ft diameter of the EBR-II footprint and it is unlikely that it would pose a risk to ecological receptors at a population level. Therefore, the metals also are eliminated as a concern. Given the approach to concentrations in the soil, the risk values calculated are most valuable for comparison

purposes. It is apparent from the evaluation of the alternatives that Alternative 4 would leave a smaller quantity of metals in the environment. However, the exposure will not be as great as modeled and care should be taken to ensure that factors involving worker risk in removing these structures are considered.

7. ALTERNATIVE ANALYSIS

This section evaluates the final end-state alternatives in this EE/CA with respect to three criteria: (1) effectiveness, (2) implementability, and (3) cost, in accordance with the *Guidance on Conducting Non-Time Critical Removal Actions Under CERCLA* (EPA 1993).

Effectiveness includes two subcriteria: protectiveness and the ability to meet removal action objectives. Protectiveness was evaluated based on (1) protectiveness of the alternative for public health and the community, (2) protectiveness of workers during implementation, (3) protectiveness of the environment, and (4) compliance with ARARs and other requirements.

Implementability is evaluated based on technical feasibility; availability of equipment, personnel, services, and disposal facilities; and administrative feasibility. Costs are estimated, including D&D costs and surveillance and maintenance costs.

Alternative 2, No Action with Continued Surveillance and Maintenance, is protective of human health and the environment because the facility is maintained so that it does not degrade to the point where contaminants become available to human or ecological receptors. Alternative 2 is only an interim measure that delays a needed future action; therefore, the alternative is only evaluated for the cost of surveillance and maintenance.

7.1 Effectiveness of the Alternatives

The two subcriteria for evaluating effectiveness are protectiveness and the ability to meet the ARARs.

7.1.1 Protectiveness

Protectiveness is the primary objective of a removal action and is a threshold criterion that must be met to recommend an alternative. The sections below address protectiveness for a future resident (public health), the onsite worker, and the environment.

7.1.1.1 ***Protectiveness: Pubic Health.*** As discussed in the human health risk assessment (Section 6.1), the total risk from Alternative 1, No Action, would exceed acceptable levels and does not meet the threshold criteria of protecting human health. Both Alternative 3 and Alternative 4 are protective of human health because they both have a lower excess cancer risk than the threshold of 1E-04. Alternative 4 removes the reactor vessel and, therefore, leaves less contamination at the EBR-II site and is more protective for a hypothetical future resident that may occupy the EBR-II site than Alternative 3.

Compliance with the ARARs also ensures that the public is protected from releases of contaminants during D&D activities (e.g., exposure from airborne radionuclides to a hypothetical receptor at the INL Site boundary must be <10 mR/yr [40 CFR 61.92] and emissions of nonradiological contaminants are less than regulatory limits [IDAPA 58.01.01.161]).

7.1.1.2 ***Protectiveness: Worker Risk.*** Specific hazards associated with implementation of Alternative 3 or Alternative 4 would be identified and mitigated using an integrated safety management process that has been shown to significantly minimize worker potential for injury. Both administrative and engineering controls will be used to protect the workers. Administrative controls include barriers and signage to prohibit nonessential personnel from hazardous work areas. Accountability of employees and close supervision of employees by competent job supervisors/foremen that match employee's abilities with the tasks to be completed are some of the administrative controls that help workers to work safely.

Engineering controls for heavy lifts will include hoisting and rigging designed to lift loads safely with significant safety margins designed into lifting lugs, slings, cranes, and jacking equipment that bear the load. In addition, a trained and experienced health and safety staff will independently monitor work activities and function as an integral part of the work planning process to ensure controls are implemented in procedures that workers are required to follow.

Under Alternative 3, industrial and radiological hazards to D&D workers are relatively routine in that they have been performed safely many times during D&D of similar buildings as part of the ICP at the INL Site. However, Alternative 4 involves many more hazards of greater magnitude and consequence than Alternative 3. Actions to address one hazard often increase a separate hazard. The radiological dose risks can be lessened by shielding with grout, but the weight of the primary coolant tank will increase, thereby increasing the industrial hazards to the workers. Alternatively, cutting contaminated systems to isolate the primary coolant tank for easier removal increases exposure to radioactive dose and radiological uptake potential. Also, working in personnel protective equipment, such as respirators and coveralls, increases the potential for accidents due to limiting the workers' ability to see and move about freely. Section 7.2.1 below describes technical challenges to removing the reactor vessel. These technical challenges provide opportunities to expose workers to hazards that are not well known or understood and the opportunity for equipment failures that put workers at increased risk of injury or worse. Alternative 4 is much more hazardous for workers from an industrial safety standpoint than Alternative 3.

The "Occupational Radiation Protection" regulation, 10 CFR 835, requires ICP to develop and implement plans and measures to maintain occupational radiation exposures at ALARA levels (10 CFR 835.101(c) and 10 CFR 835.1001). As applied to occupational radiation exposure, the ICP ALARA process does not require that exposures to radiological hazards be minimized without further consideration but that such exposures be optimized by taking into account (1) benefits arising out of the activity, (2) detriments arising from the resultant radiation exposures, and (3) controls to be implemented. The primary methods used to maintain exposures at ALARA levels are administrative controls (e.g., radiation work permits, personnel dose tracking, and access controls) and engineering controls (e.g., shielding such as grout, containment devices, and filtered ventilation systems), which are used (as appropriate) to control individual exposures to radiation.

Implementation of Alternative 4 requires workers to perform many more hazardous activities and has a greater potential for exposure to radioactive doses and uptake of radiological contamination. The Alternative 3 approach of grouting the reactor vessel in place is less complex than removing the reactor vessel under Alternative 4 and has less opportunity for workers to be exposed to the damaging effects of radiation, uptake of radiological contamination, or industrial hazards.

7.1.1.3 ***Protectiveness: Environmental Risk.*** The screening-level ecological risk evaluation, discussed in Section 6.3 indicated that soil concentrations for radionuclides of all three alternatives evaluated did not exceed the ecological-based criteria. Some metals exceeded screening levels; however, because of the conservative nature of calculating the soil concentrations and the very small area involved; therefore, ecological receptors at a population level would not be at risk. Alternative 1 is less protective of the environment than Alternatives 3 and 4 because it leaves the entire EBR-II reactor building and all of the associated contamination that could become available for uptake by an ecological receptor. Therefore, both Alternative 3 and Alternative 4 are more protective of the environment than Alternative 1, with Alternative 4 having lower concentrations of metals in the soils. Hence, Alternative 4 is more protective than Alternative 3 for ecological receptors.

7.1.1.4 ***Protectiveness: Ability to Achieve Removal Action Objectives.*** Alternative 1 does not meet the removal action objectives because it exceeds the acceptable risk threshold of 1E-04. Alternatives 3 and 4 both achieve the remedial action objectives. Alternative 4 reduces the "risk footprint" more than Alternative 3, but Alternative 3 is much more consistent with the goal of keeping workers safe from industrial hazards and keeping radiological exposure to D&D workers to ALARA levels.

7.1.1.5 ***Ability to Comply with ARARs.*** Both Alternatives 3 and 4 can be conducted to be compliant with the ARARs discussed in Section 8.1 (and provided in Table 8-1).

7.2 Implementability of the Alternatives

Implementability is evaluated based on technical and administrative feasibility and availability of equipment, personnel, services, and disposal facilities.

7.2.1 Technical and Administrative Feasibility

Alternative 3 is technically feasible. The methods for performing this activity can be planned and engineered using existing available knowledge and procedures that have been performed at the INL Site or elsewhere. The primary coolant tank with the EBR-II reactor vessel can be easily grouted in place, and the above-ground-level 7-ft portion of the reactor building containing the primary coolant tank cover, support structure, and small and large rotating plugs can be effectively incorporated into a concrete cover.

Two major obstacles provide ample technical challenges for the removal of the EBR-II reactor vessel under Alternative 4. First, the construction of the primary coolant tank and internal components does not readily allow removal of the reactor vessel. Second, the high general-area dose rates, currently estimated to be as high as 233R/hr inside the primary coolant tank, prevent necessary access to allow removal of the reactor vessel.

Two methods were evaluated for the removal of the EBR-II reactor vessel. The first method was to remove the reactor vessel from the primary coolant tank in which it was installed. To appreciate the technical challenges of removing the reactor vessel, it is important to understand how EBR-II was constructed.

The primary coolant double-wall tank was fabricated inside the reactor building and welded to the tank cover. The primary coolant tank, with the tank cover and support structure, was lowered into place empty (see Figures 2-7 and 2-8), before the internal components of the tank were installed. Once the tank was in place, workers descended on ladders into the tank to construct the internal components, including assembling the reactor vessel structure. The reactor vessel support beams, leveling plate, coolant plenums, and grid plate were bolted and/or welded to the bottom of the primary coolant tank. The reactor vessel was then installed along with the coolant lines. The sub-components were bolted and welded together to form the finished reactor. The reactor vessel was plumbed, leveled, shimmed, bolted, and then welded to the reactor leveling plate and convection plate (Figure 7-1), which is welded onto the bottom of the primary tank. The reactor internals, including the thermal baffles and two concentric rings of inner neutron shield cans were added, bolted, and welded in place. Five additional rows of the stainless steel cans were fitted around the reactor vessel, and a stainless steel mesh enclosing the neutron shield was welded into place. This was all accomplished by having workers in the primary coolant tank assembling these components.

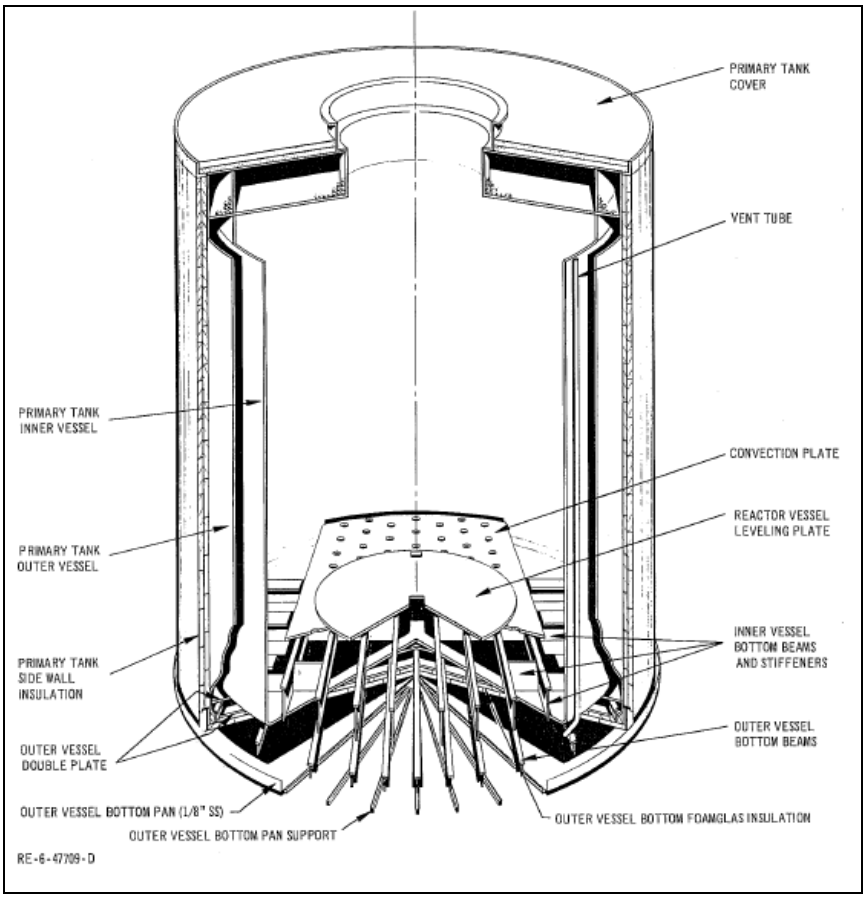

Figure 7-1. EBR-II primary coolant tank reactor vessel support components.

The reactor vessel was not constructed to be removed from the primary coolant tank. The challenge here is that no direct method exists for removing the bolts and welds that fasten the reactor vessel to the primary coolant tank because of the confined areas in the primary coolant tank and the excessive radiological dose from the activated metals associated with the reactor vessel and neutron shield. The radiological environment in the primary coolant tank is far too hazardous to have workers working inside the primary coolant tank. Accessing the bolts and welds that fasten the reactor vessel from the top with long-reach tools would require removing many of the primary coolant tank components along with the primary coolant tank cover that then would allow radiation from the interior of the tank to stream upward and also would allow contamination present in the primary coolant tank to potentially be released. Use of remote technologies, such as robotics to access the bolts and welds securing the reactor vessel, in such confined quarters is not a proven technology in the high-radiation fields present in the primary coolant tank and would likely fail.

Since the reactor vessel would be extremely difficult and hazardous to remove from the primary coolant tank, an approach that leaves the reactor vessel in place in the tank and removes the entire primary coolant tank is possible. Because the reactor vessel inside the tank has some very high radiological conditions, the outside of the primary coolant tank is affected. However, by adding a light-weight grout just above the halfway point (i.e., 16 ft) inside the primary coolant tank, the radiological doses on the outside of the tank can be reduced from approximately 2 R/hr. to near background levels. This would allow workers to work in the close proximity to the primary coolant tank as it is lifted, transported, and placed in ICDF.

One of the attributes of the construction of the primary coolant tank that lends itself to being lifted intact is that it was constructed to hang from the tank support structure. In fact, the tank is currently hanging from this support structure. As shown in Figure 2-8, the support structure is constructed from six radial I beams (spokes) and the tank is connected to each of the spokes by a hanger. This arrangement would allow hoisting equipment to be attached to the support structure and the entire primary coolant tank and internals lifted and placed on a transport trailer.

The weight of the primary coolant tank with the grout inside to reduce the radiological dose rates would be in approximately 1,500 tons (3,000,000 lb). Lifts of this magnitude have been accomplished using a system of "strand jacks." These strand jacks are diesel-powered hydraulic jacks that are mounted on top of towers assembled to the appropriate height; in this case, the primary coolant tank would need to be picked up nearly 45 ft in the air to allow placement on the transport vehicle. The primary coolant tank bottom is 27 ft in diameter; however, with the support structure being utilized as the lifting fixture, the top would be 40.5 ft across.

Although lifting the tank is possible, it is very complicated and would require an extensive amount of preparations to accomplish the lift. Preparations would include:

- Many piping and electrical (instrumentation) connections would have to be removed from the top of the primary coolant tank. Some of the lines leading from the tank are contaminated lines, so much of this work would be accomplished by workers in full contamination control gear, including respirators.

- A 3-ft, 3-in.-thick cap of concrete shielding that was poured in place completely around the top of the primary coolant tank cover, after the tank was installed, would have to be removed using heavy equipment. The reactor building dome would also be removed at this time.

- Preparing pads for the strand jacks towers to sit on. A series of four to six towers would be needed, and a highly compacted level area would have to be created for each tower.

- The strand jack towers and support equipment to the MFC would need to be mobilized to MFC. Estimates by the strand jack vendors put the semitruck flatbed loads that would need to bring the equipment to MFC at around 100 loads. This equipment would have to be staged near the job site. Currently, there is no area large enough inside the MFC security area to put this equipment.

- Installation of the strand jack system involves lifting sections of the strand jack tower using a large crane. Workers using fall protection would have to fasten these towers together. Once all towers have been installed, massive beams connecting the tops of the towers would be installed along with the lifting strand jack hydraulic units and cables.

- Rigging would then be put in place between the lifting mechanism and the primary coolant tank support structure. The load would then be ready to be lifted.

There are inherent risks for workers and property when lifts of this size and weight are performed. The location of the lift of the primary coolant tank (load) is close in to two large buildings. Each of these buildings has sensitive equipment and, in the case of MFC-765 that sits to the east of the lift site, this equipment is storing and handling nuclear materials. The other building immediately to the south of the lift is MFC-768, which houses all of the high-voltage electrical switch gear that supplies the MFC with power. Therefore, any problem with the lift could result in catastrophic results for the operating ability and infrastructure of MFC.

Once the primary coolant tank (load) is in the air, it would be transported via a rail system to the waiting transport trailer and lowered onto a multiaxeled trailer. Multiple issues would have to be addressed prior to transporting a load of this magnitude the 25 miles to the ICDF. Some of the issues are:

- A load of 1,500 tons would require a transport trailer 30 ft wide and 100 ft in length to carry the load. Trailers of this size are available and can be hauled by special tractor trucks.

- A significant construction effort would be needed to get the load from the lift site to the paved road leading from MFC. Hundreds of loads of road base material would have to be hauled, placed, and compacted to make a road that would be suitable for a load of this size and weight. The MFC security gate (North Side MFC) would need to be widened.

- Approximately 18 miles of the highway between MFC and ICDF are on U.S. Highway 20, a two-lane road that is a significant access route between outlying communities and Idaho Falls. These communities rely on this road to reach medical services and shopping. The transport of the load down this road would occur at night and be closed for approximately 8 hours. However, if any difficulties arise during moving the load, the road would be blocked until the issue(s) is resolved.

- A transport trailer 30 ft wide and 100 ft long exceeds the current design of the highway. Turns in the road where the MFC road enters Highway 20 and where the road to ICDF leaves Highway 20 would need to be widened. There are also turns at the INL Central Facilities Area and ICDF that would require modifications.

- The load that is being transported would rise to approximately 45 ft in the air, requiring electrical transmission lines and communication lines on the INL to be disconnected and dropped so the load could pass through.

- Once the load arrives at ICDF, significant modifications would be needed to the roads, including the access road into the disposal cell. The road into the disposal cell would have to be widened and made less steep.

- Prior to placing the 1,500-ton load into the disposal cell, a large load-bearing structure would be needed to placed on the cell surface to spread the load to prevent damage to the underlying disposal cell liners.

To place the primary coolant tank into the disposal cell, the strand jacking system that was utilized to lift the load at MFC would be mobilized to the ICDF (up to 100 flatbed trailers) and reassembled, and the load would be picked as before and placed on the assembled load-bearing structure. Since the load stands 38.5 ft tall, it would exceed the design height of the waste level in the disposal cell. Either another disposal cell, including leachate collection and monitoring, would have to be constructed or the current disposal cell cap would need to be raised approximately 12.5 ft in height.

All of the activities described above are believed to be technically feasible. A significant design and analysis effort would be needed to support most of these activities, and alternate ways to perform the activity may exist. However, the lifting, transport, and placement of the EBR-II reactor into a disposal facility will be a monumental task involving significant technical risk. Additional detail of the above Alternative 4 approach is provided in EDF-9455, "Technical Feasibility for the Alternative 4 EBR-II Engineering Evaluation/Cost Analysis."

Removing the EBR-II reactor vessel presents challenges that have not been encountered at the INL Site previously. Although previous reactors, such as the Engineering Test Reactor and the Power Burst

Facility reactor, were hoisted and transported to the ICDF, neither of these reactors were constructed like EBR-II nor were their radiological dose rates as high. Furthermore, EBR-II would weigh 1,300 to 1,400 tons greater than these reactors and, as previously explained, presents challenges not heretofore seen in the ICP D&D program.

7.2.2 Availability of Equipment, Personnel, and Disposal Services

For both Alternatives 3 and 4, experienced D&D personnel are currently at MFC performing other D&D and RCRA closure activities and are available to perform the activities necessary to bring EBR-II to the selected final end state. Subcontractors that specialize in heavy lifts and transporting large heavy loads are available. Adequate industrial safety and radiological control personnel are trained and available to support the work activities.

Equipment to support removing the EBR-II building and grouting the EBR-II reactor vessel is either available at the INL Site or is commercially available. Strand jack systems and heavy haul trailers are available. Special approvals to use a specialty transport trailer like the one described in the preceding section would be required from the Idaho Department of Transportation to allow transport on Highway 20.

On-Site or off-Site disposal or recycling services are available for wastes generated during completion of Alternative 3 and Alternative 4. Demolition debris that is not hazardous waste and is nonradiologically contaminated and that meets the requirements of the Idaho Solid Waste Management Rules for Tier II landfills (IDAPA 58.01.06.012) will be dispositioned at the MFC CERCLA Demolition Waste Landfill or another appropriate solid waste disposal facility.

7.3 Cost of the Alternatives

Detailed cost estimates have been prepared for Alternatives 2, 3, and 4. No costs are associated with Alternative 1, No Action, because the EBR-II reactor building would be simply left to degrade.

The estimates were prepared in accordance with *A Guide to Developing and Documenting Cost Estimates During the Feasibility Study* (EPA 2000). Costs are calculated for both D&D costs and future surveillance and maintenance expenses. In accordance with EPA guidance, the cost for the alternatives over time is calculated as present net worth costs, which are the costs in 2009 dollars.

Alternative 2 assumes maintenance of the EBR-II reactor building over an 85-year period ending in 2095. Alternative 2 would require ongoing surveillance, including routine radiological inspections and instrument checks. Maintenance includes facility repairs, maintaining the ventilation systems and heat, and periodic repainting of the EBR-II reactor building. Surveillance and maintenance costs for Alternative 2 would likely go beyond the institutional control period for an indeterminable period of time; therefore, CY 2095 was used for comparative purposes. For Alternative 3 there would be some surveillance and maintenance costs on the concrete monolith, these costs are estimated to be approximately $60,000 over the 85-year planning window and, like Alternative 2, would go beyond the institution control period used for estimating the cost. The information in the cost-estimate summary presented in Table 7-1 is based on the best available information. Changes in the cost elements are likely to occur as a result of new information and data collected during the performance of the removal action. Major changes will be documented in the form of a memorandum placed into the Administrative Record file. The cost estimate in Table 7-1 for each alternative is an order-of-magnitude engineering cost estimate that is expected to be within +50 to -30% of actual project cost. The cost estimate summary is presented in Table 7-1.

Table 7-1. Cost estimates for the viable removal action alternatives in 2009 dollars.

Cost Element	Alternative 2 No Action: Continued Surveillance and Maintenance	Alternative 3 Grouting EBR-II Reactor Vessel in Place	Alternative 4 Removal and Disposal of EBR-II Reactor Vessel at ICDF
D&D cost	$0.0 M	$15.4 M	$45.4 M
Surveillance and maintenance cost	$5.3 M (based on an annual average cost of $62.5 K)	$0.06 M	$0.0 M
Total estimated cost of the alternative	$5.3 M	$15.46 M	$45.4 M

7.4 Summary of Alternative Evaluation

No action alternatives are hypothetical, conservative, baseline assumptions that offer no reduction in toxicity, mobility, or volume of contaminants in that the sum of all identified chemical and/or radiological contamination, if not properly contained or controlled, may be released to the environment, causing an unacceptable risk to potential receptors. These baseline assumptions are for comparative purposes only and do not reflect DOE's mandate to monitor, maintain, and mitigate potential or actual hazardous or radiological constituent releases to the public or the environment from any facility or site.

Alternative 3 grouts the EBR-II reactor vessel in place, is protective of human health and the environment, and is more protective than Alternative 4 for workers, providing fewer and potentially less-consequential industrial hazards and radiological exposures. Alternative 3 costs are about one-third of the cost of Alternative 4.

Alternative 4 completely removes the EBR-II reactor vessel and leaves very little contamination at the EBR-II site, so that it is more protective for human health and the environment than Alternative 3. Alternative 4, with its complexities of dealing with multiple heavy-component lifts and the potential of high-radiological exposure, is less protective of workers than Alternative 3 and costs more.

8. RECOMMENDED REMOVAL ACTION ALTERNATIVE

This section discusses in detail the NTCRA alternative recommended in this EE/CA, presents the ARARs to which the removal action must conform, and discusses how the preferred alternative will meet the removal action objectives.

One of the primary functions of this EE/CA is to evaluate and compare proposed alternatives for risk to human health and the environment and the risk to workers performing the activities. As discussed in Section 6.1, both Alternative 3 and Alternative 4 are shown to be within the acceptable risk of less than 1 excess cancer risk in 10,000 people (less than 1E-04); therefore, worker risk is a discriminating factor in choosing the recommended end-state alternative. As stated in the Section 6, removal action objectives should be achieved "…to as practicable extent as possible in consideration of ALARA principles governing radiological exposure to decommissioning personnel, safe engineering standards . . . ." Alternative 4, removing the reactor vessel, is shown to have many more hazards of greater magnitude and consequence, including exposure of workers to radiation, than leaving the reactor vessel in place under Alternative 3.

The recommended removal action alternative is Alternative 3. Under Alternative 3, systems and structures above the reactor building floor would be demolished and most of the remaining systems and structures below floor level, including the EBR-II reactor vessel, would be grouted in place. The final end state of EBR-II under Alternative 3 is a concrete/grout monolith that contains the EBR-II primary coolant tank with internal components, including the reactor vessel. Void spaces remaining would be grouted as practicable, including the basement, sub-basement, and the interior of the primary coolant tank, resulting in encapsulation of the reactor vessel. The concrete/grout monolith will extend approximately 8 ft above ground level and will be finished with a concrete cover to facilitate drainage away from the site. Residual radioactive materials at EBR-II remaining after D&D activities are completed would stay in place and be managed under the Site-Wide Long-term Management and Control Program. Figure 8-1 is a current photo of EBR-II reactor building, and Figure 8-2 is a conceptual final end state for Alternative 3.

The recommended alternative meets the proposed removal action objectives for human health and environmental protectiveness and is cost-effective. It is also consistent with the remedial action objectives of the OU 9-04 ROD (DOE-CH 1998);, is compliant with ARARs;, and satisfies the DOE goal of reducing the "risk footprint" to as practicable extent as possible in consideration of (a) the principles of keeping exposures of decommissioning personnel to radiological hazards at ALARA levels, (b) safe engineering standards, (c) ICDF WAC, and (d) desired CERCLA site end states.

8.1 Compliance with Environmental Regulations

CERCLA (42 USC § 9621) requires the responsible CERCLA implementing agency to ensure that the substantive standards of HWMA/RCRA and other applicable laws will be incorporated into the federal agency's design and operation of its long-term remedial actions and into its more immediate removal actions. DOE-ID is the implementing agency for this NTCRA. Both DEQ and EPA concur that an NTCRA is warranted to protect human health and the environment. Through the NTCRA process, the risks presented in this document will be mitigated in a timely manner.

Figure 8-1. Current photo of EBR-II at MFC.

Figure 8-2. Conceptual end state of EBR-II under Alternative 3.

Table 8-1 lists the proposed ARARs that have been identified for this removal action. The ARARs list is based on several key assumptions:

- RCRA closure of the sodium-containing systems in the MFC-767 reactor building will be completed in compliance with the *HWMA/RCRA Storage and Treatment Permit for the Experimental Breeder Reactor-II and Buildings MFC-793E and MFC-793F located at the Materials and Fuels Complex on the Idaho National Laboratory* (DEQ 2009).

- Lead shielding will be removed from the MFC-767 reactor building prior to initiation or during this removal action through other regulatory activities intended to place the facility in an environmentally safe condition. Some lead, such as impractical-to-remove lead incidental to demolition (for example wall anchors), may remain in place or may be managed under the scope of the NTCRA as CERCLA waste and will be disposed of in the ICDF in accordance with WAC. Removed lead that cannot be recycled or reclaimed shall be declared a hazardous waste or mixed low-level waste, will be managed in accordance with the substantive requirements of the HWMA/RCRA, and will be disposed of at an off-Site disposal facility in accordance with the disposal facility WAC.

- Management of CERCLA waste generated during the removal action will be subject to meeting the ICDF WAC (DOE-ID 2009b).

- If decontamination liquids are generated, they will be disposed of at the ICDF evaporation ponds in accordance with the approved WAC. Small amounts of decontamination liquid may be solidified with absorbent and be disposed of in the disposal cells at the ICDF.

- Debris generated during this removal action may be covered with paint that contains polychlorinated biphenyls (PCBs). If encountered, such waste may trigger substantive requirements of the Toxic Substances Control Act (15 USC § 2601 et seq.). Lead-contaminated paint also may be present on demolition debris and would be subject to the substantive requirements of RCRA hazardous waste regulations. Nonhazardous low-level waste will be disposed of at the ICDF. Waste that can be demonstrated to be nonhazardous and to contain no added radiological constituents will be eligible for disposal as solid waste at an approved on-Site solid waste disposal facility. Any PCB-containing electrical equipment, such as PCB-containing light ballasts or capacitors, will be removed and disposed of off-Site at an approved disposal facility.

- Asbestos-containing material, both friable and nonfriable, may be encountered incidental to performance of this NTCRA. Friable or regulated asbestos-containing material will be subject to specific asbestos regulations and will be acceptable for disposal at the ICDF and/or, if not radiologically contaminated, at an approved on-Site solid waste disposal. Regulated asbestos will be removed and disposed of as required by 40 CFR 61.150, "Standard for Waste Disposal for Manufacturing, Fabricating, Demolition, Renovation, and Spraying Operations." Undisturbed asbestos or asbestos found in high-radiation, high-contamination, and/or inaccessible locations may be left in place.

- Mercury located in mercury fluorescent lamps is planned for removal prior to this removal action under other regulatory activities intended to place the facility in an environmentally safe condition, as are the mercury-containing electrical switches and lights. No mercury at concentrations of regulatory concern is expected to be present in the building substructure at the start of the removal action.

Table 8-1. Summary of applicable or relevant and appropriate requirements for the EBR-II non-time-critical removal action.

Requirement (Citation)	ARAR Type	Comments
Clean Air Act and Idaho Air Regulations		
"Toxic Substances," IDAPA 58.01.01.161 "Toxic Air Pollutants, Non-Carcinogenic Increments," IDAPA 58.01.01.585 "Toxic Air Pollutants, Carcinogenic Increments," IDAPA 58.01.01.586 "Environmental Remediation Source," IDAPA 58.01.01.210.16(a)	Applicable requirement	Applies to any toxic substances emitting during implementation of the removal action.
<10 mrem/yr, "Standard," 40 CFR 61.92	Applicable requirement	Applies to the waste-handling activities.
"Emission Monitoring and Test Procedures," 40 CFR 61.93	Applicable requirement	Applies to the waste-handling activities.
"Compliance and Reporting," 40 CFR 61.94(a)	Applicable requirement	Applies to the waste-handling activities.
"Standard for Demolition and Renovation," 40 CFR 61.145	Applicable requirement	Applies to any asbestos-containing materials removed during the decommissioning.
"Rules for Control of Fugitive Dust" and "General Rules," IDAPA 58.01.01.650 and IDAPA 58.01.01.651	Applicable requirement	Applies to the waste-handling activities.
Idaho Solid Waste Facilities Act		
"Applicable Requirements for Tier II Facilities," IDAPA 58.01.06.012	Applicable requirement	Applies to disposal of solid wastes.
RCRA and Idaho Hazardous Waste Management Act		
Generator Standards:		
"Standards Applicable to Generators of Hazardous Waste," IDAPA 58.01.05.006, and the following, as cited in it:		
"Hazardous Waste Determination," 40 CFR 262.11	Applicable requirement	Applies to waste that would be generated during the removal action.

Table 8-1. (continued).

Requirement (Citation)	ARAR Type	Comments
General Facility Standards:		
"Standards for Owners and Operators of Hazardous Waste Treatment, Storage and Disposal Facilities," IDAPA 58.01.05.008, and the following, as cited in it:		
"Temporary Units (TU)," 40 CFR 264.553	Applicable requirement	Waste may be treated or temporarily stored in a temporary unit prior to disposal.
"Staging Piles," 40 CFR 264.554	Applicable requirement	Waste may be temporarily staged prior to disposal.
"General Inspection Requirements," 40 CFR 264.15	Applicable requirement	Applies to a facility staging, storing, or treating hazardous waste prior to transfer to the ICDF or an off-Site facility.
"Preparedness and Prevention," 40 CFR 264, Subpart C	Applicable requirement	Applies to a facility staging, storing, or treating hazardous waste prior to transfer to the ICDF or an off-Site facility.
"Contingency Plan and Emergency Procedures," 40 CFR 264, Subpart D	Applicable requirement	Applies to a facility staging, storing, or treating hazardous waste prior to transfer to the ICDF or an off-Site facility.
"Disposal or Decontamination of Equipment, Structures, and Soils," 40 CFR 264.114	Applicable requirement	Applies to contaminated equipment used to remove, treat, or transport hazardous waste.
"Use and Management of Containers," 40 CFR 264.171-178	Applicable requirement	Applies to containers used during the removal and treatment of hazardous waste.
Land Disposal Restrictions:		
"Land Disposal Restrictions," IDAPA 58.01.05.011, and the following, as cited in it:		
"Applicability of Treatment Standards," 40 CFR 268.40(a)(b)(e)	Applicable requirement	Applies to hazardous waste and secondary waste, if treatment is necessary to meet the disposal facility's WAC or if treatment is required before placement.
"Treatment Standards for Hazardous Debris," 40 CFR 268.45	Applicable requirement	Applies to hazardous debris, if treatment is necessary to meet the disposal facility's WAC or if treatment is required before placement.
"Universal Treatment Standards," 40 CFR 268.48(a)	Applicable requirement	Applies to nondebris hazardous waste and secondary waste, if treatment is necessary to meet the disposal facility's WAC or if treatment is required before placement.
"Standards for Universal Waste Management," IDAPA 58.01.05.016		
"Standards for Large Quantity Handlers of Universal Waste," 40 CFR 273, Subpart C	Applicable requirement	Applies to management of universal wastes.

Table 8-1. (continued).

Requirement (Citation)	ARAR Type	Comments
Idaho Groundwater Quality Rules		
"Ground Water Quality Rule," IDAPA 58.01.11	Applicable requirement	The waste-handling activities must prevent migration of contaminants from the EBR-II reactor building that would cause the SRPA groundwater to exceed applicable State of Idaho groundwater quality standards in 2095 and beyond.
TSCA		
"Polychlorinated Biphenyls (PCBs) Manufacturing, Processing, Distribution in Commerce, and Use Prohibitions," 40 CFR 761	Applicable requirement	Applicable to removal, decontamination, storage, and disposal of items (including equipment) with PCB contamination.
Migratory Bird Treaty Act of 1918		
"Protection of Migratory Game and Insectivorous Birds," 16 USC 7	Applicable requirement	Applies to disturbances of nesting migratory birds.
To-Be-Considered Requirements		
"Radiation Protection of the Public and the Environment," DOE O 5400.5 Chg 2, Chapter II(1)(a,b)	TBC	Applies to the EBR-II reactor building before, during, and after the removal action. Substantive design and construction requirements would be met to keep public exposures as-low-as-reasonably-achievable.
"Radioactive Waste Management," DOE O 435.1 Chg 1	TBC	Applies to the EBR-II reactor building before, during, and after the removal action. Substantive design and construction requirements would be met to protect workers.
Region 10 Final Policy on the Use of Institutional Controls at Federal Facilities (EPA 2006)	TBC	Applies to residual waste following completion of the removal action.

ARAR	applicable or relevant and appropriate requirement	ICDF	Idaho CERCLA Disposal Facility	TBC	to be considered
CFR	Code of Federal Regulations	IDAPA	Idaho Administrative Procedures Act	TSCA	Toxic Substances Control Act
DOE	U.S. Department of Energy	PCB	polychlorinated biphenyl	USC	United States Code
EBR-II	Experimental Breeder Reactor II	RCRA	Resource Conservation and Recovery Act	WAC	waste acceptance criteria
EPA	U.S. Environmental Protection Agency	SRPA	Snake River Plain Aquifer		

8.2 Cultural Resources

Section 106 of the National Historic Preservation Act of 1966 (16 USC § 470 et seq.), as amended, requires agencies to consider the impact of undertakings on properties listed or eligible for listing in the National Register of Historic Places and to consult with the Idaho State Historic Preservation Officer and other interested parties when impacts are likely. It also requires federal agencies to invite the Advisory Council on Historic Preservation to participate in consultation when impacts may be adverse. The National Historic Preservation Act, Section 106, process has been tailored to meet the unique needs of the INL Site. Section 110 of the National Historic Preservation Act directs federal agencies to establish programs to find, evaluate, and nominate eligible properties to the National Register of Historic Places, including previously unidentified historic properties that may be discovered during the implementation of a project (36 CFR 800). In addition, the "Archaeological Resources Protection Act of 1979" (16 USC § 470aa–470mm), as amended, provides for the protection and management of archaeological resources on federal lands.

Procedures and strategies to tailor these requirements to the unique needs of the INL Site are described in the *Idaho National Laboratory Cultural Resource Management Plan* (DOE-ID 2007). The INL Cultural Resource Management Plan is implemented through a programmatic agreement between DOE-ID, the Idaho State Historic Preservation Officer, and the Advisory Council on Historic Preservation (DOE-ID 2007).

The MFC-767 EBR-II reactor building is a Category 1 historic property, eligible for nomination to the National Register of Historic Places. DOE-ID has made the decision to proceed with demolition of the facility. To mitigate the adverse impacts caused by such action, DOE-ID, through measures outlined in the INL Cultural Resource Management Plan and by the 2005 *Memorandum of Agreement Between the United States Department of Energy, Idaho Operations Office, and the Idaho State Historic Preservation Office* (DOE-ID 2005) and the 2004 Programmatic Agreement (contained in DOE-ID 2007c), has committed to the preservation of the MFC-767 building and reactor history through the completion of a Historic American Engineering Record for the facility and large-format photographs of the facility. A letter advising the State Historic Preservation Office of DOE-ID's preferred alternative for demolition of the EBR-II reactor building and vessel disposition was transmitted on October 22, 2009, and outlines the planned mitigation actions for the preferred alternative (Gallegos 2009).

8.3 Natural Resources

DOE-ID was required to review as guidance the most current U.S. Fish and Wildlife Service list for threatened and endangered plant and animal species. DOE-ID determined that none of the alternatives would impact any threatened and endangered species and also determined that formal consultation with the U.S. Fish and Wildlife Service was not required for this action.

8.4 Compliance with Disposal Facility Waste Acceptance Criteria

8.4.1 ICDF Waste Acceptance Criteria

ICDF is an on-Site disposal facility that accepts CERCLA waste generated at the INL Site. EBR-II low-level waste meets the WAC for disposal at ICDF, including the criteria not to exceed 10 nCi/g of transuranic activity.

Grout will be added as needed to waste disposed of at ICDF to reduce any void spaces to prevent subsidence in the disposal cell.

8.4.2 Achieving Removal Action Objectives

The recommended alternative meets the proposed removal action objectives for human health and environmental protectiveness and is cost-effective. It also is consistent with the remedial action objectives of the OU 9-04 ROD (DOE-CH 1998); is compliant with ARARs; and satisfies the DOE goal of reducing the "risk footprint" to as practicable an extent as possible in consideration of (a) ALARA principles governing radiological exposure to decommissioning personnel, (b) safe engineering standards, (c) ICDF WAC, and (d) desired CERCLA site end states.

9. REFERENCES

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Appendix A

EBR-II Facility Radiological Inventory (2009)

Appendix A

EBR-II Facility Radiological Inventory (2009)

Table A-1. EBR-II current (2009) radiological inventory.

Radionuclide (2009 inventory)	Total Inventory (GW Risk Alternative 1) (Ci)	Above Ground Level (HH & ECO Alternative 1) (Ci)	Below Ground Level (GW Risk Alternative 3) (Ci)	0 to 10 ft Below Ground Level (HH & ECO Alternatives 1 & 3) (Ci)	0 to 10 ft Below Ground Level (HH & ECO Alternative 4) (Ci)	Below Ground Level (GW Risk Alternative 4) (Ci)
H-3	1.74E+01	1.47E-03	1.74E+01	2.62E-03	1.66E-04	2.84E-01
Be-10	4.67E-04		4.67E-04			
C-14	8.38E+00	1.31E-06	8.38E+00			7.55E-02
Na-22	8.14E-05	1.61E-06	7.98E-05	2.88E-06	1.82E-07	
Cl-36	1.72E-01	2.80E-08	1.72E-01			3.88E-05
Ca-41	1.46E-03	2.40E-10	1.46E-03			2.40E-10
Mn-53	8.41E-04	1.38E-10	8.41E-04			1.38E-10
Mn-54	2.38E-02	1.25E-09	2.38E-02	8.05E-03	1.54E-16	1.25E-09
Fe-55	1.46E+03	2.39E-04	1.46E+03			2.39E-04
Ni-59	5.37E+01	8.59E-06	5.37E+01			3.80E-01
Co-60	7.43E+03	9.18E-04	7.43E+03			5.98E+02
Ni-63	6.02E+03	9.73E-04	6.02E+03			2.52E+01
Zn-65	2.10E-05	3.44E-12	2.10E-05			3.44E-12
Se-79	1.46E-04	2.40E-11	1.46E-04			2.40E-11
Sr-90	1.12E+01	2.04E-06	1.12E+01	5.54E+00	2.30E-07	5.03E-09
Nb-92m	2.44E-07	4.00E-14	2.44E-07			4.00E-14
Zr-93	1.05E-05	1.72E-12	1.05E-05			1.72E-12

Table A-1. (continued).

Radionuclide (2009 inventory)	Total Inventory (GW Risk Alternative 1) (Ci)	Above Ground Level (HH & ECO Alternative 1) (Ci)	Below Ground Level (GW Risk Alternative 3) (Ci)	0 to 10 ft Below Ground Level (HH & ECO Alternatives 1 & 3) (Ci)	0 to 10 ft Below Ground Level (HH & ECO Alternative 4) (Ci)	Below Ground Level (GW Risk Alternative 4) (Ci)
Mo-93	1.34E-01	2.19E-08	1.34E-01			2.93E-08
Nb-94	1.33E-01	1.50E-08	1.33E-01			1.22E-02
Tc-99	1.33E-01	4.80E-09	1.33E-01			3.17E-02
Ru-106	8.96E-07		8.96E-07			
Ag-108m	3.26E-02	4.29E-09	3.26E-02			4.29E-09
Ag-110m	6.25E-08	2.75E-17	6.25E-08	1.25E-17	3.12E-18	
Sn-121m	1.06E-03		1.06E-03			
Sb-125	3.84E-04	1.93E-07	3.84E-04	8.75E-08	2.19E-08	
I-129	1.97E-08	2.20E-15	1.97E-08			2.20E-15
Ba-133	3.23E-01	5.29E-08	3.23E-01			5.29E-08
Cs-134	5.92E-02	8.29E-09	5.92E-02	2.40E-09	6.01E-10	2.99E-09
Cs-135	9.51E-07	1.56E-13	9.51E-07			1.56E-13
Cs-137	6.29E-02	3.76E-04	6.26E-02	1.70E-04	4.25E-05	5.95E-09
Ce-144	4.38E-08		4.38E-08			
Pm-145	2.10E-04	3.44E-11	2.10E-04			3.44E-11
Sm-146	5.24E-11		5.24E-11			
Sm-151	6.41E-02	1.05E-08	6.41E-02			1.05E-08
Eu-152	4.36E+00	7.04E-07	4.36E+00			7.04E-07
Eu-154	6.53E-01	7.15E-08	6.53E-01			7.15E-08
Eu-155	7.14E-03	1.17E-09	7.14E-03			1.17E-09
Tb-158	6.45E-04		6.45E-04			

Table A-1. (continued).

Radionuclide (2009 inventory)	Total Inventory (GW Risk Alternative 1) (Ci)	Above Ground Level (HH & ECO Alternative 1) (Ci)	Below Ground Level (GW Risk Alternative 3) (Ci)	0 to 10 ft Below Ground Level (HH & ECO Alternatives 1 & 3) (Ci)	0 to 10 ft Below Ground Level (HH & ECO Alternative 4) (Ci)	Below Ground Level (GW Risk Alternative 4) (Ci)
Ho-166m	3.63E-02	5.95E-09	3.63E-02			5.59E-09
Hf-178m	1.05E-01		1.05E-01			
Pb-205	4.15E-07	6.80E-14	4.15E-07			6.80E-14
Pb-210	1.03E-12		1.03E-12			
Ra-226	1.41E-12		1.41E-12			
Ac-227	2.40E-08		2.40E-08			
Th-228	9.37E-04		9.37E-04			
Th-229	1.82E-08		1.82E-08			
Th-230	1.16E-10		1.16E-10			
Pa-231	1.69E-08		1.69E-08			
Th-232	4.31E-09		4.31E-09			6.80E-11
U-232	7.78E-07		7.78E-07			
U-233	4.19E-04	6.80E-11	4.19E-04			
U-234	2.20E-07		2.20E-07			
U-235	2.96E-05		2.96E-05			
U-236	1.36E-08		1.36E-08			
Np-237	1.44E-08		1.44E-08			
U-238	4.04E-02	1.27E-02	2.77E-02	1.11E-02	1.11E-02	2.77E-02
Pu-238	1.36E-04		1.36E-04			
Pu-239	1.22E-02		1.22E-02	4.42E-09	1.10E-09	1.92E-09
Pu-240	2.74E-04		2.74E-04			

Table A-1. (continued).

Radionuclide (2009 inventory)	Total Inventory (GW Risk Alternative 1) (Ci)	Above Ground Level (HH & ECO Alternative 1) (Ci)	Below Ground Level (GW Risk Alternative 3) (Ci)	0 to 10 ft Below Ground Level (HH & ECO Alternatives 1 & 3) (Ci)	0 to 10 ft Below Ground Level (HH & ECO Alternative 4) (Ci)	Below Ground Level (GW Risk Alternative 4) (Ci)
Pu-241	1.49E-02		1.49E-02			
Pu-242	1.23E-07		1.23E-07			
Pu-244	4.08E-16		4.08E-16			
Am-241	1.06E-03		1.06E-03			
Am-243	1.78E-07		1.78E-07			
Cm-243	2.45E-07		2.45E-07			
Cm-244	2.93E-06		2.93E-06			
Cm-245	1.03E-10		1.03E-10			
Cm-246	7.30E-12		7.30E-12			
Cm-247	3.98E-18		3.98E-18			
Cm-248	1.99E-18		1.99E-18			
Total	1.50E+04	1.67E-02	1.50E+04	5.56E+00	1.13E-02	6.24E+02

Note: Source of radiological inventory is TBL-194, 2009, "EBR-II Pre-Demolition Source Term," Rev. 0, Idaho Cleanup Project, May 2009.

ECO ecological
GW groundwater
HH human health

Appendix B

EBR-II Nonradiolgical Inventory

Appendix B

EBR-II Nonradiolgical Inventory

Table B-1. EBR-II nonradiological inventory.[a]

Nonradiological Constituent	Location(s)	Total Inventory (GW Risk Alternative 1)	Below-Ground-Level Inventory (GW Risk Alternative 3)	Below-Ground-Level Inventory (GW Risk Alternative 4)	Total Above Ground Level to 10 ft Below Ground Level (HH & ECO Alternative 1)	7 ft Above Ground Level to 10 ft Below Ground Level (HH & ECO Alternative 3)	Ground Level to 10 ft Below Ground Level (HH & ECO Alternative 4)
Aluminum	Ventilation ducting	15,110	14,230	13,000	2,110	0	0
Antimony	Wall anchors	3	1	2	2	0	0
Boron	Neutron moderator around reactor vessel	900	900	0	0	0	0
Chromium (Cr-111)	Component of stainless steel in reactor and structural components	120,563	114,929	513	33,592	35,643	95
Copper	Wiring and piping	53,950	34,500	12,550	41,800	400	400
Lead	Wall anchors	30	21	21	0	0	0
Manganese	Component of carbon and stainless steel	50,292	34,410	2,409	38,413	4,966	1,224
Nickel	Component of carbon and stainless steel	83,415	72,329	1,520	36,195	19,404	694

Table B-1. (continued).

Nonradiological Constituent	Location(s)	Total Inventory (GW Risk Alternative 1)	Below-Ground-Level Inventory (GW Risk Alternative 3)	Below-Ground-Level Inventory (GW Risk Alternative 4)	Total Above Ground Level to 10 ft Below Ground Level (HH & ECO Alternative 1)	7 ft Above Ground Level to 10 ft Below Ground Level (HH & ECO Alternative 3)	Ground Level to 10 ft Below Ground Level (HH & ECO Alternative 4)
Uranium	Depleted uranium shielding outside of primary coolant tank	265	182	182	73	73	73
Zinc	Component of galvanized steel	2,758	182	1,308	1,640	160	160

a. All inventories are expressed in pounds.

ECO ecological
GW groundwater
HH human health

Appendix C

Ecological Screening Tables

Appendix C

Ecological Screening Tables

Table C-1. Initial ecological risk screening for EBR-II alternatives.

Contaminant	Soil Concentration (pCi/g or mg/g)	Background Concentration (mg/kg or pCi/g)	Maximum Concentration > Background?	Nontoxic Metal?	Screening Value (pCi/g)	Maximum Concentration > Screening Values?	Remains a Concern?
Alternative 1							
Aluminum	3.22E+03	1.60E+04	No	Yes	Not toxic	No	No
Antimony	6.39E-01	4.80E+00	No	No	2.70E-01	Yes	No
Boron	1.92E+02	NA	NA	No	5.00E-01	Yes	**Yes**
Chromium III	2.57E+04	NA	NA	No	2.60E+01	Yes	**Yes**
Copper	1.15E+04	2.20E+01	Yes	No	2.80E+01	Yes	**Yes**
Lead	6.39E+00	1.70E+01	No	No	1.10E+01	No	No
Manganese	1.07E+04	4.90E+02	Yes	No	2.20E+02	Yes	**Yes**
Nickel	1.78E+04	3.50E+01	Yes	No	3.80E+01	Yes	**Yes**
Uranium	5.65E+01	NA	NA	No	1.66E+00	Yes	**Yes**
Zinc	5.88E+02	1.50E+02	Yes	No	4.60E+01	Yes	**Yes**
H-3	2.59E-02	NA	NA	No	3.43E+05	No	No
C-14	6.08E-04	NA	NA	No	3.94E+04	No	No
Cl-36	1.31E-05	NA	NA	No	7.84E+03	No	No
Ca-41	1.12E-07	NA	NA	No	No EBSL	No EBSL	No EBSL
Mn-53	6.47E-08	NA	NA	No	2.25E+06	No	No
Ni-59	4.02E-03	NA	NA	No	1.24E+06	No	No
Co-60	6.02E-06	NA	NA	No	1.18E+03	No	No
Ni-63	2.53E-01	NA	NA	No	1.14E+05	No	No
Se-79	1.12E-08	NA	NA	No	No EBSL	No EBSL	No EBSL

Table C-1. (continued).

Contaminant	Soil Concentration (pCi/g or mg/g)	Background Concentration (mg/kg or pCi/g)	Maximum Concentration > Background?	Nontoxic Metal?	Screening Value (pCi/g)	Maximum Concentration > Screening Values?	Remains a Concern?
Sr-90	6.62E+02	4.90E-01	Yes	No	3.34E+03	No	No
Zr-93	8.06E-10	NA	NA	No	9.95E+04	No	No
Mo-93	1.01E-05	NA	NA	No	No EBSL	No EBSL	No EBSL
Nb-94	7.01E-06	NA	NA	No	1.87E+03	No	No
Tc-99	2.25E-06	NA	NA	No	1.60E+04	No	No
Ag-108m	1.26E-06	NA	NA	No	1.82E+03	No	No
Ba-133	9.07E-08	NA	NA	No	7.34E+03	No	No
Cs-137	4.76E-02	8.20E-01	No	No	4.95E+03	No	No
Pm-145	5.78E-10	NA	NA	No	No EBSL	No EBSL	No EBSL
Sm-151	2.56E-06	NA	NA	No	No EBSL	No EBSL	No EBSL
Eu-152	4.34E-06	NA	NA	No	2.18E+03	No	No
Eu-154	4.14E-08	NA	NA	No	2.48E+03	No	No
Ho-166m	2.65E-06	NA	NA	No	No EBSL	No EBSL	No EBSL
Bi-210	7.71E-09	NA	NA	No	5.10E+03	No	No
Pb-210	7.72E-09	NA	NA	No	2.74E+05	No	No
Po-210	7.53E-09	NA	NA	No	1.84E+01	No	No
Bi-214	1.83E-08	NA	NA	No	1.99E+03	No	No
Pb-214	1.83E-08	NA	NA	No	6.78E+03	No	No
Po-218	1.83E-08	NA	NA	No	1.62E+01	No	No
Rn-222	1.83E-08	NA	NA	No	1.78E+01	No	No
Ra-226	1.83E-08	NA	NA	No	2.04E+01	No	No
Th-230	1.50E-06	1.41E+00	No	No	2.09E+01	No	No
U-233	3.19E-08	NA	NA	No	2.03E+01	No	No
Pa-234	2.62E-02	NA	NA	No	No EBSL	No EBSL	No EBSL
Pa-234m	1.64E+01	NA	NA	No	2.37E+03	No	No

Table C-1. (continued).

Contaminant	Soil Concentration (pCi/g or mg/g)	Background Concentration (mg/kg or pCi/g)	Maximum Concentration > Background?	Nontoxic Metal?	Screening Value (pCi/g)	Maximum Concentration > Screening Values?	Remains a Concern?
Th-234	1.64E+01	NA	NA	No	4.16E+04	No	No
U-234	3.94E-03	1.44E+00	No	No	2.05E+01	No	No
U-238	1.64E+01	1.40E+00	Yes	No	2.32E+01	No	No
Pu-239	4.13E-06	1.00E-01	No	No	1.89E+01	No	No
Alternative 3							
Aluminum	0.00E+00	1.60E+04	No	Yes	Not toxic	No	No
Antimony	0.00E+00	4.80E+00	No	No	2.70E-01	No	No
Boron	0.00E+00	NA	NA	No	5.00E-01	No	No
Chromium III	7.59E+03	NA	NA	No	2.60E+01	Yes	**Yes**
Copper	8.52E+01	2.20E+01	Yes	No	2.80E+01	Yes	**Yes**
Lead	0.00E+00	1.70E+01	Yes	No	1.10E+01	Yes	No
Manganese	1.06E+03	4.90E+02	Yes	No	2.20E+02	Yes	**Yes**
Nickel	4.13E+03	3.50E+01	Yes	No	3.80E+01	Yes	**Yes**
Uranium	1.56E+01	NA	NA	No	1.66E+00	Yes	**Yes**
Zinc	3.41E+01	1.50E+02	No	No	4.60E+01	No	No
H-3	1.23E+00	NA	NA	No	3.43E+05	No	No
Na-22	1.35E-03	NA	NA	No	2.31E+03	No	No
Mn-54	3.77E+00	NA	NA	No	3.53E+03	No	No
Sr-90	2.60E+03	4.90E-01	Yes	No	3.34E+03	No	No
Ag-110m	5.86E-15	NA	NA	No	1.08E+03	No	No
Sb-125	4.10E-05	NA	NA	No	8.42E+04	No	No
Cs-134	1.12E-06	NA	NA	No	1.90E+03	No	No
Cs-137	7.97E-02	8.20E-01	No	No	4.95E+03	No	No
U-238	5.22E+00	1.40E+00	Yes	No	2.32E+01	No	No
Pu-239	2.07E-06	1.00E-01	No	No	1.89E+01	No	No

Table C-1. (continued).

Contaminant	Soil Concentration (pCi/g or mg/g)	Background Concentration (mg/kg or pCi/g)	Maximum Concentration > Background?	Nontoxic Metal?	Screening Value (pCi/g)	Maximum Concentration > Screening Values?	Remains a Concern?
Alternative 4							
Aluminum	0.00E+00	1.60E+04	No	Yes	Not toxic	No	No
Antimony	0.00E+00	4.80E+00	No	No	2.70E-01	No	No
Boron	0.00E+00	NA	NA	No	5.00E-01	Yes	No
Chromium III	2.02E+01	NA	NA	No	2.60E+01	No	No
Copper	8.52E+01	2.20E+01	Yes	No	2.80E+01	Yes	**Yes**
Lead	0.00E+00	1.70E+01	No	No	1.10E+01	No	No
Manganese	2.61E+02	4.90E+02	No	No	2.20E+02	Yes	No
Nickel	1.48E+02	3.50E+01	Yes	No	3.80E+01	Yes	**Yes**
Uranium	1.56E+01	NA	NA	No	1.66E+00	Yes	**Yes**
Zinc	3.41E+01	1.50E+02	No	No	4.60E+01	No	No
H-3	7.76E-02	NA	NA	No	3.43E+05	No	No
Na-22	8.53E-05	NA	NA	No	2.31E+03	No	No
Mn-54	7.23E-14	NA	NA	No	3.53E+03	No	No
Sr-90	1.08E-04	4.90E-01	No	No	3.34E+03	No	No
Ag-110m	1.46E-15	NA	NA	No	1.08E+03	No	No
Sb-125	1.03E-05	NA	NA	No	6.02E+03	No	No
Cs-134	2.81E-07	NA	NA	No	2.04E+05	No	No
Cs-137	1.99E-02	8.20E-01	No	No	4.95E+03	No	No
U-238	5.22E+00	1.40E+00	Yes	No	2.37E+01	No	No

EBSL ecologically based screening level
NA not applicable